3.7 SEISMIC DESIGN

3.7.1 Input Criteria

3.7.1.1 Design Response Spectra

Development of the response spectra for the Maximum Possible Earthquake (0.15g) and the Maximum Probable Earthquake (0.08g) is presented in Section 2C.3.1, Table 2C.3-4 through 2C.3-6, and Figure 2C.3-7 through 2C.3-9 in Appendix 2C.

The design response spectra for the Maximum Probable and Maximum Possible Earthquake for various damping ratios (0.0, 0.005, 0.01, 0.02, 0.05, and 0.10) are presented in Figures 2C.3-5 and 2C.3-6, respectively, in Appendix 2C. These spectra are applicable to the elevations of the foundations in the station facilities.

3.7.1.2 Design Accelerograms

Development of the accelerograms used in design of station facilities is presented in Section 2C.3.4(4) of Appendix 2C. The design accelerograms were digitalized at periodic intervals of 0.01 seconds; this interval was small enough to produce accurate response spectra; see Figures 2C.3-10 through 2C.3-14 in Appendix 2C. Comparison of the design accelerograms with the recommended response spectra (see Subsection 3.7.1.1) for the damping ratios employed in the design of station facilities (0.0, 0.005, 0.01. 0.02, and 0.05) is presented in Figures 2C.3-10 through 2C.3-14 in Appendix 2C. These figures show the design accelerograms are typically larger (more conservative) than the corresponding response spectra at each damping ratio for frequencies greater than 0.5 cps; frequencies smaller than 0.5 cps are insignificant in the design of structures and equipment.

3.7.1.3 Critical Damping Factors

The percentages of critical damping for analyzing structures, systems and components are shown in Table 3.7-1. The damping values shown above the dashed line in the table and as described in subsection 3.7.2.14 are those to which the plant has been licensed.

The damping values below the dashed line have been used since 1980. However, prior to 1980, damping values for items below the dashed lines were derived by comparison with the damping values for the items above the dashed line. Higher damping values than those listed in Table 3.7-1 are allowed, provided proper justification (i.e., test results, analysis etc.) is available for specific Components or equipment. Table 3.9-10 allows the use of ASME Code Case N-411 for piping analysis. Table 3.7-4 identifies plant systems where ASME Code Case N-411 alternate damping has been utilized in the piping analysis.

The Seismic Class II structures are designed in accordance with the seismic requirements of the Uniform Building Code, Reference 30.

3.7.1.4 Site Dependent Analyses

No site dependent analyses were used in the development of the design response spectra. Section 2C.3.4(4) of Appendix 2C presents a complete description of applicability and basis of selection of the design accelerograms and corresponding response spectra. All Seismic Class I structures are founded on bedrock, on a thin layer of dense fill deposit, or on Class I granular

backfill overlying bedrock. No magnification of spectra values was considered necessary for these materials.

3.7.1.5 Seismic Re-Evaluation

During the first fuel cycle, a seismic re-evaluation of the unit was performed using a maximum possible horizontal value of 0.20g ground acceleration in lieu of 0.15g. Section 2C.3.5 of Appendix 2C presents a description of the re-evaluation. The licensee continues to use the seismic design criteria presented in Section 3.7.1, and Section 2C.3.4 of Appendix 2C.

3.7.1.6 Major Soil Supported Structures

The following list contains the major Soil-Supported structures.

No.	Structure	Class	Max. Depth of Soil (Aprx.±2')
1	Service Water Pipe Tunnel	I	5'-0"
2	Service Water System Valve Room No. 1 (Turbine Building)	I	2'-0"
3	Service Water System Valve Room No. 2 (Intake Structure)	I	5'-0"
4	Borated Water Storage Tank	I	19'-0"
5	Diesel Oil Storage Tank	II	17'-0"
7	Fire Water Storage Tank	II	20'-0"
8	Other Yard Tanks	II	20-0"
9	Cooling Tower	II	15-0"
10	Main, Aux., Start-up Transformers	II	18'-0"
11	Auxiliary Boiler Foundation	II	22'-0"
12	Electrical Manholes (Large)	I, II	17'-0"
13	Open Channel	II	16'-0"
14	Open Channel Bridge	II	4'-0"
15	Transformers at Switchyard	II	13'-0"
16	Sewage Treatment Plant	II	14'-0"
17	Switchyard Foundations	II	20'-0"
18	Relay House	II	12'-0"

3.7.1.7 Soil Spring Mathematical Model

3.7.1.7.1 Borated Water Storage Tank

A mathematical model is used having vertical, horizontal, and rotational springs representing the soil dynamic values. The damping values associated with rotation, translation and welded steel tank are considered separately but combined for composite damping.

The horizontal and vertical spring constants used for the Class I structural backfill are 3.1195×10^6 kips/ft and 4.116×10^6 kips/ft, respectively. See Subsection 3.7.2.3 and Reference 27.

The cantilever beam-type mass model is used by connecting masses to the springs, defined by the material properties of the beam model (see Figure 3.7-1).

The dynamic analysis is performed as outlined in Subsection 3.7.2.1.

3.7.1.7.2 Auxiliary Building Area - 6

The foundation for area 6 of the Auxiliary Building consists of a system of beams and reinforced concrete columns extending 27 feet through Class I structural backfill to the rock surface. This area of the Auxiliary Building is analyzed using a mathematical model as shown in Figure 3.7-2.

In the mathematical model, the soil masses between the grade slab at El. 585-0 and the surface of the rock are lumped at three points as follows: mass point 1, 9 feet above bedrock; mass point 2, 18 feet above bedrock; and mass point 3, 27 feet above bedrock at the grade slab elevation. Lumped masses at these points are calculated, including the mass of the reinforced concrete columns as well as the soil. Mass points above the grade slab at El. 585-0 are located at floor levels as outlined in Subsection 3.7.2.3.

Translational soil springs are located at mass points 1, 2, and 3. The spring constants used are calculated based on the shear rigidity of the structural backfill as a function of soil depth.

In addition to the translational springs at the soil-structure mass points, a rotational spring is used at the top of the columns (grade slab El. 585-0) to account for the rotational stiffness of the column group about its centroid. The rotational spring constant is based on the axial deformation of each column as frictional forces between the column and the surrounding soil are developed.

The mathematical model, as described herein, is assumed free to rotate at the rock surface. Lateral stability is provided by the translational soil springs. (See Figure 3.7-2)

3.7.2 <u>Seismic System Analysis</u>

3.7.2.1 Method of Analysis

3.7.2.1.1 Structures, Systems, Components, and Equipment Other Than Reactor Coolant System

The seismic analysis of the containment and the other Class I structures and equipment is based on a structural dynamic analysis using a horizontal response spectrum normalized to 0.08g for the Maximum Probable Earthquake, and 0.15g for the Maximum Possible Earthquake.

Also, the vertical ground response spectra of 0.053g for the Maximum Probable Earthquake and 0.10g for the Maximum Possible Earthquake are used to analyze the containment and other Class I structures, systems, and components.

A dynamic analysis is performed on the Class I structures to determine their behavior during an earthquake (References 20 and 21). The analysis is accomplished in five steps. The first step consists of reducing a typical structure into a mathematical model in terms of lumped masses and stiffness coefficients. The second step is to obtain the natural frequencies and mode shapes of the model.

The third step is an evaluation process to determine the proper values of damping. The fourth step determines the resulting internal forces on the typical containment using the spectrum response curve for the earthquake. The fifth step is for internal equipment located at different levels and provides a description of the earthquake environment.

In building the mathematical model (Figures 3.7-1 through 3.7-7), the locations for lumped masses are chosen at floor levels and points considered to be of critical interest. Between mass points, the structural properties are reduced to uniform segments of cross-sectional area, effective shear area and moment of inertia. The foundations of the containment and auxiliary building area 7 and 8 are located on competent rock and, consequently, are represented in the model as fixed bases. Investigations indicate that the influence of translation and rotation on the rock is small and, therefore, can be neglected. In the auxiliary building area 6, the column foundations are coupled with the surrounding soil to form a soil-structure interaction model as described in Subsection 3.7.1.7. With this information, a computerized analysis is used to form the stiffness matrix, (K), of the structure. The masses are arranged into a mass matrix, (M). See Reference 26.

The natural frequencies and mode shapes are obtained by solution of the equation below:

$$(K)(X) = w^2 (M)(X)$$

The mode shapes are plotted and examined to determine how the structure is vibrating. All modes are considered with frequencies, w, less than 30 cycles per second.

The basic values for damping of specific materials and types of construction are presented in Table 3.7-1. These values are utilized after an evaluation of the mode shapes.

3.7.2.1.2 Reactor Coolant System

The major Reactor Coolant System components (reactor vessel, steam generators, piping, pumps, and pressurizer) are analyzed using a three-dimensional composite system model as described below. The composite model is shown in Figures 5.2-1 through 5.2-3.

Detailed stress analyses of the individual Reactor Coolant System components including the vessel, piping, pumps, steam generators, and pressurizer is performed for the design bases.

A three-dimensional seismic, thermal, and dead load analysis is performed on the primary coolant system to determine piping and component nozzle stresses. An idealized mathematical model of a single loop consisting of lumped masses connected by elastic member is employed. This loop includes a reactor, a steam generator, the pressurizer, two coolant pumps with associated piping and the secondary shield wall with the attached coolant system restraints or supports. Using the elastic properties of the piping and components, a reduced flexibility matrix is generated. All flexibility calculations include the effects of torsional, shearing, bending, and axial deformations as well as changes in flexibility due to curved members and internal pressure. Flexibility factors are calculated in accordance with ANSI B31.7.

The seismic analysis utilizes the normal mode, response spectra approach with the frequencies and mode shapes being calculated by the modified Jacobi technique. The inertia forces determined for each mode are supplied mathematically to the model and the resultant displacements nozzle reactions, internal forces, and moments are obtained by taking the Square-Root-of-the-Sum-of-the-Square (SRSS) of each mode's contribution.

Thermal expansion and dead load analyses are performed using the same mathematical model as used for the seismic analysis. At selected points within the loop the internal forces and moments from thermal expansion, dead loads, and seismic loadings are combined in accordance with ANSI B31.7, or in a more conservative manner. The resulting stresses are combined with those from pressure loading and thermal gradients. ANSI B31.7 stress indices

are used for all applicable points. In addition, the effects of pressure and thermal cycling are evaluated.

Independent thermal and dynamic analyses are performed to ensure that piping connecting to the Reactor Coolant System is of proper schedule and that it did not impose forces on the nozzles greater than allowable. Small nozzles are conservatively designed and utilize ASA Schedule 160.

3.7.2.2 Criteria for Seismic Analyses

Whenever the structural mode of flexure indicates major activity of a specific material, that material damping value is employed. For these modes which indicate activity of several materials, their damping values are utilized based on a weighting process using the mode shapes and masses as outlined in the Bechtel Seismic Topical Report BC-TOP-4 (Reference 29). Damping values are assigned to each mode.

The response spectrum curve of the earthquake is discussed in Subsections 3.7.2.5, 3.7.2.6 and 3.7.2.7. The values of this curve are utilized by the standard response spectrum techniques. Acceleration values are selected from the curve for each mode, based on damping and natural frequency. Effective force, shear, and moment diagrams are computed for each mode. The shears and moments of the individual modes are combined on a root-mean-square basis Reference 25.

The criteria used for combining modal responses for shears, momenta, stresses, deflections, and accelerations are as outlined below:

a. The “square-root-of-the-sum-of-the-squares” method is used if the difference between two consecutive frequencies of two modes is more than 10 percent.

b. The “sum-of-absolute-values” is used for closely spaced frequencies, i.e., if the frequencies differ by 10 percent or less the “sum-of-absolute-values” is used.

   When the frequencies are closely spaced, they are first divided into groups in such a way that in each group the deviation in frequency between the first and last mode does not exceed 10 percent of the lower frequency. The criterion of “the-sum-of-absolute-values” is then applied to each group, and the results from all the groups are then combined according to the criterion of “the-square-root-of-the-sum-of-the-squares.”

For internal equipment located at various elevations, the earthquake environment is specified by additional analysis. This is done on a separate basis as the numerous equipment items of small mass in comparison to the building cannot feasibly be incorporated into the model (Reference 31). For internal equipment which is rigid, having natural frequencies greater than 30 cps, the maximum acceleration to be expected is obtained from an acceleration diagram generated by the response spectrum technique. For flexible equipment having natural frequencies less than 30 cps, the environment is specified by response spectrum curves developed at various elevations and other points of attachments. The floor response spectrum curves are generated by an earthquake time history analysis of the building, using the modified Helena, Montana, earthquake of October 31, 1935, E.W. component normalized to 0.08g and 0.15g accelerations for the Maximum Probable and Maximum Possible Earthquake, respectively.

If the structure, systems, or equipment are structurally simple, the dynamic model consists of one mass and one spring. Using the values of the mass and the spring constant, the natural period of the equipment is determined. The natural period, together with the appropriate damping value, is used to enter the appropriate acceleration response spectrum to obtain the equipment acceleration in units of g's. The corresponding inertia force is obtained by multiplying the weight times the acceleration.

If the structure, systems, and equipment are structurally complex, to the extent that a single-degree-of-freedom-system model does not adequately represent the action of the structure to dynamic loads, then a multi-degree-of-freedom model is used with a complete multi-degree-of-freedom analysis, as outlined in the references. A sufficient number of modes are considered to adequately represent the response of the structure.

Under certain conditions, the natural frequency of the systems or equipment is not calculated. Under these conditions, using the appropriate damping value, the peak value of acceleration response curve or the values obtained from duplicate or dynamically similar systems which have been analyzed are used to calculate the inertia forces.

For the primary coolant system, each component was initially modeled on an isolated basis (e.g., reactor vessel, steam generator, hot leg, etc.). For these models, sufficient degrees of freedom were included to yield very accurate frequencies and mode shapes for the seismically excitable frequencies, i.e., less than 30 cps. The number of degrees of freedom for each model was then successively reduced to a point where the reduced model's mode shapes and lower frequencies differed by less than 10 percent from those of the more complex model. The reduced model that retained accuracy within the 10 percent error range was then used in the total system analysis.

The licensee has investigated the seismic effects of the vertical accelerations on the structures, components, and equipment. The vertical ground motion input to the model is two thirds of the horizontal ground motion. A multi-degree-of-freedom modal analysis is made to determine vertical displacements, velocities, accelerations, shears, and moments. The vertical stresses are added directly and linearly to those stresses produced by horizontal seismic analysis in combination with stresses caused by other concurrent loads.

3.7.2.3 Soil Base Model

The mathematical models for the Containment, Auxiliary Building, and Intake Structure are of fixed base. All of these Class I structures have foundation mats either on or in sound bedrock. The investigations indicate that the influence of translation and rotation of the rock is very small, and the response on the structure is negligible for this analysis. The comparative studies for the fixed base versus the soil spring model are outlined below Reference 28.

3.7.2.3.1 Method of Analysis

The method of seismic analysis for the comparison of fixed base versus soil spring models is outlined in Subsection 3.7.2.1.

3.7.2.3.2 Model

The soil springs model consists of the three containment structures - Shield Building, Containment Vessel, and internal structures - supported on a beam representing concrete fill inside and outside the ellipsoidal base. This beam is supported by two vertical "columns"

representing the rotational soil spring. It is supported laterally by a horizontal "column" which denotes the translational soil spring (see Figure 3.7-8). The mass of the concrete fill in the ellipsoidal base is input on the beam corresponding to it. The mathematical models above the base of the structures are the same as the fixed base models (see Figures 3.7-3 through 3.7-5). The maximum probable and maximum possible earthquakes are utilized for the analyses in this comparative study.

3.7.2.3.3 Comparison of Results

Modal participation factors and frequencies are compared for Models 1, 2, and 3 of the soil springs model to the first mode of the corresponding fixed base model. For the Shield Building, Containment Vessel, and internal structures, the frequencies decreased 2.3, 0.9, and 1.4 percent, respectively. Moments and shears near the base of the Shield Building and internal structures are increased by 6 and 9 percent, respectively, while they are decreased by 3 and 2 percent, respectively, for the Containment Vessel as shown in Table 3.7-2. The time history response for both the fixed base and soil spring models compared very closely. The absolute accelerations for the various elevations are all within approximately 1 percent as shown in Table 3.7-3.

A comparison of the floor response spectra generated for the lowest floor, a middle floor, and the top floor of the internal structures for the two models shows a very close agreement. The maximum difference occurred at the natural frequency of the structure. For the lowest floor the peak increase in acceleration is less than 5 percent; for the middle floor the peak acceleration increase is 16.6 percent, and for the top floor it is increased by 14.9 percent as shown in Table 3.7-3. However, as can be seen in Figures 3.7-9 through 3.7-11, these increases are highly localized at the building primary frequency, and acceleration values for all other points on the curves are more conservative for the fixed base model. It should also be pointed out that peak accelerations, which are clearly associated with the natural frequency of the building, are enveloped and widened as specified in the Bechtel Topical Report BC-TOP-4 on Seismic Analysis. Tables 3.7-2 and 3.7-3 and Figures 3.7-9, 3.7-10, and 3.7-11 of the floor response spectra indicate that the results for the soil springs model do not differ significantly from the fixed base models; thereby confirming the use of the fixed base models.

3.7.2.3.4 Soil Supported Structure

The Borated Water Storage Tank, located 50 ft. west of the Auxiliary Building, is founded on a Class I compacted granular structural backfill. The top of the foundation is at elevation 585 ft. The foundation of the tank is approximately 6 ft. deep and 49 ft. in diameter, resting on a Class I structural backfill crushed from the onsite rock quarry. This structural backfill extends to the in-situ rock at elevation 560 ft.

The seismic analysis is performed by a mathematical model having two vertical springs and one horizontal spring in order to evaluate the translation, rotation, and torsion, if any, due to the compacted structural backfill interaction below the concrete ring foundation.

The dynamic values of the fill used for the horizontal or vertical mass model springs are shown below.

a. Poisson's Ratio = 0.40.

b. Dynamic Shear Modulus = 25200 kips per sq. ft.

c. Dynamic Young's Modulus = 70500 kips per sq. ft.

The method of dynamic analysis is outlined in Subsection 3.7.2.1. The mathematical model used for the analysis is shown in Figure 3.7-1.

The following are the formulae for computing the soil equivalent springs used for the cases of circular base.

Horizontal spring constant: $Kx = \dfrac{32(1-v)GR}{(7-8v)}$

Vertical spring constant: $Kz = \dfrac{4GR}{(1-v)}$

Where v = Poisson's ratio of backfill
G = Shear Modulus
R = Radius of circular base

3.7.2.4 Soil Interaction Analysis

The Class I structures, resting on sound rock and structural backfill, are analyzed by a lumped mass system as outlined in Subsection 3.7.2.1. The finite element method of analysis is not used for any soil coupled structures. The lumped mass model, having vertical and horizontal soil springs and utilizing the soil values outlined in Subsection 3.7.2.3, was selected in lieu of the finite element method for analysis of the borated water storage tank.

A soil-structure interaction model having translational and rotational soil springs is used in the analysis of the auxiliary building area 6 as outlined in Subsection 3.7.1.7.2.

3.7.2.5 Modal Response Spectra

The modal response spectra multi-mass method of analysis is not used to develop the floor response spectra curves, but rather, a multi-mass time history method of analysis is employed.

The differential movement of floors is considered in the pipe analysis. The flexibility of pipes at penetrations of structures and floors is ensured by providing sleeves of sufficient diameter to accommodate the expected movement.

Class I piping is designed with adequate flexibility to accept the movements of connected equipment.

3.7.2.6 Floor Response Spectra

The peak widths of the floor response spectra curves, which are clearly associated with the structural frequencies, are enveloped and widened by a minimum of ±10 percent for the

expected variations of structural properties and dampings. The Class I buildings, except the borated water storage tank, are resting on rock, therefore, the variations of soil properties and soil-structure interactions are not applicable.

3.7.2.7 Horizontal and Vertical Response Spectra

The Class I structures are analyzed in the vertical direction in a manner similar to the horizontal direction. The multi-mass model method of analysis is outlined in Subsection 3.7.2.1. The vertical accelerations of the floors obtained from the response spectrum method are considered to act simultaneously with the horizontal floor accelerations. The floor response spectrum curves for the vertical direction are developed for the equipment, components, and systems analysis. The vertical component of the acceleration is applied simultaneously with the horizontal component. The combination of response loadings for the piping is similar.

The primary coolant system was analyzed considering the effects of earthquakes occurring in three mutually perpendicular directions: i.e., two horizontal and one vertical. For each earthquake, the internal forces and moments, displacements, and accelerations were calculated at discrete points within the system for each significant mode of vibration. The combined results were obtained by taking the-square-root-of-the-sum-of-the-squares over all the modes for each parameter under consideration. The root-mean-square results for each horizontal earthquake were separately added on an absolute basis to those from the vertical earthquake. Certain components (e.g., Core Flooding Tank) are not included in the overall system analysis. They are considered separately using a lumped mass model and wall spectra (at their respective elevation) for their horizontal response. When the component is sufficiently stiff in the vertical direction, a one-mass-one-spring model is used to calculate the frequency of the component and then the vertical wall spectra is again used to obtain vertical response. This procedure yielded two distinct seismic loading cases which were individually combined with the appropriate dead weight, pressure, and thermal loadings. These combined loadings were used to determine the system stresses.

3.7.2.8 Torsional Vibration Analysis Method

The torsional effect of the earthquake on the Class I buildings is analyzed manually (hand calculations), and the stresses are combined with the stresses obtained due to vertical and horizontal analyses.

3.7.2.9 Response Spectrum Method Versus Time History Method

The comparison of the response spectrum method with the time-history method shows that the acceleration results at the selected points and floors for both the methods are in close agreement with each other. See Figures 3.7-12 through 3.7-16.

3.7.2.10 Heat Sink

The Intake Canal Forebay, approximately 700 feet long, impounds a body of water that serves as a heat sink. The dikes on each side are classified and designed as Class I structures. In Subsection 2.5.5, the slope stability and seismic effect on these dikes are discussed in detail.

3.7.2.11 Seismic Design Control Measures

The seismic analyses for Class I equipment and systems are performed according to the guidelines outlined in the project specification and are summarized herein.

The equipment suppliers are furnished with related floor acceleration response spectrum curves. The moments and shears are obtained from the acceleration values derived from these floor acceleration response spectrum curves for specified damping values.

If the equipment is structurally simple, a dynamic model comprised of one mass and one spring is assumed, using the appropriate floor response spectrum curve.

If the equipment is structurally complex, the equipment is modeled as a system of lumped masses and analyzed as outlined in Subsections 3.7.2.1 and 3.7.2.2.

If the equipment cannot be adequately analyzed by a mathematical model, the suppliers are instructed to perform a dynamic test. These tests are accomplished on a shake table with base connections identical to the actual installation configuration.

All suppliers are required to submit evidence of test or analysis to the buyer for review and approval.

B&W's Q.A. Program covers the design, analysis, and test control measure employed to assure the integrity of the NSS.

3.7.2.12 Seismic Induced Overturning Moments

The dynamic methods and procedures used in determining Class I structure overturning moments are as follows. The lumped-mass multi-degree-of-freedom system is analyzed in both the horizontal and vertical directions. Subsection 3.7.2.1 outlines the response-spectrum method of analysis used to determine overturning moments. The total resultant overturning moment due to the combined effects of horizontal and vertical seismic forces is determined by using the square-root-of-the-sum-of-the-squares method. Safety factors against overturning are calculated by dividing the stabilizing moment by the total resultant overturning moment. The smallest safety factor allowed is 1.1 for the Maximum Possible Earthquake and 1.5 for the Maximum Probable Earthquake. As outlined in Subsection 3.7.1.6, only the borated water storage tank rests on structural backfill. This backfill extends approximately 19 feet below the foundation to the underlying rock, refer to Subsection 3.7.2.3.4. The dynamic soil pressure is kept within the allowable limits specified in Appendix 2C. The soil is not considered to take any tension loading.

In determining the overturning moment, only modes equal to or less than 30 cycles per second are used. The resultant moment is obtained by combining the modal effects using the square-root-of-the- sum-of-the-squares method.

3.7.2.13 Simplified Seismic Analysis Methods

Simplified seismic analysis methods and procedures are not used for Class I structures. The equipment suppliers are given the option of doing a dynamic analysis or using a simplified seismic method by using the peak acceleration of the appropriate floor response spectrum curve in lieu of determining the frequency of the equipment with our prior approval. The following paragraph clarifies this simplified approach and reads as follows:

> "The Seller may elect not to calculate or use the natural frequency of his equipment if it is of a one mass model. By using an appropriate damping value, the peak value of the acceleration response curve must be used to calculate the inertia force. This method is considered to be conservative. For equipment having more than one mass model, the Seller must analyze the equipment according to Paragraph 2.3.2.1."

3.7.2.14 Coupled System Damping

Where required, the method outlined in the Bechtel Seismic Topical Report BC-TOP-4 is used to account for composite damping in a coupled system with different structural elements.

Although various components within a model possess different values of critical damping (example: shield wall - 2%, reactor, steam generators, and pressurizer - 1%, and piping - 1/2%), the lowest damping value was used for all components in the model by the NSSS Supplier. Reanalysis of this model utilizes the damping factors from Subsection 3.7.1.3.

3.7.2.15 Modal Period Variation

Because of the quality control measures taken and documentation required for materials such as concrete, it is felt that such parameters as the compressive strength of concrete and, correspondingly, the modulus of elasticity are always greater than that specified. Therefore, stiffness and frequencies may be somewhat greater. However, the resultant seismic forces probably decrease somewhat because all of the fundamental building frequencies for Class I structures for this power station fall on or to the right of the flat part of the ground response spectrum curves. Therefore, resulting accelerations and forces should be less than or equal to those used for the design.

3.7.2.16 Damping Factors

The damping factors used for the seismic design of all Class I structures, systems, components and equipment are summarized in Table 3.7-1 and Subsection 3.7.1.3.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Earthquake-Induced Cycles

The number of earthquake cycles used in the structural analyses of Class I structures, components, and systems is determined by multiplying the number of earthquakes that might occur in the 40-year lifetime of the station by the anticipated earthquake peak time duration for significant cycles of loading and multiplying that product by the primary frequency of the structure involved.

Appendix 2C contains the records of earthquakes at the station site. In reviewing these records, it is noted that no earthquakes have been experienced at the site with an intensity greater than Modified Mercalli V (MMV), which corresponds to a maximum ground acceleration somewhat smaller than 0.02g. However, the seismicity study indicates that there is a small probability that an earthquake of 0.06g intensity could occur. Consequently, it is concluded that only one earthquake of 0.08g Maximum Probable Earthquake and only one of 0.15g Maximum Possible Earthquake need be considered.

Further investigation of Appendix 2C indicates that the duration of strong motion during these earthquakes is approximately 3 seconds.

Assuming the most conservative viewpoint, a structure primary frequency of ten cycles per second (Hz), and by multiplying as stated above, then it follows:

$$1 \times 3 \times 10 = 30 \text{ cycles per lifetime of station}$$

However, it is conservatively proposed to use 200 cycles of significant loading for Class I structures and Nuclear Class I piping systems.

For rigid equipment having natural frequencies greater than 30 cps, the response accelerations are equal to the accelerations of the supporting structure at the appropriate elevation. The design criteria for both structures and equipment required that the calculated stresses from seismic, combined with other loads remained below yield of the material. Since the calculated stresses were below yield, no cyclic loading was considered in the design. For the small number of cycles which occur during earthquake, fatigue modes of failure of materials are not applicable for structures and equipment.

The effects of the cyclic loadings on Reactor Coolant System piping due to seismic excitation were accounted for by appropriately combining the earthquake and thermal transient stresses at discrete points within the system and calculating cumulative usage factors. For all the points examined, this factor was less than one. Consequently, the fatigue requirements of ANSI B31.7 were satisfied for the original piping; for the Replacement Hot Leg spool pieces, the fatigue requirements of ASME B&PV Code, Section III Class 1, 2001 Edition, 2003 Addenda were satisfied.

Class I mechanical equipment has been designed using a static ‘g’ loading applied both vertically and horizontally in accordance with the procedures outlined in Section 3.7.2.1. Class I active valves have been designed using a static loading with a minimum of 3.0 g applied at the center of gravity unless noted otherwise in Section 3.9. No cyclic effects were considered in the design of either the valves or equipment, as justified above.

3.7.3.2 Resonance Frequency Selection

The equipment suppliers are furnished with applicable response spectrum curves for horizontal and vertical directions in order for them to avoid the peak response.

A list of building frequencies is also given to suppliers so they can make appropriate changes in their equipment frequencies and thereby reduce resonant response. The suppliers were instructed to alter the effective mass or spring rate to avoid resonance between equipment and the building if the acceleration is too high for design of the equipment.

3.7.3.3 Modal Response Combination Method

The method used in combining modal responses for subsystem analyses is the square-root-of-the-sum-of-the-squares method.

3.7.3.4 Criteria for Combining Modal Responses

The criteria for combining modal responses (shears, moments, stresses, deflections, and accelerations), when modal frequencies are closely spaced and a response spectrum modal analysis method is used, are as follows:

a. The square-root-of-the-sum-of-squares method of combining all modal responses for moments, shears, stresses, deflections, and accelerations is used.

b. All modes having frequencies of 30 cycles per sec. (Hz) or less are used in combining these modal responses.

c. Not all modal frequencies are closely spaced for the Class I structures.

3.7.3.5 Simplified Equipment Seismic Analysis

If the natural frequency of the equipment or components of a single-degree-of-freedom system cannot be determined, the peak value of the response spectrum curve for an appropriate damping factor is used (in lieu of the one mass model analysis) for the simplified seismic analyses in order to determine the equivalent static loads.

If the equipment is structurally complex and its function is seriously affected by the earthquake, it is modeled as a system of lumped masses and analyzed as outlined in Subsections 3.7.2.1 and 3.7.2.2.

Consideration was given to all the modes having frequencies of 30 Hz or less in the analyses.

3.7.3.6 Piping Support Displacement

Curves of building displacement versus elevation are developed for each building, and/or section of a building by seismic analysis to use as an input for piping analysis. When a line runs from one building to another, such as the Auxiliary Building to the Shield Building, the worst case differential movement conditions are applied at the anchor points. A thermal analysis is run using these movements, and since the associated stresses are secondary stresses, they are included in the thermal stress which is compared to the allowable thermal stress. When the movements between different elevations in a building are significant, they are analyzed as above.

The relative displacement between floors, components, and piping of primary systems are accounted for in seismic system analysis, since all of these items are included in the analytical model as shown in Figures 3.7-2, 3.7-6 and 3.7-7. When a line (i.e., spray line) is done separately, anchor displacements are taken from the system analysis and applied to the line according to the applicable code as described in Table 3.2-2.

3.7.3.7 Combined Amplified Response

Combined horizontal and vertical amplified response loading of the piping system is solved using a computer program which inserts time varying displacements at supports, anchors, and equipment for multi-directional horizontal and vertical earthquakes.

Amplified response at different elevations and different points in the reactor coolant system are taken into account by two (2) methods:

a. The major portion of the system (major components and wall) are modeled together and tied back to the base-mat through the wall.

b. The response spectra at the point of attachment of the subsystem is calculated and used for input to the subsystem.

The vertical and horizontal responses are combined as in Subsection 3.7.2.7.

3.7.3.8 Dynamic Analysis

The dynamic response analysis for Class I piping is performed by computer programs which have a capability of multi-directional analysis in horizontal and vertical directions. The simplified dynamic analysis is not being used for Class I piping.

3.7.3.9 Torsional Effects

In order to account for torsional effects, the eccentric masses (valve operators, instruments, etc.) are coded in the model of the piping system in such a way that the effects of the eccentricity are fully accounted for.

3.7.3.10 Piping and Structural Stress Limits

All piping requiring formal analysis, buried or otherwise, is analyzed using maximum possible differential movement of support points due to differential movements between, or within, buildings. See Figures 3.7-17 and 3.7-18.

3.7.3.11 Effects of Class II Piping on Class I Piping

No distinction is being made between Class I and connected Class II piping, if it is located at the same side of an isolation anchor. They are analyzed as Seismic Class I.

3.7.3.12 Field Location of Seismic Piping Systems and Supports

The field locations of seismic supports and restraints for Seismic Class I piping and piping system components have been selected to satisfy the following two conditions: (1) the locations selected must furnish the required response to control strain within allowable limits and (2) adequate building strength for attachment of the components must be available.

The final locations of seismic supports and restraints for Seismic Class I piping, piping system components, and equipment including the placement of snubbers are checked against drawings and procedures issued by engineering. The checking of these supports and restraints is made by engineering to assure that the location and characteristics are consistent with the dynamic and static analyses of the piping.

3.7.3.13 Polar Crane Seismic Analysis

A multi-degree of freedom seismic analysis on the polar crane reveals that there is no uplift for either the bridge or the trolley. As an additional safety feature, the seismic restraints are provided on both sides of the bridge and trolley rails, to prevent dislodgement from these rails in the event of seismic excitation.

3.7.4 Criteria for Seismic Instrumentation Program

Seismic instrumentation for the Davis-Besse Nuclear Station was provided for the purpose of verifying the seismic criteria used in the engineering design of Class I equipment and structures.

3.7.4.1 Instrumentation for Earthquakes

The present earthquake instrumentation consisting of four triaxial, time-history, strong-motion accelerometers and additional peak recording accelerometers are consistent with the recommendations of Regulatory Guide 1.12 (1974). They adequately provide signals for recording the effects of an earthquake such that analysis may determine the intensity of an earthquake and derive all additional information desired to confirm the seismic design. The plant has been designed so that safety is assured for earthquakes up to and including the SSE. Earthquakes of large intensity are most accurately indicated by the triaxial strong motion accelerometer recording system. Any additional instrumentation is not warranted and increases the probability of operational errors due to excessive and possibly conflicting data taken immediately after an earthquake occurrence.

3.7.4.2 Description of Seismic Instrumentation

The triaxial accelerometers and associated recorders are used to monitor and record strong seismic motion in the north-south, east-west, and vertical directions.

Two triaxial accelerometer transducers are installed in the containment. One is rigidly mounted on the concrete foundation interior slab at El. 565. The other is rigidly fastened to the interior secondary shield wall at El. 653, directly above the lower accelerometer. These accelerometers are located at or near the center of gravity of the containment. Another triaxial accelerometer transducer is rigidly mounted to the basement floor, at El. 545, of the Auxiliary Building.

The external free-field triaxial accelerometer is located a minimum of 300 feet from the station, in a fairly isolated area, so that it will not pick up any reflective waves from the stations foundations and will be free from normal route of vehicular traffic. It is mounted on a concrete foundation, surrounded by concrete walls and is accessed via a removable weather-tight cover on the top.

Each triaxial accelerometer has a corresponding digital recorder located between the accelerometer and the control room. The accelerometer and recorder form a stand-alone system for detecting and recording seismic activity; however, the recorders are networked to provide common triggering, control room alarms, and data collection and analysis on a dedicated personal computer workstation in the control room.

When an earthquake occurs that is above the predetermined level of ground acceleration at any accelerometer, the system triggers all of the recorders to record time-history of the event.

At a minimum, the seismic trigger is adjustable from 0.005g to 0.02g. This is sensitive enough to record an earthquake which creates a low level of ground acceleration. The seismic equipment is capable of recording all major strong-motion vibrations above the predetermined starting threshold level and recording for a time period of at least 10 seconds beyond the last vibration above the triggering threshold. The recording equipment is capable of recording continuously for 30 minutes or more. Following an event, the system automatically downloads data from the recorders to the data analysis personal computer workstation, initiates alarms indicating that triggering has occurred and whether SSE or OBE levels have been exceeded, and analyzes the data for frequency-domain comparison to the design response spectra.

Peak recording accelerometers are used to verify the dynamic analysis of Class I structures, systems, and components by comparing the values with the actual recorded acceleration time-history after a seismic event.

Peak recording accelerometers are located at the following places:

a. Top of the Shield Building at Elevation 812’

b. Roof of the Auxiliary Building at Elevation 660’

c. Near the southwest corner wall of the control room at Elevation 623’

A seismically qualified control room cabinet and seismically qualified mounting of recorders and accelerometers for the seismic monitoring system allows recorded seismic events to be analyzed off site and allows the system to operate during and after a seismic event. It provides additional assurance of system operation, maintenance, and adherence to the Technical Requirements Manual (TRM).

The seismic monitoring instrumentation is maintained in accordance with written procedures and the TRM. The TRM is incorporated by reference into the USAR.

3.7.4.3 Accelerograph Readout Procedure

The locations of the accelerometers for the time-history record are so selected that they provide the base seismic input motion to the Class I structures, in particular the Reactor Vessel, containment structure internals, Auxiliary Building, and control room.

An annunciator in the control room indicates to the operator that a seismic event has activated the seismic instrumentation and that the seismic event is being recorded. Upon completion of the seismic event recording, the system automatically downloads data from the recorders to the data analysis personal computer workstation located in the control room. The data are then processed automatically and are available to the operator as time-histories or as frequency-domain response spectra.

The system provides graphical comparisons between the measured response spectra and the station design basis OBE and SSE response spectra. If a valid measured response spectra exceeds the OBE response spectra limits for any of the accelerometers, a plant shutdown is required.

The response spectra comparison is the preferred method to determine exceedance of the OBE and/or SSE limits; however, a system alarm that initiates upon exceedance of the OBE zero-

period acceleration at the containment concrete foundation accelerometer may be used as a backup.

3.7.4.4 Comparison of Dynamic Analyses

Following a Seismic Event, data from each seismic monitoring instrument is obtained to prepare a report within ten days describing the magnitude, spectrum, frequency, and resultant effects on the facility features important to safety.

Using the recorded time-history, a dynamic response check analysis is performed by using the previous mathematical models as outlined in Figures 3.7-3 to 3.7-7. The predicted and observed maximum accelerations for various elevations are compared to validate the design values.

The results of the dynamic responses of the Class I structures are not affected by the soil as described in Subsection 3.7.1.4.

3.7.5 Seismic Design Control Measures

The description of the design control measures, including necessary feedback from structure and system dynamic analyses, for the vendor - purchased Class I components and equipment is given in Subsection 3.7.2.11.

TABLE 3.7-1

Percentage of Critical Damping Factors

Item, Equipment, or Structures	Maximum Probable Earthquake	Maximum Possible Earthquake
Large diameter piping systems, pipe diameter greater than 12 in. (NOTE 1)	0.5	0.5
Smaller diameter piping systems, diameter less than or equal to 12 in. (NOTE 1)	0.5	0.5
Welded steel structures	2	2
Bolted steel structures	2	5
Reinforced concrete structures	2	4
Equipment	1	1
CMU walls	4	7
Conduit support systems	4	7
Cable tray/wireway systems	4	7
HVAC support systems	2	2

NOTE 1: Per Table 3.9-10 the ASME Code Case N-411 alternate damping method may be used for piping systems.

TABLE 3.7-2

Comparison of Fixed Base vs. Soil Springs Model for Shield Building

	Pt. Fixed Base		Soil Springs	% Change
Primary Frequency (Hz)	-	3.4	3.3	-2.3
Moments (kip-Ft)	4	893,010	934,535	+5.8
Shears (kips)	4	5790.1	6119.0	+5.7
Containment Vessel				
Primary Frequency (Hz)	-	5.4	5.34	-0.9
Moments (kip-Ft)		161,230	157,088	-2.6
Shears (kips)		1091.3	1073.	-1.7
Internal Structures - North-South				
Primary Frequency (Hz)	-	7.3	7.23	-1.4
Moments (kip-Ft)		125,170	133,022	+6.3
Shears (kips)		2795.4	3050.0	+9.1

TABLE 3.7-3

Comparison of Fixed Base vs. Soil Springs Model for Internal Structures

Time History

	Pt.	Fixed Base	Soil Springs	% Change
Absolute Acceleration(g)	32	0.253	0.258	+1.2

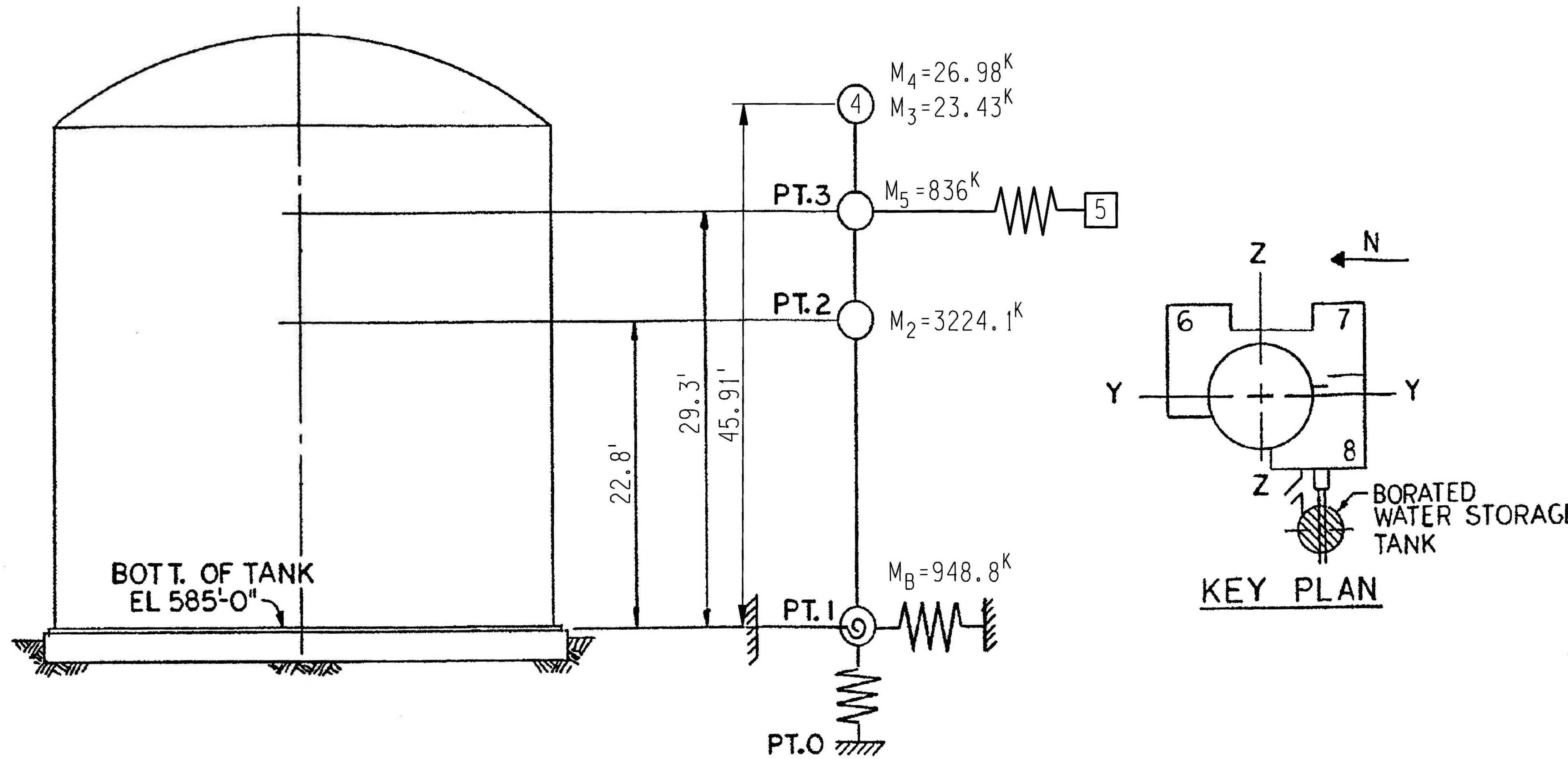
Floor Response Spectra

	Pt.	Fixed Base	Soil Springs	% Change
Peak Acceleration Damping = 2%	25	0.386	0.4037	+4.6
Peak Acceleration (g)	28	0.737	0.8595	+16.6
Peak Acceleration (g)	32	1.78	2.0454	+14.9

TABLE 3.7-4

Plant System Piping Utilizing ASME Code Case N-411 Alternate Damping

Auxiliary Feedwater
Main Steam
Low Pressure Injection
High Pressure Injection
Containment Spray
Core Flooding
Component Cooling Water
Service Water
Main Feedwater
Makeup and Purification
Decay Heat Removal
Spent Fuel Cooling
Station and Instrument Air
Radwaste/Post-Accident Sampling
Process and Post-Accident Sampling
Emergency Diesel Generator Starting Air
Emergency Diesel Generator Fuel Oil
Emergency Diesel Generator Air Intake
Steam Generator Blowdown
Steam Generator Drains
Steam Generator Instrumentation
Pressurizer Instrumentation
Pressurizer Surge Line Drain
Pressurizer Sampling
Pressurizer Auxiliary Spray
Pressurizer Spray
Reactor Vessel Gasket Monitoring
Reactor Vessel Head Vent Line
Reactor Coolant System
Reactor Coolant Drains
Reactor Coolant System Instrumentation
Reactor Coolant Pump Seal Injection
Reactor Coolant Pump Seal Return
Auxiliary Steam
Gaseous Radwaste
Containment Vent
Floor Drains
Make-up Water Treatment
Hydrogen Dilution
PORV Loop Seal Drains
PORV Line to Quench Tank



DAVIS-BESSE NUCLEAR POWER STATION
BORATED WATER STORAGE TANK-MASS MODEL
FIGURE 3.7-1

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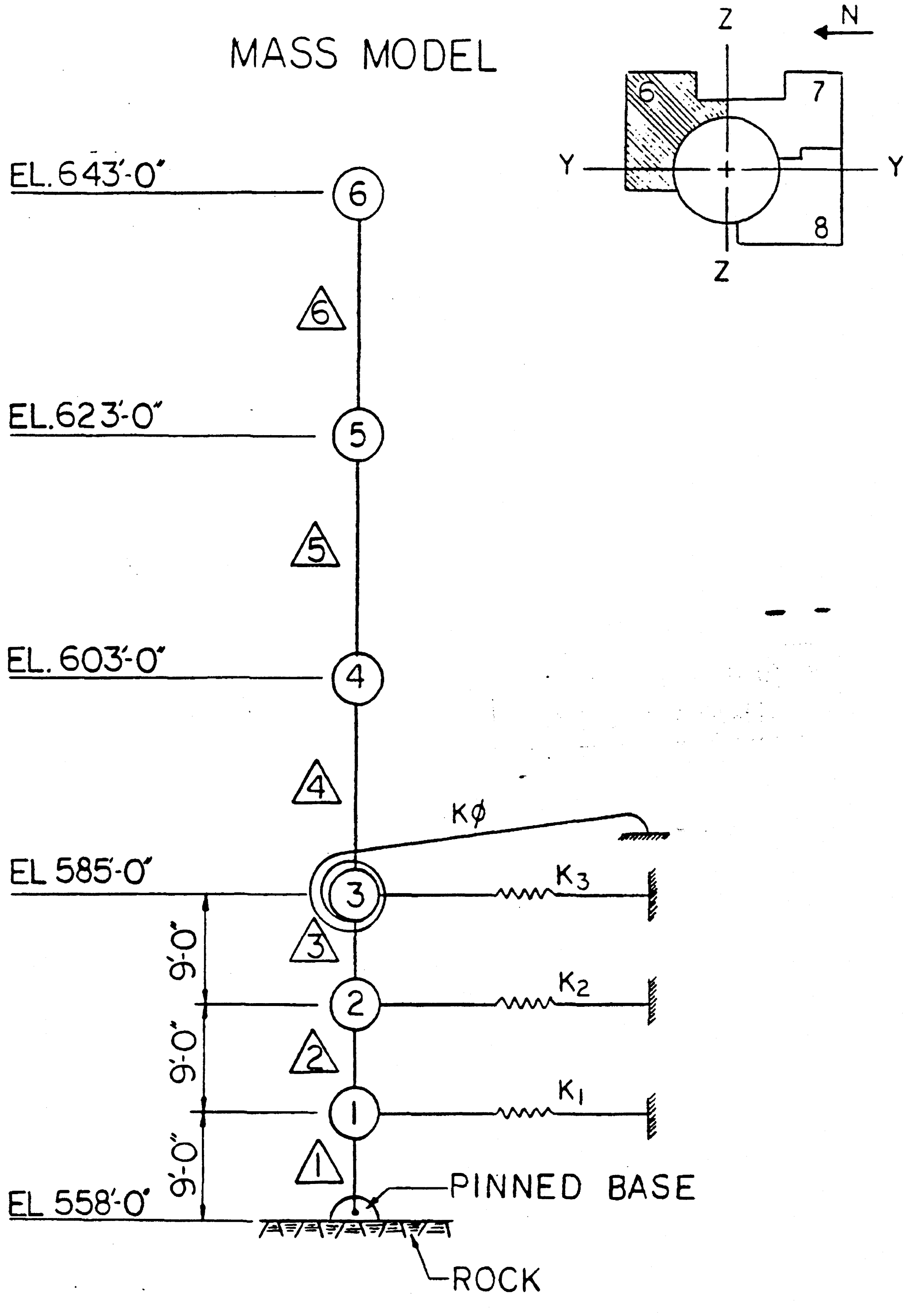
DB 07-29-16 DFN: G/USAR/UFIG371.DGN/TIF



DAVIS-BESSE NUCLEAR POWER STATION
SEISMIC SYSTEM ANALYSIS
MASS MODEL AUXILIARY BUILDING AREA 6
FIGURE 3.7-2

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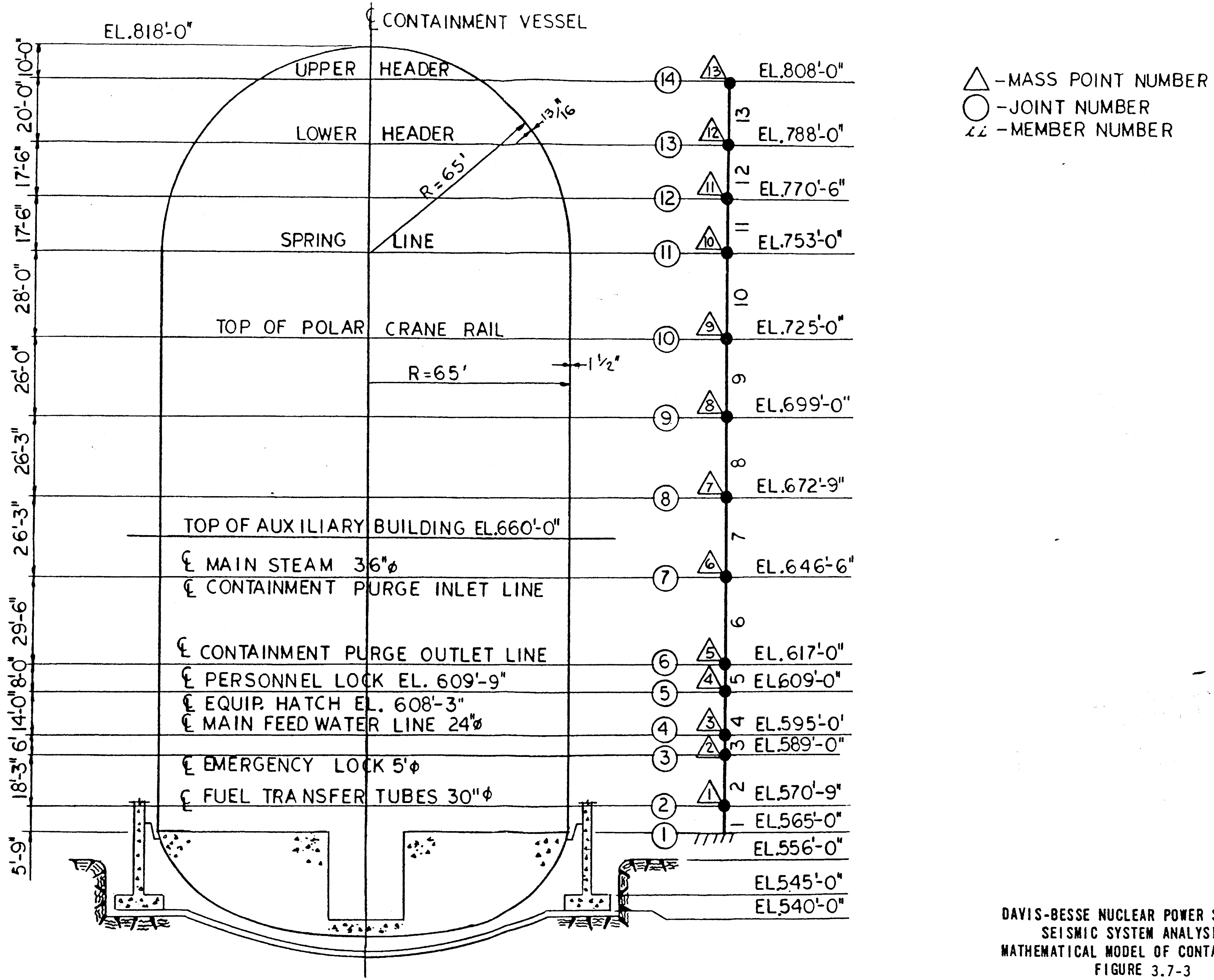

DAVIS-BESSE NUCLEAR POWER STATION
SEISMIC SYSTEM ANALYSIS
MATHEMATICAL MODEL OF CONTAINMENT
FIGURE 3.7-3

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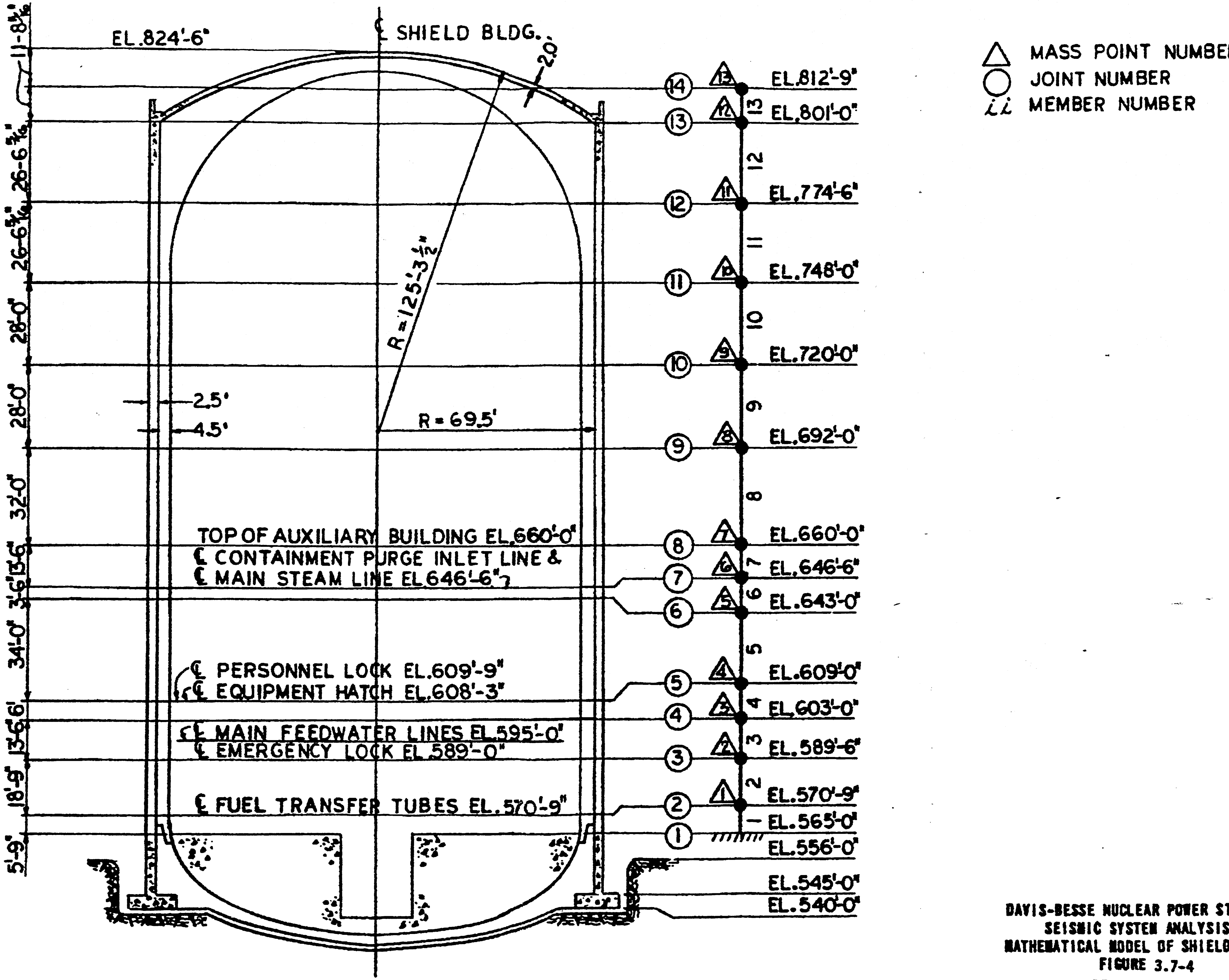

DAVIS-BESSE NUCLEAR POWER STATION
SEISMIC SYSTEM ANALYSIS
MATHEMATICAL MODEL OF SHIELD BLDG.
FIGURE 3.7-4

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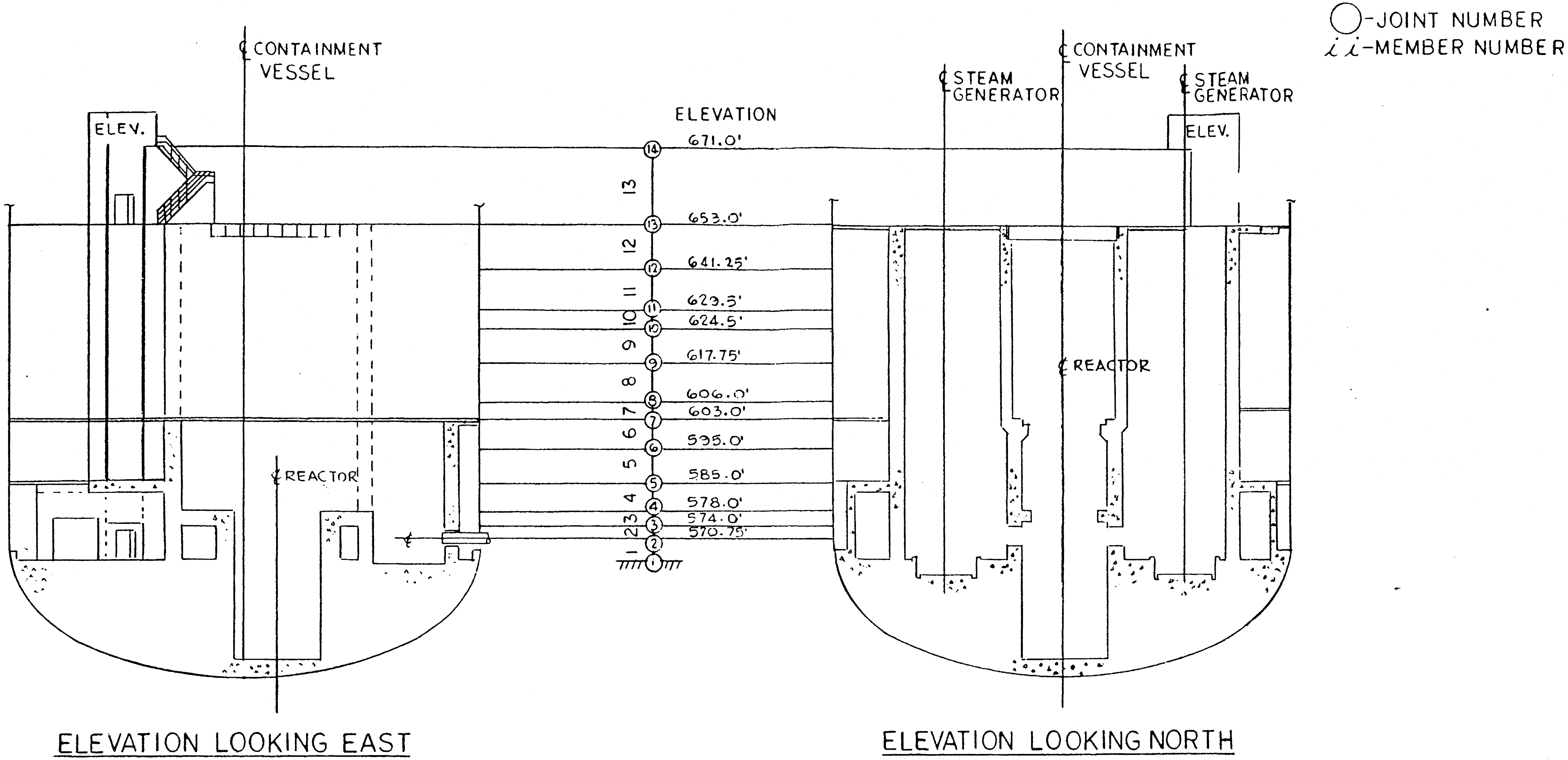

DAVIS-BESSE NUCLEAR POWER STATION
SEISMIC SYSTEM ANALYSIS - MATHEMATICAL
MODEL OF CONTAINMENT VESSEL
INTERNAL STRUCTURE
FIGURE 3.7-5

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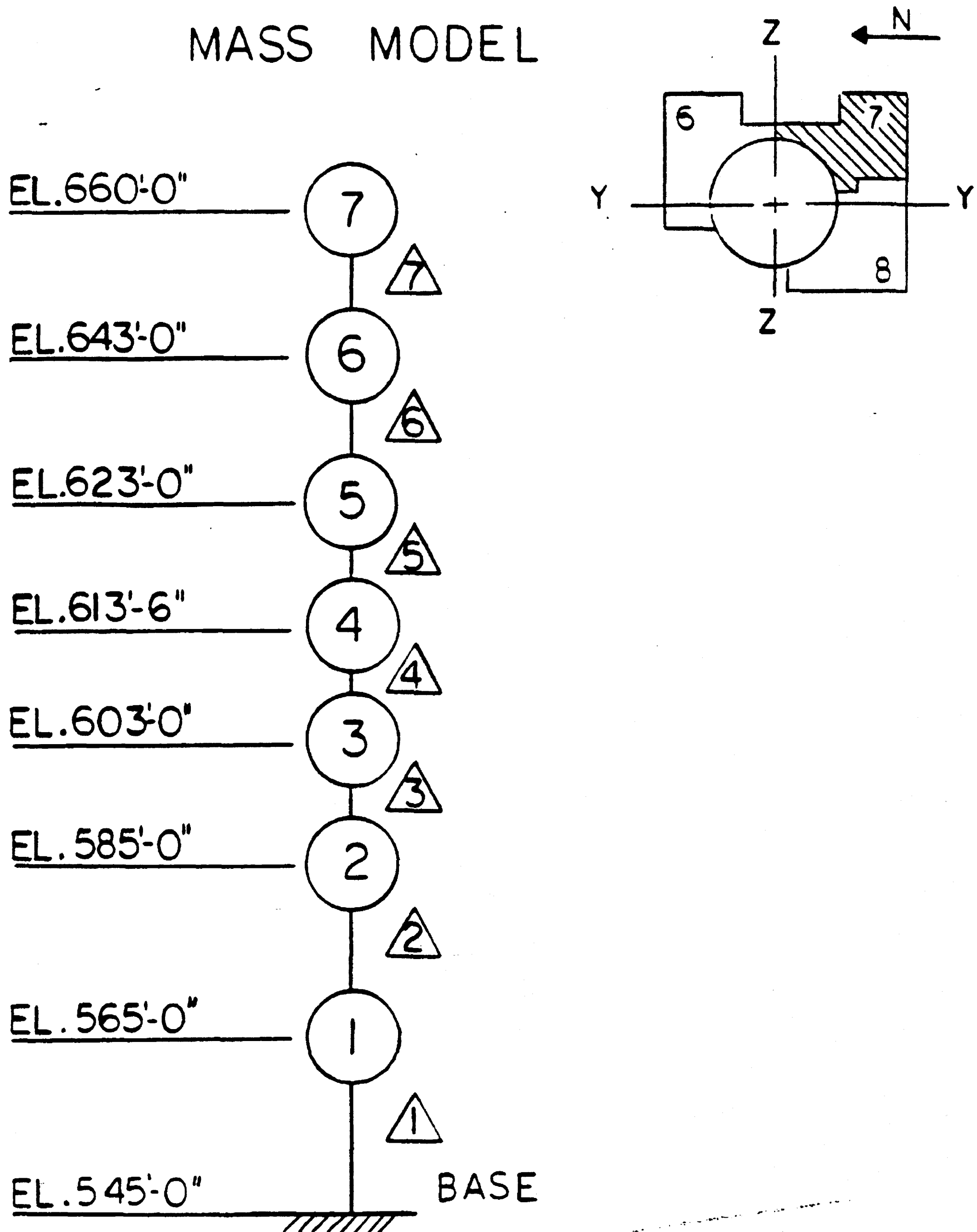

DAVIS-BESSE NUCLEAR POWER STATION
SEISMIC SYSTEM ANALYSIS
MASS MODEL AUXILIARY BUILDING AREA 7
FIGURE 3.7-6

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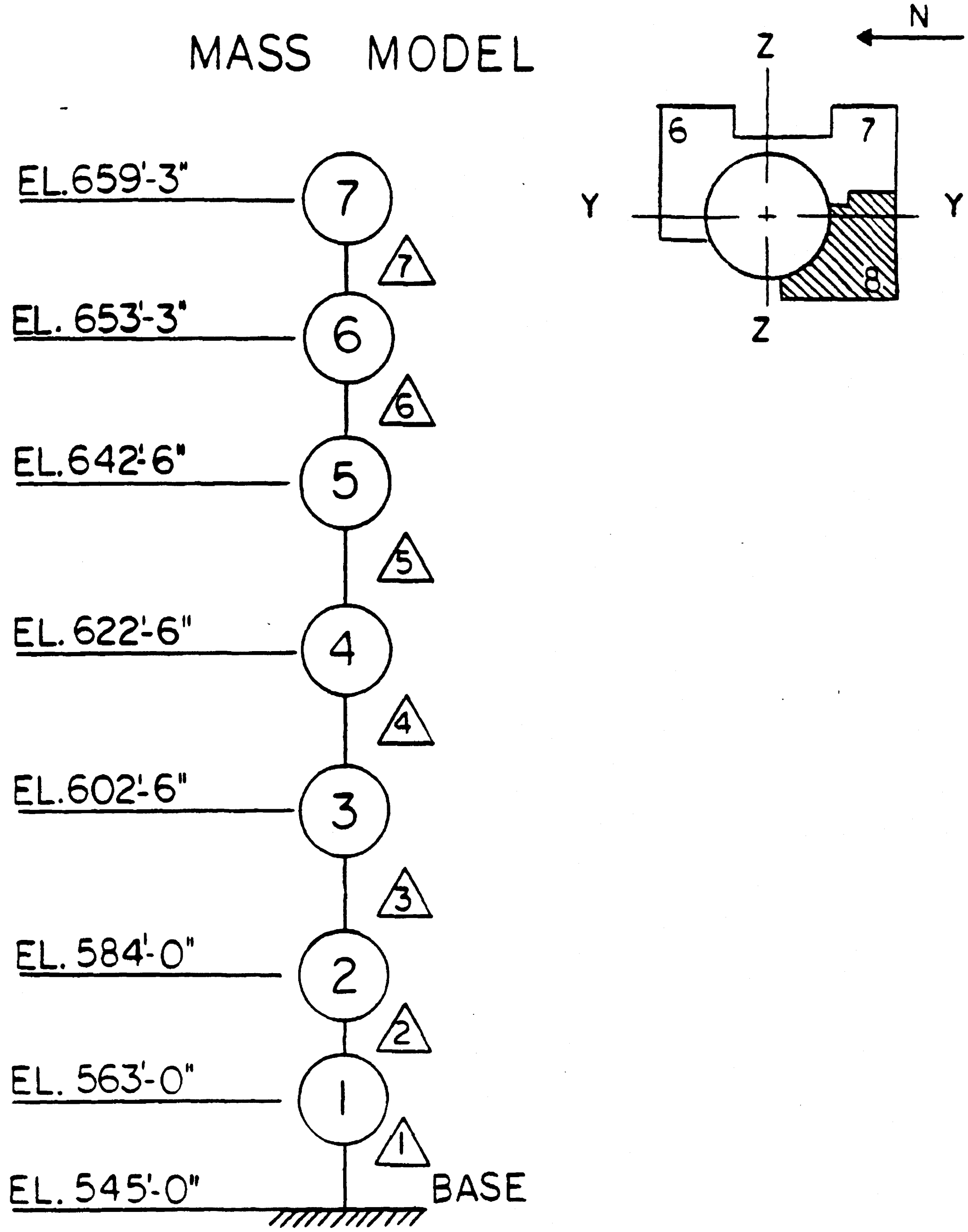

DAVIS-BESSE NUCLEAR POWER STATION
SEISMIC SYSTEM ANALYSIS
MASS MODEL AUXILIARY BUILDING AREA 8
FIGURE 3.7-7

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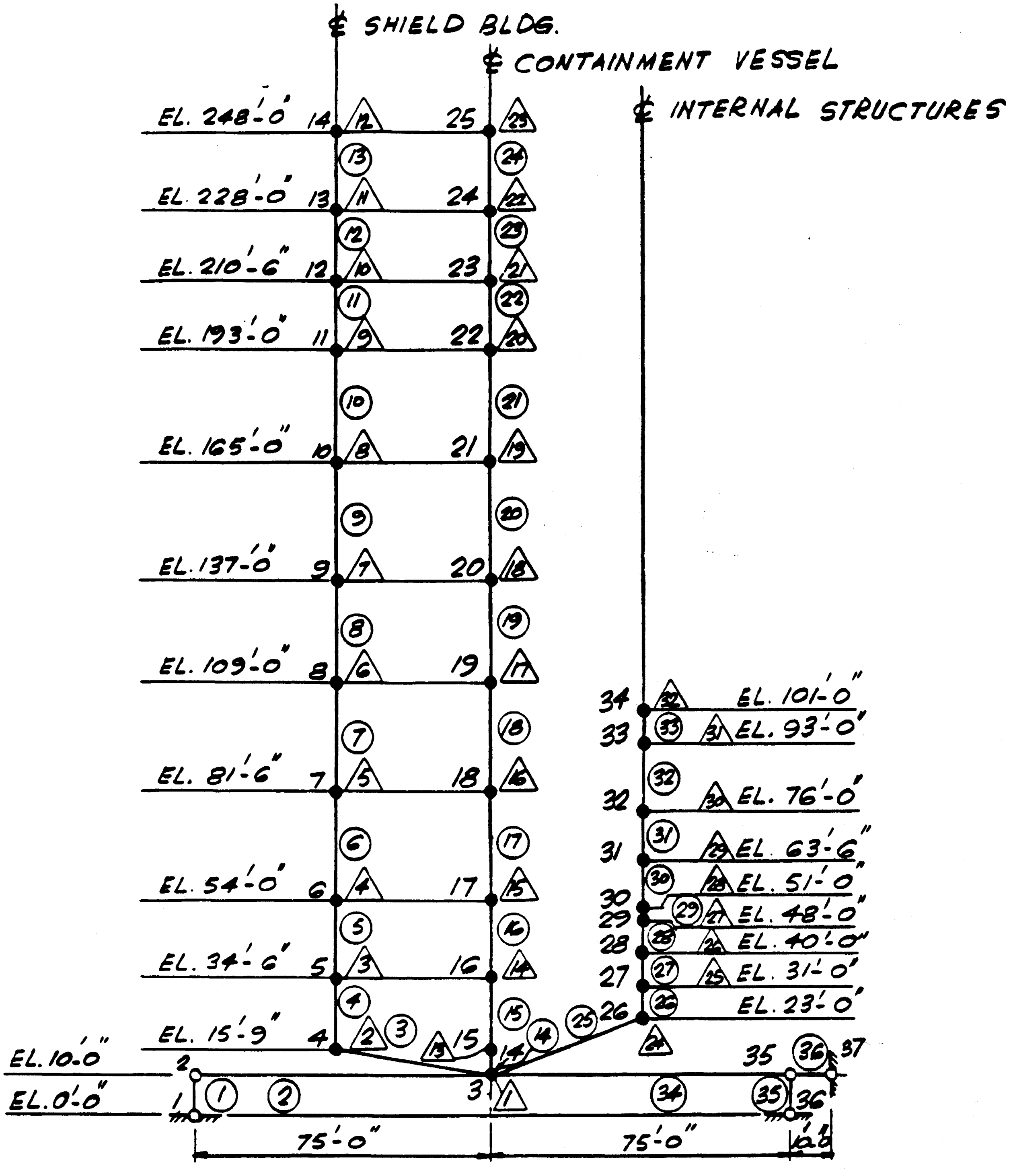
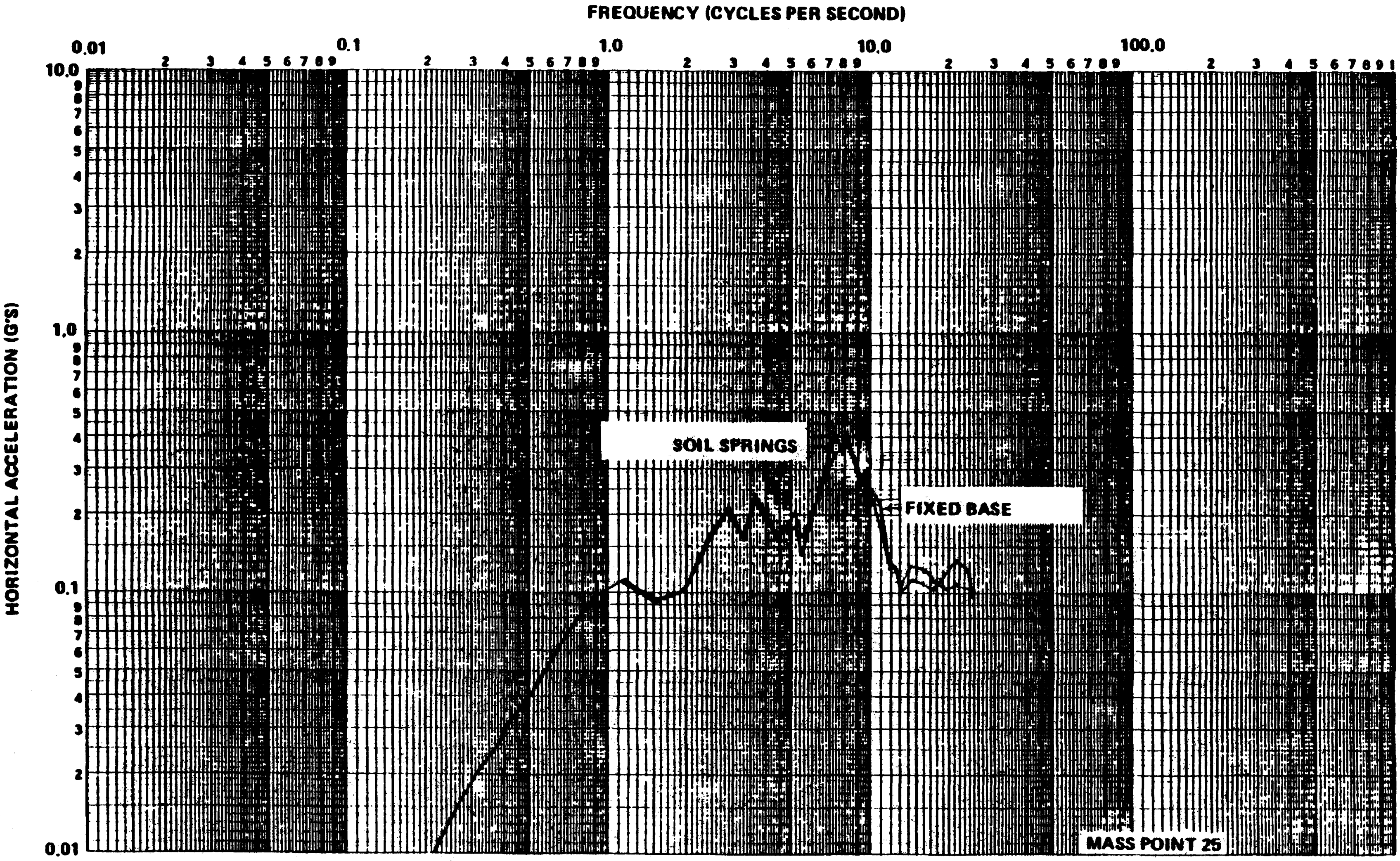


DAVIS-BESSE NUCLEAR POWER STATIO
MATHEMATICAL MODEL OF CONTMT. STRUCTURES ON SOIL SPRING
FIGURE 3.7-8



DAVIS-BESSE NUCLEAR POWER STATION
COMPARISON OF FIXED BASE MODEL RESPONSE
VS
SOIL SPRING MODEL RESPONSE
FIGURE 3.7-9

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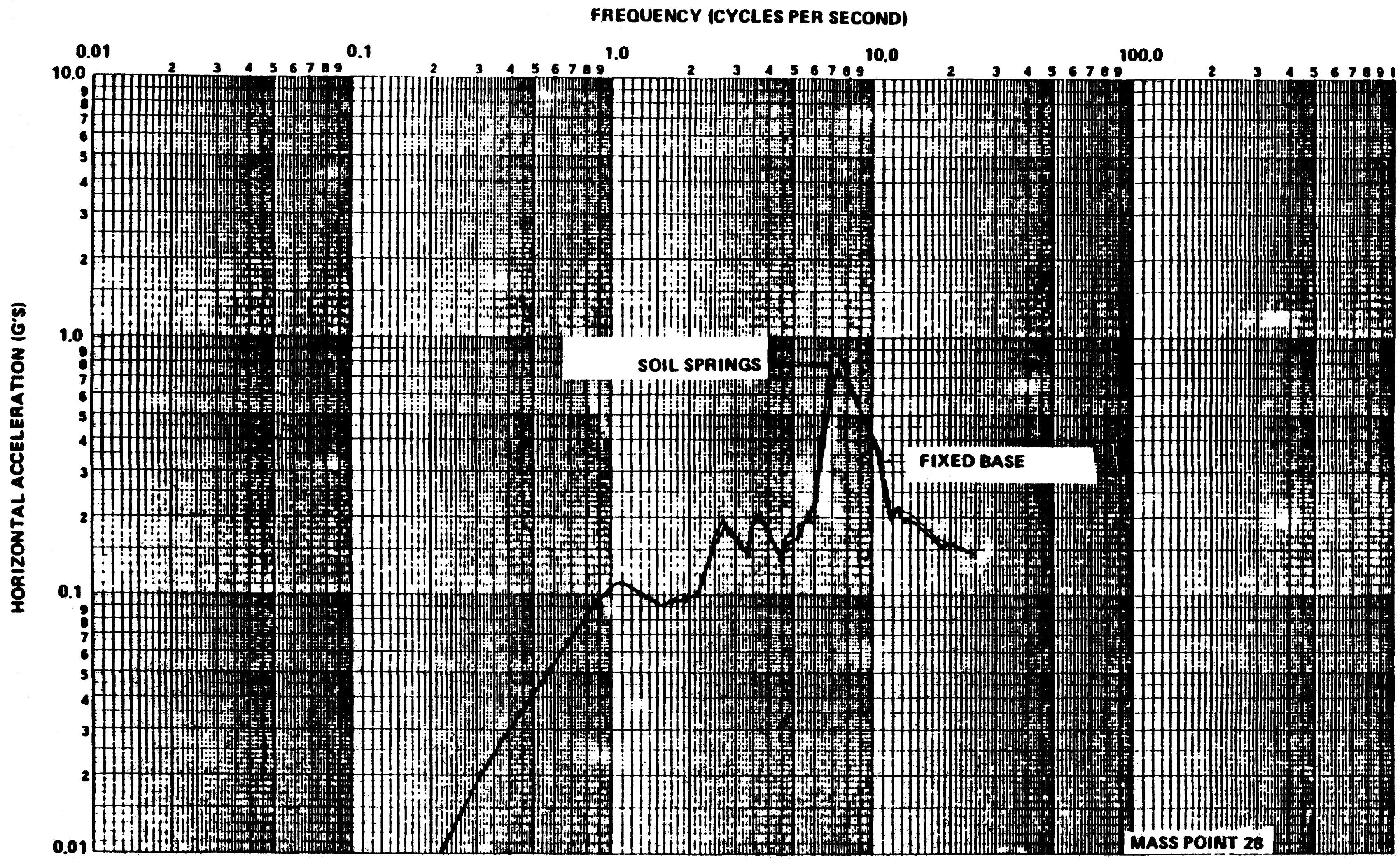

DAVIS-BESSE NUCLEAR POWER STATION
COMPARISON OF FIXED BASE MODEL RESPONSE
VS
SOIL SPRING MODEL RESPONSE
FIGURE 3.7-10

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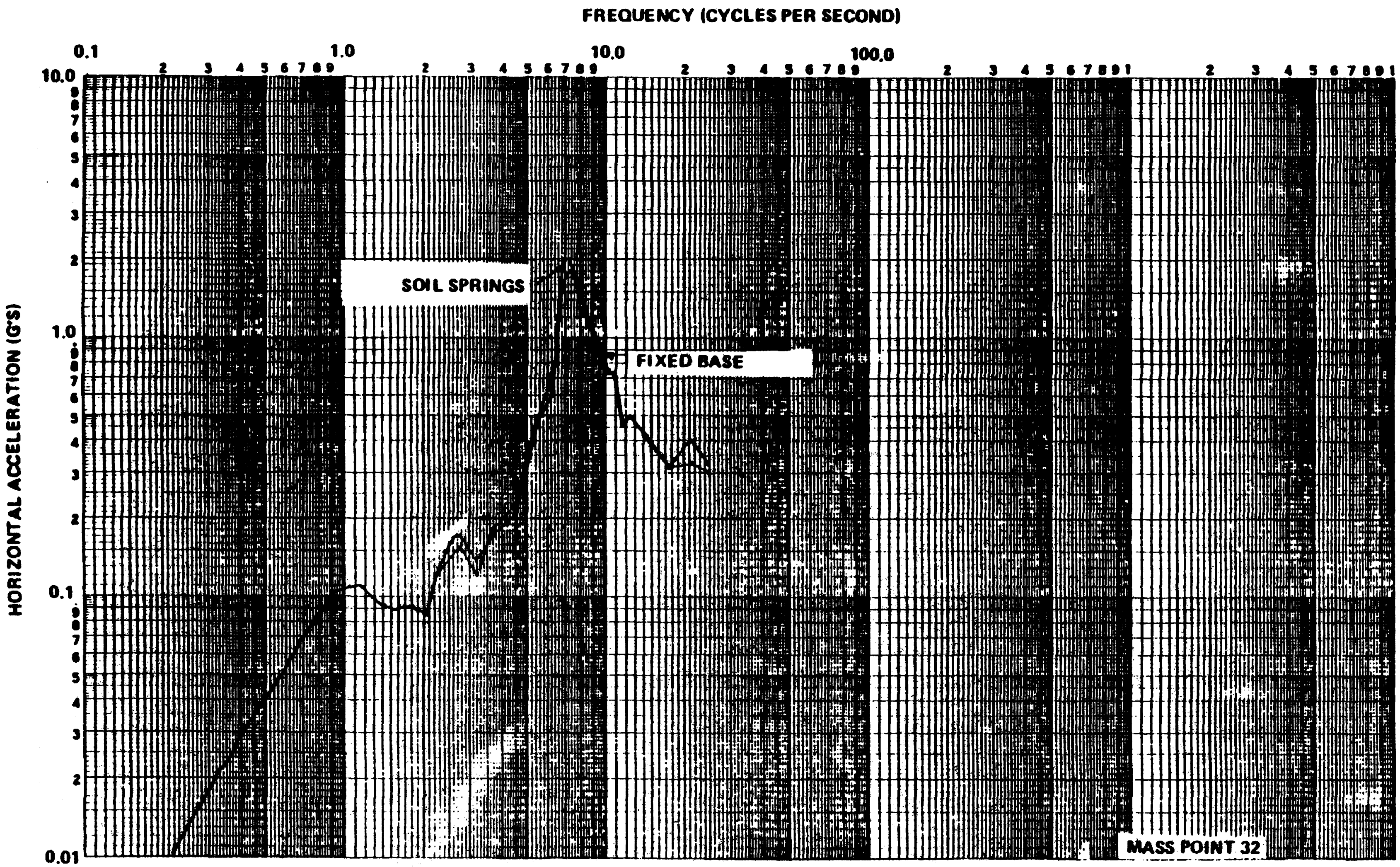

DAVIS-BESSE NUCLEAR POWER STATION
COMPARISON OF FIXED BASE MODEL RESPONSE
VS
SOIL SPRING MODEL RESPONSE
FIGURE 3.7-11

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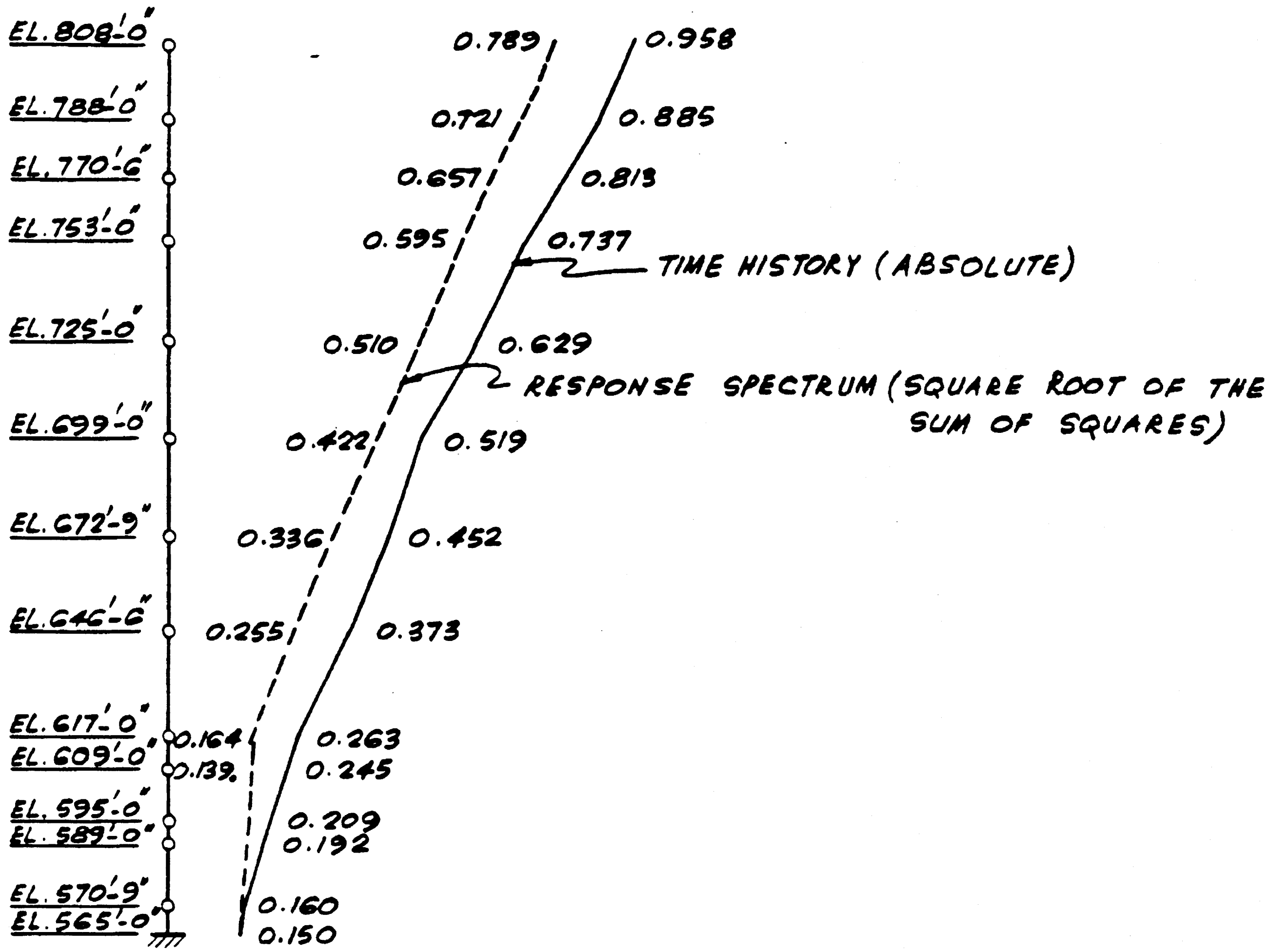

COMPARISON OF TIME HISTORY &
RESPONSE SPECTRUM ACCELERATIONS

HORIZONTAL

MAXIMUM POSSIBLE EARTHQUAKE (0.15g)

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT VESSEL - TIME HISTORY (ABSOLUTE)
& RESPONSE SPECTRUM (RMS) COMPARISON
FIGURE 3.7-12

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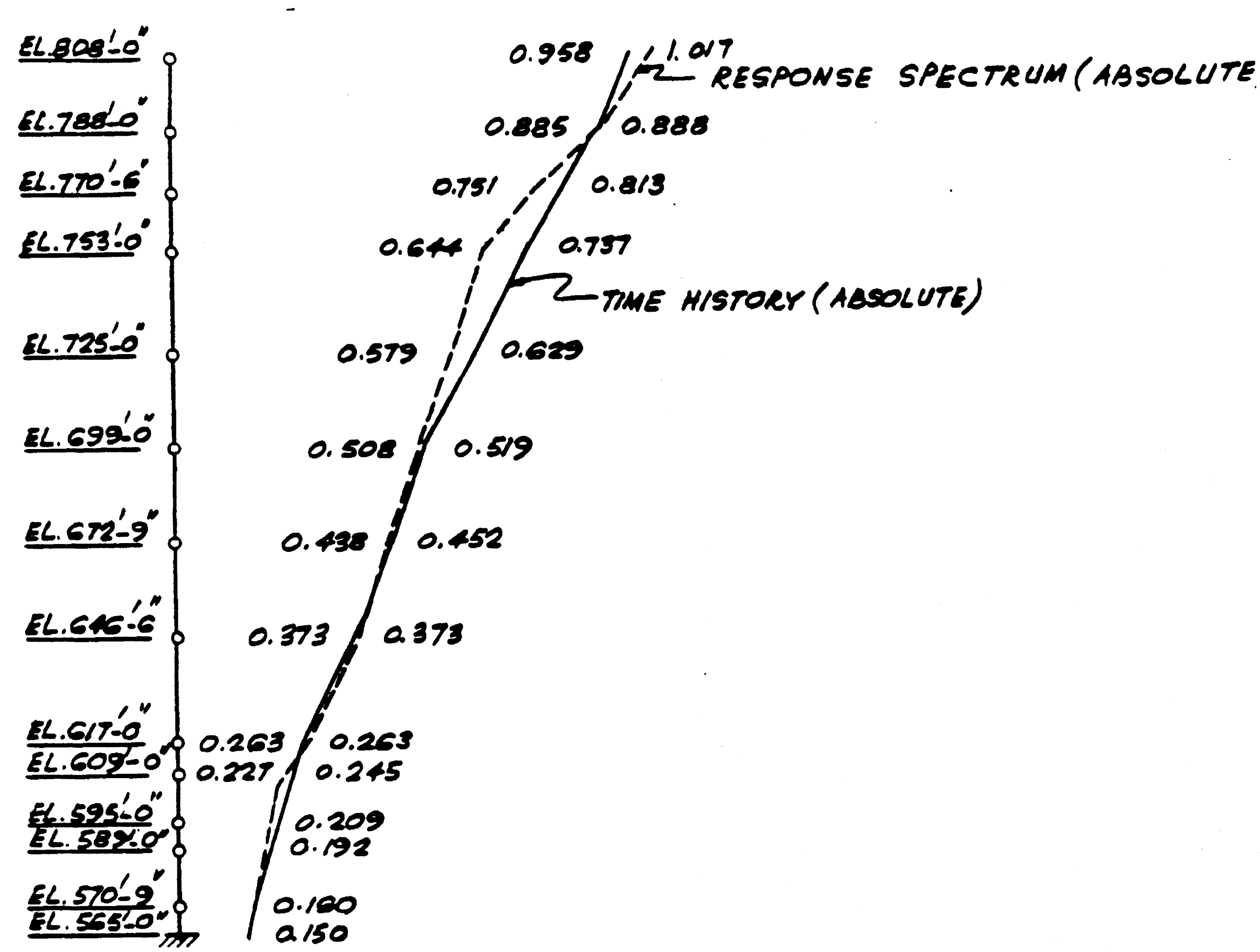

COMPARISON OF TIME HISTORY &
RESPONSE SPECTRUM ACCELERATIONS

HORIZONTAL

MAXIMUM POSSIBLE EARTHQUAKE (0.15g)

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT VESSEL - TIME HISTORY (ABSOLU
& RESPONSE SPECTRUM (ABSOLUTE) COMPARIS
FIGURE 3.7-13

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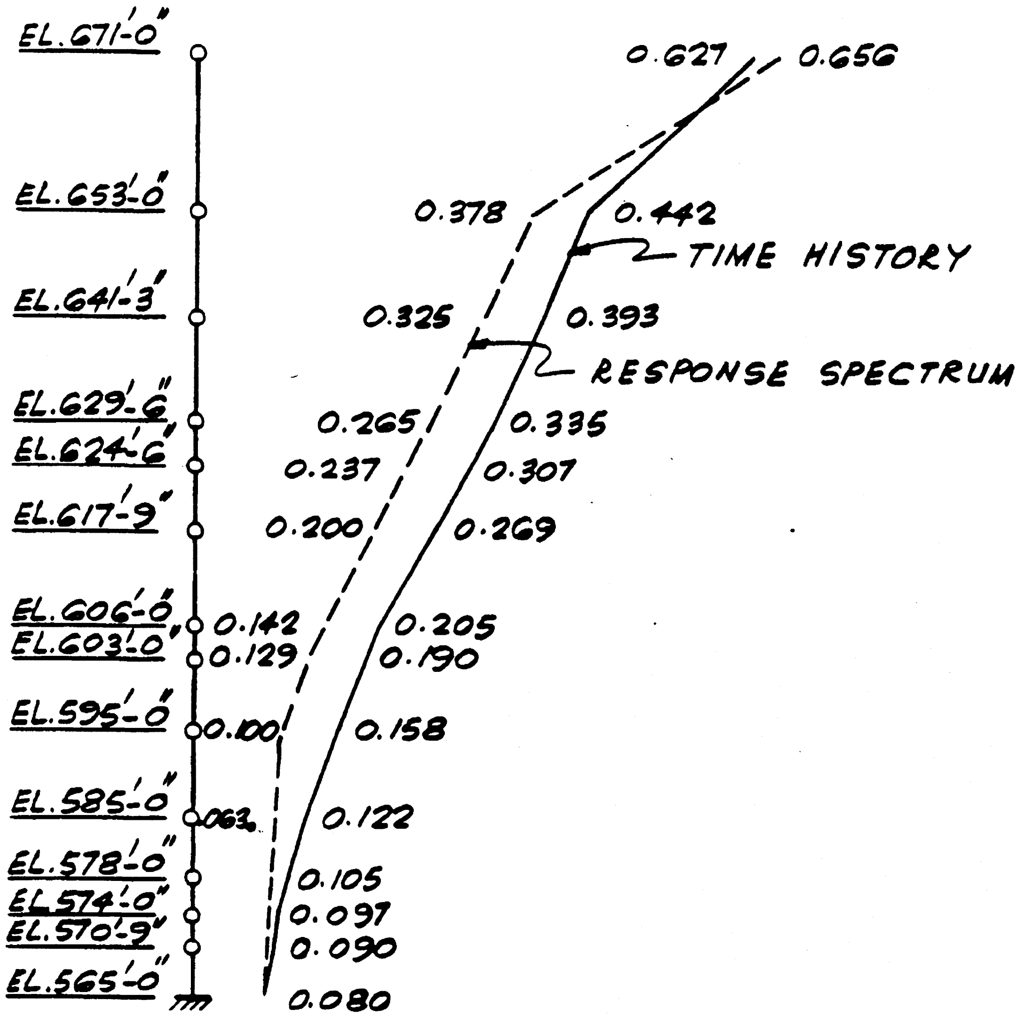

COMPARISON OF TIME HISTORY & RESPONSE SPECTRUM ACCELERATIONS

HORIZONTAL (EAST-WEST)

MAXIMUM PROBABLE EARTHQUAKE (0.08g)

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT INTERNAL STRUCTURES - TIME HIST
& RESPONSE SPECTRUM COMPARISON
(MAXIMUM PROBABLE EARTHQUAKE)
FIGURE 3.7-14

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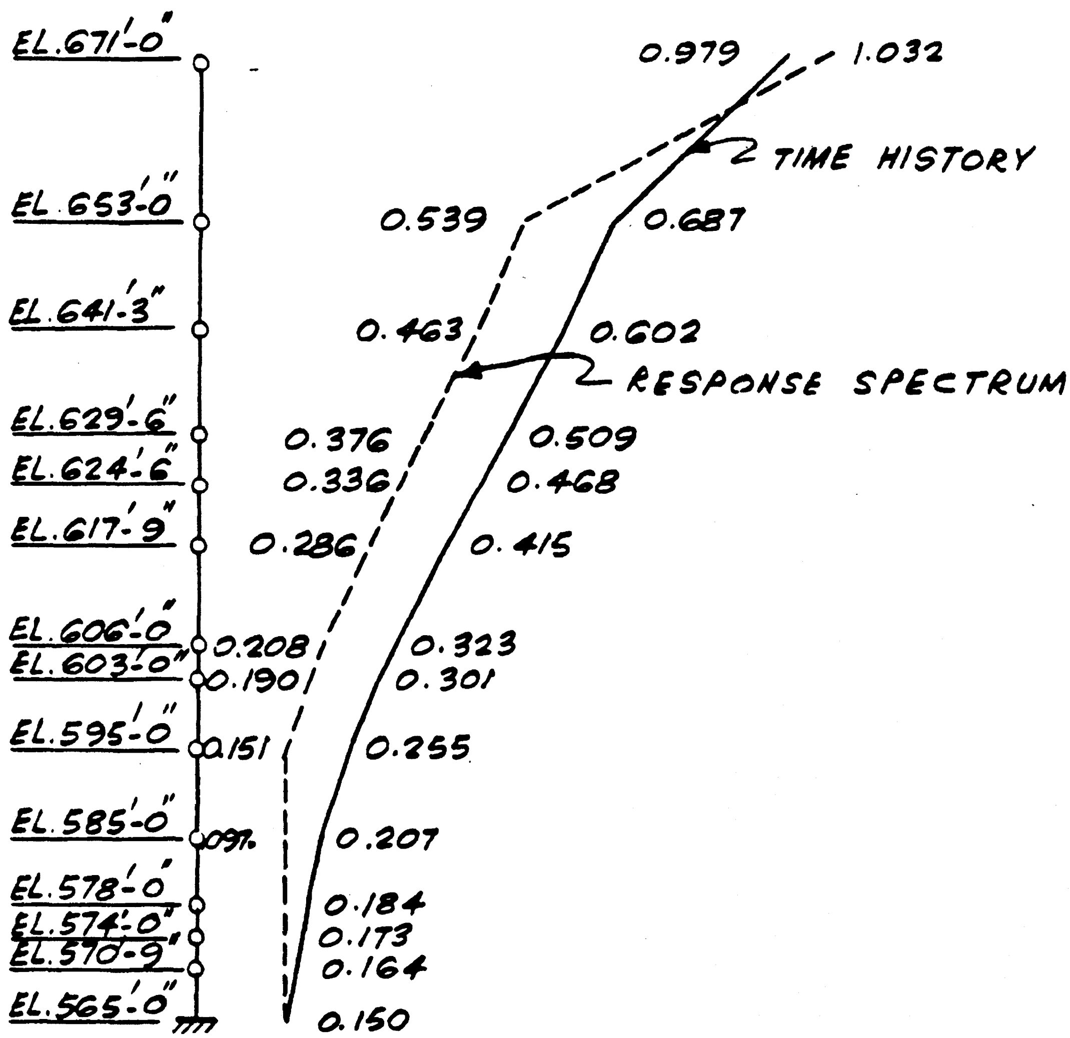
JULY 1982



COMPARISON OF TIME HISTORY & RESPONSE SPECTRUM ACCELERATIONS

HORIZONTAL (EAST-WEST)

MAXIMUM POSSIBLE EARTHQUAKE (0.15g)

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT INTERNAL STRUCTURES - TIME HISTORY & RESPONSE SPECTRUM COMPARISON
(MAXIMUM POSSIBLE EARTHQUAKE)
FIGURE 3.7-15

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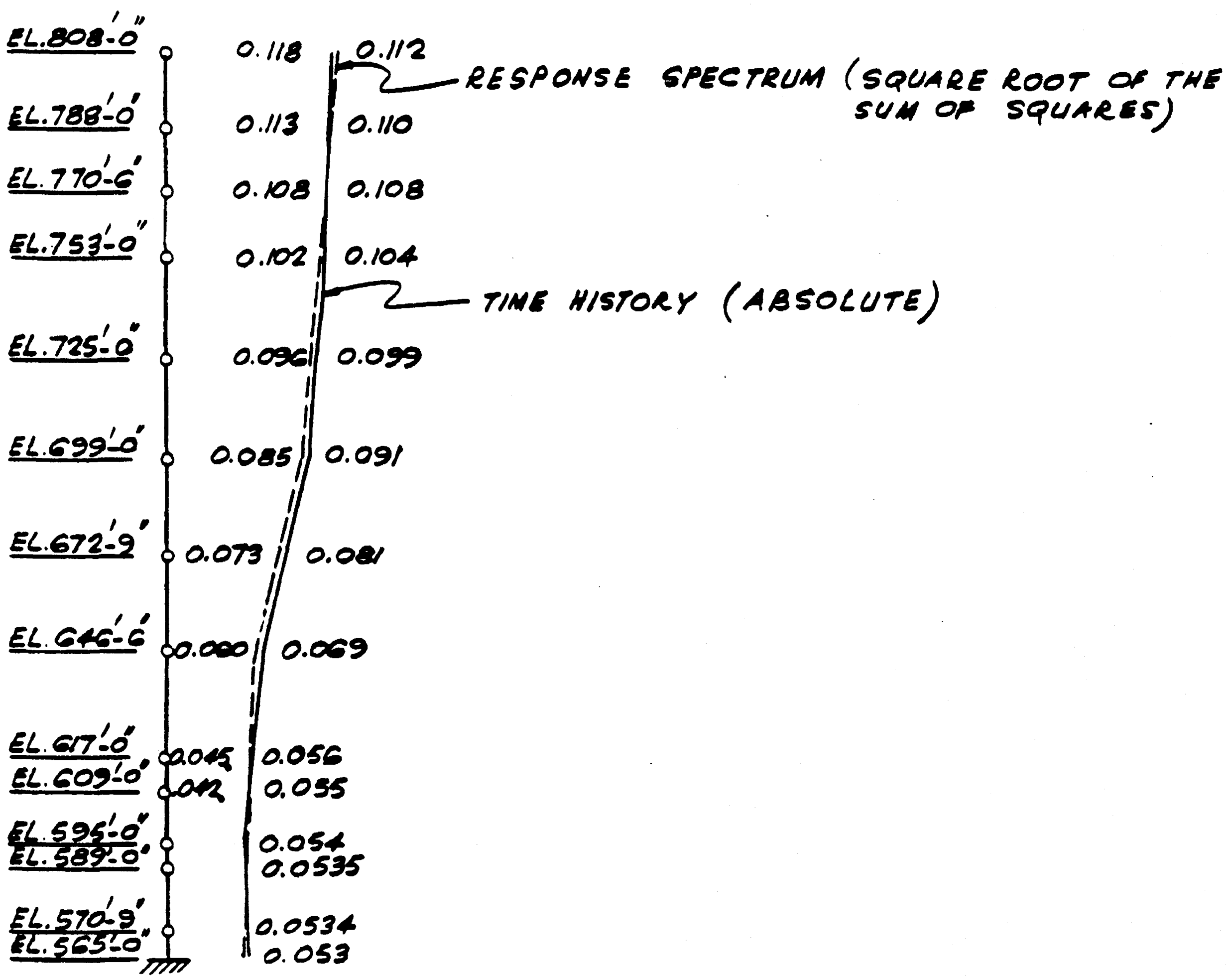

COMPARISON OF TIME HISTORY &
RESPONSE SPECTRUM ACCELERATIONS

VERTICAL

MAXIMUM PROBABLE EARTHQUAKE (0.053 g)

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT VESSEL TIME HISTORY &
RESPONSE SPECTRUM COMPARISON
FIGURE 3.7-16

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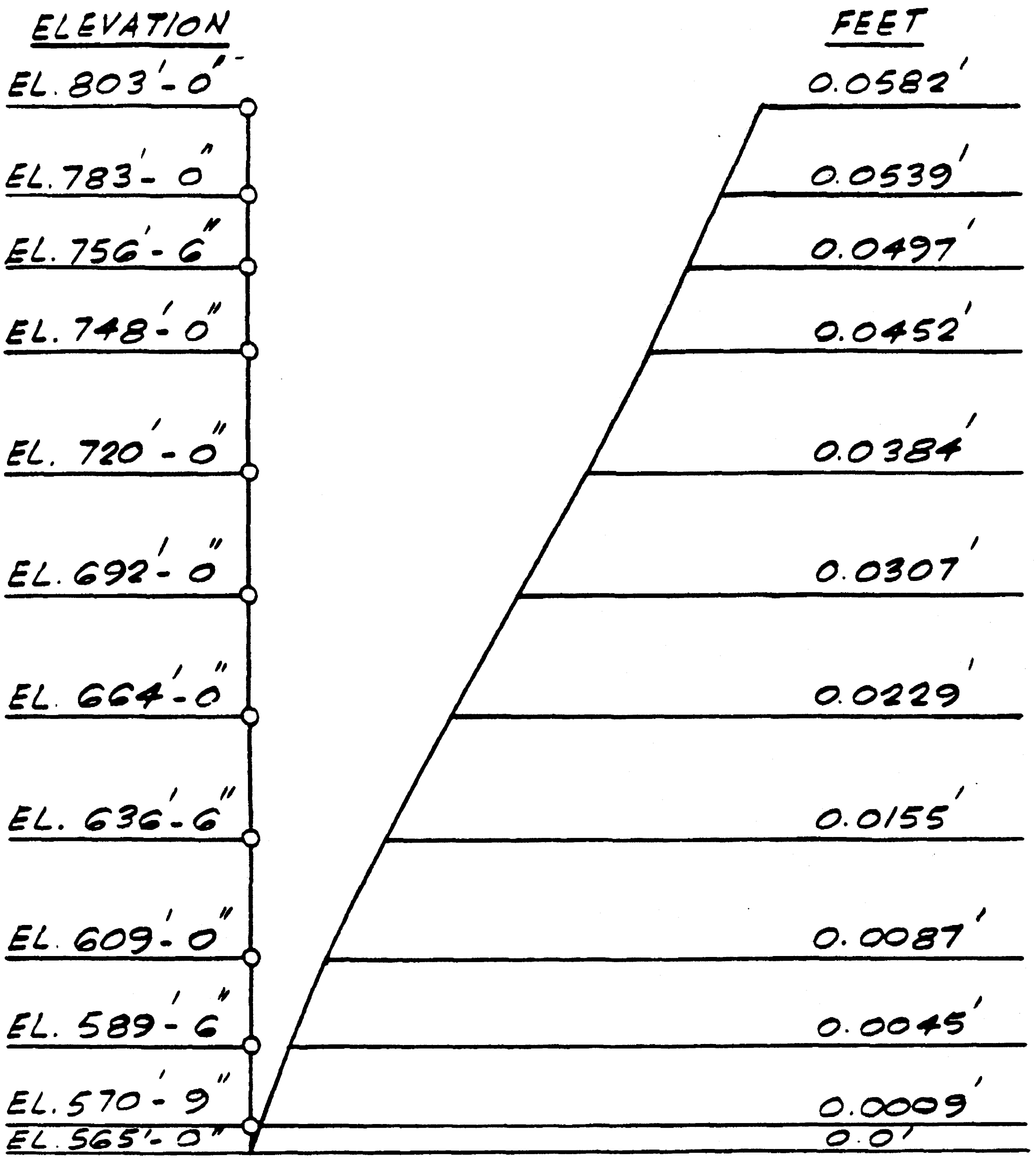

MAXIMUM POSSIBLE EARTHQUAKE (0.15g)

DAVIS-BESSE NUCLEAR POWER STATION
DIFFERENTIAL DISPLACEMENT BETWEEN SHIELD BUILDING & CONTAINMENT VESSEL
FIGURE 3.7-17

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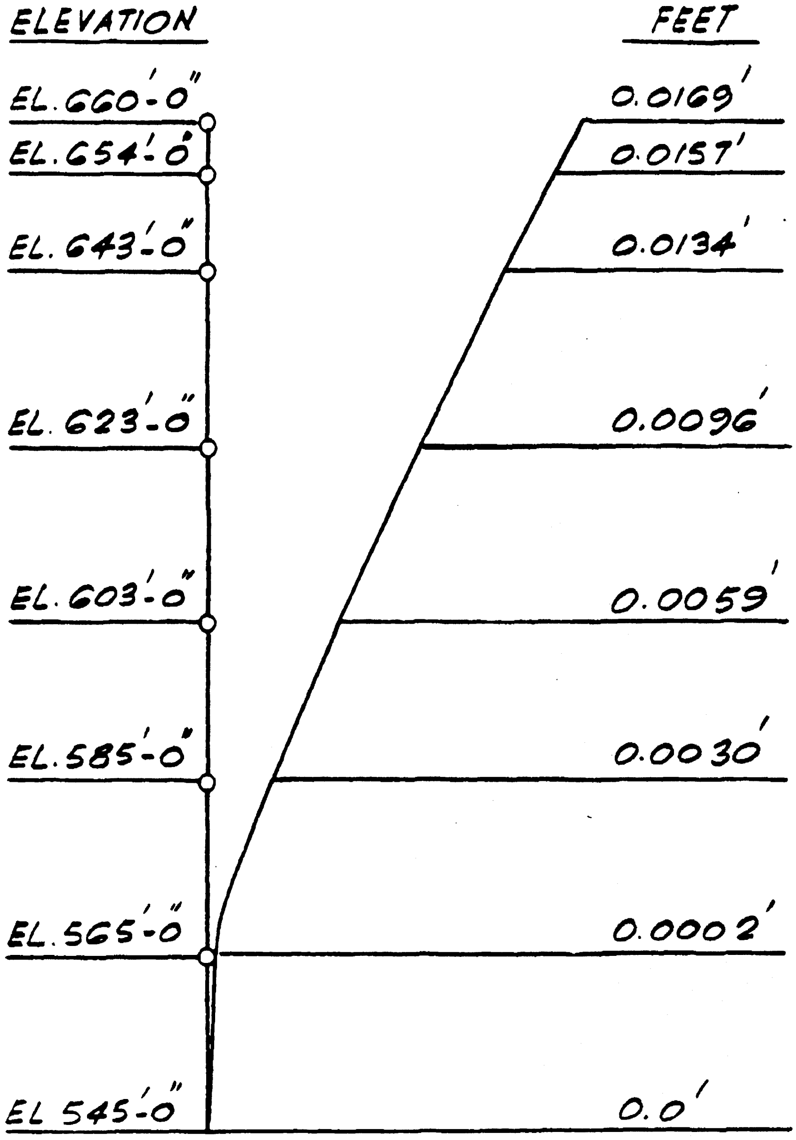

MAXIMUM POSSIBLE EARTHQUAKE (0.15g)

DAVIS-BESSE NUCLEAR POWER STATION
DIFFERENTIAL DISPLACEMENT BETWEEN SHIELD BUILDING & AUXILIARY BUILDING
FIGURE 3.7-18

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3.8 DESIGN OF SEISMIC CLASS I AND CLASS II STRUCTURES

The definitions of Class I and Class II are given in Subsection 3.2.1.1.

Piping System supports and anchors within these structures utilize loading combinations given in Table 3.9-3 for piping. Standard hangers and snubbers are designed per MSS-SP-69 for supporting piping and valves and allowable stresses as given in Table 2 of MSS-SP-58 are utilized. Nonstandard supports and anchors for piping and valves are designed per the following allowable stresses:

a. Normal Load Case
   Allowable = f_s per AISC

b. Upset Load Case
   Allowable = 1.25 f_s

c. Faulted Load Case
   Allowable = the lesser of

   i. 1.5 f_s

   ii. 0.9 F_y where F_y = yield strength of material at temperature, except that shear, F_v, shall not exceed 0.5 F_y

3.8.1 Seismic Class I Structures Other Than Containment

The Class I structures other than containment are:

1. Auxiliary Building

2. Intake Structure

3. Service Water Pipe Tunnel & Valve Rooms Number 1 and 2

4. Miscellaneous Structures:

   a. Seven Electrical Manholes (3001, 3004, 3005, 3006, 3020, 3041, and 3042)

   b. Borated Water Storage Tank

5. Emergency Diesel Fuel Oil Storage Tanks

3.8.1.1 Physical Separation

3.8.1.1.1 Auxiliary Building

The Auxiliary Building is located adjacent to both the Shield Building and Turbine Building. It is a five-story building with two levels below grade. The radwaste systems are housed in the basement. The remainder of the building is used for fuel storage and handling, auxiliary nuclear equipment and associated facilities, the control room, switchgear, and other operational facilities. The building is a reinforced concrete structure.

a. Fuel Storage and Handling Area:

The fuel storage area is located immediately adjacent to the Shield Building. This building accommodates the spent fuel storage pool and its spent fuel storage racks; cask pit, transfer pit, storage facilities for new fuel assemblies and control rods; a spent fuel cask washdown facility; and a fuel handling crane.

The 130 ton spent fuel cask crane has a single-failure-proof main hoist, which is capable of lifting the spent fuel transfer cask or shipping cask out of the cask pit and placing this cask on a trailer for transfer to the on-site Dry Fuel Storage Facility or on a spent fuel shipping cask transporter for shipment to an off-site facility. The weight of a fully loaded dry shielded canister (DSC) and transfer cask is within the capacity of the 130 ton cask crane single-failure-proof main hoist.

The spent fuel storage pool is constructed of reinforced concrete and is lined with 1/4-inch stainless steel plate to facilitate cleaning.

The design of the spent fuel cask crane prevents it from traveling over the spent fuel pool and cask pit unless a key operated by-pass switch is actuated.

The design and the safety evaluation of the spent fuel storage facility are described in Subsections 9.1.2.2 and 9.1.2.3, respectively.

New fuel assemblies are received in shipping containers and stored in the new fuel storage area, which is a separate and protected area for the dry storage of new fuel assemblies. The new fuel storage area sizing is discussed in Section 9.1.1.

The design and evaluation of the Dry Fuel Storage Facility (DFSF) components are described in the Davis-Besse Site Certified Safety Analysis Report (CSAR) for the AREVA / TN NUHOMS-24P Dry Shielded Canisters (DSCs), the AREVA / TN Standardized NUHOMS Updated Final Safety Analysis Report (UFSAR) NUH-003, Appendix U, for the 32PTH1 DSCs, and the AREVA / TN NUHOMS® EOS System Safety Analysis Report, for the EOS-37PTH DSCs. The DFSF is located within the Protected Area of the plant. The DFSF foundation is classified as not-important-to-safety in the NUHOMS UFSAR. The foundation has been designed for the applicable static loading combinations of ACI 318. The foundation was also evaluated for dynamic seismic loads.

b. Radiologically Restricted Area:

The radiologically restricted areas of the building on all five levels are grouped together next to the Shield Building to facilitate traffic control. Perimeter doors around the radiologically restricted areas, while providing radiologically emergency exit ways, are not opened from radiation uncontrolled areas without the use of a key. Emergency shower rooms and stairs within the radiologically restricted areas are provided to reduce traffic in and out of these areas and further facilitate control. Basement stairs exit into the Elevation 603 ft radiologically restricted area.

c. Control Room:

The control room and its supporting facilities, including a computer room, conference room, and test rooms as well as a kitchen and an office are grouped together in a shielded complex located so as to permit expansion, should a second unit be added to the station. This

expansion feature is further implemented by knock-out panels in the south exterior wall of the building and in other locations where connections with future structures is envisioned to be required.

The control room ceiling is an illuminative type.

d. Outside of Radiologically Restricted Areas and Equipment Rooms:

Outside of the radiologically restricted areas, but accessible to them, corridors with the appropriate fire resistance rating are provided for normal and emergency passageway as discussed in the FHAR. Three stairways, with one extending to the roof serve the corridors. All of these stairways are encased by enclosures with at least two-hour fire resistance rating.

Two elevators furnish vertical transportation between the five levels of the building. One of the elevators is for limited access only.

3.8.1.1.2 Intake Structure

The following major facilities, related to station safety and the Circulating Water System, are located in the Intake Structure:

a. Service Water pumps (Class I)

b. Cooling tower water makeup pumps (Class II) See Subsection 3.6.2.7.2.7

c. Diesel driven fire water pump (Class II)

d. Water treatment makeup pumps (Class II)

e. Traveling screens (Class II)

The reinforced concrete substructure of the Intake Structure (enclosure for the Service Water Pumps) is designed as a Class I structure. There are three floors, two of which accommodate all the pumps, traveling screens and other equipment. The third floor is used as a secondary laydown area. The Class II structural steel superstructure is provided for Class II equipment on the second floor. A 40-ton gantry crane has been provided for equipment services and maintenance.

3.8.1.1.3 Service Water Pipe Tunnel and Valve Rooms

The Service Water Pipe Tunnel is located between the Auxiliary Building and Intake Structure. This reinforced concrete tunnel is buried underground and shields the Service Water pipes and other minor pipes. Valve Room No.1 is located adjacent to the Auxiliary Building in the Turbine Building. Valve Room No.2 is located adjacent to the Intake Structure. Both Valve Rooms are single reinforced concrete rooms, housing the required valves and connections for the Service Water pipes. These structures are designed as Class I structures.

3.8.1.1.4 High Density Fuel Storage Racks

The Davis-Besse Nuclear Power Station Unit 1 high density fuel storage racks have been designed to meet the requirements for Seismic Class I structures. Detailed structural and seismic analyses of the high density storage racks have been performed to verify the adequacy

of the design to withstand the loadings encountered during installation, normal operation, the severe and extreme environmental conditions of the operating basis and safe shutdown earthquakes, and the abnormal loading condition of an accidental fuel assembly drop event.

Spent Fuel Pool Storage Racks

High density spent fuel storage racks have been installed in the spent fuel pool. These racks have been designed to meet the requirements for Seismic Class I structures. The racks are designed in accordance with ASME Code Section III, Subsection NF for Class 3 Components. The structural and seismic analyses for the racks has verified their adequacy to withstand all required loading conditions encountered during installation, normal operation, extreme environmental conditions (seismic), and abnormal loading condition of an accidental fuel assembly drop (References 88-90).

The structural response of a free-standing rack module to seismic inputs is highly nonlinear and involves a complex combination of motions (sliding, rocking, twisting, and turning). This results in impacts and friction effects. Holtec International performed the rack analysis using their proprietary “Whole Pool Multi-Rack” (WPMR) computer program. This program is used to determine the loads and displacements for each storage rack, establish the limiting relative motion between racks, and to evaluate the potential for and consequences of rack-to-rack and rack-to-wall impact. The rack cellular structure is modeled by a 3D beam, see USAR Figure 9.1-3b. The fuel assembly mass is modeled as shown in the figure.

The results of the rack analyses conclude that the spent fuel pool storage racks are adequately designed for the associated loading conditions.

For additional information about the structural analysis and design see Reference 88-90.

a. Description of the spent fuel pool racks.

Each fuel storage rack consists of either a square or rectangular array of square stainless steel storage cells spaced a nominal 9.22 inches on center in each direction. The spent fuel pool is licensed for a total of twenty one storage rack sections as shown in Figure 3.8-1.

The fuel storage racks are a freestanding cellular structure, composed of vertical cells and a common base plate. Each storage cell is nominally 9 inches square by 162 inches long with 0.075 inch walls. The vertical cells are welded together and at the common rack base plate. Each fuel assembly storage cell has a square opening with conforming lateral support and a flat horizontal bearing surface.

Loads are transmitted to the spent fuel pool floor, consisting of a reinforced concrete structure and a stainless steel liner, via four adjustable legs in the corners of the rack. The cask pit floor has been evaluated for the rack dead, thermal, seismic (vertical and horizontal), and accident loading combinations. The spent fuel pool floor, including the stainless steel liner plate, was determined to be adequate for these imposed loads (References 86 & 87).

b. Applicable Codes, Standards and Specifications

The following design codes and specifications have been used on the design/analysis of the spent fuel pool fuel storage racks.

1. ASME Code, Section III, Subsection NF, 1986

2. Specification No.C-63Q, Rev. 3

c. Loading Conditions (Reference 88):

The following loading cases and load combinations have been considered in the analysis.

Load Case 1 — Dead Weight

Under normal operating conditions the rack is subjected to the dead weight of the rack structure plus the weight of the fuel assemblies.

Load Case 2 — Dead Weight + Thermal Loading (normal)

This loading condition evaluates the stresses due to thermal loadings plus the dead weight loads on the rack structure.

Load Case 3 — Dead Weight + Thermal (normal) + OBE

This load case combines the dead weight, thermal loads due to normal operation, and OBE seismic loadings.

Load Case 4 — Dead Weight + Thermal (accident) + OBE

This loading combination evaluates the dead weight, thermal loads due to postulated abnormal conditions, and OBE seismic loadings.

Load Case 5 — Dead Weight + Thermal (normal) + Uplift

The possibility of the fuel handling bridge fuel hoist grapple or a fuel assembly getting hooked on a cell was evaluated. The uplift force for this load case was 500 pounds.

Load Case 6 — Dead Weight + Thermal (accident) + SSE

This loading condition is the same as Load Case 4, except that SSE seismic loadings were considered.

Load Case 7 — Dead Weight + Thermal (normal) + Load Drop

The possibility of dropping a fuel assembly on the rack has been evaluated. Two cases were considered: 1) a direct drop on the top of a single storage cell, and 2) a direct drop of an assembly falling straight through the length of the cell impacting the bottom plate. (References 88 and 90)

d. Structural Acceptance Criteria

The following allowable limits constitute the structural acceptance criteria used for the loading combinations described above.

Loading Combinations	Service Level
1,2,3	A
4,5	B
6	D
7	See e.3 below

The stress limits are derived from the ASME Code, Section III, Subsection NF.

e. Conclusions:

1. The results of the structural and seismic analysis document that the stresses in the racks resulting from the loadings discussed above are within the allowable stress levels of ASME III, Subsection NF for Class 3 Components.

2. The analysis has determined that there will be impact loadings due to movement/uplift of the racks under seismic loading conditions, but that the racks will not overturn. The base plates, of adjacent rack sections, will impact each other. However, the impact will have only localized effects on the plate. There will be no other impacts between either adjacent rack sections or between a rack and the spent fuel pool walls. Impact loads will be transmitted to the spent fuel pool floor, through the rack legs and base plates, due to uplift of the rack under seismic loads. The spent fuel pool floor, reinforced concrete structure and stainless steel liner plate, will maintain their structural integrity and will not be punctured.

3. The analysis of the two accidental fuel assembly drop conditions indicate that the racks will remain functional. The functionality of the racks is defined as follows: 1) For the dropped fuel assembly impacting the top of the rack there will be acceptable local structural damage above the top of the active fuel region, and there will be no buckling or collapse, 2) For the dropped fuel assembly falling directly through a storage cell the bottom base plate will have localized deformation/deflection, but the pool liner plate will remain intact.

4. Sloshing was found to be negligible at the top of the rack and is, therefore, neglected in the analysis.

3.8.1.1.5 Miscellaneous Structures

Electrical Manholes:

Five Electrical Manholes (3001, 3004, 3005, 3041, and 3042) are classified as Class I structures and are designed to withstand tornado loads. Two Electrical Manholes (3006 and 3020) are classified as Class I structures but are not required to withstand tornado loads, since the Class I E circuits contained in these manholes are for the Seismic Monitoring System and not required for safe shutdown following a tornado. All Electrical Manholes are supported on the structural granular Class I backfill.

Borated Water Storage Tank (BWST):

The BWST foundation is a reinforced concrete mat supported on the structural granular Class I backfill. This tank is located west of the Shield Building.

The BWST is not required for the safe shutdown of the plant following a tornado and its availability following a tornado is not required. It is noted that a simultaneous LOCA (when the tank's content are needed) and a tornado are not postulated to occur.

The BWST does not require protection from potential missiles since:

1. Tornado generated missiles are not considered since the BWST availability is not required following a tornado.

2. The nitrogen storage tanks, which are the nearest potential source of missiles, are orientated to preclude a missile that could adversely affect the BWST.

Evaluation of the consequences of a postulated rupture of the BWST demonstrated that expected exposures will be within the limits set by 10CFR100 and 10CFR20.

3.8.1.1.6 Influence of Seismic Class II Structures on Seismic Class I Structures

There is no significant influence of any Class II structure on the Class I structures.

The Class II Containment Auxiliary Crane, its support bases and utility skid are analyzed for SSE and found to have adequate capacity and therefore will not collapse onto the Class I West Steam Generator compartment at EL. 653', or, the Operating floor at EL. 603.

The following Class I equipment/systems or structures are protected from any possible failure of Class II structures which enclose them.

a. Class I service water pumps and piping in the Intake Structure are below the top concrete slab and the Class II intake steel superstructure. The top concrete slab is 21 inches thick and is the protecting membrane against the unlikely failure of the Class II structure. In addition, the Class II Intake Structure Gantry Crane is analyzed for SSE and found to have adequate capacity and therefore will not collapse onto the Intake Structure.

b. Class I service water piping in the Class II Turbine Building is enclosed in Class I reinforced concrete pipe tunnel that is completely surrounded by a granular compacted fill with a minimum top cover of four feet. The 10 inch concrete ground floor slab bears on the compacted fill. The reinforced concrete tunnel, 4 feet of compacted fill cover and the 10 inch concrete ground floor slab protect the Class I piping against the unlikely failure of the Class II Turbine Building superstructure.

c. A small portion of the Class I reinforced concrete Auxiliary Building supports the structural steel framing for the heater bay of the Class II Turbine Building. Multi-level steel floor framing, the elevated and ground floor concrete slabs in the heater bay, and the reinforced concrete Auxiliary Building walls and slabs protect the Class I structure from the unlikely failure of the Class II structure and/or equipment.

3.8.1.1.7 Load Combinations

The design of the above Class I structures is based on the load combinations presented in Subsection 3.8.1.3.

All of the reinforced concrete structures were designed by the Ultimate Strength Method. All of the structural steel was designed by the Working Stress Method.

3.8.1.2 Codes

The following is the list of the codes, specifications, regulations, safety guides, and other similar documents used in establishing or implementing design bases and methods, analytical techniques, material properties, and quality control provisions.

Dates noted with the listed codes, specifications, etc. are the dates of the codes, specifications etc. that were in effect during the construction phase of the plant.

Editions of the codes, specifications, etc., used in the operational phase of the plant are noted in the design specifications, drawings and/or procedures that govern plant modification activities.

3.8.1.2.1 American Society for Testing and Material (ASTM)

A-36	-	Structural Steel
A-48	-	Gray Iron Castings
A-53	-	Welded & Seamless Steel Pipe
A-185	-	Welded Steel Wire Fabric For Concrete Reinforcement
A-283	-	Low and Intermediate Tensile Strength Carbon Steel Plates of Structural Quality
A-307	-	Low Carbon Steel Externally and Internally Threaded Standard Fasteners
A-325	-	High Strength Bolts For Structural Steel
A-446	-	Zinc Coated (Galv.) Steel Sheets of Structural Quality, Coils and Cut Lengths
A-615	-	Deformed Billet-Steel Bars For Concrete Reinforcement
A-992	-	Standard Specification for Steel for Structural Shapes for use in Building Framing
C-31	-	Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field
C-33	-	Concrete Aggregates
C-39	-	Test for Compressive Strength of Molded Concrete Cylinders
C-94	-	Ready-Mix Concrete
C-131	-	Test for Resistance for Abrasion of Small Size Coarse Aggregate
C-138	-	Test for Unit Weight, Yield, and Air Content of Concrete
C-144	-	Aggregate for Masonry Mortar
C-150	-	Portland Cement
C-156	-	Test for Water Retention by Concrete Airing Materials
C-171	-	Sheet Materials for Curing Concrete
C-205	-	Specification for Portland Blast Furnace Slag Cement
C-207	-	Hydrated Lime for Masonry Purposes
C-260	-	Air Entraining Admixtures for Concrete
C-289	-	Test for Potential Reactivity of Aggregates
C-295	-	Practice for Petrographic Examination of Aggregates for Concrete

C-478	-	Pre-cast Reinforced Concrete Manhole Sections
C-498	-	Perforated Clay Drain Tile
C-595	-	Standard Specification for Blended Hydraulic Cements
C-618	-	Fly Ash and Raw or Calcined Natural Pozzolans for use in Portland Cement Concrete
C-989	-	Standard Specification for Slag Cement for use in Concrete and Mortars
D-423	-	Test for Liquid Limit of Soils
D-424	-	Test for Plastic Limit and Plasticity Index of Soils
D-448	-	Standard Sizes of Coarse Aggregate for Highway Construction
C-698	-	Test for Moisture-Density Relations of Soils
D-994	-	Preformed Expansion Joint Filler for Concrete
D-1190	-	Concrete Joint Sealer, Hot-Poured Elastic Type
D-1556	-	Test for Density of Soil in Place by the Sand-Cone Method
D-1751,2	-	Preformed Expansion Joint Fillers for Concrete Paving and Structural Construction
D-1850	-	Concrete Joint Sealer Cold-Application Type
D-2049	-	Test for Relative Density of Cohesionless Soils

3.8.1.2.2 American Concrete Institute (ACI)

Code 211 (613-54)	-	Recommended Practice for Selecting Proportions for Concrete
Code 301-66	-	Specifications for Structural Concrete for Buildings
Code 304-59 (614-59)	-	Recommended Practice for Measuring, Mixing, and Placing Concrete
Code 305-59 (605-59)	-	Recommended Practice for Hot Weather Concreting
Code 306-66	-	Recommended Practice for Cold Weather Concreting
Code 307-69 (505-54)	-	Specifications for the Design and Construction of Reinforced Concrete Chimneys
Code 315-65	-	Manual of Standard Practice for Detailing Reinforced Concrete Structures
Code 318-63	-	Building Code Requirements for Reinforced Concrete Structures
Code 347-68	-	Recommended Practice for Concrete Formwork
Code 349-85	-	Code Requirements for Nuclear Safety Related Structures

3.8.1.2.3 American Water Works Associated (AWWA)

C-201-66	Standard for Fabricated Electrically Welded Steel Pipe
C-203-66	Standard for Coal-Tar-Enamel Protective Coatings for Steel Water Pipe
C-206-62	Standard for Field Welding of Steel Water Pipe Joints
C-207-55	Standard for Steel Pipe Flanges
C-208-59	Standard for Dimensions for Steel Water Pipe Fittings
M-11	Manual for Steel Pipe-Design and Installation
D-100-67	Standard For Steel Tanks

3.8.1.2.4 American Association of State Highway Officials (AASHO)

T-26-51	Method of Sampling and Testing Water
M-36-64	Corrugated Metal Pipe for Sewers
M-73-67	Specifications for Cotton Mat Surfacing for Concrete
M-190-65	Corrugated Metal Pipe for Sewers

3.8.1.2.5 American Society of Mechanical Engineers (ASME)

a. Boiler and Pressure Vessel Code:

Section III	Nuclear Vessels
Section VIII	Pressure Vessels
Section IX	Welding Qualifications
Section XI	In-service Inspection of Nuclear Reactor Coolant Systems

b. Material Specifications:

SA 193	B8 Alloy Steel Bolting Materials (High Temp)
SA 299	Carbon-Manganese-Silicon Steel Plates for Pressure Vessels
SA 320	Gr. L43 Alloy Steel Bolting Materials (Low Temp)
SA 333	Seamless and Welded Steel Pipe (Low Temp)
SA 350	Gr. LF1 & Forged Carbon Steel (Low Temp)
SA 516	Gr. 70 Carbon Steel Plates for Pressure Vessels

3.8.1.2.6 American Welding Society (AWS)

D 1.0-66	Standard Code for Arc and Gas Welding in Building Construction
D 1.3	Structural Welding Code - Sheet Steel
D 2.0-69	Specifications for Welded Highway and Railway Bridges
D 5.2-67	Standard for Steel Tanks, Stand Pipe Reservoirs, and Elevated Tanks for Water Storage
D 7.0-62	Standard for Field Welding of Steel Water Pipe Joints
D 9.1	Specification for Welding of Sheet Metal
D12.1-61	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction
D1.4-98	Structural Welding Code — Reinforcing Steel

3.8.1.2.7 Steel Structures Painting Council (SSPC)

Surface Preparation Specifications:

SP-1	Solvent Cleaning
SP-2	Hand Tool Cleaning
SP-3	Power Tool Cleaning
SP-5	White Metal Blast Cleaning
SP-6	Commercial Blast Cleaning
SP-10	Near White Blast Cleaning

3.8.1.2.8 Others

a. Uniform Building Code (Reference 30)

b. Ohio Building Code (OBC)

c. American Institute of Steel Construction (AISC):

1. Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

2. Specification for the Design of Light Gauge Cold-Formed Steel Structural Members.

d. American Petroleum Institute (API):

1. Specification No. 620 - Recommended Rules for Design and Construction of Large, Welded Tanks

2. Specification No. 650 - Welded Steel Tanks for Oil Storage

e. American Society of Civil Engineers (ASCE):

1. Paper No. 3269 “Wind Forces on Structures”

f. Basic Fire Protection for Nuclear Power Stations Agencies:

1. American Nuclear Insurers (ANI)

2. Nuclear Electric Insurance Limited (NEIL)

g. Corps of Engineers:

1. Specification CRD-513 Rubber Water Stops

2. Specification CRD-572 PVC Water Stops

h. Federal Specifications:

1. AAA - S-121D - Scales Testing

2. TTP - 86 (e) - Paint-Red Lead Base Ready-Mix

i. American National Standards Institute (ANSI):

1. B31.1.1-1967 Specification for Pressure Power Piping

j. U.S. Atomic Energy Commission (AEC):

1. ORNL-NSIC22, US-80 Missile Generation and Protection in Light Water-Cooled Power Reactor Plants (Reference 19)

k. Nuclear Construction Issues Group (NCIG):

1. NCIG-01 Rev. 2 Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants.

2. NCIG-03 Rev. 0 Training Manual for Inspectors of Structural Welds at Nuclear Power Plants Using the Acceptance Criteria of NCIG-01.

3.8.1.3 Load Combinations

The load combinations which were applied in the design and analysis of the above structures are described below, except for the spent fuel pool area of the Auxiliary Building. The spent fuel pool area of the Auxiliary Building has been re-analyzed for additional spent fuel storage racks using the load combinations of the Standard Review Plan Section 3.8.4, as described in References 84 and 87.

3.8.1.3.1 Class I Structures

a. Operation during Normal and Maximum Probable (Smaller) Earthquake Conditions:

For loads encountered during normal station operation, Class I structures are designed in accordance with the following load combinations:

1. Concrete

$U = 1.5D + 1.8L$

$U = 1.25(D+L+H_o+E) + 1.0T_o$

$U = 1.25(D+L+ H_o+W) + 1.0\ T_o$

$U = 0.9D+1.25(H_o+E) + 1.0T_o$

$U = 0.9D+1.25(H_o+W) + 1.0\ T_o$

In addition, for ductile moment resisting concrete space frames, shear walls, and braced frames, the following load combinations were used:

$U = 1.4(D+L+E) + 1.0\ T_o + 1.25\ H_o$

$U = 0.9D+ 1.25E+ 1.0\ T_o + 1.25\ H_o$

For structural elements carrying mainly earthquake forces, such as equipment supports:

$U = 1.0D+ 1.0L+ 1.8E+ 1.0\ T_o + 1.25\ H_o$

2. Yield Capacity Reduction Factors:

The yield capacity of all load carrying structural elements was reduced by a yield capacity reduction factor (ø) as given below. This factor provided for the possibility that small adverse variations in material strengths, workmanship,

dimensions, control, and degree of supervision while individually within required tolerance and the limits of good practice, occasionally may combine to result in under-capacity. This factor was applied to the design of the concrete structures by using the Ultimate Strength Design Method.

Yield Capacity Reduction Factors:

ø = 0.90 for concrete in flexure.

ø= 0.85 for shear/diagonal tension, bond, and anchorage in concrete.

ø= 0.75 for spirally reinforced concrete compression members.

ø= 0.70 for tied compression members.

ø= 0.90 for fabricated structural steel

ø= 0.90 for reinforcing steel in tension (excluding splices).

ø= 0.90 for reinforcing steel in tension with mechanical splices.

3. Structural Steel

Steel structures were designed by the Working Stress Design Method to satisfy the following loading combinations without exceeding the specified stresses:

D + L	Stress Limit = f_s
D + L + T_o + H_o + E	Stress Limit = 1.25 f_s
D + L + T_o + H_o + W	Stress Limit = 1.33 f_s

In addition, for structural elements carrying mainly earthquake forces, such as struts and bracings:

D + L + T_o + H_o + E	Stress Limit = f_s

b. During Accident and Maximum Possible (Larger) Earthquake Conditions:

The Class I structures in general are proportioned to maintain elastic behavior when subjected to various combinations of dead loads, thermal loads, seismic and accident loads. The upper limit of elastic behavior is considered to be the yield strength of the effective load-carrying structural materials. The yield strength for steel (including reinforcing steel) is considered to be the guaranteed minimum given in appropriate ASTM specifications. The yield strength for reinforced concrete structures is considered to be the ultimate resisting capacity as calculated from the ‘Ultimate Strength Design’ portion of the ACI-318-63 code.

1. Concrete:

   Concrete structures satisfy the most severe of the following loading combinations:

   $U = 1.0D + 1.0L + 1.25\,E + 1.0\,T_A + 1.0\,H_A + 1.0R$

   $U = 1.0D + 1.25\,E + 1.0\,T_A + 1.0\,H_A + 1.0R$

   $U = 1.0D + 1.0L + 1.0\,E' + 1.0\,T_O + 1.25\,H_O + 1.0R$

   $U = 1.0D + 1.0L + 1.0\,E' + 1.0\,T_A + 1.0\,H_A + 1.0R$

   $U = 1.0D + 1.0L + 1.0\,W' + 1.0\,T_O + 1.25\,H_O$

2. Structural Steel:

   Steel structures satisfy the most severe of the following loading combinations without exceeding the specified stresses:

$D + L + R + T_O + H_O + E'$	Stress Limit = $1.5\,f_s$*
$D + L + R + T_A + H_A + E'$	Stress Limit = $1.5\,f_s$*
$D + L + W^1 + T_O + H_O$	Stress Limit = $1.5\,f_s$*

   *Maximum allowable stress in bending and tension is $0.9F_y$. Maximum allowable stress in shear is $0.5F_y$.

3.8.1.3.2 Class II Structures

a. Concrete:

   Load factors and combinations as specified apply. All other design, except load factors and combinations, is as specified in the ACI Standard 318-63. Significant thermal loads are included. The following load factors and combinations were used for ductile moment space frames and shear walls:

   $U = 1.4\,(D+L+E) + 1.0\,T_O + 1.25\,H_O$

   $U = 0.9\,D + 1.25E + 1.0\,T_O + 1.25\,H_O$

b. Structural Steel:

Steel structures were designed based on the following loading combinations without exceeding the specified stresses:

D + L	Stress Limit = f_s
$D + L + T_O + H_O + E$	Stress Limit = 1.33 f_s
$D + L + T_O + H_O + W$	Stress Limit = 1.33 f_s

3.8.1.3.3 Notations

U = Required ultimate load capacity

D = Dead load of structure and equipment plus any other permanent loads contributing stresses, such as soil or hydrostatic loads. An allowance is also made for future permanent loads.

L = Live load and piping loads.

R = Force or pressure on structure due to rupture of any one pipe.

T_O = Thermal loads due to temperature gradient through wall under operating conditions

H_O = Force on structure due to thermal expansion of pipes under operating conditions.

T_A = Thermal loads due to temperature gradient through wall under accident conditions.

H_A = Force on structure due to thermal expansion of pipes under accident conditions.

E= “Maximum Probable (Smaller) Earthquake” resulting from ground surface acceleration of 0.08g in the horizontal direction.

E’ = “Maximum Possible (Larger) Earthquake” resulting from ground surface acceleration of 0.15g in the horizontal direction.

W = Wind load. (Wind velocity 90 mph at 30 ft above ground. See ASCE 3269 for increase due to gusts and height.)

W’ = Tornado load including differential pressure.

ø = Capacity reduction factor. (Defined in ACE-318-63 code, Section 1504)

f_s = Allowable stress for structural steel. (Defined in AISC, Section 1.5)

F_y = Yield strength for steel. See subsection 3.8.1.3.1.b.

3.8.1.4 Analytical Techniques for Loads

The analytical techniques for all Class I structures are divided into the following areas:

1. Structure, piping and equipment dead loads.
2. Live loads.
3. Earthquake loads (Maximum Possible and Maximum Probable)
4. Lateral Earth pressure loads.
5. Tornado loads.
6. Hydrodynamic loads.
7. Thermal loads.
8. Pipe rupture loads, pipe whipping loads, and jet forces.
9. Missile loads.
10. Pressure loads.

3.8.1.4.1 Dead Loads

Dead loads consist primarily of the weight of the concrete, block walls, structural steel, interior partitions, equipment, major piping, electrical conductors, cable trays, and ventilating ducts.

3.8.1.4.2 Live Loads

Live loads for the design for the structural framing consist of predicted loads for intended use of the structure and those loadings recommended in the applicable codes:

a. Floor loads.

b. Snow loads.

c. Spent fuel cask loads.

3.8.1.4.3 Earthquake Loads

All Class I structures are designed for a Maximum Probable Earthquake of 0.08g in the horizontal direction and a Maximum Possible Earthquake of 0.15g in the horizontal direction. The vertical ground accelerations are 0.053g for the Maximum Probable Earthquake and 0.10g for the Maximum Possible Earthquake and are considered acting simultaneously with the horizontal acceleration. The method of analysis is outlined in Subsection 3.7.2.1.

3.8.1.4.4 Lateral Earth Pressure Loads

Active and passive lateral earth pressure loads due to soil backfill around Class I structures are considered using the theories of soil mechanics and then combined with other loads.

Backfill pressure around seismic class I structures has been calculated, based on Coulomb's theory, as follows:

$$p = k_a wh$$

where k_a = coefficient of active earth pressure
w = unit weight of soil (lbs/ft³)
h = depth of soil (ft)

$$k_a = \frac{\cos^2\ \text{ø} - \beta)}{\cos^3\beta\left[1+\left[\dfrac{\sin\ \text{ø}\ \sin(\text{ø}-\sigma)}{\cos\beta\ \cos(\sigma-\beta)}\right]^{1/2}\right]^2}$$

where ø = angle of friction. Other angles are explained in Figure 3.8-13.

Additional pressures due to earthquakes were used in accordance with "Manuals-Corps of Engineers, U.S. Army, EM 1110-2-2502."

The earthquake effect of soil pressure was taken into account by increasing the static pressure by 20 percent. The increased water pressure caused by an earthquake was calculated by using Westergaard's parabola (ref. 5) as shown in Figure 3.8-14.

$$P_e = \frac{2}{3} C_e \alpha h_2^2$$

where: P_e = the additional total water pressure

α = the ratio of the earthquake acceleration to g

C_e = a factor depending on physical conditions, principally the height and the earthquake period (varies from 52 to 61)

h_2 = the depth of water

3.8.1.4.5 Tornado Loads

Tornado loads for the Class I structures are outlined in Subsection 3.3.2.

3.8.1.4.6 Hydrodynamic and Hydrostatic Loads

Hydrodynamic forces on Class I structures are discussed in Section 3.4.

3.8.1.4.7 Thermal Loads

Thermal loads resulting from the differential temperatures between operating and construction, and between operating and pipe rupture are considered in the design of the Class I structures. The design load combinations are given in Subsection 3.8.1.3.

3.8.1.4.8 Pipe Rupture Loads, Pipe Whipping Loads, and Jet Forces

All Class I pressure-containing pipes, 2 inches or more in diameter, are anchored and restrained against piping accident loads such that, the Class I structures, systems, and components are not damaged. For a further discussion see Section 3.6.

3.8.1.4.9 Missile Loads

All external and internal missiles and their analyses are discussed in Section 3.5.

3.8.1.4.10 Pressure Loads

The pressure built-up due to the rupture of the pressure containing pipes and the external pressure drop due to tornado forces are considered independently in the design of all Class I structures. See Subsections 3.8.2.3 and 3.3.2.

3.8.1.5 Principal Design Methods

The principal design methods for all Class I structures are as follows:

Ultimate Strength Design Method per ACI Code 318-63 is used for the design of reinforced concrete structures. The load combinations and factors other than ACI Code are shown in Subsection 3.8.1.3.

The water temperature in the spent fuel pool will be maintained below 150°F for long term fuel storage conditions. The 150°F temperature limit for the pool is in accordance with the requirements of ACI 349, Appendix A. The thermal-hydraulic analysis for the spent fuel pool storage racks has determined that there is a potential for short term (less than 100 hours) water temperatures of about 154°F with component cooling water (CCW) at 97°F (Reference 91). This short term elevated temperature is less than the 350°F short term temperature limit specified in Appendix A of ACI 349. See USAR Section 9.1 for information about cooling the water in the spent fuel pool.

Working Stress Method in accordance with AISC “Manual of Steel Construction,” Sixth Edition is used for the design of steel structures. For load combinations and factors see Subsection 3.8.1.3.

The calculated maximum stresses and strains, based on the Ultimate Strength Design, for the Class I structural components are as follows:

STRUCTURE	Stress, ksi		Strain, in/in	
	Concrete	Rein. Steel	Concrete	Rein. Steel
Auxiliary Building Walls:	2.930	.70	0.00010	.00026
Intake Structure	3 .586	.55	0.00048	.00188
Valve Rooms:				
No. 1	3.613	.70	0.00044	.00171
No.2	3.519	.36	0.00030	.00191
Service Water Pipe Tunnel	3.598	.60	0.00034	.00164

The maximum stress and strain for structural steel in the Auxiliary Building are 22.2 ksi (axial), 1.23 ksi (bending) and 0.00081 in/in., respectively.

The values presented above are historical in nature. These values were calculated from the results of the computer programs used for the original design of the structures. The current design basis calculations should be consulted for the actual stress/strain values of these structures.

In general, all structures are proportioned to maintain elastic behavior under the worst credible load combination. The upper limit of elastic behavior is considered to be the yield strength of the effective load-carrying structural material.

The following are the allowable stresses and strains used in design of the structures:

a. Concrete:

f'_c= 5000 psi, @ 28 days	f'_c = 4000 psi, @ 28 days
E_c = 4,000,000 psi	E_c = 3,600,000 psi
n = 7	n = 8
f_c= 0.85 f'_c	f_c= 0.85 f'_c
ε_c = 0.003 in/in (Ultimate)	ε_c = 0.003 in/in (Ultimate)

The allowable shear, bond, and anchorage stresses recommended by ACI Code 318-63 were applied.

b. Structural Steel:

F_u = minimum ultimate strength 58,000
F_y = minimum yield strength, 36,000 psi or as described in this subsection
F_b = Flexural Stress, 24,000 psi (compact shapes, adequately braced)
F_v = shear stress, 14,500 psi
E_s = 29,000,000 psi
ε_s = 0.00124 in/in

NOTE: These values are for ASTM A-36. Corresponding values for other structural steels are in accordance with the AISC "Manual of Steel Construction."

c. Reinforcing Steel:

Billet, ASTM-A615-Grade 60
F_u = minimum ultimate strength, 90,000 psi
F_y = minimum yield strength, 60,000 psi
E_s = 29,000,000 psi
ε_s = 0.00207 in/in

d. Horizontal Dowels Between Shield Wall and Internal Concrete Fill:

Billet, ASTM-A615-Grade 40
F_u = minimum ultimate strength, 60,000 psi
F_y = minimum yield strength, 40,000 psi
E_s = 29,000,000 psi
ε_s = 0.00207 in/in

See Subsection 3.8.1.3 for other allowable stresses related to each loading combination.

There was no deformation permitted by design that could interfere with the functional capacity of any Class I structure or component which interacts with other structures.

The maximum deformation was considered and analyzed for each Class I structure. These structures were completely separated from each other by expansion joints which were filled with a resilient type of material that would permit relative movement (unrestrained) between these adjacent structures.

3.8.1.6 Principal Construction Material

3.8.1.6.1 Concrete

Type I and II - Low Alkali Portland Cement	ASTM-C-150
Aggregates	ASTM-C-33
Mortar Sand	ASTM-C-144
Ready Mix Concrete	ASTM-C-94
Flyash	ASTM-C-618
Slag	ASTM-C-595 / ASTM-C-989
Water	AASHO-T-26

3.8.1.6.2 Structural Steel

Main and Miscellaneous Steel	ASTM-A-36 or ASTM A-992
Embedded Metal	ASTM-A-36
Anchor Bolts	ASTM-A-36
High Strength Bolts	ASTM-A-325
Crane Rails	ASTM-A-1
Liner Plate (Stainless Steel)	AISI Type 304

3.8.1.6.3 Reinforcing Steel

Bars #3 through #18	ASTM-A-615 Grade 60

3.8.1.6.4 Others

All other secondary and miscellaneous materials used were as per standard industry specifications. See Subsection 3.8.1.2 for complete list of Codes and Standards used in this design. Non-standard materials were not used in the construction of the station.

No procedures involving unusual techniques or quality control standards in excess of normal construction practices were used in the construction of the station.

The Quality Control/Quality Assurance program is discussed in Chapter 17.

3.8.1.7 Test Procedures

3.8.1.7.1 Users Tests

To augment the testing performed by manufacturers or suppliers of various construction materials, considerable on-site testing is conducted on random samples of these materials in the form of user's tests.

a. Reinforcing Steel:

Random samples of reinforcing steel bars are selected for the purpose of making chemical analysis checks and mechanical property tests. The testing methods and minimum strength requirements are found in ASTM A615, Standard Specification for Billet-Steel Bars for Concrete Reinforcement. The sampling plan and acceptance criteria are found in the project specification.

b. Cadweld Splices:

The testing of reinforcing bars spliced by Cadweld "T" series sleeves is conducted on either production or sister splice samples. Production splice samples are those cut from in-place reinforcement while sister or companion splice samples consist of bars 3 ft. in length spliced in sequence with, and adjacent to, the production work in the identical position to the production splice. The sampling schedule and acceptance standards are presented in the project specification.

Appendix 3B.9.0 discusses the requirements of Cadweld splices during plant operations phase.

c. Cement:

Cement, conforming to ASTM C 150 specification for Portland Cement, is tested for the chemical and physical requirements of this standard at a frequency of every 5000 cubic yards of concrete produced. In addition, the on-site testing laboratory assigns a representative to the supply source whenever rail cars are loaded for delivery to the storage silos designated for this project. The representative oversees the loading and sampling of the cars by the supplier and obtains split samples of each car loaded. Independent testing is performed, while shipment is in transit, by the supplier and the on-site testing laboratory to satisfy the basic acceptance standard of 3,000 psi minimum on the 7-day mortar cube breaks. No cement is transferred to the project silos until the test results confirm this strength value. Complete testing for the chemical and physical requirements including the 3-, 7-, and 28-day compressive strength tests or mortar cubes is performed by the supplier for each rail car delivery and the certified test results are forwarded to the job site.

d. Fly Ash/Slag:

Testing of the Fly Ash is performed by the on-site testing laboratory in complete accordance with ASTM C 311, Methods of Sampling and Testing Fly Ash. Every 100 tons of Fly Ash used by the Central Concrete Mix Plant are sampled for the physical and chemical determinations except for silicon dioxide, magnesium oxide, and available alkalies in the Fly Ash which are required for every 1000 tons of Fly Ash used. All test results are compared with the chemical and physical requirements of Pozzolan Class F (Fly Ash) listed in ASTM C 618, Specifications for Fly Ash or Calcined Natural Pozzolans. Any Fly Ash delivered to the site and found through testing to be unsuitable was removed from the premises.

Per ACI 318-63, slag is to meet the requirements of ASTM C 205, Specification for Portland Blast Furnace Slag Cement. ASTM C 205 was withdrawn and replaced by ASTM C 595, Standard Specification for Blended Hydraulic Cements. ASTM C 595 covers slag mixed with cement. The slag component is to meet the requirements of ASTM C 989, Standard Specification for Slag Cement for Use in Concrete and Mortars, to comply with ASTM C 595.

e. Concrete Aggregates:

Fine and course aggregates, conforming to ASTM C33 Specifications for Concrete Aggregates, are sampled for testing during the progress of the work in accordance with the ACI Manual of Concrete Inspection (SP-2). The following user tests are performed on every 5,000 tons of aggregates delivered to the job site:

Method of Test	ASTM Designation
L.A. Abrasion	C 131-66
Potential Reactivity (Chemical)	C 289-66
Soundness	C 88-63

In addition, routine aggregate testing is performed for every 100 tons of each aggregate size delivered consisting of:

1. One sand sample for gradation from incoming material.

2. One coarse aggregate sample for gradation from incoming material of each nominal size group.

Incoming aggregates are unloaded but remain separated from the approved stockpiles in use until gradation tests indicate that the material is acceptable.

Whenever the above samples fail to meet the gradation requirements, two additional samples of the incoming material are taken. Failure of one of these samples to meet the gradation requirements is cause for rejection of the material represented.

Periodically, the stockpiles in use are randomly sampled and gradation tests performed to determine if the material in the stockpiles has become unduly segregated. Occasional sampling and testing is performed on the fine and coarse aggregate from the supply source stockpiles.

One test of sand for organic impurities is performed weekly.

f. Concrete Cylinders:

All samples of the concrete for casting cylinders are taken from the discharge of the stationary mixer at the central concrete mix plant except for pumped concrete which requires samples taken at the truck discharge. Occasional sampling is required at the point of discharge (truck or pump-line) to ascertain slump, air, temperature and strength comparisons with figures obtained from the routine sampling points, i.e., mix plant for conventionally-placed concrete and truck discharge for pump-placed concrete.

Samples are made, cured and tested in accordance with ASTM C31, Method of Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field, and ASTM C 39, Method of Test for Compressive Strength of Molded Concrete Cylinders. Cylinders improperly made and cured are discarded without testing.

For structural concrete, six (6) cylinders are made for each 100 yards of concrete placed, and two tested at 7, two at 28 and two at go days. When a correlation of test data is established for each mix, the 90-day test cylinders may be discontinued.

Acceptance standards for the concrete is in accordance with the ACI Standard 318-63, Building Code Requirements for Reinforced Concrete, Paragraph 504.

If the compressive strength tests fail to meet the above requirements, it may be required to conduct load tests or obtain cores by drilling and testing for those portions of the work where the questionable concrete has been placed. If those tests should indicate that the concrete does not conform to the Code, the poor concrete must be removed and replaced or other corrective measures taken.

g. Compacted Structural Backfill:

A continuous program of backfill testing is performed by the on-site testing laboratory during backfill placement. In all work areas, where manual and/or machine compaction is performed, at least one test is made each day in each work area. (Class I areas are so indicated on the design drawings.)

Structural backfill must meet the specified water content of 3 to 12 percent by dry weight of the material, must be placed in loose lift thicknesses of 6 inches or 12 inches depending on the granular material with approximately the following sieve analysis requirements:

Sieve Size	Total Percent Passing
2 1/2 inch	100
1 inch	60-100
No. 100	0-30
No. 200	0-5

The compaction values must satisfy the requirement of Alternate “A” or Alternate “B” as described below:

1. Alternate “A” Density Test

Each lift of granular backfill material is compacted to at least 98 percent Standard Proctor Density as determined in ASTM Specification D698, Method D, latest revision, or 80 percent minimum relative density as determined in ASTM Specification D2049, latest revision, whichever test method is applicable for the material.

2. Alternate “B” Plate Load Test

A plate load test, defined in the project specification, is performed on compacted granular backfill in random areas to ascertain a maximum plate deflection of 0.06 inch under a 10,000 LB plate load.

All density, moisture and/or plate load tests are made at a depth of the entire lift thickness. All compliance testing is performed at the job site by a qualified representative familiar with standard methods of compaction testing and moisture determination. The testing technician performs moisture density relationships to determine any variation in density and in moisture content of the backfill.

If any areas of compaction appear to be doubtful, the next lift is not placed until testing is completed. If the density or plate load deflection does not meet requirements of the Project Specification, or if the moisture is not within the required range, or if both occur, the test is considered a failure. When a failure occurs, either of two courses of action is followed.

1. Make two retests in the near vicinity of the failing test. If both retests pass the requirements, the area is accepted. If one or both retests fail, the area is considered a failure and Item 2 is followed.

2. The area is reworked until a passing test is achieved.

   If either the density or moisture requirements or both cannot be met, or the plate load test deflection is higher than permitted the material is removed and discarded.

h. Earthwork for Canal Dikes and Barge Slip:

In addition to continuous inspection performed by on-site testing laboratory during construction of the Class I canal dikes, water content test samples are obtained. Random samples are taken from in-situ fill and borrow sources to determine water content for each 8000 square feet area.

Soil to be used as Class I canal dike fill mist be placed and compacted within the water content range which is defined as the ratio of insitu water content to optimum water content minus 5 percentage points. The insitu water content is determined in accordance with ASTM Specification D2216-66 and the optimum water content is determined in accordance with ASTM Specification D 698-66T, Method A on non-air dried samples.

To ensure that the pre-construction minimum shear strength of the soil in dikes is obtained, a minimum of four field vane shear strength and sensitivity tests of the insitu soil in dikes is made in the Class I barge slip road fill.

i. Reinforced Concrete Pressure Pipe Testing:

Field pressure tests are performed on each joint in accordance with AWWA M11, Section 16.3.3 after completing the installation of a section of the line, and before the joints are covered. As the joint tightness is assured, the backfilling commences.

The completed pipeline is Subjected to a 24-hour soaking period prior to commencing hydrostatic testing to record leakage under the following conditions: (a) 70 psig for one hour, and (b) 30 psig for 10 hours. The leakage rate for the above pressures must not exceed 25 gal/in. of diameter per mile of pipe per 24 hours.

j. Welded Steel Circulating Water Pipe Testing

After completing the installation of a pipeline, or a section of the line, and before the joints are covered, an air tightness test of the joints is made in accordance with AWWA M11, Subsection 16.3.3.

Field pressure tests are performed on all piping to 70 psig for one hour and 30 psig for 10 hours to satisfy a no leakage requirement for the welded and bolted-flanged joints. Any piping, fittings or welds failing the test are repaired and retested.

3.8.1.7.2 Qualification of Welders

All welding in Class I structures is performed strictly by welders or welding operators who have been previously qualified by tests as prescribed in AWS D2.0 or ASME Section IX of the Boiler and Pressure Vessel Code. The qualification test records for each qualified welder are maintained in a file at the job site.

3.8.1.7.3 Nondestructive Testing Requirement

To determine the presence of surface and internal discontinuities in structural steel and welds, the following nondestructive examinations are used:

a. Magnetic Particle or Liquid Penetrant:

Conducted in accordance with Appendix VI or Appendix VIII, respectively, of ASME Section VIII of the Boiler and Pressure Vessel Code or AWS D2.0-69, Paragraphs 607(e) and 607(f), respectively, for designated fillet welds.

b. Ultrasonic:

Conducted in accordance with Appendix U of ASME Section VIII of the Boiler and Pressure Vessel Code or AWS D2.0-69, Appendix C, for designated fillet and penetration welds. Conducted in accordance with ASTM Specification A435 or ASTM A578 for structural plates.

c. Radiography:

Conducted in accordance with Paragraph UW-51 of ASME Section VIII of the Boiler and Pressure Vessel Code or AWS D2.0-69, Appendix B for designated butt welds.

d. Visual

Conducted in accordance with the Visual Weld Acceptance Criteria (VWAC) for Structural Welding at Nuclear Power Plants, Revision 2 for structural weldments that are under the perview of AWS D1.1 or other non-ASME class structures. VWAC will not be used for inservice inspections required by Section XI of the ASME Code.

e. Vacuum Box Leak Testing:

Conducted in accordance with approved contractor’s procedures for accessible seam welds in the refueling canal and spent fuel pool.

f. Pipe Joint Leak Testing:

Conducted in accordance with AWWA M11, Section 16.3.3 for all pipe joints in the circulating water system. This air pressure test is performed on the double-gasket bell and spigot joints for concrete pipe as well as the standard welded joints for steel pipe.

3.8.2 Containment Structures

The containment for the station consists of three basic structures: a steel containment vessel, a reinforced concrete Shield Building, and the internal structures. Figure 3.8-3 shows a typical section through the containment structures. The Containment Vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom which houses the reactor vessel, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, reactor coolant pumps, steam generators, core flooding tanks, letdown coolers, and normal ventilating system. It is completely enclosed by a reinforced concrete Shield Building having a cylindrical shape with a shallow dome roof. An annular space is provided between the wall of the Containment Vessel and the Shield Building, and clearance is also provided between the Containment Vessel and the dome of the Shield Building. The Containment Vessel and Shield Building are supported on a concrete foundation founded on a firm rock structure. With the exception of the concrete under the Containment Vessel there are no structural ties between the Containment Vessel and the Shield Building above the foundation slab. Above this there is unlimited freedom of differential movement between the Containment Vessel and the Shield Building. The containment internal structures are constructed of reinforced concrete and structural steel. These structures are isolated from the containment vessel by steel grating panels with sliding supports which allows free differential movement between the internal structures and the vessel. The internal structures are supported by the massive concrete fill within the Containment Vessel bottom head.

3.8.2.1 Containment Vessel

3.8.2.1.1 Description

The non-field stress relieved containment vessel is constructed in a two-stage operation and in a manner that conforms to the ASME Boiler and Pressure Vessel Code, Article 14, N-1411. The vessel inside diameter is 130 feet and the net free volume is approximately 2,834,000 ft^3. The cylindrical shell and bottom head thickness, exclusive of reinforced areas, is 1 1/2" with a dome thickness of 13/16". The 180-ton polar crane is supported from the cylindrical vessel shell by a 14' -6 1/2" deep by 5'-11" wide circular crane girder. Access to the containment is provided by an equipment hatch (Figure 3.8-4), a personnel air lock (Figure 3.8-5) and an emergency air lock (Figure 3.8-6). Electrical and Mechanical Penetrations are provided for services to the containment. Figures 3.8-7 through 3.8-12 show typical penetration details.

3.8.2.1.2 Design Bases

Postulated Accident Conditions:

The containment system is designed to provide protection for the public from the consequences of any break in the reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe assuming unobstructed discharge from both ends. Pressure and temperature behavior subsequent to the accident is determined by calculations evaluating the combined influence of the energy sources, heat sinks and engineered safety features. These are discussed in detail in Chapter 15.

The containment system also provides protection for the public from the radiological consequences of a hypothetical accident discussed in Chapter 15. The containment design, along with the engineered safety features provided, ensure that the exposure of the public resulting from a hypothetical accident is below the guidelines established by 10CFR100.

Energy and Material Release:

The sources available for the release of energy and materials into the containment systems are:

a. stored heat from the reactor core and internal structures;

b. fission coastdown and decay heat from the reactor core;

c. stored heat in the materials of the reactor coolant system;

d. reactor coolant and its contained corrosion and fission products;

e. fission products from the fuel elements in the core.

The amount of energy contributed by the major sources during various types of accidents is discussed in Chapter 15.

Contribution of Engineered Safety Features:

The design, application, and evaluation of the engineered safety features are discussed in Chapter 6. Their effectiveness is treated in Chapter 15. Their relationship to the containment design is summarized in this section.

Engineered safety features systems are provided to minimize the consequences of postulated accidents by removing heat from the fuel, inserting negative reactivity into the reactor, decreasing the pressure in the Containment Vessel by removing thermal energy, and removing radioactive material that may leak into the Shield Building from the Containment Vessel. The principal engineered safety features are:

a. Emergency Core Cooling

b. Containment Cooling

c. Containment Isolation

d. Emergency Ventilation

e. Safe Shutdown

f. Containment Vessel Combustible Gas Control

Design Leakage Rates:

The Containment Vessel was tested at the conclusion of construction and after all penetrations have been installed to verify that the design leakage rate associated with an internal pressure of 38 psig does not exceed 0.5 percent of the containment contained weight of air and vapor in 24 hours. The analysis in Chapter 15 shows that this is more than adequate to meet the guidelines of 10CFR100.

3.8.2.1.3 Codes

The Containment Vessel is a Class B vessel as defined in the ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, Paragraph N-132, 1968 through Summer Addenda 1969.

The design, fabrication, quality control, inspection, and testing of the containment vessel complies with the requirements of the ASME Boiler and Pressure Vessel Code, Section II Materials, Section III Nuclear Vessels, Subsection B “Requirements for Class B Vessels,” Section VIII Unfired Pressure Vessels, and Section IX Welding Qualifications.

The containment vessel’s external pressure has been re-evaluated in accordance with the ASME Boiler and Pressure Vessel Code Section III, Subsection NE, 1986.

The Containment Vessel is code stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-1610.

The design internal pressure for the containment vessel has been specified in accordance with the provisions of Section III of the ASME Boiler and Pressure Vessel Code. The design requirements for Class B Vessels are contained in Article 13 of the ASME Code.

The containment vessel has been pressure tested in accordance with the rules of ASME Boiler and Pressure Vessel Code, Section VIII, UG-100 and Section III, N-1314(d). The test pressure is 1.25 times the design internal pressure of 36 psig.

The design of the Containment Vessel is based on permissible stresses as set forth in the applicable codes.

Piping is designed, fabricated, erected, and tested in accordance with ANSI Piping Code B31.1 or ASME III Class B.

Penetrations conform to the applicable requirements of Section III of the ASME Boiler and Pressure Vessel Code.

Crane runway girders and brackets, temporary columns, ladders and accessories are designed, fabricated, and erected in accordance with the requirements of the Specification for Design, Fabrication and Erection of Structural Steel for Buildings, of the American Institute of Steel Construction, except that attachments to the vessel are in accordance with Section III (Subsection B) of the ASME Code.

The Containment Vessel design and construction also meets applicable requirements of state and local building codes.

3.8.2.1.4 Loading

Load combinations used in the design of the Containment Vessel provide for all loading conditions during construction, operation and testing. The design of the vessel provides for movement of the vessel and supports due to expansion and contraction. Local stresses due to the restraint at the point of embedment were analyzed and found to be acceptable. The Containment Vessel in its final supported condition is capable of transmitting all superimposed loads from the internal structures through the bottom head into the foundation below.

Design Loads:

a. Dead Loads:

Dead Loads consist of the dead weight of the Containment Vessel and its appurtenances, the weight of internal concrete, and the weight of structural steel and miscellaneous building items within the Containment Vessel.

Densities used for dead load calculations are as follows:

1. Concrete: 143 lb/cu ft
2. Steel reinforcing: 489 lb/cu ft using nominal cross section areas of reinforcing bar sizes
3. Steel containment vessel: 489 lb/cu ft
4. Structural Steel: 489 lb/cu ft

b. Loss of Coolant Accident Loads:

This load is determined by analysis of the transient pressure and temperature effects that could occur following a break of a reactor coolant pipe. Breaks up to and including a double-ended break of the largest reactor coolant pipe are considered. The analysis is presented in Section 6.2.

c. Operating Loads:

Operating Loads include the following:

1. Gravity loads of all equipment and piping, including contained fluid
2. Weight of water in the refueling pool and fuel transfer canal
3. Loads resulting from the restraint of that part of the vessel which is embedded in concrete
4. Crane loads

Equipment loads are those specified on the drawings supplied by the equipment manufacturers.

Floor live loadings are assigned for the design of internal floors consistent with their intended use.

d. External Pressure Load:

The original design documented that the Containment Vessel is capable of withstanding an external pressure differential of 0.50 psi in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, UG-28. The external pressure loading on the Containment Vessel has been re-evaluated using ASME Boiler and Pressure Vessel Code, Section III, subsection NE. This re-evaluation has determined that the allowable Containment Vessel external design pressure loading is 0.67 psi. The Containment Vessel is vented as required to eliminate pressure fluctuations caused by air temperature changes during various operating modes. This is accomplished through ventilation purge connections which are normally closed while the reactor is in operation.

Automatic vacuum relief devices are also used to prevent the Containment Vessel from exceeding the external design pressure in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article 16. Multiple vacuum breakers are used to relieve pressure from the shield building into the containment in case the Containment Vessel is subjected to excess external pressure. These valves ensure that external pressure differential on the Containment Vessel does not exceed 0.50 psi.

See Table 3.6-6 for supplemental information on the Containment Vessel Structural capacity (increased capacity based on Letter, Toledo Edison to NRC, Serial No. 1992, “Containment Vessel Structural Capacity External Pressure”, 11/26/1991).

e. Temperature and Pressure:

The Containment Vessel is designed for the following temperature and pressure conditions:

1. Maximum Internal Pressure - 40 psig
2. Design Internal Pressure - 36 psig 264°F max.
3. Leakage Rate Test Pressure - 38 psig

4. Maximum Negative Pressure - 0.5 psig*

5. Lowest Service Metal Temperature - 30°F

6. Maximum Operating Ambient Temperature - 120°F

7. Maximum Operating Internal Temperature - 120°F

8. Pneumatic Test Pressure is 1.25 Times the Design Pressure - 45 psig

*The design allowable negative pressure is re-evaluated using equations given in ASME B&PV Code, Section III, Subsection NE, 1986 as 0.67 psig.

The containment vessel and penetrations are designed to withstand tornado depressurization due to the Auxiliary Building/Annulus. The amount of depressurization is determined analytically as described in Section 3.3.2.2.1.

f. Impingement Force and Missiles:

1. Jet Forces

In addition to the pressure specified above, the Containment Vessel is designed to withstand the following jet forces acting from any direction on the part of the vessel specified without causing a rupture of the vessel. Local membrane stresses do not exceed 1.5 of yield strength for the impingement forces specified combined with other loads in accordance with N-417.10 (a) and (e) of ASME Code, Section III, Summer 1969 Addenda.

Location	Jet Force	Interior Area Subject to Jet Force
Bottom Head	None	
Cylinder		
(1) Primary Coolant	2,220,000 lbs.	870 ft² (50° to normal)
(2) All piping systems other than primary coolant system. (Refer to Section 3.6)		
Dome		
(1) Primary Coolant	2,220,000 lbs.	4250 ft² (5.4° to normal)
(2) All piping systems other than primary coolant system.		

2. Missiles

The Containment Vessel is also designed to withstand missiles representative of those that could originate within. The missile considered for design purposes has the following properties: Material - steel, weight - 9.0 lbs., missile impact area - 1.0 sq. in., velocity (max.) - 135 ft/sec., and a kinetic energy of 2560ft. lbs.

g. Wind Loads:

The Containment Vessel and its temporary support columns are designed for a wind load of 90 mph during construction and prior to the time that the Shield Building concrete roof dome is completed.

h. Seismic Loads:

1. Seismic Design:

The base for the seismic design are the horizontal ground accelerations of 0.08g (Maximum Probable Earthquake) and 0.15g (Maximum Possible Earthquake). Vertical ground motions are two thirds of the horizontal ground motion.

A multi-degree of freedom dynamic analysis is performed on the Containment Vessel in both horizontal and vertical directions. The analysis envelops the maximum displacement, acceleration, shear, and moments at all significant levels on the vessel.

The mathematical model accurately represents points of critical interest on the vessel and its appurtenances. Since the vessel is founded on competent rock, the model assumes a fixed base. A vertical seismic ground acceleration equal to 2/3 of the horizontal ground acceleration is assumed to be acting simultaneously with the lateral forces. Vertical stresses were added directly and linearly to other concurrent stresses.

2. Maximum Possible Earthquake:

The Containment Vessel is designed for the Maximum Possible earthquake response based on horizontal ground acceleration of 0.15g acting concurrently with a vertical ground acceleration of 0.10g. When this seismic load is combined with other loads, the stresses do not exceed the ASME Code allowable stresses for emergency condition. In accordance with the ASME Code, the Maximum Possible Earthquake is considered a faulted condition. Since the stress limits under a faulted condition require a complicated stress analysis, the stress limits for emergency conditions (which are more conservative) are used for convenience and simplicity in designing the vessel.

3. Personnel Locks, Escape Lock, and Equipment Hatch:

The personnel lock, escape lock, and equipment hatch are designed for seismic conditions. The value of the horizontal acceleration acting at each lock or door is a minimum of 0.25g horizontal and 0.10g vertical acceleration. This seismic load is combined with dead load, temperature and pressure effects without exceeding the stresses allowed by the ASME Boiler and Pressure Vessel Code. Appropriate floor response spectra were used to find the acceleration corresponding to the primary frequencies of the locks and equipment hatch. These proved to be the governing acceleration in all cases.

In addition, the locks and hatch were investigated for the effect of the Maximum Possible Earthquake. Under this condition, stresses do not exceed those allowed by the ASME Boiler and Pressure Vessel Code for emergency condition.

i. Gravity Loads:

The gravity loads of the nuclear steam supply system and supports, the internal structure, equipment and operating loads bear on the lower vessel head are transmitted through the lower head to the concrete foundation. Other gravity loads supported by the Containment Vessel are transmitted from the cylinder walls into the bottom vessel head and thence to the concrete foundation.

The gravity loads include the following:

1. The weight of the steel shell and its appurtenances.

2. The weight of major equipment and foundations for the Nuclear Steam Supply System including Reactor Vessel, Steam Generators, Reactor Coolant pumps, Quench Tank, Letdown Coolers, Core Flooding Tank, Pressurizer, and other associated equipment: (12,000,000 lbs.)

3. The weight of operating platforms and their structural supports.

4. The loads on miscellaneous attachments to the Containment Vessel as shown on the drawings including the polar crane.

j. Live Loads:

1. Polar bridge crane operating loads.

2. Penetration loads.

   - Floor load of 200psf or two 2000 lb. axle loads with a tread distance of 2ft. 6in. and a wheel base of 4ft. 6in. on the passage area of the personnel lock. The emergency personnel lock is designed for a floor load of 200psf only.

   - Water load during refueling or from the spray system: (3,010,000 lbs.)

   - Floor load reaction at equipment hatch: (500psf or two 25 kip wheel loads, whichever is greater)

3. Snow loads during construction: (40psf.)

4. Construction loads of 200psf from forming and shoring the Shield Building roof dome. The bottom head of the Containment Vessel was supported on concrete during this load application.

5. Temporary construction load from rigging during erection of Steam Generators and Reactor Vessel.

Load Combinations:

The Containment Vessel is designed for the loading combinations listed in the following sections:

a. Pressure Test:

1. Dead load of Containment Vessel and appurtenances.
2. Pneumatic Test Pressure (1.25 x design pressure) at ambient temperature (notless than 60°F).
3. The weight of contained air.
4. Maximum Probable Earthquake.

b. Leak Test:

During this test, the Containment Vessel is supported on concrete fill.

1. Dead load of Containment Vessel and appurtenances.
2. Leak test pressure.
3. Maximum Probable Earthquake.

c. Construction:

1. Dead load of Containment Vessel and appurtenances.
2. Construction load of temporary shoring for the concrete Shield Building dome supported on pads of the Containment Vessel dome. The vessel dome is designed for a load of 200psf.
3. Weight of 180 ton capacity polar bridge crane in place with lifted loads.
4. Temporary reaction forces from rigging during erection of steam generators.
5. Gravity load of equipment and foundations on Containment Vessel bottom head.
6. Maximum Probable Earthquake.

d. Normal Operating:

1. Dead load of Containment Vessel.
2. Gravity load of equipment, foundations, platforms, and water for refueling.
3. Polar bridge crane load with lifted loads as shown on the drawings.
4. Operating pressure and temperature, internal and external.
5. Horizontal and vertical Maximum Probable Earthquake or maximum possible earthquake effects.
6. Dead loads on welding pads.
7. Live loads on personnel lock and equipment hatch.
8. Penetration loads (normal operating pipe thrusts, moments, and shears).

e. Loss of Coolant Accident (LOCA) Condition:

1. Dead load of Containment Vessel arid appurtenances.
2. Gravity load of equipment, foundations, platforms and water for accident spray system.
3. Polar bridge crane weight.
4. Accident pressure and accident temperature.
5. Horizontal and vertical maximum probable or maximum possible earthquake.
6. Dead load on welding pads.
7. Penetration loads as shown on the drawings (accident loads).
8. Internal missile loads.

f. Pipe Rupture Other Than LOCA:

1. Same as LOCA condition.
2. Same as LOCA condition.
3. Same as LOCA condition.
4. Operating pressure and temperature.
5. Same as LOCA condition.
6. Same as LOCA condition.

7. Jet impingement force due to a double ended rupture or a side- split rupture of any piping system other than primary coolant system.

g. Internal Missile Load:

A missile having the following characteristics has been specified for design purpose:

Material: Steel

Weight: 9.0 lbs.

Missile Impact Area: 1 sq. in.

Maximum Velocity: 135 ft/sec.

Kinetic Energy: 2560 ft.-lb

3.8.2.1.5 Containment Vessel Analytical Techniques

Containment Vessel

For axisymmetric loadings, the membrane theory of thin shells is used to analyze the Containment Vessel except where discontinuities exist.

A cantilevered beam of lumped masses is modeled to determine seismic forces. However, a finite-element computer program, “YALE,” has also been used to find the circumferential stresses.

Horizontal displacements at various elevations are also obtained from the same output, based on this data. A plot showing the distortion of the cross-section (ovaling) is made. It is evident from this plot that the distortion of the cross-section is insignificantly small.

For thermal stress and discontinuity stress due to all other loads, “YALE” computer program is used again to analyze the Containment Vessel.

The “YALE” computer program is again used to analyze the Containment Vessel for thermal stress and discontinuities stress due to all other loads.

A finite element computer program, Bechtel’s CE 779, is used to analyze the vessel shell for jet impingement force.

The Containment Vessel also meets the requirements for Paragraph N-451.1, Section III of the ASME code. Therefore, the inclusion of cyclic or fatigue analyses in the design of Containment Vessel can not be justified.

The reactor vessel closure head and once through steam generator replacement projects of 2011 and 2014 created temporary access openings in the Containment Vessel wall. The dead load redistribution effects due to the creation and restoration of the original construction opening, as well as the 2011 temporary access opening, are evaluated by performing a construction sequence analysis using the finite-element program, GT-STRUDL. No dead load

redistribution effects occur due to creation and restoration of the 2014 opening since it is within the boundaries of the 2011 opening.

Restoration of the Containment Vessel for these projects includes welded attachments to the vessel wall. The finite element program, ANSYS, is used for the ASME qualification analysis of the restored Containment Vessel in the vicinity of the temporary opening. The ANSYS analysis is performed for the applicable loading conditions, described in Section 3.8.2.1.4, to verify the actual stresses meet the allowable design stresses described in Section 3.8.2.1.6.

Equipment Hatch, Personnel Lock, and Emergency Lock:

a. Seismic Analysis:

For analytical purposes each lock has been assumed to be a beam cantilevered from the main body of the Containment Vessel. It is further assumed to vibrate in three independent directions as described in the following:

1. Case I:

   Lock vibration in the meridional plane of the vessel due to vertical earthquake. This condition imposes a moment on the shell in the meridional plane of the vessel.

Equipment Hatch, Personnel Lock and Emergency Lock:

a. Seismic Analysis:

For analytical purposes each lock has been assumed to be a beam cantilevered from the main body of the Containment Vessel. It is further assumed to vibrate in three independent directions as described in the following:

1. Case I:

   Lock vibration in the meridional plane of the vessel due to vertical earthquake. This condition imposes a moment on the shell in the meridional plane of the vessel.

2. Case II:

   Lock vibration in the circumferential plane of the vessel due to the combined effects of vessel translation and angular oscillation resulting from an applied horizontal earthquake acting perpendicular to the plane of the lock. This condition imposes a moment on the shell in the circumferential plane of the vessel.

3. Case III:

   Lock vibration radial to the vessel due to an applied horizontal earthquake acting in the plane of the lock. This condition imposes a radial load on the shell.

Stresses are determined at the lock to insert and insert to shell junctions for the above three cases. Since Cases I and II, and I and III can occur simultaneously, stresses are added vectorially for these combinations and other applicable shell stresses, acting concurrently.

In order to determine the seismic accelerations acting on the lock and consequently the induced loads in the shell, lock response data is developed from specified ground time history.

This ground time acceleration data is input to program 1044 of CB&I for each of the three cases which in turn yields vessel acceleration at the lock elevation as a function of time for each case.

Again, using CB&I program 1044 these lock time histories are applied to a series of single-degree-of-freedom systems having a range of natural periods to yield response spectrum curves of acceleration vs. natural period, one for each of the three cases.

Since the lock is a one-degree-of-freedom system, after its natural periods of vibration are calculated for the longitudinal, circumferential, and radial directions of the containment vessel the lock accelerations are determined from the above described response spectrum curves. Equivalent dynamic loads of vibration are obtained by multiplying the accelerations by the mass of the lock. By applying these accelerations at the center of gravity of the lock, forces, moments, and shears transmitted to the shell are determined.

The Containment Vessel is reinforced locally at the lock penetration in accordance with the area replacement requirement in Paragraph N-454 of ASME, Section III. By inputing external forces and moments due to dynamic loads in the CB&I program No.1027, stresses due to this local effect, as well as combined loads, are obtained at the lock, insert, and shell itself. The CB&I program used in this analysis is based on Welding Research Council Bulletin No. 107.

b. Structural Analysis:

The Equipment Hatch, Personnel Lock, and Escape Lock are in accordance with the ASME Code, Section VIII, paragraph UG 27, UG 29, UG 33, and UA6 which are in turn based on the membrane theory of thin shell. Where geometrical discontinuities exist, a finite element computer program, “YALE,” is used to find stresses at discontinuities. The floor system within each lock is designed by the conventional beam-slab method.

Pipe Penetrations:

All penetrations (including inserts) are checked for area replacement in accordance with Paragraph N-1330 of ASME Section III by CB&I computer program 772 and by hand calculations.

Operating loads are applied on a penetration of interest and on adjacent penetrations which are on cardinal lines of the penetration within a distance of 2 $(RT)^{½}$; (68.5”).

Penetrations are then analyzed for stresses in the vessel shell by the CB&I program No. 1036M using the method outlined in Welding Research Council Bulletin No.107. They are added to the initial stresses produced by internal pressure.

Pressure produces a complex state of stress in the shell and penetration at their intersection. As a rational means of estimating these stresses Paragraph N-451 (B) of Section III has been used as a guide. This paragraph assumes that in the vicinity of a penetration reinforced in accordance with ASME rules, maximum membrane pressure stress does not exceed 1.0 Sm and the maximum surface stress does not exceed 1.5 Sm.

3.8.2.1.6 Containment Vessel Design Stresses

Basic Design Stresses

Allowable design stresses are based on Figure N-414 of ASME Code, Section III, Summer 1969 Addenda with the Sm, Sy, and Su having the following values:

Physical Properties for Materials
To Be Used for Pressure Parts or Attachment to Pressure Parts

Material Specification	Su Minimum Ultimate Tensile (ksi)	Sy Minimum Yield at Ambient (ksi)	Sy Minimum Yield at 264°F (ksi)	Sm ASME Code Allowable Stress Intensity at 264°F (ksi)
Plate				
SA299 (1” & Under)	75	42	37.56	18.75
SA299 (Over 1”)	75	40	35.82	18.75
SA516 GR 70	70	38	34	17.5
SA516 GR 60	60	32	28.65	15
Pipe				
SA333 GR 6	60	35	31.32	15
SA333 GR 1	55	30	26.87	13.75
SA106 GR B	60	35	31.32	15
SA312 Type 304	75	30	23.41	14.2
Forgings				
SA350 LF-2	70	36	-	17.5
SA182 F304	70	30	20	14
Bolting				
SA193 B7	125	105	-	25
SA194 GR 7	-	-	-	-

Physical Properties for Materials to be Used for Non Pressure Parts

Material Specification	Minimum Ultimate Tensile (ksi)	Minimum Yield at Ambient (ksi)	Minimum Yield at 264°F(ksi)
A36	58	36	-
SA516 GR 70	70	38	33.8
A53 GR B	60	35	-
A106 GR B	60	35	31.2

Section III of the ASME Code, Paragraph N-412 applies to localized stress regions. Paragraph N-412 states that the distance over which total membrane stress intensity exceeds 1.1 Sm may not extend more than 0.5 $(RT)^{1/2}$ and may not be closer than 2.5 $(RT)^{1/2}$ to another region

where the total membrane stress intensity exceeds 1.1Sm. R is the mean radius of the vessel and t is the vessel wall thickness.

Allowable Buckling Stresses:

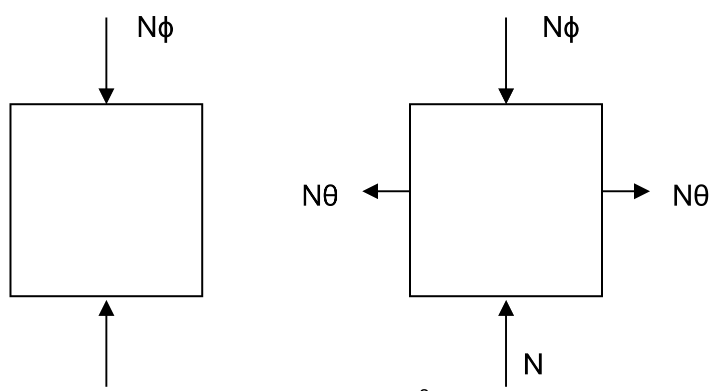
a. Hemispherical Head:

Compressive stress resultants in the top head are compared to the allowables obtained from the paragraphs entitled “Biaxial Compression-Equal Unit Forces” and “Biaxial Compression-unequal Unit Forces” of the Welding Research Council Bulletin No. 69, Biaxial Stress Criteria for Large Low Pressure Tanks. Using these allowables for the spherical dome is based on the assumption that the dome acts as a cylinder with the radius equal to the radius of the dome.

Three cases are considered.

1. For a uniaxial compressive stress resultant and for biaxial unequal tensile and compressive stress resultants.

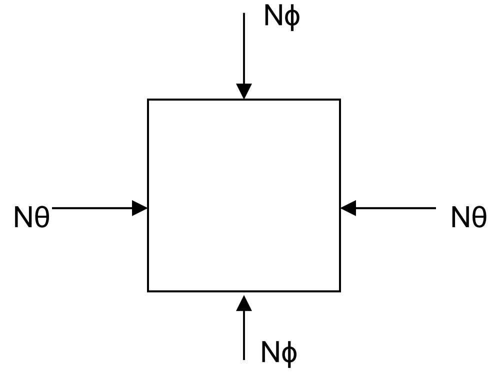
i.e.	$N\phi$ (-) Compressive $N\theta$ -0	or	$N\phi$ (-) Compressive $N\theta$ (+) Tensile



$$N\phi \quad \text{Allow} = 1.8 \times 10^{6} \frac{t^2}{R}$$

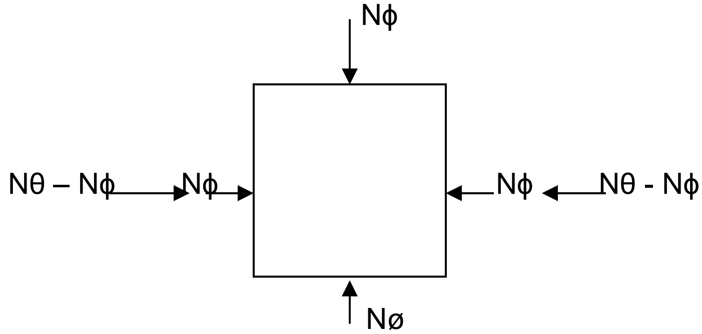
2. For biaxial equal compressive stress resultants.

   i.e. $N\phi$ (-) compressive = $N\theta$ (-) compressive



$$N\varnothing \text{ Allow} = 0.9 \times 10^6 \frac{t^2}{R}$$

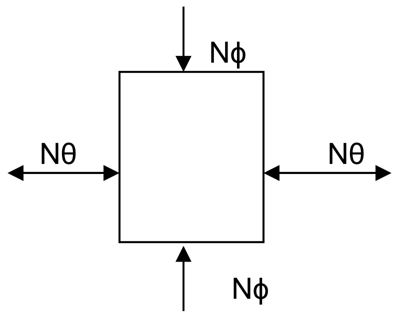
3. For biaxial unequal stress resultants. This case is treated as the summation of the uniaxial condition with equal stress.



$$\frac{N\theta - N\phi}{1.8 \mathrm{x} 10^6 \frac{t^2}{R}} - \frac{N\phi}{0.9 \mathrm{x} 10^6 \frac{t^2}{R}} < 1$$

Meridional or Axial Stress:

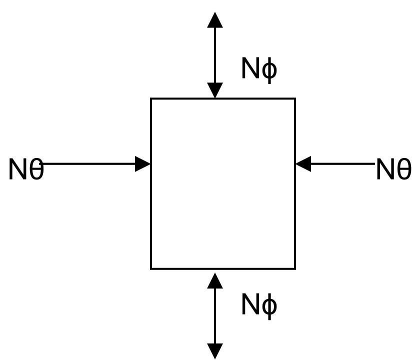
The maximum allowable compressive stress used in the design of cylindrical shells subjected to loadings that produce longitudinal compressive stress is in accordance with Section VII, Paragraph UG-23 (b).



Nθ can be tensile or compressive

Circumferential Stress:

Generally speaking, circumferential compression results from external pressure loading. The criteria of Section VIII, Paragraph UG-28 are used to analyze circumferential buckling. These rules provide a safety factor of 4.0 against shell buckling.



Nϕ can be tensile or compressive

Allowable Weld Stresses:

a. ASME Allowable Weld Stresses:

Full Fusion:

Weld allowables are in accordance with Subsection B of the ASME Code, Section III. Same as parent metal.

Partial Depth Groove Welds:

Allowable stress on the effective depth is:

An inspection factor x load factor x Sm of weaker material

The inspection factor used is 0.8

The load factors used are:

1.0 for load perpendicular to axis of weld

0.875 for any combination of perpendicular and parallel loads

0.75 for a load parallel to the axis of weld

For simplicity an allowable of 0.8 x 0.75 x Sm = 0.6Sm is used for all partial groove welds except where a higher allowable is required and is permissible as indicated above.

Fillet Welds:

Per UW-18 (d), Section VIII:

Allowable stress = 0.55Sm (of weaker material) on min leg which is equivalent to 0.78Sm on throat

b. AISC Allowable Weld Stresses:

Fillet Weld:

Allowable Stress = 15,800 psi on throat (AISC 1.5.3.1)

Groove Welds:

Per 1.5.3.2 AISC

Calculated Stresses:

Based upon analytical techniques as described in Subsection 3.8.2.1.5, Actual Stresses due to all load combinations as described in Subsection 3.8.2.1.4 are found to be within the allowable stress limits except at the personnel lock where the insert is found to be overstressed by 1%. This overstress is considered to be within the accuracy of the analytical method and is therefore acceptable.

3.8.2.1.7 Materials

The Containment Vessel shell is fabricated from steel plate conforming to ASME Specification SA-299. This material also conforms to ASME Specification SA-300 requirements except that impact requirements are as specified in ASME Code, Section III, N-1211. The minimum service temperature of plate materials is 30°F.

The equipment hatch, personnel lock, and emergency lock are fabricated from either SA-299 plate material or SA-516 grade 70 plate materials. Impact and service temperature properties are similar to the previous paragraph.

Penetrations, which are integral parts of the containment vessel, are made from materials conforming to either ASME Specification SA-333, Grade 6 or Specification SA-312, Type 304.

Seamless pipe conforms to ASME Specification SA-333. Welded pipe meets the same requirements as the vessel shell except that the plate is SA-516, Grade 70.

Bolting Materials conform to one of the following materials:

a. Austenitic Stainless Steel Bolts are manufactured to ASME Specification SA-193-B8.

b. Ferritic Steel Bolts are manufactured to ASME Specification SA-320, Grade L43 or SA 193-B7.

Forgings conform to Specification SA LF1 or LF2.

Gaskets are inflatable or solid type rings. Gasketing material is silicone rubber or EPDM compound (approved by Engineering) having a minimum service life of 24 months.

3.8.2.1.8 Containment Vessel Preoperational Testing and Quality Control

General Requirements:

Test, code, and cleanliness requirements accompany each specification or purchase order for materials and equipment. Tests to be performed by the supplying manufacturers are enumerated in the specifications together with the requirements for test witnessing by inspectors. Fabrication and cleanliness standards, including final cleaning and sealing, are also described together with shipping procedures. Standards and tests are specified in accordance with applicable regulations, recognized technical society codes, and current industrial practices.

The containment vessel manufacturer is required to submit design calculations, drawings, and weld procedures to the applicant for review by his engineer before the performance of any work. This review and review of work during construction ensure compliance with applicable codes and specifications.

All welders and welding procedures are qualified in strict accordance with, and meet the requirements of, the ASME Boiler and Pressure Vessel Code, Section IX. Prior to the start of welding operations, the vessel manufacturer provides the applicant and his engineer with copies of the qualified welding procedure specifications and reports of the results of the qualification tests for each welder or welding operator.

All longitudinal and circumferential welds in the shell of the Containment Vessel are double-welded full penetration butt joints. All butt joints in any accessories subject to the ASME Boiler and Pressure Vessel Code are full penetration welds. All welds subject to the ASME Boiler and Pressure Vessel Code are 100 percent radiographed or otherwise examined in accordance with the ASME Boiler and Pressure Vessel Code. Welds which cannot be radiographed, or where the interpretation of radiographs would be open to doubt, are examined by the magnetic particle, liquid penetrant, or ultrasonic method.

In manual arc-welding, the electrodes are of the low hydrogen type. All welding filler metal has mechanical properties which are similar to the base metal. All automatic welding is done by the submerged-arc process.

Preheat in accordance with the ASME Boiler and Pressure Vessel Code is applied to all seams whose thickness exceeds 1-1/4 inch regardless of the surrounding air temperature. Preheat at 100°F is applied to thinner seams if the surrounding air temperature falls below 40°F and/or the surfaces to be welded are damp.

Charpy V-Notch impact tests are made on material, weld deposit, and the base metal weld heat affected zone, employing a test temperature of not higher than 0°F. The requirements of the ASME Boiler and Pressure Vessel Code, Paragraph N-1211 are met for all materials under jurisdiction of the Code.

Impact tests of weld deposit and base metal weld heat affected zones are made for each welding procedure, requiring ASME Boiler and Pressure Vessel Code, Section IX qualifications.

Specimen removal from the test weld conforms to the requirements of ASME Boiler and Pressure Vessel Code, Section IX and removal of the impact specimens is in accordance with Paragraph N-541.3.

On completion of the Containment Vessel fabrication and after the penetration internals are installed and the construction opening is closed, pneumatic tests are performed in accordance with the applicable requirements of the ASME Boiler and Pressure Vessel Code to demonstrate the integrity and leak tightness of the completed vessel. The bottom head of the Containment Vessel is Halide Leak tested in accordance with Section III, Article 14, Paragraph N- 1411 of the ASME Boiler and Pressure Vessel Code prior to placing interior and exterior concrete fill.

Locks:

A soap bubble inspection test is conducted with the vessel pressurized to 5 psig. Soap suds are applied to all weld seams and gaskets, including both doors of the personnel locks. A second soap bubble inspection test is performed at 36 psig upon completion of the over-pressure test in accordance with the requirements of the ASME Boiler and Pressure Vessel Code. After successful completion of the initial soap bubble test, a pneumatic pressure test is made on the Containment Vessel and each of the personnel locks at a pressure of 45 psig. Both the inner and outer doors of the personnel locks are tested at this pressure. The test pressure in the Containment Vessel is maintained for at least one hour. The test pressure is maintained on each individual lock door for at least one-half hour. Following a successful completion of the over-pressure test, a leakage test at 36 psig pressure is performed on the Containment Vessel with the personnel air lock inner doors closed. Pressure is maintained for whatever length of time is required to demonstrate full compliance with the air tightness requirements. The leakage rate is determined by the “Absolute Method” which consists of measuring the pressure loss with respect to time. Measurements during the test include pressure, temperature, and water vapor content of the air. Verification tests, which consist of imposing a known leak rate on the containment and comparing the measured leak rate with the known valve, are run after each leakage rate test. The tests conform to Appendix J to 10 CFR 50.54 (o) “Reactor Containment Leakage Rate Testing for Water Cooled Power Reactors.” Leakage Rate Testing procedures are described in detail in Subsection 6.2.1.4. The leakage rate during any 24-hour period does not exceed 0.5 percent of the total contained weight of air.

The tests of the locks include operational testing and an overpressure test.

After completion of the locks, including all latching mechanisms and interlocks, each airlock is given an operation test consisting of repeated operation of each door and mechanism to

determine that 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.

The locks are pressurized with air to 45 psig. All welds and seals are observed for visual signs of distress or noticeable leakage. The lock pressure is then reduced to 36 psig, and a soap 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.

The internal pressure of the lock is reduced to atmospheric pressure and all leaks are repaired after which the lock is again pressurized to 36 psig with air and all areas suspected or known to have leaked during the previous test are retested by above soap bubble technique. This procedure is repeated until no leaks are discernible by this means of testing.

Penetrations:

Penetration closure devices for electrical and main steam and main feedwater piping penetrations are purchased by written specification from suppliers with tested closure devices for similar service. Performance data from prototype closures of similar or identical design are required as part of vendor qualifications.

Pipe penetrations which must accommodate thermal movement are provided with expansion bellows. The bellows expansion joints are designed to withstand containment vessel maximum internal pressure and can be checked for leak tightness when the containment vessel is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized to the maximum internal pressure to permit testing the individual penetrations for leakage at any time.

Penetration welds to the Containment Vessel can be leak tested by pressurizing the entire Containment Vessel.

Non-welded interfaces in electrical penetrations are also provided with double seals which are separately tested. The test taps and seals are so located that the leakage tests of the electrical penetrations can be conducted without entering or pressurizing the Containment Vessel.

All containment closures which are fitted with resilient seals or gaskets are separately tested to verify leak-tightness. The covers on flanged closures are provided with double seals and with a test tap which allows pressurizing the space between the seals without pressurizing the entire containment system. In addition, provision is made so that the space between the airlock doors can be pressurized to full containment vessel maximum internal pressure.

3.8.2.1.9 Post-Operational Testing and Inspection

Leakage Rate Testing:

Periodic leakage rate tests of the Containment Vessel and leak tests of the testable penetrations are conducted to verify their continued leak-tight integrity. The method and frequency of these tests are as described in the Containment Leakage Rate Testing Program, which has been established in accordance with Technical Specifications.

Surveillance of Structural Integrity:

A steel shell pressure containment vessel was designed, fabricated, inspected and pressure tested in accordance with the ASME Boiler and Pressure Vessel Code and protected by the concrete Shield Building offers continued structural integrity over the life of the unit. The vessel receives a code stamp from an authoritative body and represents the most recent development in the techniques of pressure vessel design and fabrication that are backed up by years of research, testing, and successful in-service experience. Inservice Inspections of the Containment Vessel are conducted in accordance with the requirements of Subsection IWE of the 2007 Edition, 2008 Addenda, of ASME Section XI as modified by 10CFR 50.55a or by relief granted in accordance with 10CFR 50.55a.

3.8.2.1.10 Penetrations

Design Bases:

All containment penetrations have the following design characteristics in order to maintain the desired containment integrity:

a. Penetrations are capable of withstanding the maximum internal pressure which would occur due to the postulated rupture of any pipe inside the Containment Vessel.

b. Penetrations are capable of withstanding the applicable jet forces associated with the flow from a postulated rupture of the pipe in the penetration or adjacent to it, while still maintaining the integrity of containment.

c. Penetrations are capable of safely accommodating all thermal and mechanical stresses which may be encountered during all modes of operation and testing.

The materials used for penetrations, including the personnel access air locks, the equipment access hatch, the piping and duct penetration sleeves and the electrical penetration sleeves, conform with the requirements set forth by the ASME Boiler and Pressure Vessel Code. In accordance with this Code the penetration materials meet the necessary nil ductility transition impact values as specified in Section III, N-30 of the ASME Boiler and Pressure Vessel Code.

Electrical Penetrations:

Modular type penetrations are used for all electrical conductors passing through the steel containment vessel. The penetration modules are hollow cylinders through which the conductors pass. Each electrical penetration assembly is pressurized with dry nitrogen to maintain and monitor integrity, and to prevent the intrusion of moisture into the penetration. Figure 3.8-12 shows typical electrical penetrations. The penetration assemblies are installed in penetration sleeves welded into the wall of the Containment Vessel which are provided with bolting flanges. Mounting of the assemblies to the sleeves is accomplished by bolting. The headers through which the cables pass are hermetically sealed. All materials used in the design are selected for resistance to all possible environment conditions.

As shown in Figure 3.8-12 two details of electrical penetrations are provided to meet the following requirements:

a. Typical Medium Voltage Power Penetration

b. Typical Low Voltage Power Control & Instrumentation Penetration

Each assembly is sealed and tested at the factory for leakage. The only work that needs to be done in the field is the bolting of the assemblies into the sleeves. Where necessary, the outer sides of the penetration headers are protected against contamination by accumulations of dirt and moisture.

Cable penetrations through the Shield Building are made through relatively leak tight bulkhead type cable seals. Sufficient cable slack is provided in the annulus to allow for differential expansion between the Containment Vessel and the Shield Building.

For further detail and analysis of the electrical penetrations, refer to Subsections 8.3.1.1.8(i) and 8.3.1.2.29.

Piping Penetrations:

Piping penetrations are divided into two general groups:

a. Type 1 Large diameter, high energy, hot piping

b. Type 2 General piping small diameter lower energy piping

Commercially available components are utilized in both types of penetrations and are assembled to fit the physical arrangement conditions present. The penetration designs have been utilized on other installations.

Type 1 penetrations are the main steam and main feedwater lines. Figure 3.8-7 shows the main steam line containment vessel penetration and Figure 3.8-8 shows the main feedwater line containment vessel penetration.

Each main steam and main feedwater containment penetration consists of the following major components:

1. Process Pipe (Main steam or main feedwater pipe)

2. Guard Pipe

3. Flued Head

4. Penetration bellows assembly

To ensure system integrity, all pressure retaining parts are designed in accordance with the requirements of ASME Code, Section III, Class 2 components, designed and analyzed to Seismic Class I, protected against missiles and restrained so that passive failure of one component does not damage adjacent components. In addition a strict quality assurance program is applied on all pressure retaining parts, supports, and restraints to ensure that material and workmanship meet specifications.

1. Process Pipe:

The process pipe is made of welded carbon steel and is welded to the flued head.

2. Guard Pipe:

The guard pipe is made of welded carbon steel and is designed to contain the full pressure of the process pipe including jet effects. In case of passive failure of the process pipe, the guard pipe contains the process fluid and discharges it into the containment past the secondary shield wall and so it protects the containment vessel and the penetration bellows assembly from jet effects and over pressurization. One end of the guard pipe is welded to the flued head while the other end is open to the containment.

3. Flued Head:

The flued head is made from forged carbon steel. It is designed to contain the full pressure of the process fluid in the areas adjoining the process pipe and guard pipe and full containment pressure in parts adjoining the penetration bellows assembly. In addition the flued heads are anchored and restrained at the Auxiliary Building and they are designed and analyzed to be capable of carrying loads resulting from the failure of the process pipe. The anchor and restraint of the flued head prevents axial displacement of the guard pipe and the penetration bellows assembly.

The main steam and feedwater flued heads are designed in such a manner that the stresses resulting from pipe failure loads do not exceed the stresses allowable by the ASME Code, Section III, Paragraph NE-3320 (Summer 1972 Addenda) in the portion of the flued head indicated in Figure 3.8-10a. Results of the analysis for various loading combinations are as follows:

Applied load combination	Allowable Stress* Values	Max. Calculated Stress Intensity M.S.	F.W.
Axial force and internal pressure	1.0 Sm 19,300	10,388	9,230
Bending moment and transverse shear internal pressure	1.5 Sm 28,950	27,816	24,386
Torsional moment and transverse shear and internal pressure	1.5 Sm 28,950	25,340	20,527
Bending moment and axial force and internal pressure	1.5 Sm 28,950	21,454	26,459

*Sm from ASME Section III, Summer 1972 Addenda, Table I-10.1 for SA-508 Class I at 600F.

The main steam and feedwater guard piping is designed to contain the full pressure of the process pipe including jet effects. A longitudinal break in the process pipe, at the weld between the process pipe and the flued head, has been postulated and the stresses in the guard pipe due to the jet effect are calculated and compared to the material's yield stress. The maximum calculated stresses are as follows:

	Maximum Stress (psi)	Allowable Stress-Sy (psi)
Main steam	17,158	28,100
Main feedwater	24,377	25,900

Access to the welds joining the process pipe to the flued head is not possible, as may be seen on Figure 3.8-10a. Therefore, inservice inspection of the welds cannot be performed. However, periodic hydrostatic tests performed on the welds will indicate any cracks formed in the welds.

4. Penetration Bellows Assembly:

The penetration bellows assembly allows differential movement between the Containment Vessel and the Auxiliary Building. The penetration bellows assembly is an extension of the containment and is designed for containment pressure and displacement resulting from thermal expansion and seismic movements.

The penetration bellows assembly consists of the following components:

a. Two metallic bellows expansion joints

b. Interconnecting pipe between bellows

c. End pipe pieces

d. Protective sleeves

a. Metallic Bellows Expansion Joints:

The bellows expansion joints are fabricated of stainless steel material and are designed to contain full containment pressure. They are of two ply construction, meaning that there are two independent concentric bellows elements seal-welded at each end. The two ply construction provides the means for the required frequent testing of the bellows expansion joints for leakage without pressurizing the whole containment vessel. There are two test connections on each bellows.

b&c. Interconnecting Pipe and End Pipe Pieces:

The interconnecting pipe and end pipe pieces are fabricated of welded carbon steel and designed for full containment pressure and for forces resulting from the displacement of the bellows.

d. Protective Sleeves:

Each of the bellows has a removable protective sleeve whose only function is to protect the bellows from mechanical damage during construction and maintenance periods. As the

protective sleeves are not part of the pressure boundary they are not made in accordance with code requirements. Glass covered portholes installed on the protection sleeves permit visual examination of the bellows without removal of the sleeves. It is essential to have removable sleeves to implement closer visual examination of the bellows when necessary and to have special leak tests performed.

An additional component, for the main steam penetrations only, is the low pressure seal. Its function is to seal the annular space between the Containment Vessel and the Shield Building. It consists of carbon steel piping components and a rubberized expansion joint. Because of its function, there are no code requirements for this equipment. Failures of parts of the penetrations are analyzed as follows:

1. Failure of the process pipe outside the Shield Building and beyond the flued head:

Flued head anchors and supports are designed for pipe rupture condition. Thus flued head supports absorb any axial, torsional, or lateral loads so that pipe rupture does not result in any additional displacement for the expansion bellows and no force is transmitted to either the expansion bellows or to the Containment Vessel.

2. Failure of the process pipe inside the guard pipe:

The jet force is deflected by the guard pipe which is designed to take all loads from the process pipe rupture. Fluid leaving the process pipe flows in the annulus between the process pipe and guard pipe and is discharged into the Containment Vessel behind the secondary shield wall. Thus there is no possibility that a fluid jet could damage either the Containment Vessel or the bellows. Axial loads are taken up by the flued head and its supports and lateral loads are taken up by the flued head and the secondary shield wall. No forces or displacement are transmitted to the Containment Vessel or the bellows from this type of failure.

3. Failure of one ply of the expansion bellows:

Failure of one ply of the two ply expansion bellows presents no hazard since each ply is designed to maintain containment integrity.

4. Failure of the flued head:

The flued head is designed in accordance with the requirements of ASME Code, Section III, Class 2 piping components. There is a finite element analysis performed on the flued heads for normal operating conditions and, in addition, for pipe failure conditions. For the above mentioned reasons, failure of the flued head or its supports is not postulated.

5. Failure of the guard pipe:

The guard pipe is designed for a back up service in case of failure of the process pipe. During normal operating conditions it does not carry a load. For the above mentioned reasons, failure of the guard pipe is not postulated.

6. Failure of the low pressure seal:

The function of the low pressure seal is to allow for maintenance of a slight negative pressure in the annulus area between the Containment Vessel and Shield Building after an accident. With

proper periodic servicing and considering the small differential pressure it is postulated that a sufficient leakage path cannot develop to prevent maintenance of the negative pressure.

The rubberized asbestos expansion joint is removable for ease of maintenance and replacement. The life expectancy of the rubberized expansion joint is 10 years.

7. Failure of structural supports and pipe rupture restraints of the flued heads:

Structural supports have been designed with a conservative safety factor and this mode of failure is not postulated.

Type 2 penetrations are welded directly to the Containment Vessel nozzle or through double flued head, butt weld cap, or flat plate. Flued heads are used for penetrations, where the configuration is such that forces and moments are transmitted from the differential movement between the penetrating pipe and the Containment Vessel. Butt weld cap and direct welding the penetrating pipe are used, where there is no differential movement between the Containment Vessel and the penetrating process pipe. Flat plate is used for below grade penetration where there is no differential movement between the Containment Vessel and the penetrating process pipe, due to the concrete embedding of both.

The Containment Vessel nozzles, flued heads, butt weld caps, and flat plates are designed to carry loading due to failure of the penetrating process pipe and to the differential movement between the Containment Vessel and the penetrating pipe.

All flued heads, butt weld caps, flat plates, and penetrating pipes are designed in accordance with the requirements of the ASME Code, Section III, Class 2 Components, designed and analyzed as Seismic Class I, protected against missiles and restrained so that passive failure of one component does not damage adjacent components. In addition a strict quality assurance program is applied to ensure that material and workmanship meet specifications.

A typical Type 2 penetration utilizing flued head is shown on Figure 3.8-10.

Where necessary, the penetrating pipes are anchored and restrained to limit the movement of the pipe relative to the Containment Vessel as well as not to exceed permissible stress levels in the Containment Vessel penetrations. The limitation of movement is utilized to ensure that the design limits of the flexing portions of the penetration are not exceeded during station operation, test, or post-accident condition.

All pipes leaving the penetration room, or where there is no penetration room the Shield Building, are sealed by a flexible rubberized expansion joint. These seals are designed to satisfy the leakage criteria for the Shield Building. They allow the maintenance of a slight negative pressure in the Shield Building and the penetration rooms after an accident, while allowing for the differential movement between the building and the penetrating pipe. With proper periodic servicing and the small differential pressure, it is postulated that a sufficient leakage path cannot develop to prevent maintenance of the negative pressure. The seals are designed for ±3 psi pressure difference. The rubberized expansion joint is removable for ease of maintenance and replacement. The life expectancy of the expansion joints is 10 years.

Figure 3.8-11 shows installation of the low pressure seals for the different process pipe penetrations.

Equipment and Personnel Access:

An equipment hatch is provided as shown in Figure 3.8-4. This is a welded steel assembly, with a double-gasketed, flanged and bolted cover. Provision is made to pressurize the space between the double gaskets to 40 psig. One personnel lock and one emergency lock are provided as shown in Figures 3.8-5 and 3.8-6. These are welded steel assemblies. Each lock has two double gasketed doors in series. Provision is made to pressurize the space between the gaskets. The doors are mechanically interlocked to ensure that one door cannot be opened until the second door is sealed. Provisions are made for deliberately violating the interlock by the use of special tools and procedures under strict administrative control. Each door is equipped with quick acting valves for equalizing the pressure across the doors. The doors are not operable unless the pressure is equalized. Pressure equalization is possible from every point at which the associated door can be operated. The valves for the two doors are properly interlocked so that only one valve can be opened at one time, and only when the opposite door is closed and sealed. Each door is designed so that with the other door open, it can withstand and seal against design and testing pressures of the containment vessel. There is visual indication outside each door showing whether the opposite door is open or closed. Provision is made outside each door for remotely closing and latching the opposite door so that in the event that one door is accidentally left open it can be closed by remote control. The access locks have nozzles installed which permit pressure testing of the lock at any time.

An interior lighting system and a communications system are installed; these systems are capable of operating from the essential power supply.

Fuel Transfer Penetrations:

Two fuel transfer penetrations are provided to transport fuel rods between the refueling canal and the spent fuel pool during refueling operations of the reactor. Each penetration consists of a 30-inch diameter stainless steel pipe installed inside a 42-inch sleeve. The inner pipe acts as the transfer tube. Provisions are made to provide integrity of containment, allowance for differential movement between structures and prevent leakage through the transfer tubes in the event of an accident. Figure 3.8-9 shows the Fuel Transfer Penetration.

Flexible Closures at Penetrations:

The design criteria for the flexible closures of the personnel lock, emergency lock, and the equipment hatch allow for all temperature and pressure transients that could be experienced during the life of the station, the postulated tornado phenomena, or the LOCA. The flexible closures also accommodate the differential movements caused by either earthquakes and/or expansion during normal operating cycles or LOCA. The closure meets the single failure criterion. The closure is designed for a temperature range of 30°F to 150°F, a pressure differential of +3 psi between the annular space and the penetration, both acting concurrently with the maximum differential movements that could be experienced with either earthquake and/or LOCA.

Penetration Design Criteria:

The larger penetrations such as main steam and feedwater are anchored in the Auxiliary Building floor. Flexible bellows type connections are provided at the Containment Vessel shell thus allowing for all differential movements between the two structures. All significant loads including vibrational loads are isolated from the Containment Vessel by the bellows.

Penetration sleeves for smaller pipes are anchored in the Containment Vessel shell. These sleeves and the vessel shell are designed to take the maximum resisting bending and torsional moments which can be transmitted to the shell by plastic yielding of the pipe. The sleeve and vessel are also designed for a thrust and shear equal to 1.5 PxA which are developed for the particular penetration service. Bending is considered to act concurrently with shear, and torsion to act concurrently with thrust. The Containment Vessel penetration is then designed in accordance with Bulletin No. 107 of the Welding Research Council.

Vibrational loads are treated in accordance with Paragraph N-415 of the ASME Boiler and Pressure Vessel Code.

Pipe Breaks Within the Shield Building Annular Space:

Main steam and feedwater lines which pass through the annular space are provided with guard pipes attached to the lines which direct the effects of the pipe rupture into the Containment Vessel, thereby protecting the annular space from the effects of large pipe rupture.

Smaller pipe ruptures have been investigated and found to produce pressure build-up effects less severe than the criterion used to size the vacuum breakers. The Containment Vessel shell is designed for the combined effects of pressure buildups, jet impingement forces, and pipe reaction forces acting concurrently caused by a double ended failure of any single unshielded pipe passing through the annular space.

The allowable local stresses in the Containment Vessel are within those allowed by the ASME Boiler and Pressure Vessel Code for Class B Nuclear Vessels.

In the event that any pipe does not meet the above criteria, guard pipes are provided to direct the effects of the pipe rupture into the Containment Vessel.

3.8.2.1.11 Containment Vessel Painting

The containment vessel is coated with an inorganic zinc primer followed by an epoxy topcoat.

For additional information pertaining to containment coatings see Section 6.1.1.

3.8.2.1.12 Containment Vessel Relief Valves

Inside the containment annulus space ten vacuum relief valves are arranged in two groups of five penetrations on the Containment Vessel. They open when a vacuum begins to develop within the Containment Vessel. The valves are of the swinging-disc type with a self-alignment feature to permit the disc to seat squarely after each opening. When closed the valve holds against a pressure and temperature of 45 psig and 264°F, respectively. The vacuum relief valves are designed to start opening at 0.15 psid. These valves are part of a penetration isolation system which is discussed in Subsection 6.2.4.2. They are designed to ASME Section III, Class 2 and are Seismic Class I. The pressure loss through this system is below the 0.50 psi maximum differential allowed across the containment vessel.

3.8.2.2 Shield Building

3.8.2.2.1 Description

The Shield Building is a reinforced concrete structure of right cylinder configuration with a shallow dome roof. An annular space is provided between the steel containment vessel and the interior face of the concrete shield building of approximately 4.5 feet to permit construction operations and periodic visual inspection of the steel containment vessel. The volume contained within this annulus is approximately 678,700 cu. ft. The Shield Building has a height of 279.5 ft. measured from the top of the foundation ring to the top of the dome. The thicknesses of the wall and the dome are approximately 2.5 ft. and 2 ft., respectively. The design bases for shielding requirements for operational radiation protection are discussed in Chapter 12.

3.8.2.2.2 Design Bases

The Shield Building completely encloses the Containment Vessel, the personnel access openings, the equipment hatch, and that portion of all penetrations that are associated with primary containment. The design of the Shield Building provides for (1) biological shielding, (2) controlled release of the annulus atmosphere under accident condition, and (3) environmental protection of the Containment Vessel.

Adequate reinforcing is placed in the concrete walls, dome, and foundation to control cracking due to concrete shrinkage and temperature gradients. The loading combinations, as stated in Subsection 3.8.2.2.4, provide a design basis to ensure that the Shield Building suffers no loss of shielding or containment function due to seismic or tornado events.

The design of the Shield Building ensures an elastic behavior of steel reinforcement during a Maximum Possible Earthquake controlling cracking of concrete and impairment of leaktight integrity.

The personnel and equipment hatch openings and the major piping penetrations through the Shield Building are designed such that all the anticipated loads are carried by frame action around the openings in accordance with Welding Research Council Bulletin #102. This frame action is achieved by adding sufficient reinforcement around the perimeter of the openings. Diagonal bars at each corner of the opening are added to provide the horizontal and vertical shear resistance.

Normal Operating Conditions:

The normal ambient temperature in the annular space is set by heat loss through the steel containment vessel shell and concrete shield building. The steel containment vessel metal temperature can be maintained above 30°F during reactor operation.

Loss of Coolant Accident Conditions:

Following a loss of coolant accident, heat transferred to the air in the annular space could cause a pressure rise. Conservative assumptions for temperature transmission to the space, and pressure drop in the Emergency Ventilation System are used in sizing the ventilation system. Following this initial pressure transient, the Shield Building is maintained at a minimum negative pressure of 1/4-inch water gauge. The Shield Building structure is analyzed also to ensure

adequate strength to accommodate thermal stresses resulting from thermal gradients produced by the temperature transients.

Loading Conditions:

The following loadings are considered in the design of the Shield Building:

a. Structures dead load

b. Loss of coolant accident load

c. Live Load

d. Wind Load

e. Tornado load

f. Uplift due to buoyant forces

g. Earthquake loads

h. External missiles

i. Equipment load

j. Thermal load

k. Earth load

Penetrations:

The Shield Building and penetration room penetrations for piping, ducts, and electrical cable are designed to withstand the normal environmental conditions which may prevail during station operation and also to retain their integrity during postulated accidents.

The openings in the Shield Building and penetration rooms, including personnel access openings, equipment access openings and penetrations for piping, ducts, and electrical cable are designed to provide containment which is as effective as the Shield Building and consistent with the leakage rate specified. The penetration rooms for the mechanical piping are open to the Shield Building annulus. Sealing of the penetration room to the Shield Building is provided by flexible membranes designed to accommodate relative motion of the two structures.

Bulkhead type doors equipped with gasket seals and positive closure devices are provided for personnel access. A bolted, sealed cover is provided at the equipment opening.

Sealed penetrations are provided for steam lines, cooling water, vents, and other services. Flexible seals or expansion joints are installed where necessary to accommodate pipe movements.

Possible deterioration of seals on the doors and penetrations are detected and remedied through periodic inspections and tests.

Electrical cable penetrations through the Shield Building are made through leaktight cable seals.

Flexibility of all cables is provided between the Shield Building and the Containment Vessel so that no damage can occur to the cables or structures due to differential movement between the two structures.

All redundant controls, instrumentation, and power circuits are physically separated so that no duplicate circuits terminate at the same penetration canister at the Containment Vessel. A more complete discussion of penetrations is presented in Subsection 3.8.2.1.

Corrosion Protection:

For concrete structures, minimum concrete protection for reinforcement conforms to the ACI Standard 318-63.

Based upon the recommendations of ACI Code, corrosion protection for the reinforcing steel in the Shield Building is provided by positioning reinforcing steel to allow 2 inches minimum clearance between the steel and any concrete face on the shield building wall.

Waterproofing membrane is used around that portion of the Shield Building below the groundwater level.

3.8.2.2.3 Codes

The reinforced concrete shield building is designed in accordance with ACI 307-69, Specification for the Design and Construction of Reinforced Concrete Chimneys, and is checked by the Ultimate Strength Design Method in accordance with ACI 318-63. Load combinations specified in ACI 307-69 provide the design basis of the shield building. In addition, the shield building is checked to see that all load combinations with load factors as discussed in Appendix 3A are satisfied by using the Ultimate Strength Design Method.

Subsection 3.8.2.2.7 describes in detail the standard industry specifications which are used in the selection of materials.

The design bases for structures for normal operating conditions are governed by the applicable building codes. The design bases for specific systems and equipment are stated in the appropriate SAR section. The basic design criterion for the design basis accident and seismic conditions is that there be no loss of function if that function is related to public safety.

The design of structures and facilities conforms, but is not limited, to the applicable codes and specifications listed below.

1. Uniform Building Code (UBC), 1967 Edition.

2. American Institute of Steel Construction (AISC) “Specification for the Design, Fabrication and Erection of Structural Steel for Buildings” - Sixth Edition.

3. American Iron and Steel Institute (AISI) “Specification for the Design of Light Gage Cold- formed Steel Structural Members” - 1961 Edition.

4. American Concrete Institute (ACI) “Building Code Requirements for Reinforced Concrete” - (ACI-318-63 and ACI-318-70).

5. American Welding Society (AWS)

- AWS D1.0-66	“Standard Code for Arc and Gas Welding in Building Construction”
- AWS D12.1-61	“Recommended Practices for Welding Reinforcing Steel, Metal Inserts and Connection in Reinforced Concrete Construction”
- AWS D1.4-98	“Structural Welding Code - Reinforcing Steel

6. AEC Publication TID 7024 “Nuclear Reactor and Earthquakes” - governs the seismic design of all Class I structures.

7. American Society of Civil Engineers (ASCE) Paper No. 3269, “Wind Forces on Structures” - governs wind design requirements.

8. American Nuclear Insurers - Nuclear Electrical Insurance Limited (ANI-NEIL) Basic Fire Protection for Nuclear Power Plants.

9. American Concrete Institute (ACI) “Specification For the Design and Construction of Reinforced Concrete Chimneys” - (ACI-307-69).

3.8.2.2.4 Loading

Design Loads:

a. Dead Load:

Dead load consists of the dead weight of the Shield Building, the Containment Vessel, grout under the Containment Vessel, the foundation slab, the weight of structural steel, and miscellaneous building items within the Containment Vessel.

Densities used for dead load calculation are as follows:

1. Concrete: 143 lb/ft^3

2. Steel Reinforcing: 489 lb/ft^3 using nominal cross section areas of reinforcing bar sizes

3. Steel Containment Vessel: 489 lb/ft^3

b. Loss of Coolant Accident Load:

The loss of coolant accident load is determined by analysis of the pressure and temperature transients in the annulus during a loss of coolant accident. The Emergency Ventilation System will keep the pressure in the annulus within design limits under this condition.

c. Live Load:

Live load on the dome is uniformly applied to the top surface of dome at an assumed value of 40 lb. per horizontal plan projection square foot. Live load is not considered in conjunction with loss of coolant accident load, and is intended only to ensure structural adequacy of the roof.

d. Wind Load:

Wind loading for the Shield Building is based on Figure 1 (b) ASCE Paper 3269, "Wind Forces on Structures" (2), using the fastest mile of wind for a 100-year period of recurrence. This results in 90-mph basic wind at 30ft. above grade. In addition, this paper is used to determine shape factors, gust factors, and variation of wind velocity with height. See Section 3.3 for additional information.

e. Tornado Load:

Tornado loads are computed using the following (see Section 3.3):

- Design wind speed of 300 mph
- Design pressure drop of 3.0 psi
- Tornado missiles per Table 3.3-2, Table 3.5-2 and Section 3.5

f. Uplift Due to Buoyant Forces:

Uplift forces which are created by the displacement of groundwater by the structure is accounted for in the design.

g. Seismic Loads:

Seismic loads are computed using the following: (See Section 3.7)

1. Maximum Probable Earthquake horizontal seismic ground acceleration considered is 0.08g.

2. Maximum Possible horizontal seismic ground acceleration considered is 0.15g.

3. Vertical component of 2/3 of the horizontal ground acceleration is applied simultaneously with the horizontal acceleration.

h. External Missiles:

The Shield Building is designed to withstand, without loss of function, a tornado driven missile as described in Section 3.5.

Load Combinations:

The Shield Building is a reinforced concrete structure except that a temporary steel frame was used for shoring the concrete dome during construction and was removed upon completion of

the dome. Therefore, the steel frame was designed only for dead load plus live load and the allowable stress was limited to 1.0 f_s in accordance with AISC Steel Construction Manual.

The Shield Building is designed by the ultimate strength method for evaluation by ACI 318-63 as well as the working stress method by ACI 307-69.

The loading combinations described in Subsection 3.8.2.3.4 are used for the ultimate strength design, and the following loading combinations are also considered in the working stress design method in accordance with ACI 307-69.

a. T_o

b. D. + T_o

c. D. + W

d. D. + E

e. D. + T_o + E

f. D. + T_o + W

Note: For notations, see Subsection 3.8.2.3.4.

3.8.2.2.5 Analytical Techniques

The membrane theory of thin shells is used to analyze the spherical dome and the cylindrical wall of the Shield Building in accordance with Article I-2, Analysis of Cylindrical Shells, and I-3, Analysis of Spherical Shells, of the ASME Code, Section III, 1968.

At places where changes in geometry or loading exist, an analogy of a beam on elastic foundations is made in conformance with Article I-7, Discontinuity Stresses, of the ASME Code, Section III, 1968.

For seismic load, a cantilevered beam of lumped masses is modeled to determine seismic forces. Refer to Subsection 3.7.2.1 for a more detailed description.

For thermal load, the cracking effect of reinforced concrete is considered as suggested by ACI 307-69. This cracking is limited by using a relatively large amount of reinforcing steel. Based upon ACI Publication SP-20, “Causes, Mechanism and Control of Cracking in Concrete,” under the operating conditions the maximum width of cracks is found to be 0.009 inch, which is less than the allowable (0.01 inch) permitted by the ACI Standard Building Code.

In the design of foundations, only the reinforced portion which is outside of the Containment Vessel is considered to act effectively as a spread footing.

The rest of the concrete mat is considered as dead weight. The portion inside the Containment Vessel is entirely reinforced to act as a rigid mat. It is designed to be capable of taking any unbalanced shear and vertical force which may be imposed on it by the outside footing.

All pipe penetrations and the equipment hatch, personnel lock and emergency lock are

structurally isolated from the Shield Building either by using expansion joints, bellows, or by providing an adequate gap between structures.

Welding Research Council Bulletin No. 102 is followed in analyzing stress distributions due to load combinations at openings in the cylindrical shield wall. Adequate reinforcement is provided to strengthen areas where required by the analysis.

3.8.2.2.6 Design Stresses

Stresses for Ultimate Strength Design:

A value of 4000 psi is used for f'_c and a value of 60,000 psi is used for F_y in the ultimate strength design In accordance with the ACI-318-63 Code. The ultimate resisting capacity so obtained is found to be greater than the external moment and forces due to the loading combinations described in Subsection 3.8.2.2.4.

The smallest safety margin provided on the shield building is greater than the ACI-318-63 Code requirements by 3 percent. This applies to the vertical reinforcement at the base floor under the load combination of $U=1.0D+1.0L+1.0E'+1.0T_A+1.0H_A+1.0R$.

Consequently, it can be concluded that the Shield Building has satisfied structurally all the loading requirements as described in Subsection 3.8.2.2.4.

Stresses for Working Stress Design:

The allowable stresses for each loading combination listed in Subsection 3.8.2.2.4 are found in accordance with Paragraph 4.9, ACI 307-69 as follows:

a. f_c = 1600 psi f_s = 24 ksi
b. f_c = 1600 psi f_s = 24 ksi
c. f_c = 1000 psi f_s = 15 ksi
d. f_c = 1500 psi f_s = 18 ksi
e. f_c = 2680 psi f_s = 32.4 ksi
f. f_c = 2680 psi f_s = 32.4 ksi

Using techniques described in Subsection 3.8.2.2.5, the most severe loading combination is found to be $D. + T_o + E$. However, the stress in the reinforcement is only 24.6 ksi, which is much less than the allowable 32.4 ksi. Therefore, it can be concluded that the shield building has also satisfied the stress requirement of ACI 307-69.

3.8.2.2.7 Materials

Concrete and reinforcing steel used in the construction of the Shield Building is described in Subsection 3.8.2.3.7.

Structural steel, miscellaneous steel and steel decking are described in Subsection 3.8.1.

3.8.2.2.8 Preoperational Testing

General Requirements:

Appropriate ASTM Material Specifications are cited in the Building Specifications for all construction materials which describe the testing and basis for acceptance of materials. Standards and tests are specified in accordance with applicable regulations and current building practices.

The testing of concrete and reinforcing bars is accomplished by an independent testing laboratory whose primary business is to perform such testing and who can show proof of the required knowledge and facilities to perform the specified tests and report accurate results. This testing company examines local aggregates and cement, takes samples of concrete mixes and makes the required field tests.

Reinforcing steel samples of each bar size from each heat are supplied by the reinforcing steel supplier. These samples are tested based upon ASTM specifications. The user inspection and testing of reinforcing is as follows:

a. Material:

All user-tests of reinforcing steel are in accordance with ASTM Specifications. Tests include one tension and one bend test per heat or per mill shipment, whichever is less, for each diameter bar. Test samples are obtained at the fabrication station. High strength bars are clearly identified prior to shipment.

b. Fabrication:

Visual inspection of fabricated reinforcements is performed to ascertain dimensional conformance with specifications and drawings.

c. Placement:

Visual inspection of in-place reinforcements is performed by the placing inspector to ensure dimensional and location conformance with drawings and specifications.

Inspections are performed as necessary to verify compliance with specifications.

Because of the loads considered in the design of the shield building, standard building construction and quality control practices are satisfactory. Special attention is given to obtaining good leak-tightness.

Leak Tightness:

The Emergency Ventilation System is used to draw a negative pressure on the annulus, thus ensuring adequate leak tightness of the Shield Building and penetration rooms.

3.8.2.3 Containment Vessel Internal Structures

3.8.2.3.1 Description

The containment internal structures are comprised of the reactor cavity, the primary shield wall, the secondary shield wall, the refueling pool, the operating floors miscellaneous equipment supports, stairs, and service missile shields. The primary coolant system, including the reactor, steam generators, pressurizer, and reactor coolant pumps, is supported by these structures. Shield walls and floors are constructed of reinforced concrete. Structural steel frames and columns support the floors and transmit loads to the foundations. Metal decks provide support for the concrete floors during concrete placement.

3.8.2.3.2 Design Bases

Structures which house or support the basic systems are designed to sustain the Class I loading combinations as described in Subsection 3.8.2.3.4.

Design Bases to be Applied:

a. The structures can sustain all operating loads, thermal loads, seismic loads, and thermal deformations with a reduction of yield capacity as shown in Appendix 3C.

b. Loads and deformations resulting from a loss of coolant accident and its associated effects is restricted such that the propagation of the failure to any other system is prevented. In addition, a failure in one loop of the nuclear steam supply system does not cause failure in the other loop.

Loss of Coolant Accident Loads and Associated Effects:

a. Thrust loads resulting from rapid mass release from a pipe break in any system.

b. Pressure buildup in locally confined areas.

c. Jet forces resulting from the impingement of the escaping mass upon adjacent structure.

d. Pipe whipping following a break in the pipe.

e. Rapid rise in ambient temperature and accompanying rise in ambient pressure.

f. Internal missile loads as described in Section 3.5.

The pressure buildup within a sub-compartment resulting from a pipe break is determined by using Bechtel’s peripheral computer program COPRA. Following the LOCA, high enthalpy water flows out of both ends of the ruptured pipe, flashing partly to steam. As the pressure builds up within the compartment, the steam-air-water mixture flows through openings into the main containment.

The time-dependent pressure differential between compartments following the LOCA and the thrust forces resulting from the pipe break are calculated based on the rate of blowdown.

Seismic analysis for the interior structures conforms to the appropriate procedures outlined in Section 3.7. Where concrete structures, such as the primary shield wall, are subjected to sustained internal heat buildup, air cooling is provided to keep the internal temperatures below acceptable design limits.

Internal Missile Protection Features:

High-pressure Reactor Coolant System equipment which could be the source of missiles is suitably screened either by the concrete shield wall enclosing the reactor coolant loops or by the concrete operating floor to block any passage of missiles to the containment vessel walls. The steam drum, which forms an integral part of the steam generator, represents a mass of steel which provides protection from missiles originating in the section of the containment within the shield wall and below the operating floor. A structure is provided over the control rod drive mechanisms to intercept any missiles generated from a hypothetical fracture of a housing.

Missile protection is provided to comply with the following criteria:

a. The containment vessel is protected from loss of function due to damage by such missiles as might be generated in a loss of coolant accident for break sizes up to and including the double-ended severance of a reactor coolant pipe.

b. The engineered safety features and components required to maintain containment integrity are protected against loss of function due to damage by missiles.

During the detailed station design, the missile protection necessary to meet the above criteria was developed and is implemented using the following considerations:

a. Components of the Reactor Coolant System are examined to identify and to classify missiles according to size, shape, and kinetic energy for purposes of analyzing their effects.

b. Missile velocities are calculated considering both fluid and mechanical driving forces which could act during missile generation.

c. The structural design of the missile shielding takes into account both static and impact loads.

d. The Reactor Coolant System is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with a double-ended rupture of a reactor coolant pipe and designed to stop the missiles.

3.8.2.3.3 Codes

The concrete internal structures are designed in accordance with American Concrete Institute Standard Building Code Requirements for Reinforced Concrete (ACI-318-63) Ultimate Strength Design Method. Structural steel design conforms to the requirements of the American Institute of Steel Construction Manual of Steel Construction (AISC), seventh edition. Other applicable codes are specified in Subsection 3.8.2.2 for the Shield Building.

3.8.2.3.4 Loading

Design Loads:

a. Dead Load:

Dead load consists of the dead weight of the concrete walls and floors, the structural steel frames, the Nuclear Steam Supply System and all other equipment within the containment. Densities of materials used for dead load calculations are given in Subsection 3.8.2.2.4.

b. Live Loads:

Live loads on the floors encompass all modes of operation and maintenance of the plant. Loads are conservatively estimated considering the heaviest loading the floor might experience during the plant life time. Floors are designed to support construction operations such as transporting the nuclear supply system and its components into their final locations.

c. Wind and Tornado Loads

Because the internal structures are fully contained within the shield building and containment vessel, wind and tornado loads are not experienced.

d. Seismic Loads:

The internal structures are designed to carry out their functions during seismic experiences. Section 3.7 details the seismic analytical method used.

e. Loss of Coolant Accident Loads:

For the original design of the plant, the pressure differentials resulting from a loss of coolant accident were considered in designing all structural elements within the containment. Analyses which develop pressure differentials across walls and floors are made considering the mass-energy release from a double-ended pipe rupture. By using computer code COPRA, the maximum pressure differential which the reactor cavity may experience during a LOCA is calculated to be 195 psi. However, the actual pressure differential used in the design of the reactor cavity is 225 psi, which is based on a more conservative preliminary analysis.

The Reactor Coolant System has been evaluated using the criteria of the Standard Review Plan 3.6.3, Leak-Before-Break (LBB) evaluation procedures. This criteria, in conjunction with General Design Criterion 4 (GDC-4) of 10CFR50 Appendix A, allows the exclusion of the dynamic effects of a postulated pipe rupture. LBB excludes cold leg and hot leg breaks from the Reactor Vessel (RV) cavity pressurization analysis but does not exclude the Core Flood Line (CFL) break. The CFL break will be the limiting LOCA event in the RV cavity. By using computer code CRAFT2, the maximum pressure in the RV cavity from a CFL break would be 92 psi with the Permanent Canal Seal Plate installed.

For the original design of the plant, thermal gradients were developed by a heat-transfer computer program, Tiger V, which determines the time-dependent temperature profile across the primary shield wall. By comparing the output of COPRA and Tiger V, it can be observed that the time lag between the pressure buildup in the reactor cavity and the temperature build-

up inside of the primary shield wall is so large that loads can be treated as non-concurrent conditions for design purposes.

The same conclusion also holds true for all other containment internal structures. COPRA is again used to determine the maximum pressure differential across the wall of steam generator compartments (secondary shield wall). In addition to a maximum pressure differential of 86.5 psi, the effect of a concurrent jet force of 1.6 $x10^{6}$ lbs due to a rupture of the primary coolant system is considered in the design of the steam generator compartments.

The pressurizer is located within one of the steam generator compartments.

Those jet forces which act concurrently with the pressure effects are combined with the applicable operating loads.

f. Internal Missiles:

A spectrum-of missiles which might develop during accident conditions was investigated. Vital structures and equipment are protected from missile effects by separation of the components, or by providing suitable missile shielding. (See Section 3.5.1)

g. Load Combinations:

Class I equipment and systems located in Class II structures have reinforced concrete enclosures designed to withstand the loads for Class I structures.

All steel structures are designed by the Working Stress Method. All reinforced concrete structures are designed by the Ultimate Strength Method.

Class II structures are designed in accordance with design methods of referenced codes and standards, with prudent engineering practice, and in accordance with applicable codes. The area is in Seismic Zone No. 1 (UBC-1967).

1. Notations:

U = Required ultimate load capacity.

D = Dead load of structure and equipment plus any other permanent loads contributing stresses, such as soil or hydrostatic loads. An allowance is also made for future permanent loads.

L = Live load and piping loads.

R = Force or pressure on structure due to rupture of any one pipe.

T_o = Thermal loads due to temperature gradient through wall under operating conditions.

H_o = Force on structure due to thermal expansion of pipes under operating conditions.

T_A = Thermal loads due to temperature gradient through wall under accident conditions.

H_A = Force on structure due to thermal expansion of pipes under accident conditions.

E = “Maximum Probable Earthquake” resulting from ground surface acceleration of 0.08g.

E’ = “Maximum Possible Earthquake” resulting from ground surface acceleration of 0.15 g.

W = Wind load. (Wind velocity 90 mph at 30 ft above ground. See ASCE 3269 for increase due to gusts and height.)

W’ = Tornado load including differential pressure.

Φ = Capacity reduction factor. (Defined in ACI-318-63 Code, Section 1504. See Appendix 3C for discussion.)

f_s = Allowable stress for structural steel. (Defined in AISC, Section 1.5)

F_y = Yield strength for steel. See Subsection 3.8.2.3.4.g.2.

2. Class I structures and equipment support during normal operation condition:

For loads encountered during normal station operation, Class I structures, systems, and equipment are designed in accordance with design methods of referenced codes and standards. Seismic design is in accordance with Section 3.7.

Concrete:

Reinforced concrete structures are designed for ductile behavior, whenever possible; that is, with steel stresses controlling.

Design of concrete structures satisfies the most severe loading combinations, based on the load factors shown below:

$U = 1.5\ D + 1.8\ L$

$U = 1.25\ (D + L + H_o + E) + 1.0\ T_o$

$U = 1.25\ (D + L + H_o + W) + 1.0\ T_o$

$U = 0.9\ D + 1.25\ (H_o + E) + 1.0\ T_o$

$U = 0.9\ D + 1.25\ (H_o + W) + 1.0\ T_o$

In addition, for ductile moment resisting concrete space frames, shear walls and braced frames:

$U = 1.4\ (D + L + E) + 1.0\ T_o + 1.25\ H_o$

$U = 0.9\ D + 1.25\ E + 1.0\ T_o + 1.25\ H_o$

For structural elements carrying mainly earthquake forces, such as equipment supports:

$$U = 1.0\ D + 1.0\ L + 1.8\ E + 1.0\ T_o + 1.25\ H_o$$

Yield Capacity Reduction Factors:

The yield capacity of all load carrying structural elements is reduced by a yield capacity reduction factor (ϕ) as given below. The justification for these numerical values is given in Appendix 3C. This factor provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, control, and degree of supervision while individually within required tolerance and the limits of good practice occasionally may combine to result in under-capacity.

Yield Capacity Reduction Factors:

Φ = 0.90 for concrete in flexure.

Φ = 0.85 for shear/diagonal tension, bond, and anchorage in concrete.

Φ = 0.75 for spirally reinforced concrete compression members.

Φ = 0.70 for tied compression members.

Φ = 0.90 for fabricated structural steel.

Φ = 0.90 for reinforcing steel in tension (excluding splices).

Φ = 0.90 for reinforcing steel in tension with mechanical splices.

Structural Steel:

Steel structures shall satisfy the following loading combinations without exceeding the specified stresses:

1. D+L: Stress Limit = f_s

2. D+L+ T_o + H_o + E: Stress Limit = 1.25 f_s

3. D+L+ T_o + H_o + W: Stress Limit = 1.33 f_s

In addition, for structural elements, carrying mainly earthquake forces, such as struts and bracings:

D + L + T_o + H_o + E: Stress Limit = f_s

During Accident and Maximum Possible Earthquake Conditions:

The Class I structures, systems, and equipment are in general proportioned to maintain elastic behavior when subjected to various combinations of dead loads, thermal loads, seismic loads,

and accident loads. The upper limit of elastic behavior is considered to be the yield strength of the effective load-carrying structural materials. The yield strength F_y for steel (including reinforcing steel) is considered to be the guaranteed minimum given in appropriate ASTM Specifications. The yield strength for reinforced concrete structures is considered to be the ultimate resisting capacity as calculated from the “Ultimate Strength Design” portion of the ACI-318-63 code.

Concrete:

Concrete structures satisfy the most severe of the following loading combinations:

$$U = 1.0\ D + 1.0\ L + 1.25\ E + 1.0\ T_A + 1.0\ H_A + 1.0\ R$$

$$U = 1.0\ D + 1.25\ E + 1.0\ T_A + 1.0\ H_A + 1.0\ R$$

$$U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ T_o + 1.25\ H_o + 1.0\ R$$

$$U = 1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ T_A + 1.0\ H_A + 1.0\ R$$

$$U = 1.0\ D + 1.0\ L + 1.0\ W' + 1.0\ T_o + 1.25\ H_o$$

Structural Steel:

Steel structures satisfy the most severe of the following loading combinations without exceeding the specified stresses:

1. $D + L + R + T_o + H_o + E'$: Stress Limit* = 1.5 f_s

2. $D + L + R + T_A + H_A + E'$: Stress Limit* = 1.5 f_s

3. $D + L + W' + T_o + H_o$: Stress Limit* = 1.5 f_s

*Maximum allowable stress in bending and tension is 0.9 F_y. Maximum allowable stress in shear is 0.5 F_y.

Stress in some of the materials may exceed yield strength. If this is the case an analysis is made to determine that the energy absorption capacity of the structure exceeds the energy input. The resulting deflection or distortion is reviewed. The containment vessel and engineered safety features system components are protected by barriers from all credible missiles which might be generated from the primary system. Local yielding or erosion of barriers is permissible due to jet or missile impact, provided there is no general failure.

The final design of the missile barrier and equipment support structures inside the containment is reviewed to ensure that they can withstand applicable pressure loads, jet forces, pipe reactions, missiles and earthquake loads without loss of function. The deflections or deformations of structures and supports are checked to ensure that the functions of the containment and engineered safety features equipment are not impaired.

3. Class II Structures and Equipment Supports:

Concrete:

Load factors and combinations as specified apply. All other design, except load factors and combinations, is as specified in the ACI Standard 318-63. Significant thermal loads are included. The following load factors and combinations are used for ductile moment resisting space frames and shear walls:

$$U = 1.4 (D + L + E) + 1.0 T_o + 1.25 H_o$$

$$U = 0.9 D + 1.25 E + 1.0 T_o + 1.25 H_o$$

Structural Steel:

Steel structures satisfy the following loading combinations without exceeding the specified stresses:

1. D + L: Stress Limit = f_s

2. $D + L + T_o + H_o + E$: Stress Limit = 1.33 f_s

3. $D + L + T_o + H_o + W$: Stress Limit = 1.33 f_s

3.8.2.3.5 Analytical Techniques

Reactor Cavity (Primary Shield Wall):

The reactor cavity is considered as a thick-walled cylinder subjected to axisymmetrical loads which are described in Subsection 3.8.2.3.4. A complete description of the analysis can be found in Chapter VI, "Deformations Symmetrical About An Axis", of "Strength of Materials" by S. Timoshenko.

The portion of the reactor cavity which is penetrated by reactor coolant pipes is treated differently. Those areas between penetrations are considered as columns spanning from base floor to El. 574'-0.

The primary shield wall is structurally isolated from the primary coolant pipes by providing an adequate gap. However, LOCA pipe restraints are embedded in the primary shield wall. These restraints limit the movement of the reactor during the postulated LOCA.

Steam Generator Compartments:

Steam generator compartments are analyzed and designed as a sequence of fixed end simple beams to sustain all loading combinations described in Subsection 3.8.2.3.4. Bending moments and axial forces which represent the reaction of the adjacent beam are considered to act concurrently.

By multiplying the horizontal acceleration determined from the seismic analysis by the unit weight of the wall, a unit seismic force is obtained. This force, in turn, is considered as a uniformly distributed load in designing and analyzing the steam generator compartments.

“Advanced Reinforced Concrete” by C.W. Dunham is used as a reference in designing large slabs for concentrated loads. Reinforcing steel requirements are determined for the most critical section of each fixed-end beam and is used through the beam section.

Steam Generator Supports:

Steam Generator stability is ensured by a system of upper and lower supports which are designed for LOCA loads concurrent with maximum seismic loads while allowing unrestricted movement due to thermal expansion of the Reactor Coolant System. Due to leak before break, large break LOCA loads are not considered in the structural design of the steam generator supports. The largest LOCA load considered in the structural design is a surge line break. The upper support consists of a heavy steel rigid frame structure, spanning between the secondary shield walls, which is attached to four trunnions mounted on the upper steam generator tube sheet. Vertical thermal and rotational motion are allowed by mounting the trunnions in machined blocks.

The lower steam generator support consists of a lubrite ball joint which allows free rotation and radial sliding. Transverse motion is guided by tightly fitted lubrite bearings. This system accommodates all thermally induced motions without imposing significant restraints on the reactor coolant system. Transverse lateral loads are supported directly by the transverse lubrite bearings. Radial lateral loads have been analyzed and do not require additional support. Vertical dynamic loads are transferred through the lubrite ball joint directly to the concrete foundation by holddown bolts.

Pressurizer Support:

The pressurizer support, which is anchored on the steam generator compartment wall, is designed and analyzed as a steel truss. All loading combinations as described in Subsection 3.8.2.3.4 are satisfied.

A conventional working stress design approach is employed in conformance with the “Steel Construction Manual” of AISC.

Floor Systems:

Conventional beam and column design methods are used for all floor systems. No composite action between concrete slab and supporting beam is assumed for simplicity in design.

Reactor Vessel Supports:

The reactor vessel is supported by four groups welded plate girders, each group is composed of two steel plate girders (see Figure 3.8-16). These girders are designed utilizing all possible combinations of dead, live, thermal and seismic loadings as described in Subsection 3.8.1. All of these loads are transmitted by the girders already to the primary shield walls, except LOCA loadings, which are designed assuming that the welds on the reactor nozzle beam supports fail in shear, allowing the reactor to move horizontally until impacting occurs on the pipe (LOCA) supports in the primary shield walls.

Polar Crane Supports:

The polar crane is supported by a continuous box-type girder attached to the Containment Vessel as shown in Figure 3.8-15. The girder is approximately 14-1/2 feet deep by 5 feet wide and rigidly attached to the Containment Vessel, enabling it to transfer design loads including earthquake-induced shears directly to the Containment Vessel.

The crane support girder is connected to the Containment Vessel by means of welding. For type and size of the welds, see Section A-A of Figure 3.8-15.

3.8.2.3.6 Design Stresses

Concrete Structures:

The ultimate strength design method is employed for all internal concrete structures. A value of 5,000 psi is used for f'_c and a value of 60 ksi is used for F_y in calculating the ultimate resisting strength of the internal structure. Stress reduction factors are also used to determine the ultimate resisting capacities as described in Appendix 3C. The most severe loading combination is the combination of

$$1.0\ D + 1.0\ L + 1.0\ E' + 1.0\ T_A + 1.0\ H_A + 1.0\ R$$

However, in no case is the ultimate strength provided by the internal structure less than the calculated external force.

Steel Structures:

Structural steel design stress limits are specified for different loading combinations in Subsection 3.8.2.3.4.

In designing the upper steam generator supports, the calculated combined bending and tensile stress of a center member is found to be 31.9 ksi under the load combination of maximum earthquake plus LOCA. The safety factor against the allowable limit of 32.4 ksi is therefore 1.01. All other steel structures have larger safety factors.

In summary, all steel structures satisfy the loading requirements as specified in Subsection 3.8.2.3.4.

3.8.2.3.7 Materials

Basically, two materials are used for containment internal structures and the Shield Building. They are:

a. Concrete, and

b. Reinforcing steel

Detailed specifications and working drawings for these materials and their installation are of such scope as to ensure that the quality of work is commensurate with the necessary integrity of the containment.

Basic specifications for these materials, for their procurement, and for their delivery to the structure are described in the following sections.

Concrete:

All concrete work is in accordance with ACI-318-63, Building Code Requirements for Reinforced Concrete, and ACI-301-66, Specifications for Structural Concrete for Buildings. Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement, fly ash/slag, and water. Admixtures are added to improve the quality and workability of the plastic concrete during placement and to retard the set of the concrete. Aggregates conform to Standard Specifications for Concrete Aggregate, ASTM Designation C33. Fine aggregate consists of sharp, hard, strong, and durable sand, free from adherent coatings, clay, loam, alkali, organic material, or other deleterious substances.

Acceptability of aggregates is based on the following ASTM tests. The tests are performed by a qualified commercial testing laboratory.

Test	ASTM
Los Angeles Abrasion	C-131
Clay Lumps Natural Aggregate	C-142
Material Finer No. 200 Sieve	C-117
Mortar Making Properties	C-87
Organic Impurities	C-40
Potential Reactivity (Chemical)	C-289
Potential Reactivity (Mortar Bar)	C-227 (if necessary after performance of C-289)
Sieve Analysis	C-136
Soundness	C-88
Specific Gravity and Absorption	C-127 C-128
Petrographic Examination	C-295

Aggregate testing is carried out as follows:

a. Sand Sample for Gradation (ASTM C-33 Fine Aggregate).

b. Organic Test on Sand (ASTM C-40)

c. 3/4 Inch Sample for Gradation (ASTM C-33 Size No. 67).

d. 1-1/2 Inch Sample for Gradation (ASTM C-33 Size No. 4).

e. Check for Proportion of Flat and Elongated Particles.

Cement for all concrete except the Shield Building is Type II low alkali cement as specified in “Standard Specifications for Portland Cement,” ASTM Designation C-150 and is tested to comply with ASTM C-114. The Shield Building has Type I cement above grade.

An equivalent of 15 percent of the weight of cement is replaced by fly ash in concrete except in concrete used in slip-form work. Shield Building laminar cracking repair replacement concrete does not include fly ash as this is a replacement material for slip form work concrete which did not include fly ash.

Fly Ash is specified as ASTM C-618-68 Class F, Fly Ash and Raw or Calcined Natural Pozzolans for use in Portland Cement Concrete, and is tested to comply with ASTM C-311, Sampling and Testing Fly Ash for Use as an Admixture in Portland Cement Concrete.

An equivalent of 40 percent of the weight of cement is replaced by slag in concrete for the Shield Building laminar cracking repair. Slag is acceptable for this repair as opposed to fly ash as noted above due to its material properties, quality and consistency.

Per ACI 318-63, slag is to meet the requirements of ASTM C 205, Specification for Portland Blast Furnace Slag Cement. ASTM C 205 was withdrawn and replaced by ASTM C 595, Standard Specification for Blended Hydraulic Cements. ASTM C 595 covers slag mixed with cement. The slag component is to meet the requirements of ASTM C 989, Standard Specification for Slag Cement for Use in Concrete and Mortars, to comply with ASTM C 595.

Water for mixing concrete is clean and free from any deleterious amounts of acid, alkali, salts, oil, sediment, or organic utter, and it is required to pass ACI 313 requirements.

A water-reducing agent is employed to reduce shrinkage and creep of concrete. Admixtures containing chlorides are not used.

Concrete mixes are designed in accordance with ACI 613, using materials qualified and accepted for this work. Only mixes meeting the design requirement specified for containment building concrete are used. Trial mixes are tested in accordance with applicable ASTM Codes as indicated below:

Test	ASTM
Making and Curing Cylinders in Laboratory	C-192
Air Content	C-231
Slump	C-143
Bleeding	C-232
Compressive Strength Tests	C-39

The concrete has a design compressive strength of 4000 psi at 28 days for the Shield Building walls, dome, and building foundation and 5000 psi at 28 days for the Internal Structures.

Concrete samples are taken from the mix according to ASTM C-172, Sampling Fresh Concrete. From these samples, cylinders for compression testing are made. They are stripped within 24 hours after casting and marked and stored in the curing room. These cylinders are made in accordance with ASTM C-31, "Tentative Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field."

Slump, air content, and temperature measurements are taken when cylinders are cast. Slump tests are performed in accordance with ASTM C-143, "Standard Method of Test for Slump of Portland Cement Concrete." Air Content tests are performed in accordance with ASTM C-231, "Standard Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method." Compressive strength tests are made in accordance with ASTM C-39, "Method of Test for Compressive Strength of Molded Concrete Cylinders."

Evaluation of compression test is in accordance with ACI 214-65.

Design and construction complies with appropriate provisions of ACI 318-63 for concrete cover over conventional reinforcement with the following exceptions:

a. Bar sizes greater than No. 11 and Cadweld splice sleeves at formed faces in contact with soil; 3 inch minimum clear cover.

b. Bar sizes No. 11 or less at formed faces in contact with soil; 2 inch minimum clear cover.

c. 2 inch minimum cover for reinforcement in the shallow dome and 3 inch clear cover for reinforcing steel in the vertical wall and foundation of the shield building. During the Reactor Pressure Vessel Head Replacement in 2002, it was found that a localized area on the exterior face of the shield building wall has less than 3 inch clear cover. This condition was evaluated and found acceptable. Clear cover over reinforcing steel in the vertical wall of the Shield Building, no less than the values specified in ACI 318-63, has been evaluated and found acceptable for subsequent maintenance activities.

Reinforcing Steel:

Main reinforcing steel in the internal structures and the Shield Building consists of deformed billet steel bars conforming to ASTM A-615-68 Grade 60.

Horizontal dowels between shield wall and internal concrete fill are deformed billet steel conforming to ASTM-A-615, Grade 40.

Mill test results are obtained from the reinforcing steel supplier for each heat of steel to show proof that the reinforcing steel has the specified composition, strength, and ductility.

All user-tests of reinforcing steel are in accordance with ASTM-A-615 specification. Tests include: (1) one tension and one bend test (random diameter size selected) from each 50 tons of number 11 bar size or smaller delivered and (2) one tension test for each number 14 and number 18 bar size from each 100 tons delivered at the job site. High strength bars are clearly identified prior to shipment.

The following are requirements for splices in reinforcement:

a. For bar sizes No. 11 and smaller, splices are made in accordance with ACI 318-63 and ACI 318-70 as applicable.

b. For bar sizes greater than No. 11, mechanical or welded splices are used.

c. All mechanical (Cadweld) splicing is in accordance with Appendix 3B.

Welding of reinforcing steel is performed by qualified welders in accordance with AWS D12.1-61, "Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction" or AWS D1.4-98, "Structural Welding Code - Reinforcing Steel" but tacking welding is not permitted.

Shield Building Wall Openings:

For the reactor pressure vessel head replacement project in 2002, a temporary access opening was created through the Shield Building wall. The wall opening was restored by replacing the removed segments of reinforcing steel and placing replacement concrete. The approved concrete mix was designed in accordance with the "method of trial mixes" in ACI 318-99 and using the recommended mix proportioning/design practices in ACI 211.1-91 (re-approved 1997). Concrete placement and repair was in accordance with ACI 301-99. The replacement concrete had a specified compressive strength of 4000 psi at seven days, to expedite achieving the original shield building design strength of 4000 psi at twenty-eight days. The concrete strength for the approved mix design was maintained by the water/cement ratio limitations. The concrete slump was controlled using a high-range water-reducing admixture to enable increased workability without an increase in the established water/cement ratio of the mix. This ensured that the quality and strength of the hardened concrete was not adversely affected by the permitted concrete slump range (2"- 6" at point of placement). The cement utilized conformed to ASTM C150, Type I. To the extent possible, existing reinforcing steel removed during the creation of the opening was reinstalled during the restoration using mechanical (Cadweld) splices in accordance with Appendix 3B.9.0. New reinforcing bars, where needed, conformed to ASTM A615, Grade 60.

For the reactor pressure vessel head replacement project in 2011 and the Steam Generator Replacement Project in 2014, temporary construction openings were created in the Shield Building wall. The wall openings were restored by replacing the reinforcing steel and placing replacement concrete. The new reinforcing bars conformed to ASTM A615, Grade 60. As allowed by ACI 318-63, mechanical couplers were used in lieu of Cadweld® splices. The mechanical couplers meet the tensile strength requirements of ACI 318-63, and were installed in accordance with the requirements of ASME B&PV Code, Section III, Division 2, 1995 Edition with respect to qualification requirements of splice operators (CC-4333.4), tensile testing of splice samples, including sister splices (CC-4333.5.2), tensile testing frequency (CC-4333.5.3) and examination (CC-5320). The concrete mix used to close the openings was designed in accordance with the ACI 318 "method of trial mixes," using the recommended mix proportioning/design practices of ACI 211.11. Concrete replacement and repair was performed in accordance with ACI 301. Replacement concrete has a specified compressive strength of 4000 psi at 7 days to expedite achieving the required shield building design strength of 4000 psi at 28 days. Concrete slump was controlled using a high range water-reducing admixture, which enhanced workability of the concrete without affecting the water-to-cement ratio. The admixture

process ensured that the quality and strength of the hardened concrete was not adversely affected by the permitted slump range. The cement used conformed to ASTM C150, Type I.

A concrete repair was performed on the 2011 Shield Building temporary construction opening restoration using Five Star early-strength structural concrete that meets the 4000 psi requirement. Compressive strength testing of this repair was successfully performed in accordance with ASTM C109.

For the Laminar Cracking project concrete is removed and replaced to eliminate laminar cracking. This project also includes the option to remove and replace reinforcement. Replaced reinforcement will be spliced mechanical couplers in accordance with the criteria and description of splicing for the 2014 Steam Generator Replacement Project.

3.8.3 Computer Programs

A number of computer programs were used in the structural and seismic analysis of Seismic Class I structures. These programs are divided into two groups:

1. Computer programs used by Bechtel Corporation

2. Computer programs used by Chicago Bridge and Iron Company

Tables 3.8-3 and 3.8-4 provide the title, traceability and capability for these computer programs.

TABLE 3.8-2

NOT USED

TABLE 3.8-3

Computer Programs Used in Structural and Seismic Analyses by Bechtel Power Corp.

Program	Title	Document Traceability	Program Capabilities
CE 309	STRESS	PICC[1]	Generates flexibility matrix or stiffness matrix for structural models
CE 591	Spectra Analysis	PICC	Calculates and plots response spectra for earthquake accelerogram
CE 611	Time-History Response Analysis	PICC	Computes time-history response for structures subjected to earthquake
CE 617	Modes and Frequencies Extraction	BPC[2]	Extracts modes and frequencies from stiffness or flexibility matrix and diagonal mass matrix
CE 641	Response Spectrum Analysis	BPC	Spectral response analysis of simple cantilever structures
CE 655	Stress Dynamic Analysis	PICC	Extracts flexibility or stiffness matrix from CE-209, “STRESS”
CE 779	SAP	PICC	Perform linear elastic analyses of three-dimensional structural systems
CE 029	GT-STRUDL	BPC[3]	Performs finite-element structural analysis
CE 498	ANSYS	BPC[3]	Performs finite-element analysis of three-dimensional structural systems

[1] Pacific International Computing Corp., San Francisco, California

[2] Bechtel Power Corp., San Francisco, California

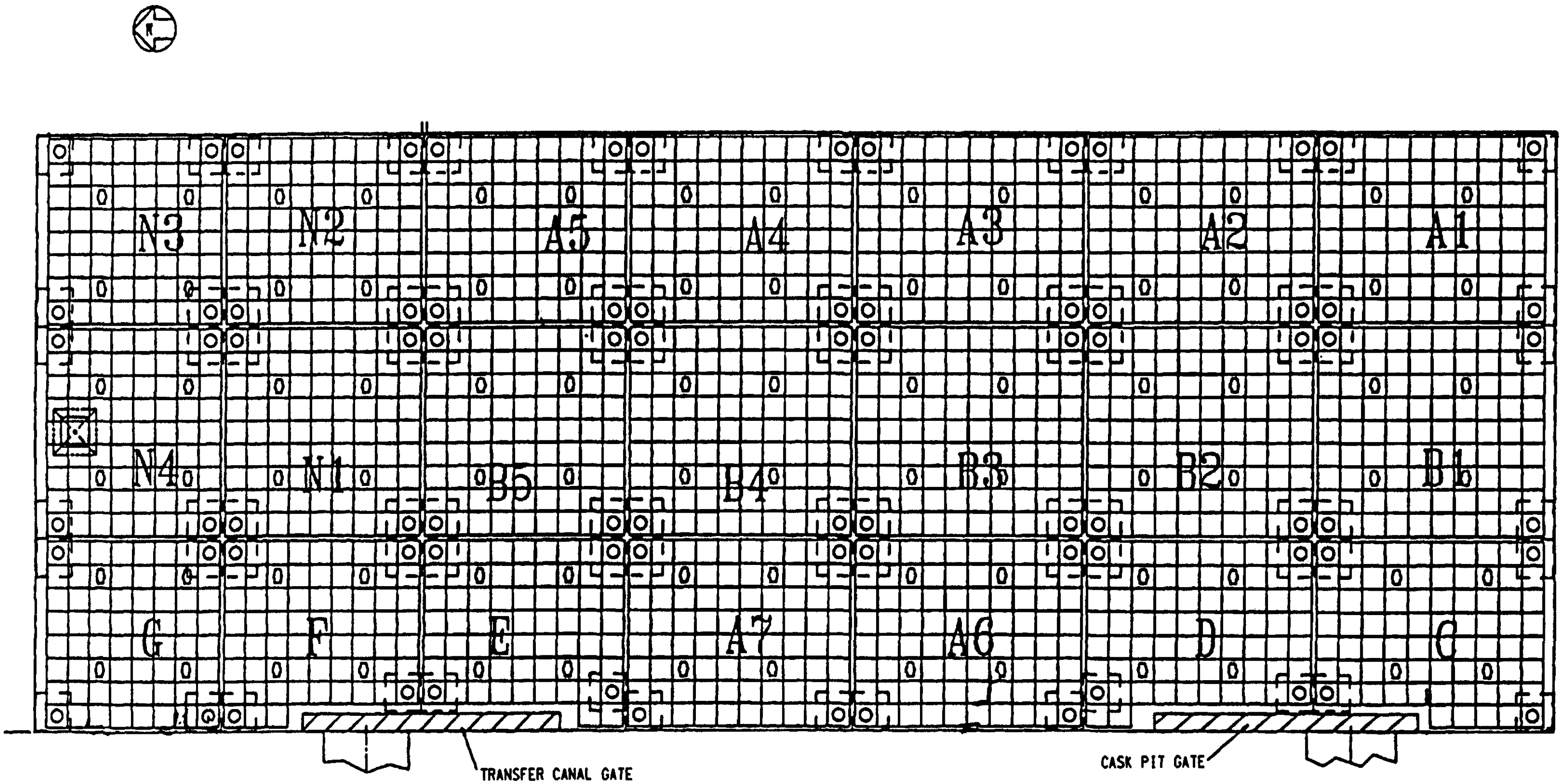
[3] Bechtel Power Corp., Frederick, Maryland

TABLE 3.8-4

Computer Programs Used in Structural and Seismic Analyses by Chicago Bridge and Iron Company

Program	Title	Document Traceability	Program Capabilities
405	Ring Analysis	CB&I[1]	Analyzes ring with constant moment of inertia and modulus of elasticity
772	Nozzle Reinforcing	CB&I	Checks adequacy of plates used for nozzle reinforcing
7-8IN	Shell stresses at Discontinuities	CB&I	Analyzes shell stresses at discontinuities
1017	Modal Analysis of Structures	CB&I	Calculates mass and stiffness matrices, undamped natural periods, and maximum modal responses
1027	Stress Intensities	CB&I	Analyzes stress intensities in a sphere or cylinder around an externally loaded round or square attachment
1036M	Stress Intensities in Jumbo Insert Plates	CB&I	Analyzes stress intensities in jumbo insert plates with multiple penetrations
1044	Seismic Analysis of Vessel Appendages	CB&I	Evaluates the maximum elastic differential accelerations between an appendage and a vessel due to excitations of the vessel

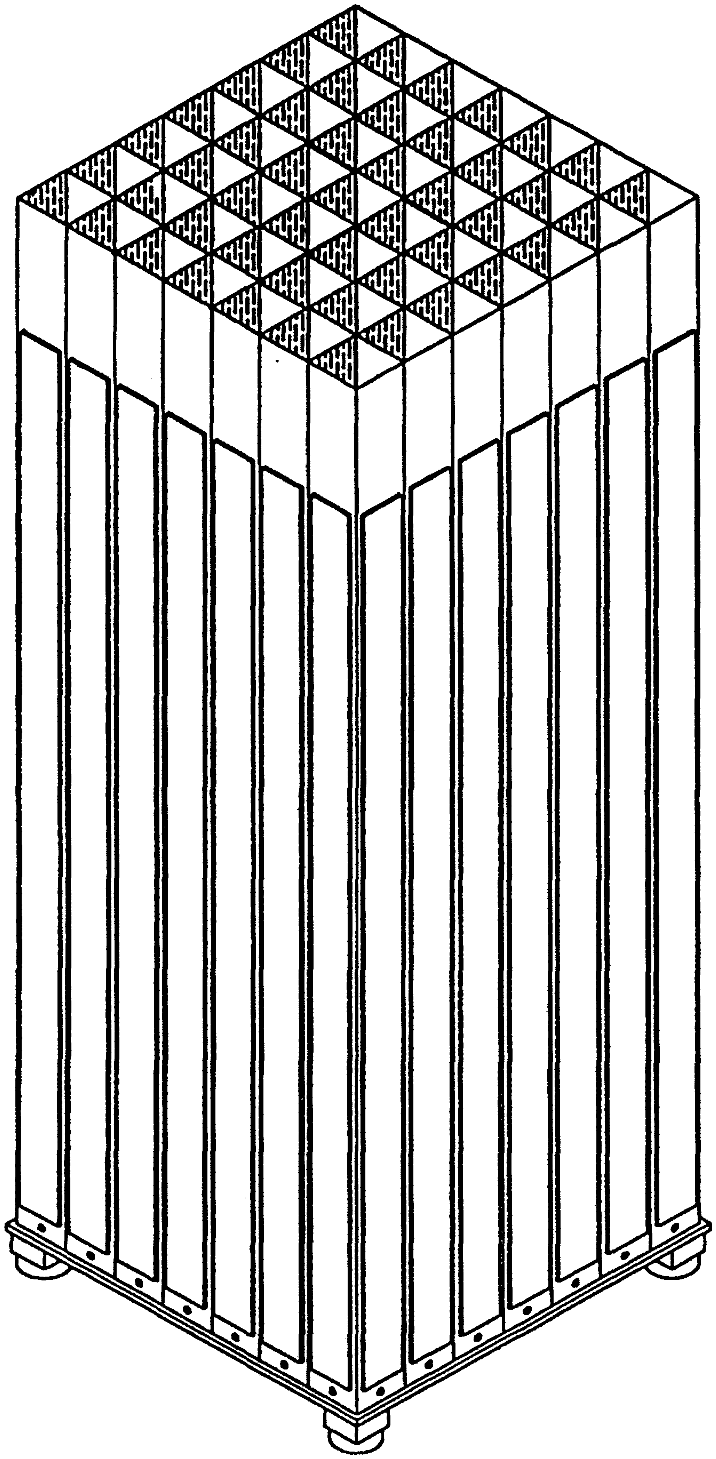
[1] Chicago Bridge & Iron Co., Memphis, Tennessee



POOL LAYOUT FOR DAVIS-BESSE FUEL POOL

DAVIS-BESSE NUCLEAR POWER STATION
SPENT FUEL STORAGE RACK
ARANGEMENT PLAN
FIGURE 3.8-1

REVISION 23
NOVEMBER 2002



DAVIS-BESSE NUCLEAR POWER STATION

TYPICAL FUEL STORAGE RACK

FIGURE 3.8-2

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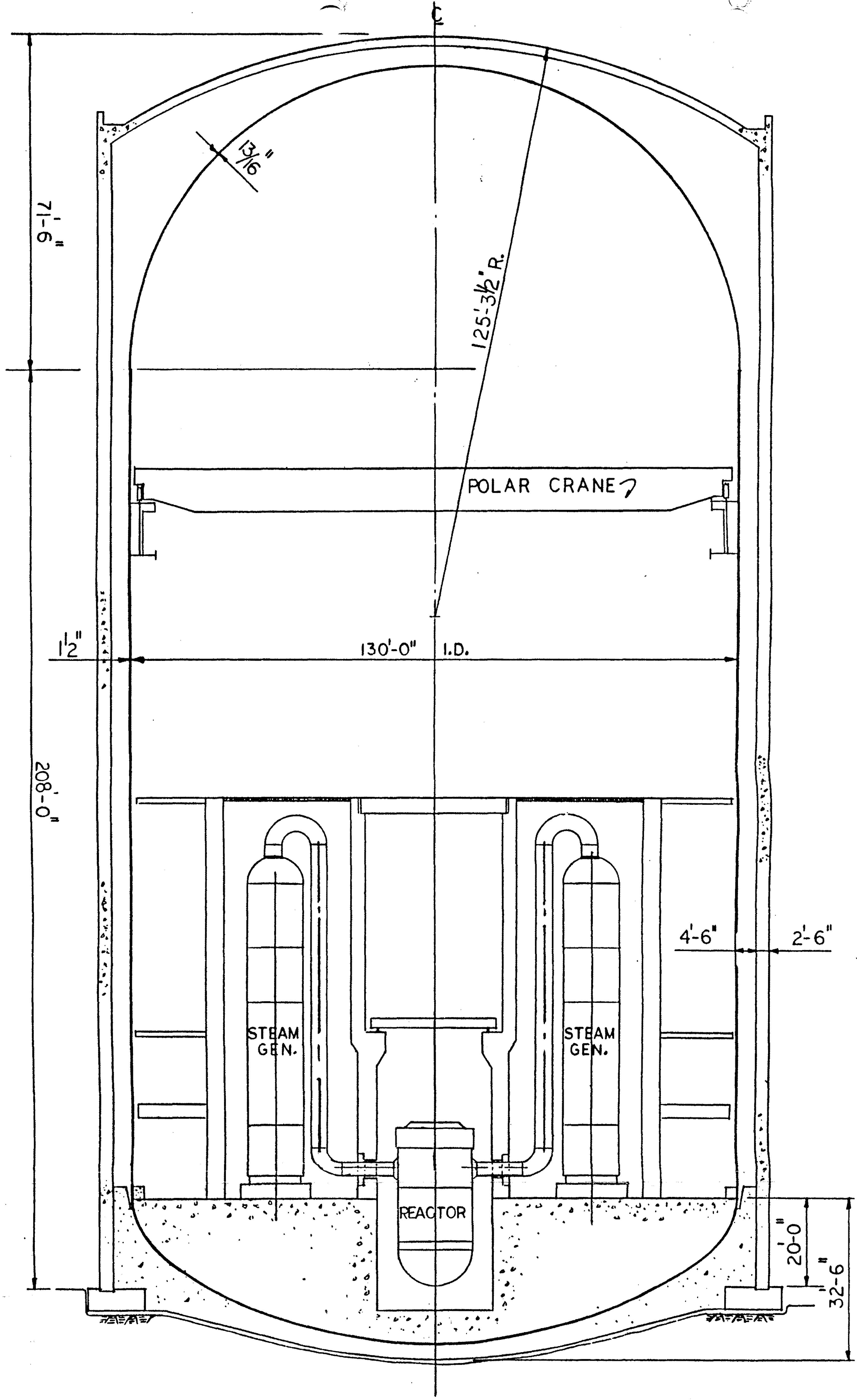
11-04-02 DFN=/G/USAR/UFIG382.DGN/CIT

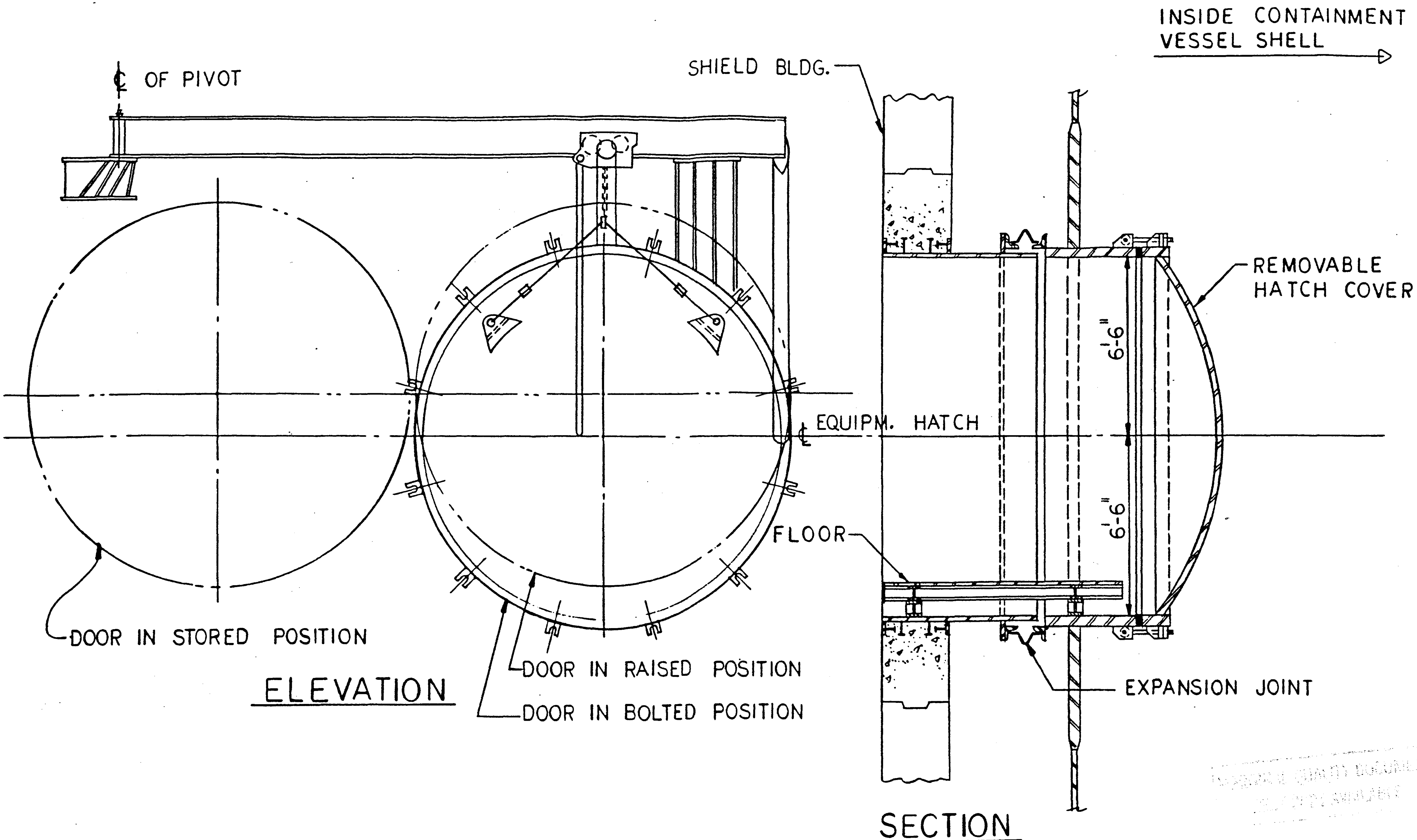


TYPICAL SECTION

DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT STRUCTURE - CONTAINMENT VESSEL
AND SHIELD BUILDING TYPICAL SECTION

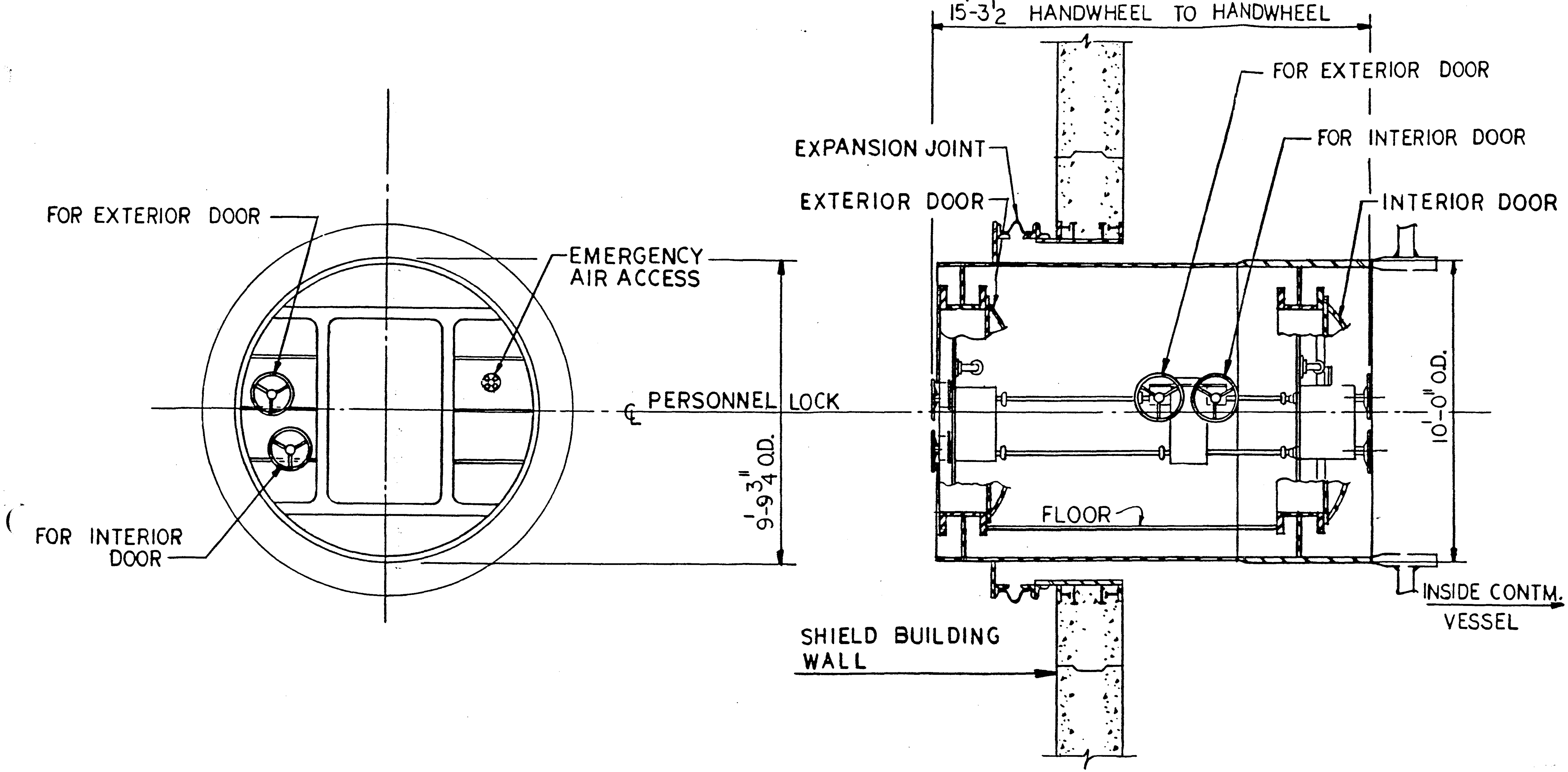
FIGURE 3.8-3

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT STRUCTURE EQUIPMENT HATCH DETAIL
FIGURE 3.8-4

REVISION 0
JULY 1982

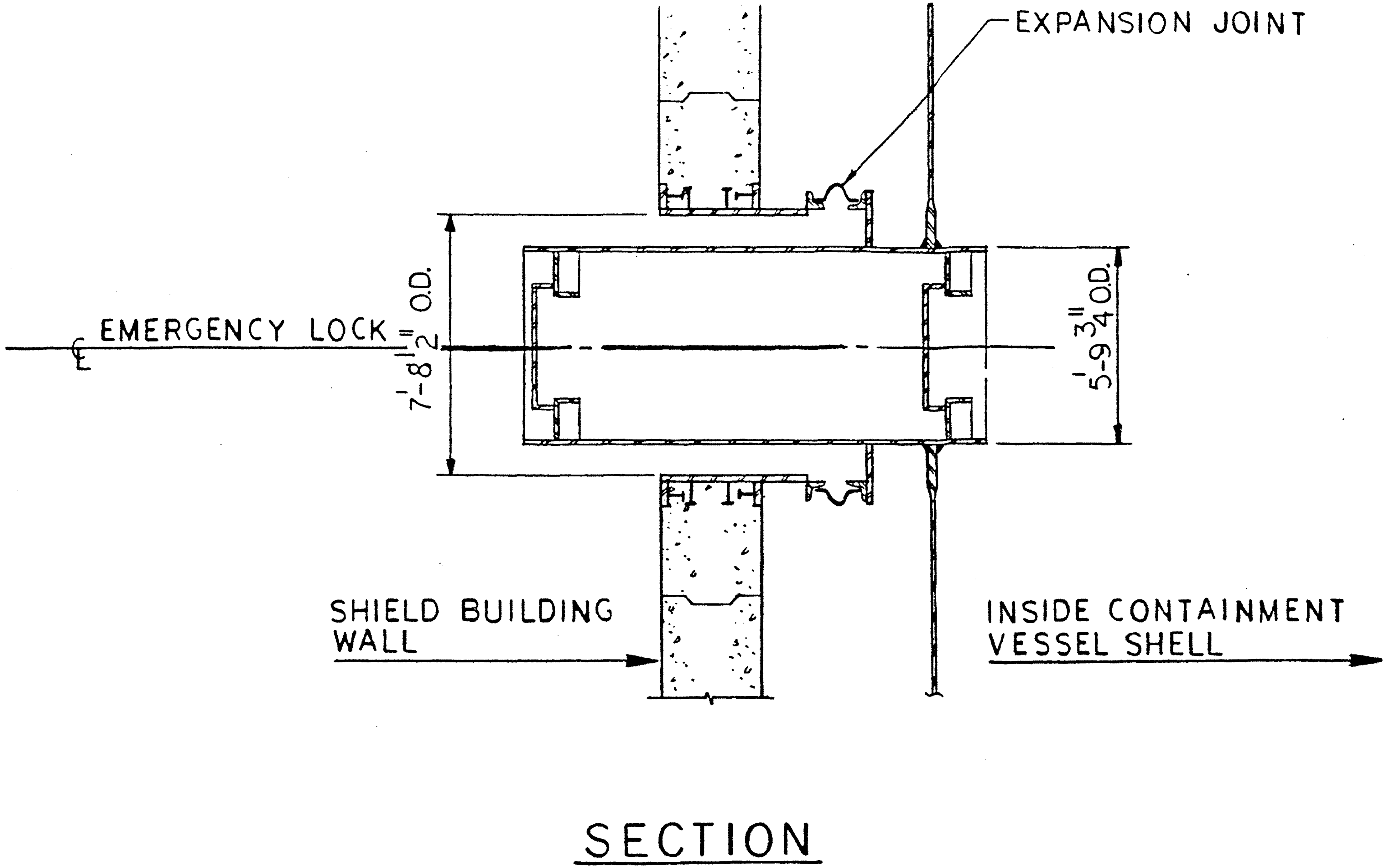


ELEVATION

SECTION

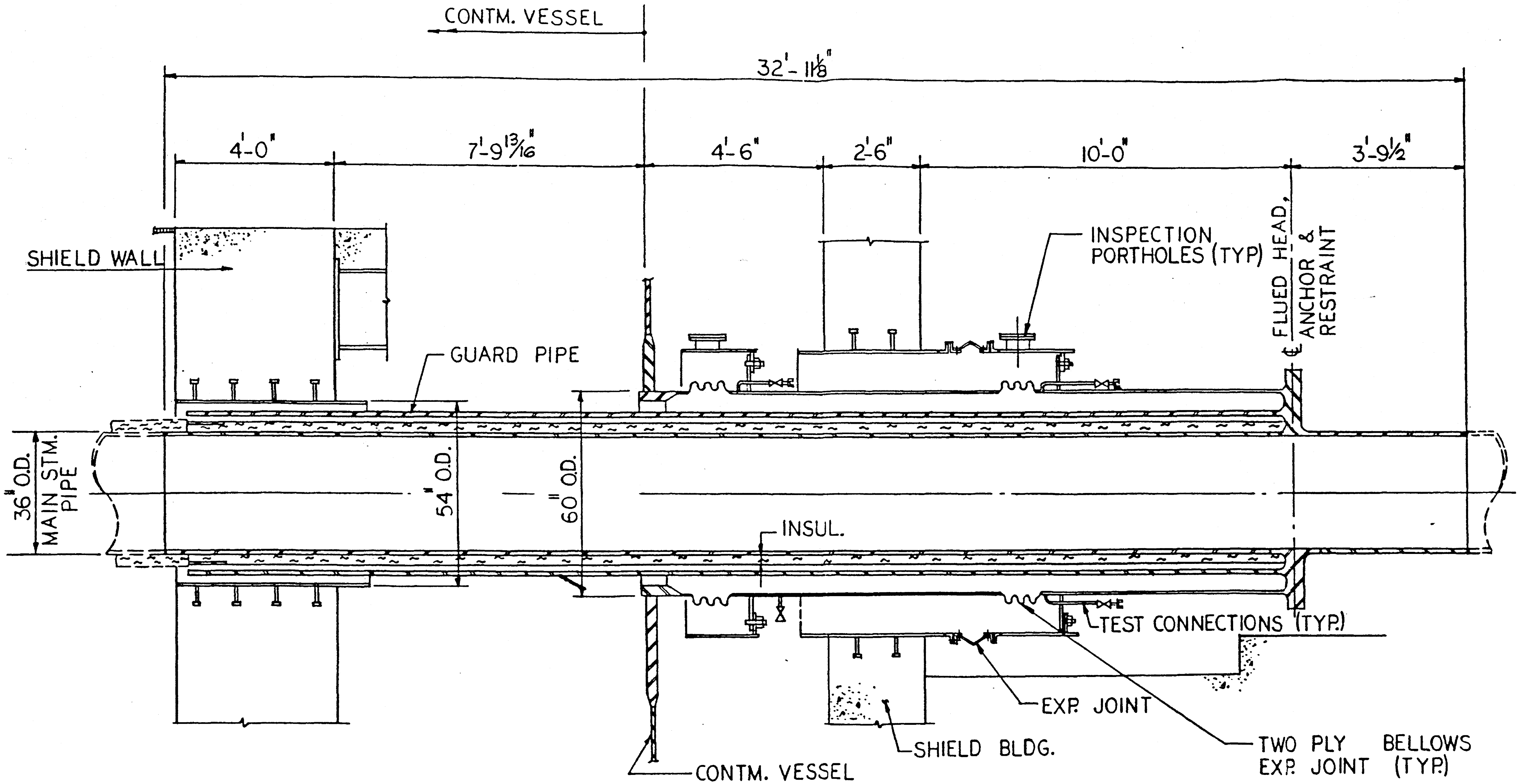
DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT STRUCTURE PERSONNEL LOCK DETAIL
FIGURE 3.8-5

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT STRUCTURE EMERGENCY LOCK DETAIL
FIGURE 3.8-6

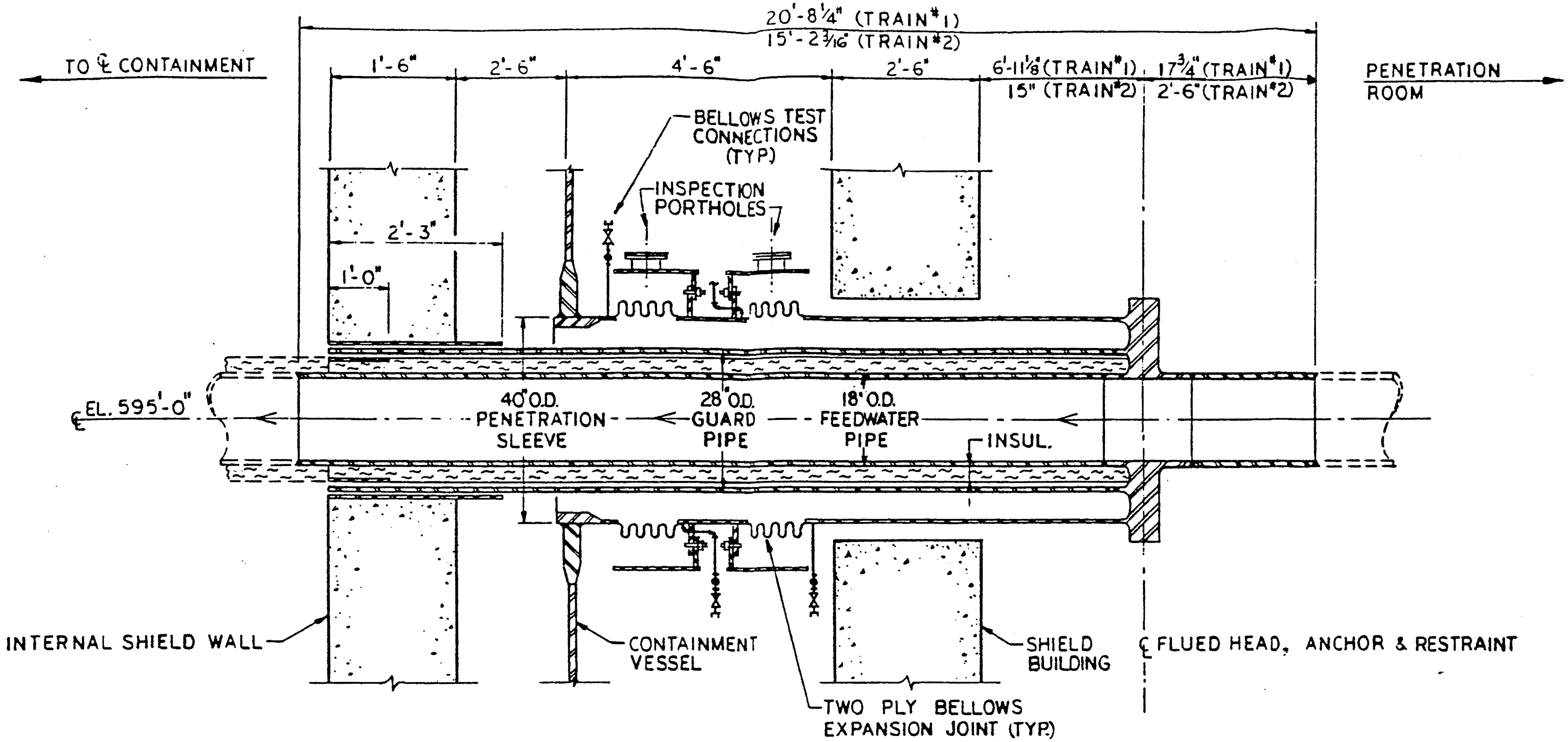
REVISION 0
JULY 1982



MAIN STEAM LINE CONTAINMENT PENETRATION
PENETRATION # 39 & 40

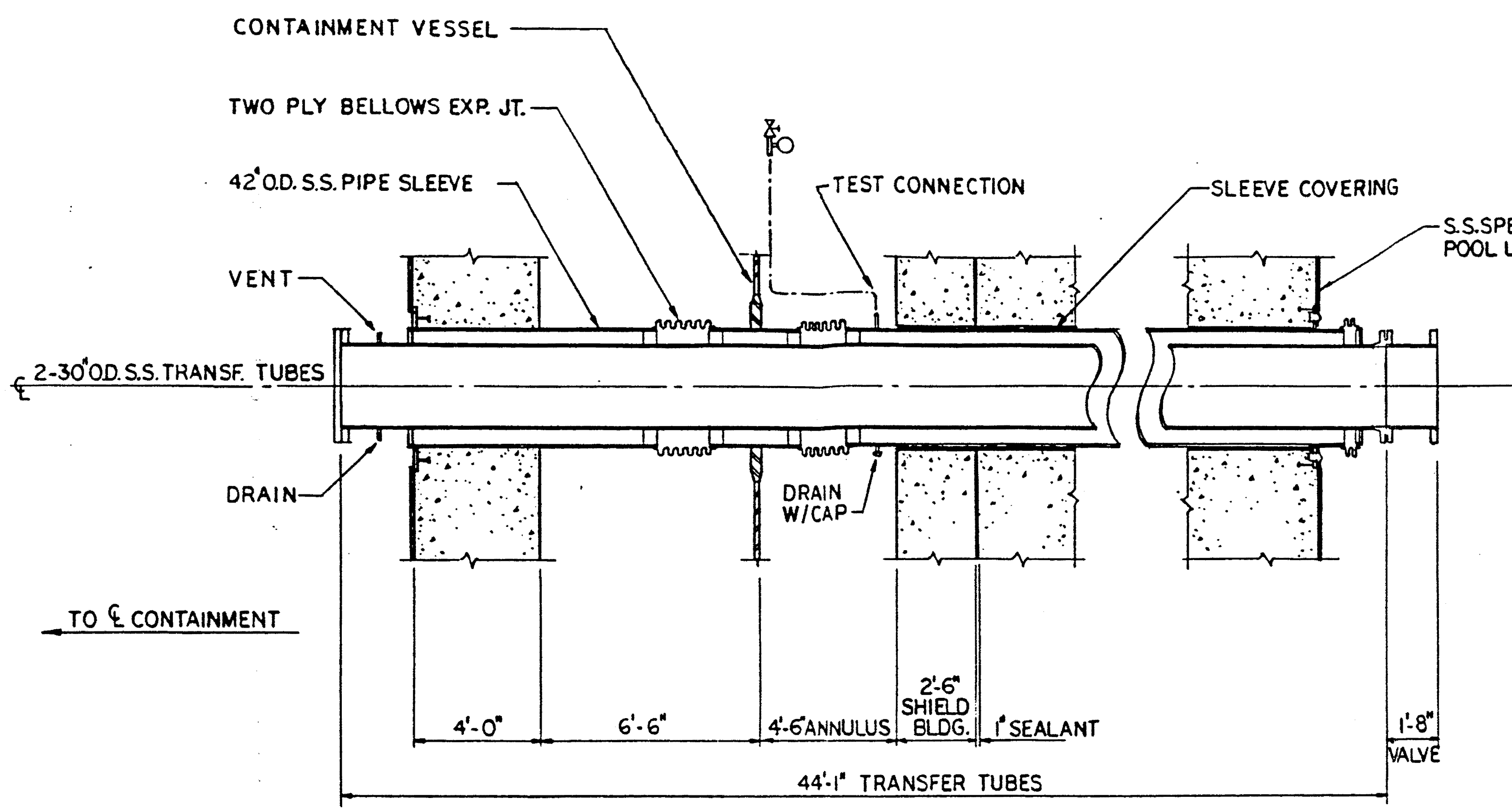
DAVIS-BESSE NUCLEAR POWER STATION
MAIN STEAM LINE CONTAINMENT VESSEL
PENETRATION DETAILS
FIGURE 3.8-7

REVISION 0
JULY 1982



PEN. 38, TRAIN #1
PEN. 37, TRAIN #2

DAVIS-BESSE NUCLEAR POWER STATION
MAIN FEEDWATER LINE CONTAINMENT VESSEL
PENETRATION DETAILS
FIGURE 3.8-8

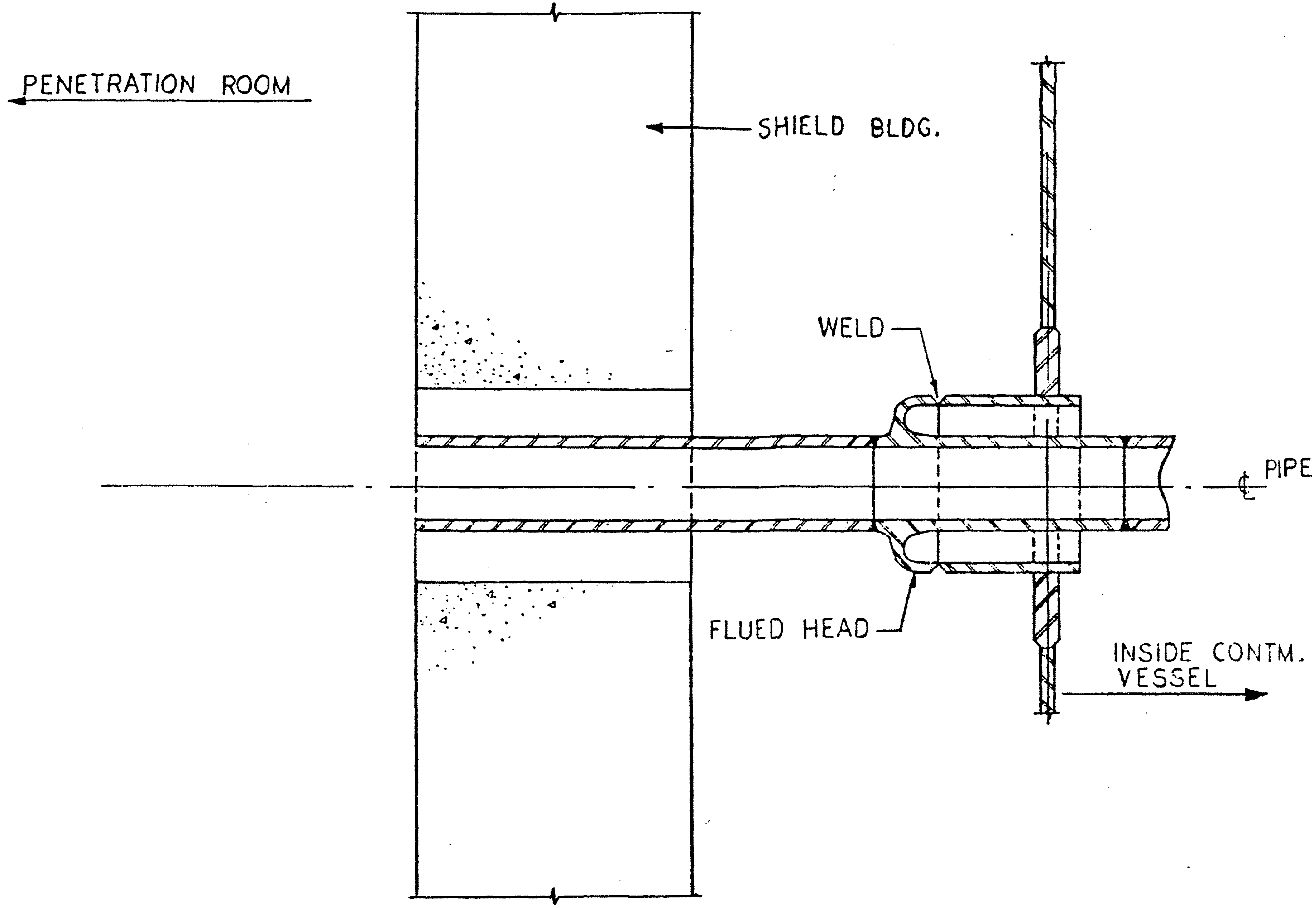


FUEL TRANSFER TUBES - CONTAINMENT PENETRATIONS

PEN. 23 & 24

DAVIS-BESSE NUCLEAR POWER STATION
FUEL TRANSFER TUBES CONTAINMENT VESSEL
PENETRATION DETAILS
FIGURE 3.8-9

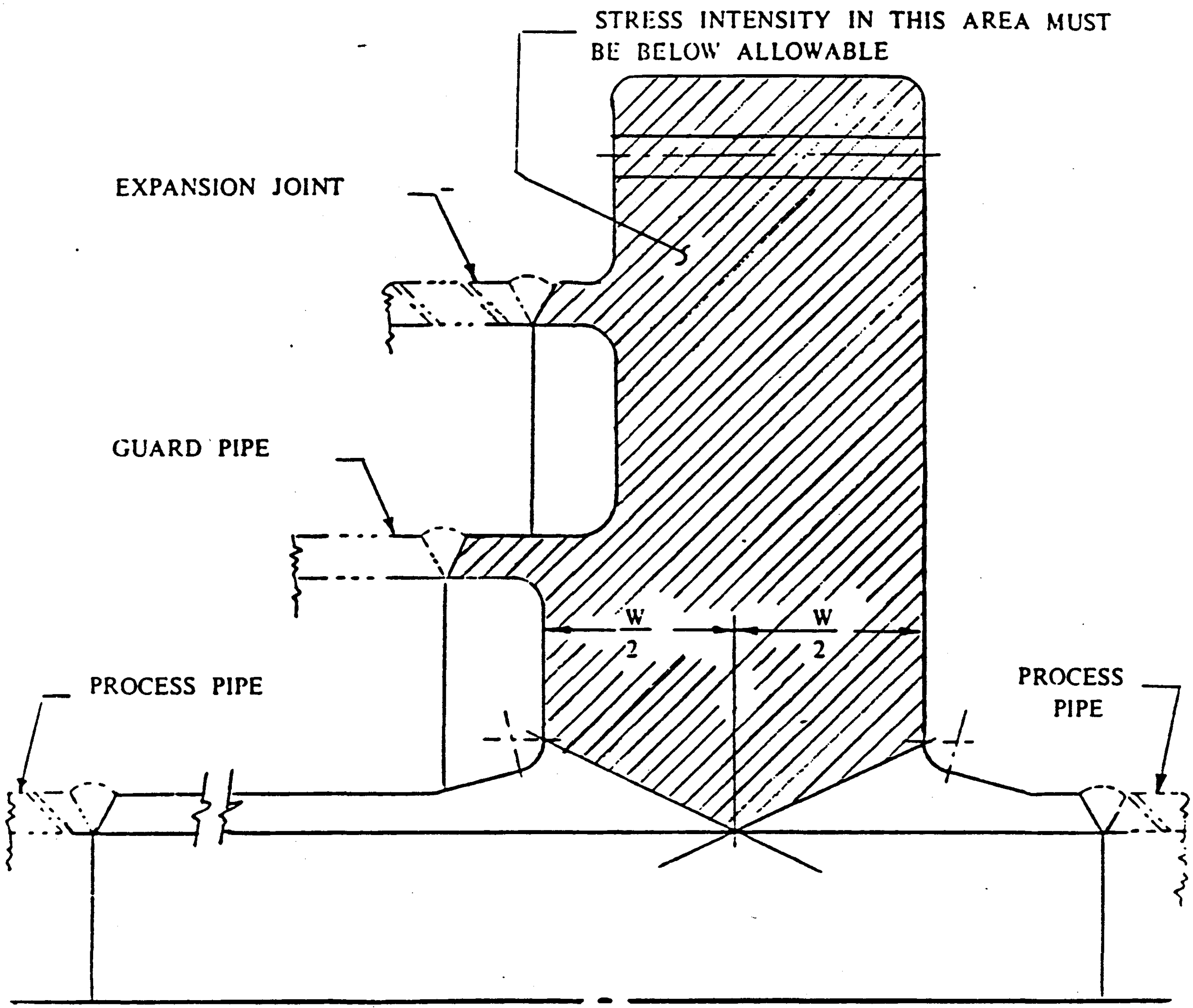
REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT VESSEL - FLUED HEAD PENETRATION DETAILS

FIGURE 3.8-10

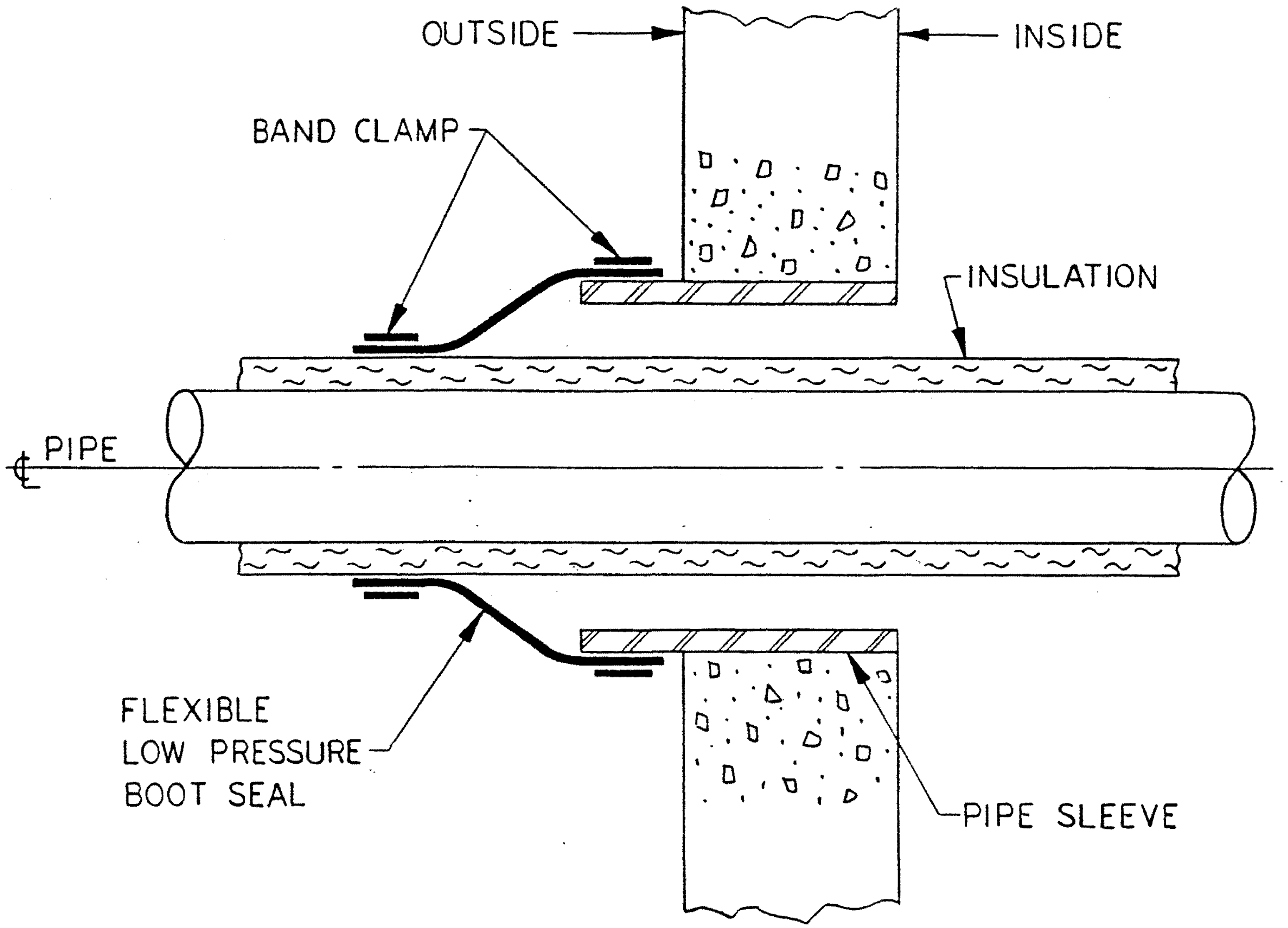
REVISION 0
JULY 1982



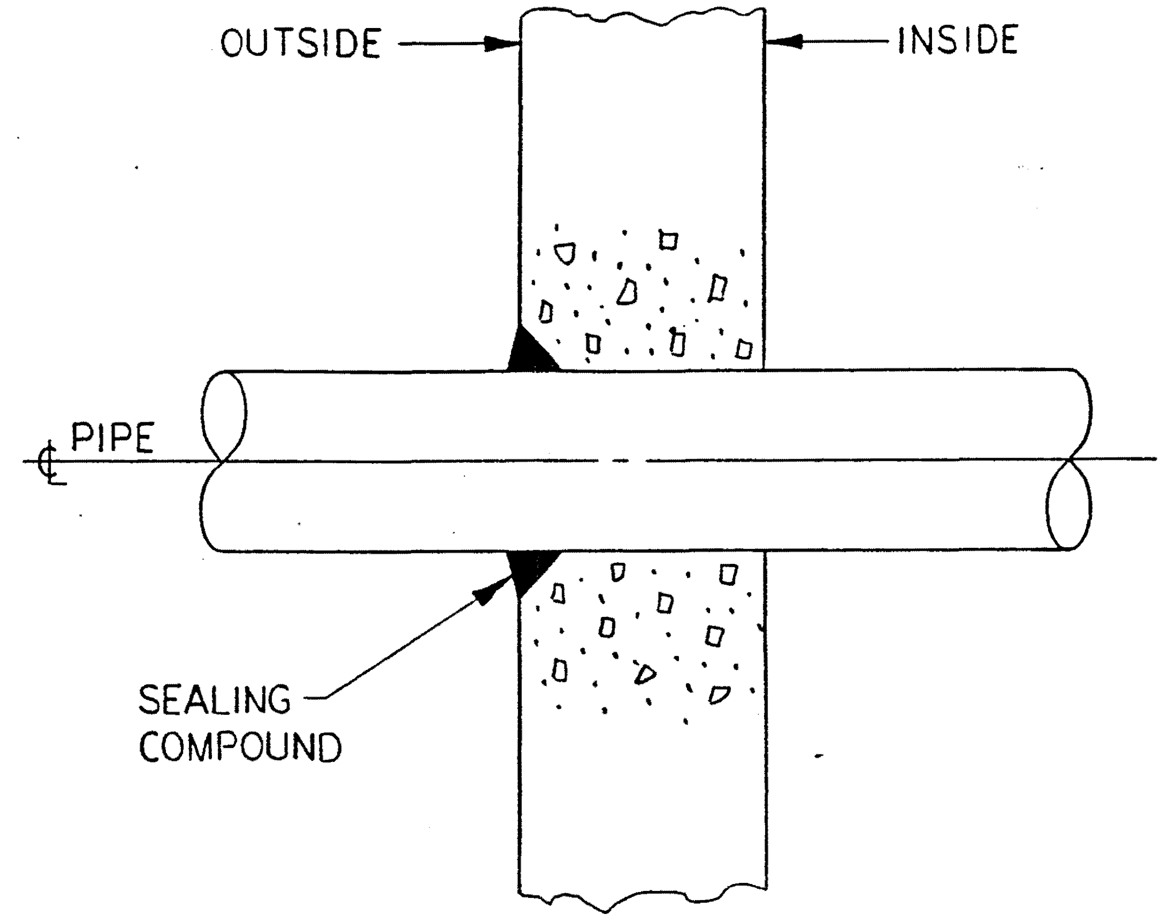
HALF SECTION - TYPICAL

DAVIS-BESSE NUCLEAR POWER STATION
MAIN STEAM AND
FEEDWATER FLUED HEADS
FIGURE 3.8-10A

REVISION 0
JULY 1982



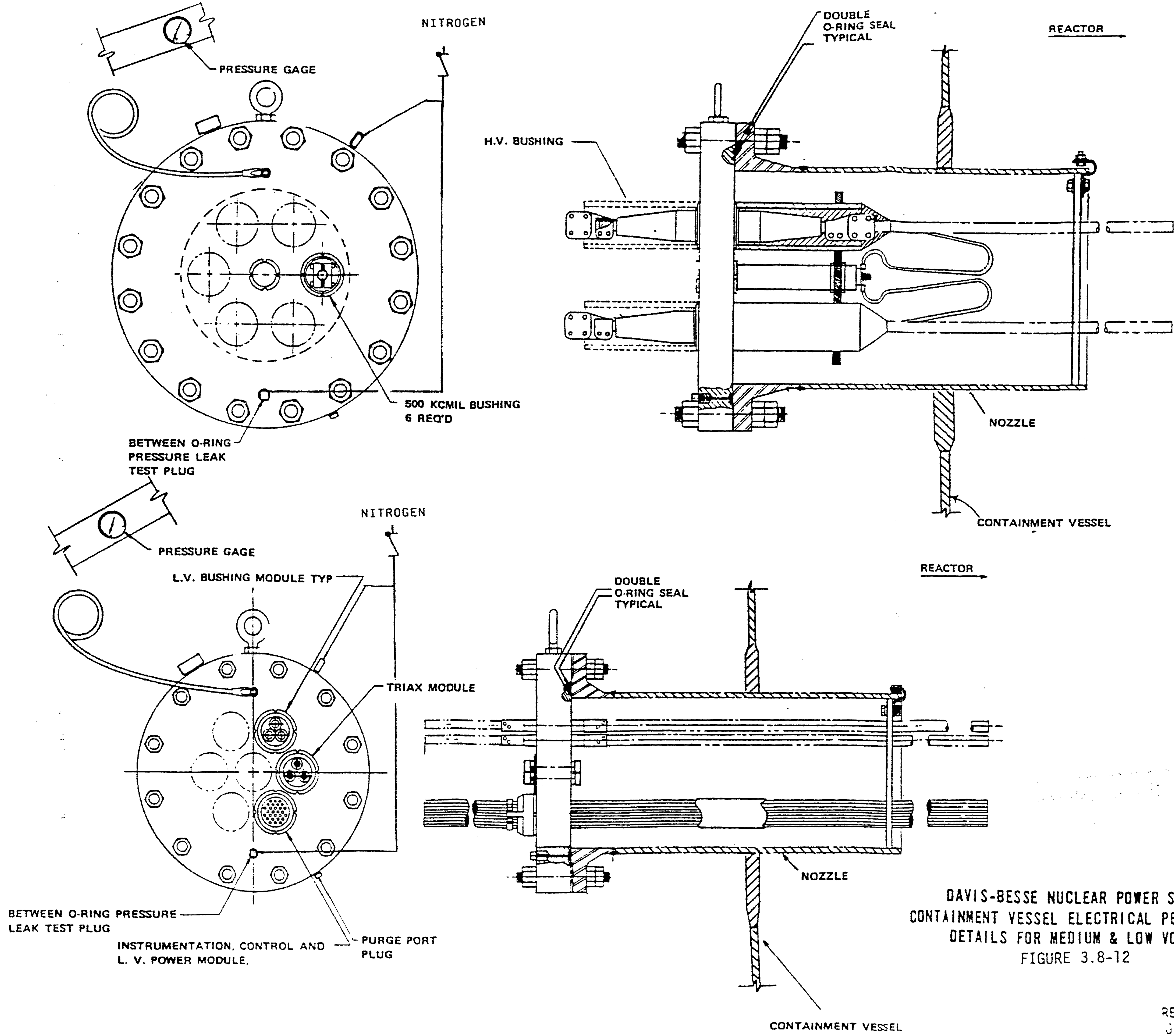
HOT PIPE
PENETRATION



COLD PIPE
PENETRATION

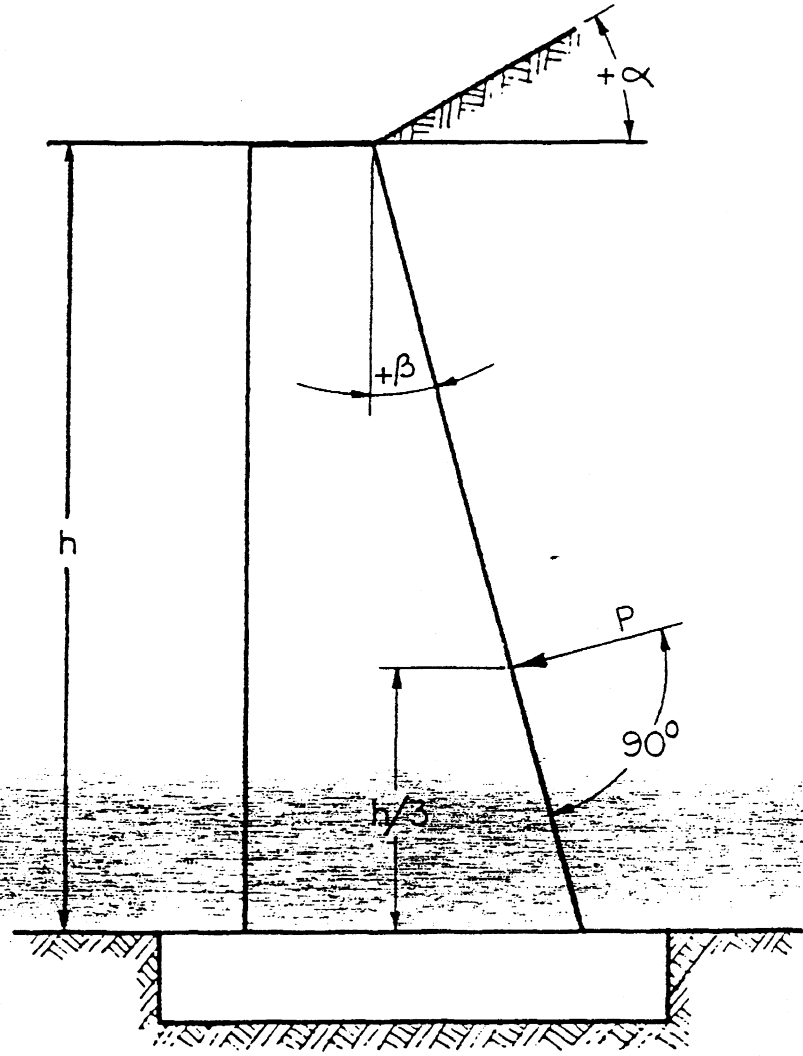
DAVIS-BESSE NUCLEAR POWER STATION
PIPING PENETRATIONS
THROUGH NEGATIVE PRESSURE
AREA BOUNDARY
FIGURE 3.8-11

REVISION 0
JULY 1982



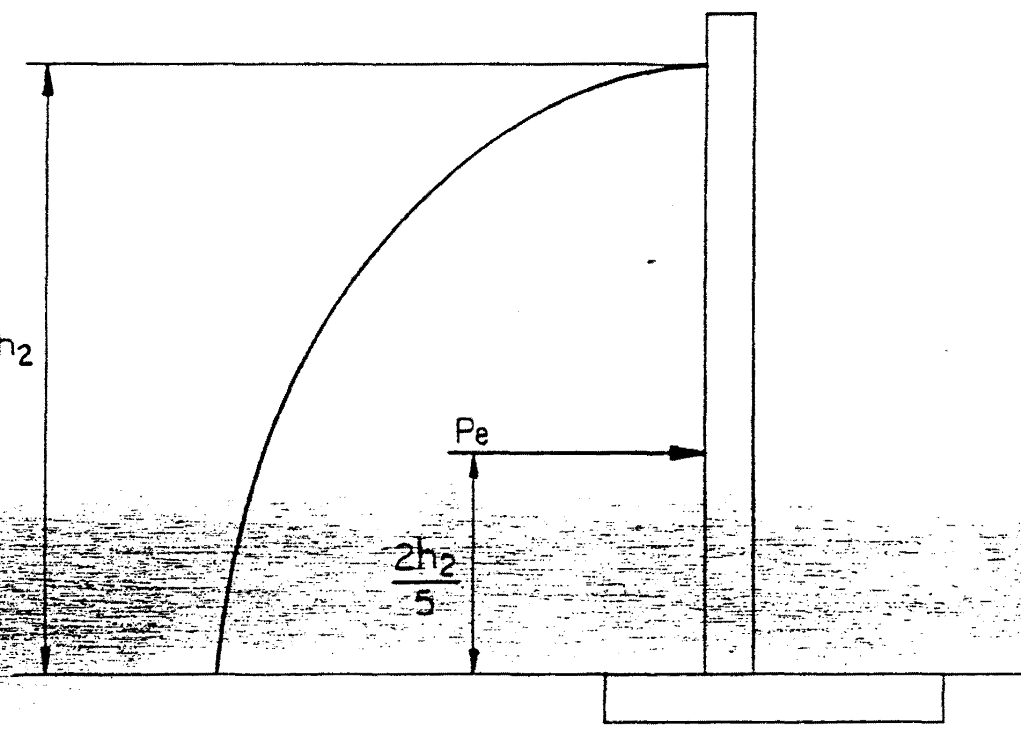
DAVIS-BESSE NUCLEAR POWER STATION
CONTAINMENT VESSEL ELECTRICAL PENETRATION
DETAILS FOR MEDIUM & LOW VOLTAGE
FIGURE 3.8-12

REVISION 12
JULY 1990

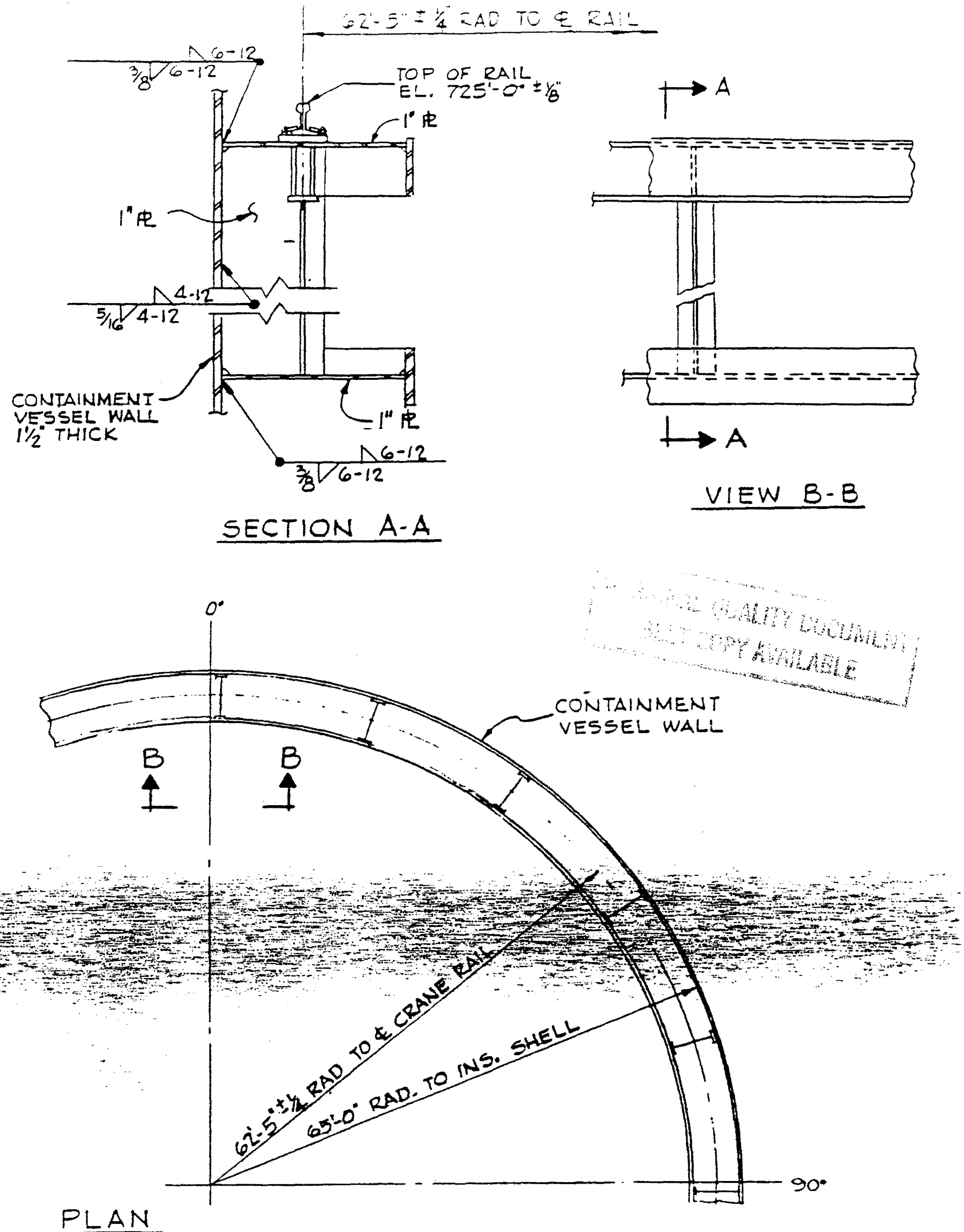


DAVIS-BESSE NUCLEAR POWER STATION
FOUNDATION WALL
FIGURE 3.8-13

REVISION 0
JULY 1982

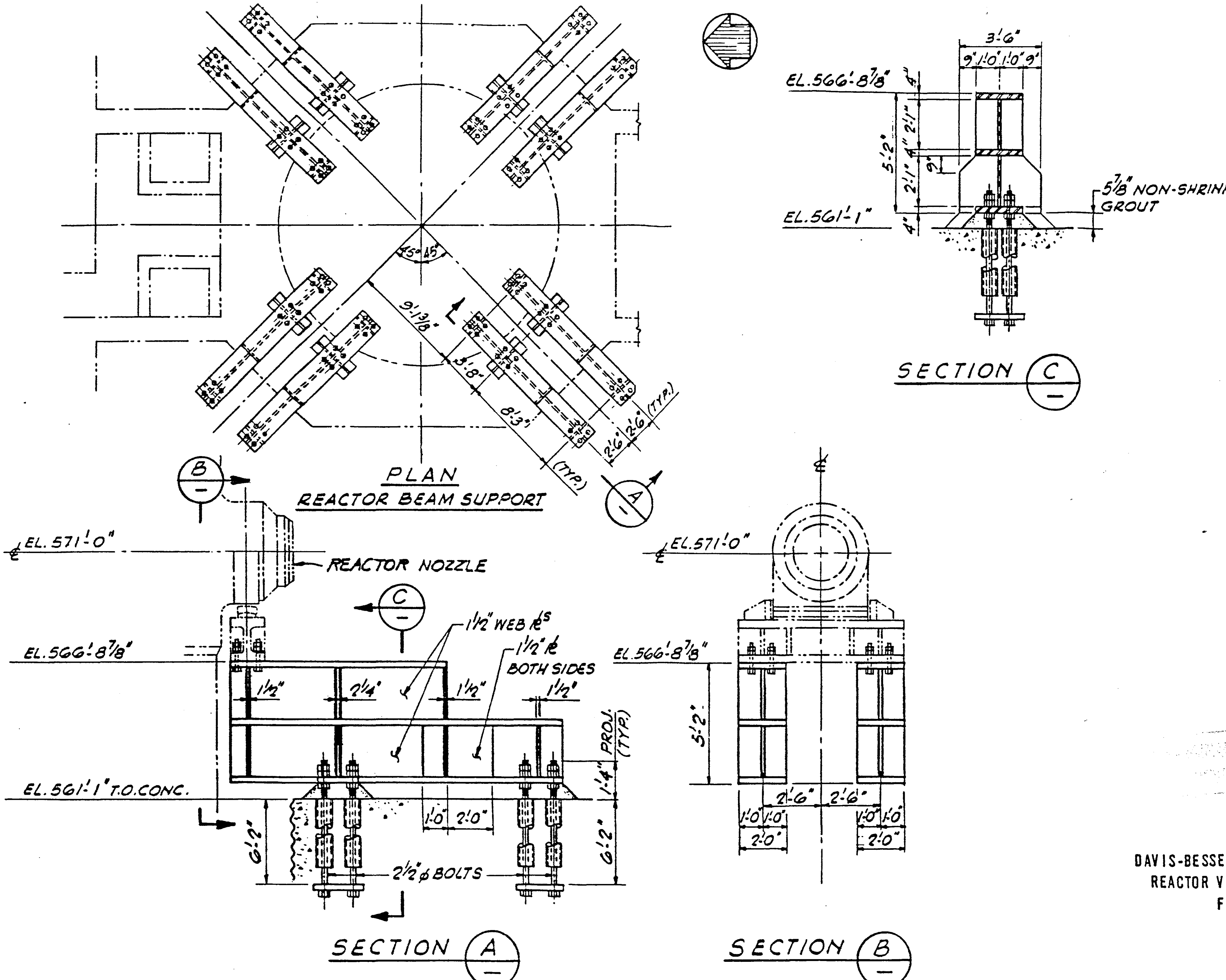


DAVIS-BESSE NUCLEAR POWER STATION
WESTERGAARD PARABOLA ON PRESSURE
FIGURE 3.8-14

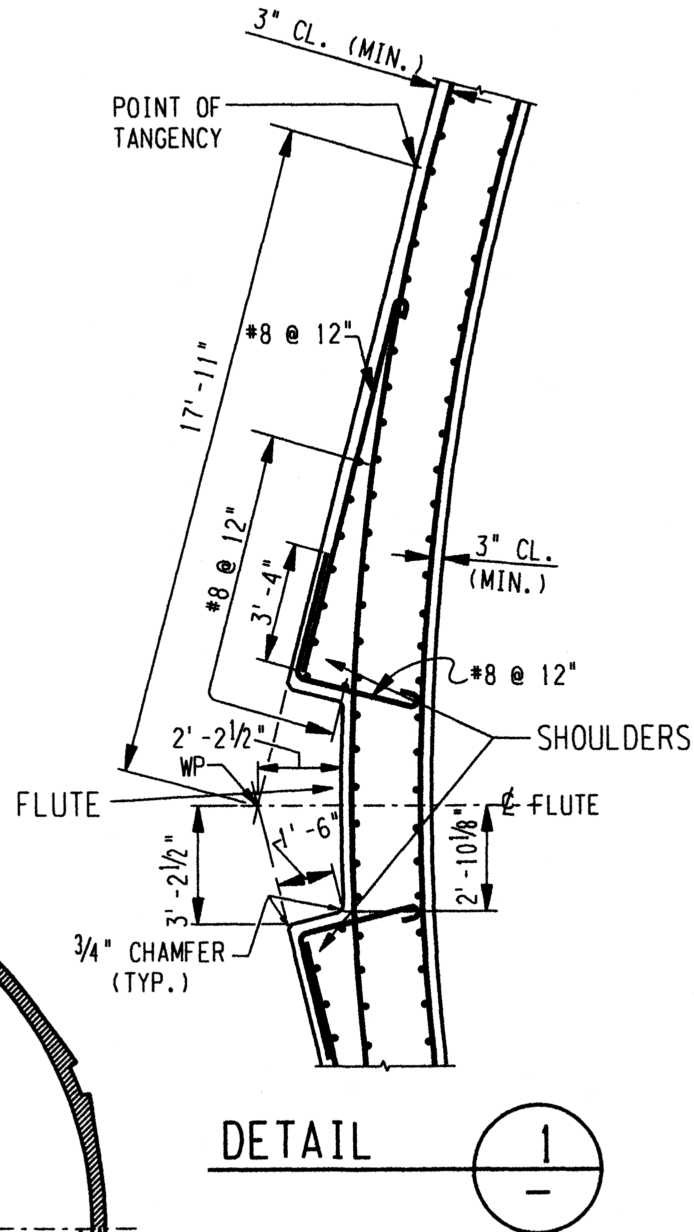


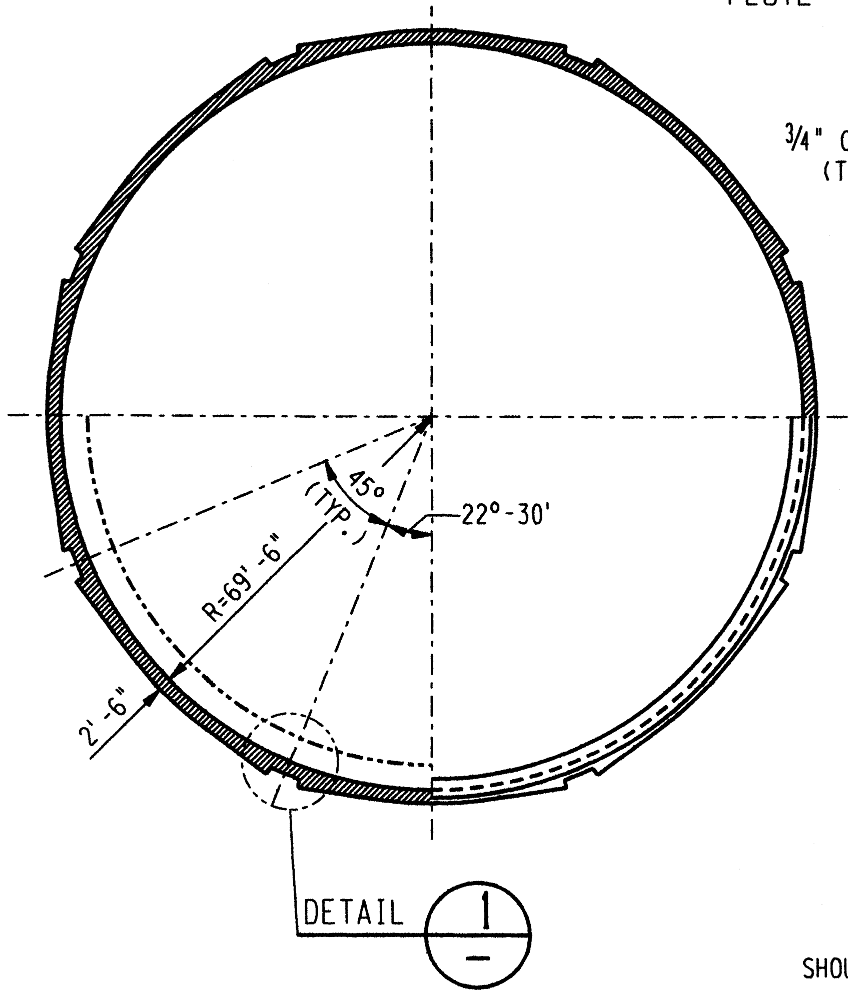
DAVIS-BESSE NUCLEAR POWER STATION
POLAR CRANE GIRDER DETAIL
FIGURE 3.8-15

REVISION
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
REACTOR VESSEL BEAM SUPPORT
FIGURE 3.8-16





DAVIS-BESSE NUCLEAR POWER STATION
SHIELD BUILDING ARCHITECTURAL FLUTE
SHOULDER AREA DETAILS (BASED ON DRAWING C-110)
FIGURE 3.8-17
REVISION 30
OCTOBER 2014

DB 02-08-14 DFN: G/USAR/UFIG3817.DGN

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Dynamic System Analysis and Testing

3.9.1.1 Vibration and Expansion Preoperational Test Program

The vibration operational test program, required by Paragraphs NB-3622.3, NC-3622 and ND-3611 of the ASME B and PV Code, Section III, was conducted simultaneously with the startup testing program. Piping dynamics effect testing was conducted on the following systems:

a. High Pressure Injection System

b. Low Pressure Injection System

c. Pressurizer Relief (Electromatic valve only)

d. Atmospheric Relief for Main Steam

e. Monitor Turbine Trips (Main steam piping from steam generator to first anchor after main steam isolation valve)

Measurements were taken at selected locations which provide sufficient data to ensure dynamic effects are within acceptable levels. The data at the selected locations was gathered during hot functional testing, safety system actuation tests, and power ascension tests; it was a result of valve openings and closures, starts and pump trips and turbine trip testing.

The following acceptance criteria were used to verify that piping displacements are within acceptable levels.

a. The displacement does not exceed the calculated displacement for the applicable condition.

b. If no calculated displacement has been determined for a specified test, the measured displacement is used in an analysis to verify that allowable stresses have not been exceeded.

If the measured displacements exceed a predetermined displacement, or if the measured displacement is verified by subsequent analysis to result in stresses above allowables, restraints were installed or relocated to reduce stresses to acceptable levels.

A piping preoperational expansion and restraint test (Ref. Initial Tests, Chapter 14) was conducted on the following systems:

Main Steam System
Main Feedwater System
Auxiliary Feedwater System
High Pressure Injection System (from the Reactor Coolant System loop to the first anchor)
Decay Beat Removal System
Reactor Coolant Makeup System (from the Containment Vessel to the Reactor Coolant System Loop)

Measurements were taken at selected spring hanger and snubber locations. In addition, pipe movements are measured at points of high stress and significant movement in the piping system.

The following acceptance criteria were used to verify the correct design and installation of the piping system:

a. There is no evidence of blocking for the displacement of any system piping or components caused by thermal expansion of the system.

b. The piping thermal deflection does not exceed the calculated thermal deflection, and the piping moves in the proper direction.

c. Piping hanger loading does not exceed the calculated hot hanger setpoints.

d. For Nuclear Class I piping the preoperational test program verifies that the loads and stresses on the piping does not exceed those predicted by analysis. Since these loads and stresses are the input for the fatigue analysis, the cumulative usage factor for fatigue does not exceed the code allowable.

If the measured deflections did not meet the criteria listed above, the flexibility analyses were reverified. If the code allowable stresses are exceeded, restraints were installed or relocated to reduce the stresses to acceptable levels.

Subsequent to plant startup, industry experience found that higher than expected stresses occurred in other facilities' pressurizer surge lines due to thermal stratification and thermal striping. NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification" required re-evaluation of the stresses and cumulative usage factors for the Pressurizer Surge Line. Topical Report, BAW 2127 (Reference 78), "Final Submittal for Nuclear Regulatory Commission Bulletin 88-11 'Pressurizer Surge Line Thermal Stratification'" with Supplements 2 (Reference 81) and 3 (Reference 82) describe the results of the revised evaluation.

3.9.1.2 Seismic Design of Class I Mechanical Equipment

The seismic design and qualification of Class I mechanical equipment is described in Section 3.7.

3.9.1.3 Dynamic Analysis Forcing Functions

Dynamic analysis of the reactor internals during normal operating and anticipated transients is based on the velocity distribution within the vessel. The loading conditions include pump startup and coastdown, and one, two, three, or four pumps in operation.

Cross flow on cylindrical components causes periodic vortex shedding forces, but the designs are such that the forcing frequency is much less than the natural frequency of the structure. Flow-induced vibrations can also be induced by parallel flow, and methods exist for predicting the amplitudes of motion and thus, the cyclic stresses. The results of analytical work to confirm the adequacy of internals for normal reactor operation are contained in B&W Topical Reports BAW-10051, Rev.1, BAW-10037, Rev. 2, BAW-10038, Rev.1 and BAW-10039, Supp.1 Rev.1.

Additional confirmation of the structural adequacy for the reactor internals and the analytical procedures were obtained from an extensive vibration measurement program on an earlier design, very similar to that used at Davis-Besse. This measurement program is described in B&W Topicals BAW 10038 Rev.1 and BAW 10039. Thus, no instrumentation system to record flow-induced responses is deemed necessary for Davis-Besse. A visual inspection of the internals following hot functional testing was made in accordance with Safety Guide No. 20.

3.9.1.4 Preoperational Test Program

There are similarities between the dynamic analyses performed on internals under normal operating and LOCA loadings. Some of the data from in-air vibration testing and vibration measurements during hot functional testing on an earlier design was used to confirm predominant resonant frequencies calculated for the thermal shield, core support shield, and core barrel cylinder. Knowledge of these frequencies permits more accurate calculation of the dynamic load factors used in analysis of the shells under LOCA pressure differentials. At present, the LOCA analysis, as reported in Appendix 4A includes enough conservatism to account for the uncertainties in the fundamental ring mode frequencies of the core support structure.

3.9.1.5 LOCA Structural Response

A detailed description of the structural response under LOCA loadings that assures the adequacy of the internals is given in Appendix 4A.

For loss of coolant conditions, the Reactor Coolant System was analyzed on a component basis. The dynamic response of major components such as the reactor, steam generator, pressurizer, and reactor coolant pumps was calculated using discrete mass planar models.

The computer program used determines the behavior of planar complex structures which may be impacting each other while being subjected to dynamic loads. Each major component is represented by discrete masses, connected together by elastic beams, linear springs and/or gapped nonlinear springs. Uniform or non-uniform damping may be used with the elastic beams. The non-linear springs may have viscous, hysteretic, or coulomb damping.

Due to leak-before-break (LBB), large break LOCA loads are not considered in the structural design of the replacement OTSG. LOCA and secondary side break cases that are considered in the structural design are: surge line break and main steam line break. These breaks are found to envelope all breaks not qualified by LBB.

Output of the program consists of time histories of the deflection, velocities, and accelerations of the lumped masses, as well as internal forces and moments in the beam and spring elements. It also provides the solution of natural frequencies and mode shapes of an elastic system. As an example, a model of the steam generator used in calculating the dynamic loads under LOCA conditions is shown in Figure 3.9-1. Lumped mass models of this type were used for the reactor, pressurizer, and reactor coolant pump.

Shown in Figure 3.9-2 is a typical time history plot of the loading on the steam generator upper support under LOCA conditions. Loads on other points of interest are calculated by the code and may be obtained as time histories.

The applied thrust due to a pipe rupture is obtained in the form of a thrust time curve similar to Figure 3.9-3. Double-ended ruptures and pipe splits are taken at the points causing the maximum possible loadings on the component being analyzed.

The effects of asymmetric LOCA loads on the components and equipment within the reactor vessel (RV) sub-compartment have been analyzed by the B&W Owners Group (BWOG) for the B&W 177-Fuel Assembly plants. These components are as follows: RV, service support structure (SSS), CRDMs, and RV internals. The RV internals consist of the following sub-assemblies: core support cylinder, lower grid assembly, flow distributor assembly, core barrel assembly, thermal shield, plenum assembly, and fuel assemblies. The results of the BWOG analyses and plant specific evaluations are contained in Reference 74 and 75. The Reference 74 and 75 results, in conjunction with Davis-Besse specific submittals, have been reviewed by the NRC. In Reference 76 the NRC issued their safety evaluation report for the Davis-Besse plant where the NRC concluded that the assessment of asymmetric loads performed for Davis-Besse were acceptable and had demonstrated with reasonable assurance that asymmetric LOCA loads would not prevent the plant from safely reaching a cold shutdown condition.

An integrated head assembly (IHA) was installed during outage 17M. The IHA was installed as a replacement service support structure (SSS). Since the reactor coolant system (RCS) is nozzle supported, the impact of changes in loading resulting from the IHA is primarily absorbed by the RV nozzle supports; therefore, installation of the IHA has negligible impact on the remainder of RCS loop major components (i.e., steam generators, pressurizer and reactor coolant pumps) and their associated subcomponents. An RCS structural analysis that considers installation of an IHA has determined that, with an IHA installed, the reactor vessel (RV) nozzles, RV supports, and RV internals are qualified for seismic and LOCA loadings. BWSPAN is the computer program used in this structural analysis. Also, the IHA-related structural analysis and the previous analyses performed by References 74 and 75 supplement Appendix 4A. The IHA-related structural analysis and References 74 and 75 performed load analyses where all loads were shown to be bounded by previously existing design loads. Appendix 4A performed stress analyses that were not re-performed by either the IHA-related structural analysis or References 74 and 75. Furthermore, the IHA-related structural analysis qualifies all RV internals via load comparisons of Leak-Before-Break (LBB) loads to those loads in Reference 74 supplemented by Reference 75, which are large break LOCA loads. In all cases, the loads from the IHA-related structural analysis were shown to be bounded by the loads from Reference 74 supplemented by Reference 75. The evaluations performed in the IHA-related analysis for the reactor pressure vessel supports incorporate use of LBB criteria, which eliminates LOCA loadings on the supports. Adoption of the LBB criteria was per References 97, 98, 104, 104a and 105.

3.9.1.5.1 Location and Type of LOCA Pipe Ruptures

Following the adoption of Leak-Before-Break criteria in 1990 (Reference 98), the original design basis of defining specific break locations and break types in the primary piping system is no longer applicable. The discussion below reflects the original plant design and is provided for historical purposes only.

This Section defines the location and type of LOCA pipe ruptures in the Davis-Besse Unit 1 primary piping system. The criteria used to define location and type of LOCA pipe ruptures are outlined in Subsection 3.6.2.

Pipe rupture locations and types are determined for breaks in the hot leg and the upper and lower cold leg pipes. Results are based on simplified pipe stress analysis per Section III of the ASME Code for Boilers and Pressure Vessels and its predecessor USAS B31.7-1969.

As stated in the criteria, breaks in the Class I piping are postulated to occur at terminal ends of the pressurized portions of piping runs, and at locations where the maximum stress ranges for normal and upset plant conditions and for Operating Basis Earthquake (OBE) exceed $2.4S_m$ (where S_m is the allowable design stress intensity specified by ASME) by both Equations 10 and 13 in Paragraph NB-3653 of the ASME Code, Section III, and at locations where the cumulative usage factor (U) exceeds 0.1 (or U/. 1 > 1.0). Only Equation 13 and usage factor results are reported herein. Dead weight, thermal, and seismic moment loadings used to calculate Equation 13 stresses are taken from Revision 1 of the "Piping Final Design Document for Toledo Edison Co. & C.E.I." May 1975.

The cumulative usage factors reported are also taken from Revision 1 of the stress report since these numbers are valid for the purpose of determining break locations.

Break locations shown in Figure 3.9-5 for the upper and lower cold legs are shown only for the P1A1 RCS loop. Due to the symmetrical nature of the RCS, breaks are postulated in the P1A2 loop at corresponding points to those shown for the P1A1 loop.

As stated in Subsection 3.6.2, guillotine ruptures are automatically postulated at the terminal ends of piping runs. To determine breaks at intermediate locations, ASME Section III Equations 10 and 13 (NB-3653), and the cumulative usage factor are employed. At locations where stress calculated by Equation 10 exceeds $2.4\ S_m$, Equation 13 is investigated. At any location where Equation 13 exceeds $2.4\ S_m$ a break (guillotine/split) is specified. Also, at any location where the cumulative usage factor U (see ASME, NB-3222.4 (5)) divided by .1 exceeds 1.0, a break is specified. Where no breaks are required to be postulated by application of the above stress and usage factor criteria, at least two breaks are postulated at separate locations selected on the basis of highest cumulative usage factor or stress intensity. Once a break location is specified, Subsection 3.6.2.5.10 outlines the procedure for determining the break type and orientation. Stress and cumulative usage factor calculations are carried out using computer code T3PIPE which uses pipe properties, moments loadings (from finite-element analyses of the RCS using computer code ST3DS), specified pressures, temperature gradients, etc., as input. Results are based on comparing Equation 13 for the worst earthquake combination, that is, the maximum resultant moment of the combined horizontal and vertical earthquake (X + Y or Z + Y whichever is greater). Stress indices associated with Equation 13 are per USAS B31.7-1969.

Table 3.9-1 summarizes the results of stress and usage factor calculations employed to determine break locations. For each node point shown in Figure 3.9-4, the ratio of calculated stress (Equation 13) to $2.4\ S_m$ is reported. Also reported is the ratio of the fatigue usage factor to 0.1 (U/.1). Since the ratio of calculated stress to $2.4\ S_m$ is not exceeded (except at terminal end which is already a specified break) and the ratioed fatigue usage factor is less than 1.0 at all locations, only guillotine breaks are postulated at 2 intermediate locations (in each piping run) of highest stress intensity (maximum ratio to $2.4\ S_m$).

3.9.1.6 Analytical Methods

The calculation of stresses for the primary piping and major components of the Reactor Coolant System was performed in accordance with ANSI B31.7, and the ASME Code Section III,

respectively. Seismic stresses for the integral supports were calculated on elastic basis and were evaluated in accordance with the ASME Code, Section III, or the AISC Structural Steel Manual, Seventh Edition.

LOCA plus Maximum Possible Earthquake stresses for the integral supports were evaluated as follows on a per component basis using the loadings from the methods described in Subsections 3.9.1.5 and 3.7.2.1.

Although the Reactor Vessel has one set of supports for carrying normal and seismic loads and another set of supports for carrying LOCA loads, some LOCA loading is imposed on the seismic supports. However, the design of the seismic supports is such that the only LOCA loads of any consequence which can be transmitted to the support acts downward. Under these conditions stresses are within the elastic range.

The stresses in the steam generator supports are within the elastic range under the combined action of LOCA and seismic loads except in highly localized areas where plastic deformation does not invalidate the design.

It should be noted that per the criteria in USAR Section 3.6.2.2.1, dynamic effects from a postulated pipe rupture in the RCS can be excluded.

3.9.2 ASME Code Class 2 and 3 Components

Major safety-related components constructed to the ASME Boiler and Pressure Vessel Code are listed in Table 3.9-2.

3.9.2.1 Design Bases

The design pressure, temperature, and loading condition that provides the basis for design of these components is specified in Table 3.9-2.

3.9.2.2 Design Loading Combinations

The components listed in Table 3.9-2 are designed to withstand the more severe of the following load combinations.

a. Maximum Probable Earthquake Conditions (0.08g).

The load combinations include gravity load, operating loads, applicable operating temperatures and pressures combined with the simultaneously applied horizontal and vertical design earthquake inertia forces.

The horizontal inertia forces are obtained by performing a dynamic analysis using acceleration response figures. Stresses in the structural portions do not exceed the allowable working stress limits accepted as good practice as set forth in the appropriate design standards. If the code permits stress increases for load combinations which include earthquake loads, such stress increases do not apply. The resulting deflections do not prevent normal operation of the equipment.

b. Maximum Possible Earthquake Conditions (0.15g).

The load combinations include gravity loads, operating loads, applicable temperatures and pressures combined with the simultaneously applied horizontal and vertical maximum earthquake inertia forces. The resulting deflections do not prevent operation of the equipment nor exceed 0.8 times the deflections which would cause loss of function of the equipment.

3.9.2.3 Design Loadings

The combinations of design loadings indicated in Subsection 3.9.2.2 are shown in Table 3.9-3. Stress limits associated with each design loading combination are also shown in Table 3.9-3.

The design criteria for pump and vessel supports are given in Table 3.9-3. These supports are analyzed as part of the components as indicated in Subsection 3.9.2.9.

3.9.2.4 Allowable Loads

The component allowable loads were not exceeded by the combination of design loadings; therefore, the stress limits did not result in inelastic deformation.

3.9.2.5 ASME and ANSI Code Case Interpretations

Applicable ASME and ANSI Code Case Interpretations are shown in Table 3.9-2.

3.9.2.6 Active Pump and Valves

Major safety-related ASME Code active pumps and valves established during initial plant design are shown in Table 3.9-2. The designation of active components since initial design has been controlled by the Inservice Testing Program. Criteria used to ensure that components functioned as designed and that stresses are below yield stress are as follows:

for pumps:

a. Seismic calculations.

b. Attached piping thermal and seismic loading.

c. Hydrotesting.

d. Performance testing.

for valves:

a. Seismic calculations.

b. Hydrotesting.

c. Designed to ANSI B16.5.

3.9.2.6.1 Functional Testing of Active Pumps and Valves

The primary means of ensuring operational readiness of ASME Class 1, 2, and 3 active pumps and valves are the Inservice Testing (IST) and Inservice Inspection (ISI) Programs. These programs address periodic functional verification, and testing following repair, replacement, and modification activities on these components. Other testing activities are conducted, as applicable, to provide additional assurance of the reliability of ASME Class 1, 2 and 3 active pumps and valves. These activities are established, specified, conducted, and documented in accordance with Technical Specifications and established plant programs and procedures, and include surveillance and periodic testing, pre-and post-maintenance testing, post-modification testing, modification testing, and maintenance testing.

3.9.2.7 Design Bases and Criteria for Pipe Whip

The design bases and the criteria used to ensure the protection of all critical systems and the containment from effects of pipe whip are presented in Section 3.6.

3.9.2.8 Pressure Relieving Device Design

The main steam safety valves are mounted on a section of each main steam line which has an increased wall thickness to reduce bending and compressive stresses. The safety valve mounting nozzles are machined from forgings and the inside diameters sized so as not to affect safety valve flow capacity or set pressure. The nozzle wall thickness is based on stress limits stipulated in the ASME Code, Section III.

The discharge of the safety valves are alternately oriented 180 degrees apart in order to minimize the steam line torsional moment imposed by discharge thrusts. In addition, a short radius elbow is used on each valve discharge to reduce the moment arm.

The main steam line outside the containment is analyzed using a piping flexibility analysis computer program. The analytical model contains all relief valves and imposes the full safety valve discharge forces at the discharge pipe exits. The computer output contains forces, moments, and stresses in the main steam line and connected piping, and loads and moments at the anchors and supports. The maximum stresses are below the allowable stresses given in Section III of the ASME Code. The complete analysis includes all operating and seismic loads.

All overpressure relief valves and their connected piping (i.e., headers, header connections, and discharge piping) are designed to withstand the following conditions without exceeding the applicable codes primary stress allowable: maximum loads due to valve discharge thrust, internal pressure, dead weight, and earthquake applied simultaneously. When more than one relief valve is attached to a piping system, the loads, due to all relief valves discharging simultaneously, are applied to the system along with the above mentioned primary loads. In addition, the loads from the most critical combination of valves discharging are applied. The local stresses in the main steam line outside the containment, at the connection of the relief valves, were computed as specified in “Welding Research Council Bulletin 107” and contained below-the-allowable primary stress level.

3.9.2.9 Analytical Methods and Criteria

The analytical methods and criteria used to evaluate stresses in all pumps and valves are provided in the following Subsections (for component numbers refer to Table 3.9-2). Specific results are included in Engineering calculations.

3.9.2.9.1 Main Steam Isolation Valves (No. 1)

The seismic loading is defined as 3 gravities in any direction. For the valve installed in line, the upper-structure is at a 45 degree angle to horizontal. Thus the worst loading condition would be the force acting through the c.g. of the upper structure perpendicular to centerline of upper structure. This total force would be the seismic loading plus the component of gravity acting at a 45 degree angle. The worst loading condition for the bypass assembly which is horizontal would be 3.0g's plus the full effect of gravity acting perpendicular to the plane of by-pass.

There are seven components which are analyzed. The first is the tie rods which are sized to provide the valve with an acceptable natural frequency. Next, the mounting plate and spring guide tubes (yoke tubes) are analyzed, using Castigliano's theorem. Finally, the bonnet bolting, the yoke tube bolting, the by-pass assembly, and the weld end prep are analyzed.

Tie Rod Sizing:

The tie rods are sized so that the elongation caused by the maximum expected seismic force is less than that needed to meet the required natural frequency of 20 cps. Different values for the force absorbed by the tie rods are substituted into the formula for loading in the upperstructure until the deflection found for the whole upperstructure is approximately the same as that due to the elongation of the tie rods. The resulting stresses in the upper structure are then found for the final values obtained. The effect of the tie rods is maximized by preloading them to one half of maximum load. This means at maximum loading the two tie rods opposing the seismic force is doubled in loading while the two tie rods in the other directions are reduced to zero loading.

Upper Structure:

The plane of analysis selected was based on the plane in which the cylinder mounting flange and spring guide tubes have a minimum moment of inertia. Use of Castigliano's theorem yields the reaction loads and moments, from which stresses and deflections were calculated. The maximum stresses in the spring tube guide and flange were calculated for both the seated and backseated valve positions.

Deflection:

As a check that the valve functions during seismic loading, the deflection is calculated. The most critical deflection for the valve is movement of the cylinder mounting flange in a direction parallel to the flange. Castigliano's theorem was used to calculate the deflection. The deflections are 0.1373 inches (valve seated) and 0.1364 inches (valve backseated), which do not affect valve operation.

Natural Frequency of Upper Structure:

Highest deflection of analysis is for the seated position which gave a value of 0.1373 inch. The actual value is between the value found for tie rod analysis of 0.136 inch and the value of 0.1373 inch, but the higher value is used to be conservative. The calculated natural frequency is 20.3Hz.

Stress Acting on Bonnet Bolts:

The forces were calculated per Appendix E of ASME, Section III. The maximum tensile stress is 27,969 psi.

Thermal stresses in the studs are assumed to be negligible because of the clearance between studs and bonnet plate, and both materials have the same coefficient of thermal expansion.

The allowable stress for bolts per paragraph NB-3232.1 in ASME III is twice the design stress when preload, pressure, and thermal stresses are combined. The design stress for ASME SA 540 Gr B23 is 40,000 psi at 100°F.

The combined stress is the larger of the preload stress or pressure stress since thermal stress is zero.

Standard preload = 45,000 psi
Pressure stress = 27,969 psi
Thermal stress = 0

Since 45,000 psi is the larger of the two stresses and since 45,000<80,000 psi, the bonnet bolt stress level is acceptable.

Stress in Bonnet to Yoke Bolting:

The maximum bending, shear, and stress intensity were calculated for the valve in both the seated and backseated positions. The stress intensities were 20,654 psi and 22,060 psi, respectively. The bolts are prestressed to 45,000 psi. Since the actual load is below the preload, the bolts see no more stress than that caused by the preload.

Weld End Analysis:

The sectional modulus of the weld endwall is 1.92 times as great as that of the pipe. This means that any stresses of concern in the weld end area would be critical in the pipe before being critical in the weld end.

Results

Component	Material	Maximum Stress psi	Allowable Stress psi
Tie rods	AISI 4140	29,863	100,000
Guide tubes	AISI 1015/1025	----	35,000
Backseated	----	14,858	---
Seated	—-	14,108	---
Mounting flange	AISI 4140	-----	100,000
Backseated	----	38,064	---
Seated	----	37,130	---
Bonnet Bolts	SA540GRB23	45,000	80,000
Bonnet/Yoke bolts	A-574	45,000	140,000

3.9.2.9.2 Motor Operated Valves

a. Gate valves (Nos. 3, 6, 7, 20, 23, 27, 31, 38, 41, 52, 54, 57, 58, 68, 73, 75, 76, 77, 78, 81, 84, 85, and 86)

Traces taken during the El Centro earthquake of May 1940 established a duration of some 30 seconds to attenuated disappearance. The requirement for a valve to remain functional during an earthquake is necessary in view of the fact that many valves must complete their operation within a time span of 10 seconds, while the earthquake may last 30 seconds.

Relative to the effects of earthquakes on piping systems, it has been established that the maximum acceleration which any point in a piping system could experience before the pipe would be overstressed would be 2.0g for the maximum possible earthquake. If one point moves with 2.0g in each of three directions, the resultant would be 3.5g.

It is almost impossible that any point would experience 2.0g simultaneously in each of three directions. Therefore, a lower resultant of 3.0g is considered to be a safe value to use for most stress analysis. For components which are not of a compact design, such as a motor operated valve, the 3.0g may be used for stress analysis provided that the frequency of vibration for the motor is much greater than the predominate frequency of the building.

These valves are designed to withstand an inertial load of 3.0g in any direction, in addition to normal operating loads. The frequency of vibration for extended parts of the valve is greater than 20 cps.

Stresses are calculated in valve components due to the dynamic loads by a seismic load and normal closing. The combined stresses in the yoke arms are determined by the “Maximum Distortion Energy Theorem” by Huber (32) which has proven to give theoretical results very close to experimental results.

The rigidity of the yoke in three directions plus the torsional rigidity are determined. From this, the frequency of the system is calculated using Dunkerley’s equation (33). This is compared to the natural frequency.

b. Butterfly valves (Nos. 10, 13, 16, 19, 39, and 51)

Three “combined-stress-values, “the maximum and minimum normal stress, and the maximum shear stress are calculated for each valve area (plane) being analyzed. Only the total tensile stress and the total shear stress are used as input in the combined-stress-formulas. The valve body is considered to be rigidly mounted in the pipeline and is stronger than the mating pipe. Previous calculations have shown that the stresses due to seismic loading on the internal parts of the valve (disc, shaft, etc.) are insignificant. Therefore, these specific parts are not analyzed. The three combined stress values for each plane are calculated using the “Principal Stress Formulas.” The “Maximum Shear Stress Theory” of failure is used as specified in Section III of the ASME Boiler and Pressure Vessel Code, Paragraph NB-3212.

The total load at any of these planes consists of the normal operating load, plus the weight of those components supported by the critical area, multiplied by the seismic g-factor. The latter is considered to act at the center of gravity of the assembly involved.

The effect of the seismic inertial load acting on a plane is reduced as the distance from the valve centerline to that plane increases, because less components are involved. The valve body is considered to be rigidly and integrally connected to the pipeline.

Bolt Systems:

Bolt systems are subjected to both tensile and shear stresses. Tensile stresses result from bending moments caused by the seismic inertial load acting at the center of gravity of the assembly of components supported by these bolts and the actuator force. The magnitude of this stress is greater in the bolts where the maximum strain occurs, and the strain produced in the bolts is proportional to their distance from the edge of the component, which acts as a pivot line. This pivot line is perpendicular to the action line of the force so it can be either vertical or horizontal, depending on which load is under study.

The total shear stress is found by combining all forces causing shear stresses, then dividing by the bolt area. Some forces act in direct shear and others in torsion.

The combined stress values are calculated using the “Principal Stress Formulas.”

Actuator Brackets:

The tensile and shear stresses of the most severely loaded cross sections of the first three brackets are calculated.

The total tensile stress is a summation of three individual tensile stresses caused by the vertical and horizontal bending moments and the actuator force.

The total shear stress (due to direct seismic shear forces and torsional moments) is a summation of individual shear stresses caused by the vertical and horizontal seismic inertial loadings and the actuator output.

The combined stress values are calculated using the “Principal Stress Formulas.”

Based on the calculations, the primary steady-state stresses, when combined with the inertial loading resulting from the response to a ground acceleration of 3g acting in the vertical and 3.0g acting in the horizontal planes simultaneously, produce combined stresses which are safely within the yield stresses of the construction materials, both in tension and in shear.

Also, calculations verify that the extended parts of each valve assembly have a natural frequency of vibration greater than 20 cps.

c. Globe valves (No. 33)

Since the valve has been Code classified, the rules for design are those stated in Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components.

Since the valve is to be installed such that the distance from the body to the top of the actuator is extended, the length of this extension is the length of the appurtenance, and these parts--bonnet, flange, and actuator--are the appurtenances.

In accordance with the Code, the piping system, and not the valve, is limiting. In this case, the piping system must be considered the limit with respect to the valve.

On this basis, an analysis can be performed with respect to the effect of seismic loadings on:

1. Actuator yoke
2. Bonnet to actuator yoke Connection
3. Body-bonnet bolted flanged connection

The seismic forces (3.0g) are considered as acting through the center of gravity of the extended masses. The vertical and horizontal forces and moments based on operational and seismic loads are calculated.

Actuator Yoke Stresses:

This analysis of the columns in the actuator yoke is limited to those simple stresses created by the horizontal and vertical seismic loads and the normal operating load developed by the valve actuator. All axial and flexial loads are considered as acting simultaneously. Only maximum stress conditions are analyzed. From this, the bending stress, axial stress, and combined stress are calculated.

Yoke-Bolt Stress:

The bolt force due to the overturning moment, the axial force, and total tensile stress is calculated. To determine the natural frequency, the actuator yoke is analyzed as a cantilever beam. The lump parameter technique is followed in which the weight is considered as acting through the center of gravity and the vibrations are attributed to the elasticity of the yoke legs.

Body-Bonnet Flange Bolt Stress:

The stresses resulting from earthquake effects are included with weight, pressure, or other applied loads. The design pressure used in calculations to determine the minimum bolt cross-sectional area is the ‘flanged design pressure’ (PFd) as defined in NB-3647.1 of Section III. Having determined this pressure, the design bolt load is determined. This is compared with the minimum initial bolt load to seat the gasket. From this, the maximum bolt area is determined.

d. Globe valves (Nos. 29 and 69)

The analytical method is as indicated in paragraph (a) above.

e. Gate valves (Nos. 21, 22. 62, and 67)

The analytical method is an indicated in paragraph (a) above.

f. Ball Valve (No. 17)

Since the valve has been Code classified, the rules for design are those stated in Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components.

Since the valve is to be installed such that the distance from the body to the top of the actuator is extended, the length of this extension is the length of the appurtenance, and these parts--adapter plate and actuator--are the appurtenances.

In accordance with the Code, the piping system, and not the valve, is limiting. In this case, the piping system must be considered the limit with respect to the valve.

On this basis, an analysis can be performed with respect to the effect of seismic loadings on:

1. Actuator mount
2. Actuator to body connection
3. Body-actuator bolted connection

The seismic force of 3.0g is considered as acting through the center of gravity of the extended masses. The vertical and horizontal forces and moments based on operational and seismic loads are calculated.

To determine the natural frequency, the actuator is analyzed as a cantilever beam. The lump parameter technique is followed in which the weight is considered as acting through the center of gravity and the vibrations are attributed to the elasticity of the valve body throat.

The results of this analysis are as follows:

Component	Material	Actual stress, psi	Allowable stress, psi
Adapter plate bolts	ASTM-193 B7	16,347	25,000
Actuator bolts	ASTM-193 B7	8241	25,000
Valve throat	SA-216 WCB	8012	17,500

Natural frequency = 125 cps

g. Diaphragm Valve (No. 66)

The loading conditions, design data, methods, assumptions, allowable stress, and natural frequency are similar to the discussion in Subsection 3.9.2.9.8e.

h. Three-way Ball Valve (No. 72)

The appurtenances of the valve are the bracket and the actuator. Treating the actuator as a lumped mass acting at its center of gravity, the natural frequency of the system is found to be 74cps.

The seismic force of 3.0g is considered as acting through the center of gravity of the extended masses. The vertical and horizontal forces and moments based on operational and seismic loads are calculated. The results of this analysis are as follows:

Component	Material	Actual stress, psi	Allowable stress, psi
Bolts	Steel	3.124	----
Bracket	ASME SA-36	3,925	12,600

i. Gate Valves (No. 82)

These valves are Code Class 2 and the rules for their design are those stated in Section III of the ASME Boiler and Pressure Vessel Code.

The design of these valves is such that the actuator is located an extended distance from the valve body center line, therefore an analysis of these extended portions is required for seismic and normal operating loads. The valves are designed to withstand an inertial load of 3.0g in any direction, in addition to normal operating loads. The frequency of vibration for the extended components of the valve is greater than 33 cps.

Stresses due to dynamic loads caused by seismic and normal closing are calculated in valve components using classical equations for shear, axial and bending stress. For pressure retaining components, the resulting stress levels are compared to the design limits specified in the ASME B&PV Code. For non-pressure retaining components, stress levels are limited to a multiple of minimum specified material yield strength.

The analysis performed determines the effect of the previously described loading on the following valve components.

1. Actuator mounting
2. Yoke
3. Yoke clamp
4. Body neck
5. Body

j. Globe valve (No. 83)

The analytical method as indicated in paragraph (i,) above.

k. Stop-check valves (Nos. 79 and 80)

Stresses in the valve components due to seismic and operating loads are determined using classical bending, tensile and shear stress formulas. The valves are designed for a 3.0g inertial load in any direction in addition to normal operating loads. The resulting stress levels are compared to seventy percent of the minimum specified material yield strength. The natural frequency of vibration for the extended masses of the valve is greater than 20 Hz.

l. Gate Valves (No. 74)

These valves are Code Class 2 and the rules for their design are those stated in Section III of the ASME Boiler and Pressure Vessel Code.

The design of these valves is such that the actuator is located an extended distance from the valve body center line, therefore an analysis of these extended portions is required for seismic and normal operating loads. Based on the piping system stress analysis results, a seismic acceleration of 1.5g in the x, y and z directions is considered as acting through the center of gravity of the extended mass. The valves are designed to withstand this inertial load in conjunction with the design operating loads. The fundamental frequency of vibration for the extended components of the valve/operator assembly is greater than 33 cps.

Stresses due to dynamic loads caused by seismic and design operating loads are calculated in valve components using classical equations for shear, axial and bending stress. For pressure retaining components, the resulting stress levels are compared to the design limits specified in the ASME B&PV Code. For non-pressure retaining components, stress levels are limited to a multiple of minimum specified material yield strength at the operating temperature.

In conclusion, the actual valve accelerations are enveloped by the above accelerations, the valve/actuator fundamental frequency is greater than 33 cps and no stresses generated exceeds the material allowables.

3.9.2.9.3 Auxiliary Feed Pumps (No. 4)

The natural frequency of vibration was determined for the pump supports so that the seismic accelerations could be determined using the frequency response spectra at the appropriate pump elevation.

The hold-down bolting and doweling were investigated to determine if they were of adequate sizes to withstand gravitational, nozzle, and seismic loads. Stresses in the pump feet were calculated to verify that the dowel pins would not fracture the pump feet.

The deflection of the pump shaft due to seismic loading was calculated to verify that the running clearances would be maintained and thus fulfill the no loss of function criteria.

The determination of seismic effects on a structure using dynamic methods requires simplifying assumptions and idealization to formulate a problem that lies within the capability of known methods of solution. These simplifications, in effect, involve the substitution of a model for the structure, and the response determined is that of the model. To calculate the natural frequencies, it is assumed that the pump, hold-down bolts, and pedestals are an equivalent spring-mass system where strength of materials is used to obtain the spring constants.

The following coordinate system was used: The x-axis is parallel to the pump shaft and is positive toward the driver end. The y-axis is vertical and is positive upward. The z-axis is perpendicular to the pump shaft, and the positive direction is such that a right-handed coordinate system is defined. The origin of this coordinate system is the center of the rectangle formed by the four holddown bolts. Also, the plane defined by these four bolts is referred to as the bolting shear plane.

Vertical Frequencies Computational Procedure:

The pedestals and bolts are represented by springs with spring constants k_1 and k_2, respectively. It has been established for many structural materials that within certain limits the elongation of the pedestals and bolts is proportional to the force. This simple linear relationship between the elongation and the force is Hooke's experimental law

$$\delta y = \frac{F\ell}{AE} \qquad (1)$$

Rearranging Equation (1), the spring constant is defined as

$$k = \frac{F}{\delta y} = \frac{EA}{\ell} \qquad (2)$$

The following table gives explicit definitions of the symbols used in Equation (2).

	Pedestal	Bolt
F (lbs)	Force producing elongation	Force producing elongation
δy (in)	Elongation of pedestal in y-direction	Elongation of bolt in y-direction
E (psi)	Modulus of elasticity	Modulus of elasticity
A (in^2)	Cross-sectional areas of pedestal at midpoint	Root area of bolt
ℓ (in)	Height of pedestal	Length of bolt allowed to stretch

Equation (3) is used for translational motion in the y-direction:

$$k_y = \frac{4k_1k_2}{k_1 + k_2} \qquad (3)$$

where the springs at the corners of the rectangle formed by the holddown bolts are replaced by one spring having translational spring constant k_y.

Similarly, for rotational motion about the two horizontal axes x and z, two effective springs given by rotational spring constants are (Reference 34)

$$k_{yx} = \frac{4a^2k_1k_2}{k_1 + k_2} \qquad (4)$$

and

$$k_{yz} = \frac{4a^2k_1k_2}{k_1 + k_2} \qquad (5)$$

where the following definition for the double subscript is used in the above values of k. The first subscript denotes the direction of the applied force, and the second subscript denotes which axis about which the rotation takes place.

Once the spring constants are obtained, k_y, k_{yx}, and k_{yz}, it is a simple matter to obtain the natural frequencies of vibration which are given by the following three relationships:

$$f_y = \frac{1}{2\pi}\left[\frac{k_y}{M}\right]^{1/2} \qquad (6)$$

$$f_{yx} = \frac{1}{2\pi}\left[\frac{k_{yx}}{I_{mx}}\right]^{1/2} \qquad (7)$$

and

$$f_{yz} = \frac{1}{2\pi}\left[\frac{k_{yz}}{I_{mz}}\right]^{1/2} \qquad (8)$$

where I_{mx} and I_{mz} are the mass moments of inertia of the pump about the x and z axes, respectively.

Horizontal Frequencies Computational Procedure:

In the calculation of the horizontal natural frequencies of vibration, the pedestals and bolts were considered as cantilevered beams. For a force in the x-direction, the deflection is given by

$$\delta x_n = \left[\frac{1}{n}\right]\left[\frac{F\ell^3}{EI}\right] \qquad (9)$$

Rearranging Equation (9), the spring constant is defined as

$$k_n = \frac{F}{\delta x_n} = \frac{nEI}{\ell^3} \qquad (10)$$

where explicit definitions of the symbols used in Equation (10) are given in the following table.

	Pedestal	Bolt
F (lbs)	Force producing deflection	Force producing deflection
δx_n (in)	Pedestal deflection	Bolt deflection
n	n = 3 for an end loaded beam	n = 8 for a uniformly loaded beam
E (psi)	Modulus of elasticity	Modulus of elasticity
I (in^4)	Area moment of inertia	Area moment of inertia
ℓ (in)	Height of pedestal	Length of bolt allowed to deflect

In this model, it is assumed that the pedestals are end loaded beams and the bolts to be uniformly loaded beams. For a force in the z-direction, the analysis proceeds in an analogous manner. It is easily shown that the effective spring constant for translational motion in the x-direction is

$$k_x = \frac{4k_3k_8}{k_3 + k_8} \qquad (11)$$

Similarly, for rotational motion about the vertical axis an effective spring with a rotational spring constant is given by (Reference 34)

$$k_{xy} = \frac{4b^2 k_3 k_8}{k_3 + k_8} \qquad (12)$$

For motion resulting from a force parallel to the z-direction, calculate a translational spring constant, k_z, and a rotational spring constant, k_{zy}.

Given the spring constants, the horizontal frequencies are calculated in an analogous manner to the vertical frequencies. From the response spectrum curves, the accelerations reach their steady-state values for frequencies greater than 35 Hz. Since the calculated natural frequencies are greater than this value, the seismic forces were calculated on the basis of the steady-state region of the frequency spectra.

Pump Mounting Stresses:

A typical pump foot consists of a hold down bolt hole and a locating dowel hole. The maximum stresses of the pump mounting are found in the hold down bolts and dowels because their tensile and shear area are much less than the area of attachment between the foot and the case.

Since the two dowels are located in tight fitting holes which permit vertical motion and the four bolts are installed in clearance holes, it is assumed that shear forces are transmitted only to the dowels and tensile forces are transmitted only to the bolts. After determining the forces and moments on the pump, they are translated from their point of application to the center of the bolting shear plane. The vertical force and the moments about the horizontal axes contribute only to the tensile load. Similarly, the horizontal forces and the moment about the vertical axis contribute only to the shear load.

The vertical force, F_y, is divided equally between the four bolts.

The moments about the horizontal axes, M_x and M_y, are each assumed to be a couple consisting of two forces of equal magnitude and opposite direction. The sum of the three tensile forces acting at each bolt is evaluated, and the largest is divided by the root area of the bolt to obtain the tensile stress.

The horizontal forces, F_x and F_z, are divided equally between the two dowels. The moment about the vertical axis, M_y, is also assumed to be a couple. The forces comprising this couple are added vectorially to the other shear forces. The resultant forces are evaluated, and the largest one is divided by the cross-sectional area of the dowel to obtain the shear stress.

Determination of Loads due to Natural Phenomena:

The horizontal and vertical forces are given by

$$F_h = \frac{w}{g} a_h \qquad (13)$$

and

$$F_v = \frac{w}{g} a_v - W \qquad (14)$$

where:

F_h = horizontal force

F_v = vertical force

W = pump weight

g = acceleration due to gravity

a_h = horizontal seismic acceleration

a_v = vertical seismic acceleration

The shear stress in the dowels and the tensile stress in the bolts are calculated using the previously outlined methodology. The horizontal forces were taken in the direction that maximized the stresses.

The stress values calculated are now compared with the prescribed values shown below. All numerical values of yield and tensile stresses are taken from ASME Section III (1971). Prescribed shear stresses are computed by taking 0.8 of the applicable tensile or yield stress.

Maximum Probable Earthquake

Stress	Calculated Value	Prescribed Value
Tensile (psi)	-1,098	7,000 (a)
Shear (psi)	2,574	20,000 (b)

Maximum Possible Earthquake

Stress	Calculated Value	Prescribed Value
Tensile (psi)	-973	27,000 (c)
Shear (psi)	4,554	75,600 (d)

(a) Prescribed value = allowable tensile stress
(b) Prescribed value = 0.8 x allowable tensile stress
(c) Prescribed value = 0.9 x yield stress
(d) Prescribed value = 0.8 x 0.9 yield stress

Examination of the above table shows that in all cases the calculated stress values are less than the prescribed stress values, and therefore the hold down bolting and doweling are sufficient to maintain alignment under loading due to natural phenomena.

Calculation of Shaft Deflection:

The shaft deflections are found by using a digital computer program. The method of calculation consists of dividing the shaft into one inch segments for which all diameters are noted. The diameters and location of bearings and sleeves are recorded. In addition, the weight and location of the impellers are also recorded. After determining the load and moment of inertia for each segment, the deflections are found by numerical integration.

The program takes into account the stiffening effect of any hubs or sleeves by adding a fraction of the moment of inertia of the hub or sleeve to the moment of inertia of the shaft. The “stiffness coefficient” which determines the amount to be added may take values from 0.0 to 1.0, depending on the type of fit. Timoshenko (ref.14) suggests taking the coefficient equal to unity for cases where the vibrational stresses are small and the shrink fit pressure is sufficient to prevent any relative motion. On the other hand, Stepanoff (ref. 35), without differentiating between shrink and loose fits, suggests a value between 0.5 and 0.65. This program determines the deflection for “stiffness coefficients” of 0.0, 0.5, and 1.0. It was considered, as does Reference (4), that the most representative deflections are those for K=0.5.

To determine the deflection of the pump shaft under seismic loadings, it is assumed that the seismic accelerations act directly at the bearings. The applied seismic forces were transmitted to the rotating element through the bearings, assuming no dissipation due to deformation of the pump case. The results of the calculation give a shaft deflection of 0.001 inch for a value of the “stiffness coefficient” equal to 0.5. The running clearances for the pump are much greater than 0.001 inch, and therefore there is no interference.

In summary, the vertical and horizontal frequencies of vibration were calculated and were found to be greater than 35 Hz. The accelerations from the frequency response spectra were used to calculate the forces due to natural phenomena from which the stresses in the pump hold down bolts and dowels were examined and found to be within the allowable limits. The shear stress in the pump foot was also examined and found to be less than the allowables. The maximum shaft deflection resulting from seismic loading does not cause the rotating element to rub or seize, thus fulfilling the no loss of function criteria due to seismic loading.

3.9.2.9.4 Atmospheric Vent Valves (No. 5)

In the primary pressure containing elements, stress limits and evaluation were those prescribed by ANSI 16.5. This is in accordance with the rules given in ASME Section III, Pressure Vessel Code, Article MC-3500.

In other structural elements, including body and bonnet, flanges, and disc retaining bolts, stress criteria considered were those recommended in ASME Section III, Pressure Vessel Code, Article iii-3100.

The valve structure is assumed to be rigidly mounted around the outlet flange. To determine the static deflection, the body is considered rigid, and deflection takes place in the outlet pipe, yoke and actuator.

Using 3g and the normal operating force, the vertical and horizontal forces are calculated. From this, the bending stresses are determined.

Considering the weight of the body to be evenly distributed over its entire length, the total static deflection and natural frequency are calculated. The results shown below are for the most highly stressed zones in the valve due to seismic and normal loadings:

Section	Material	Allowable stress psi	Actual stress psi
Outlet Pipe Extension	SA105	17,500	11,673
Body to Bonnet Bolting	SA193 Grade B7	25,000	19,244
Yoke	A500 Grade B	41,400	3274

Natural frequency = 36 Hz

3.9.2.9.5 Main Steam Safety Valves (No. 8)

There are a total of eighteen Main Steam Safety Valves. These valves are of two sizes, 6"x 8" and 6" x 10". The valves are located in plant Areas, 7 and 8 as described in Section 3.7

Design Conditions:

1115 psig at 565°F, back-pressure 0 psig
ASME Section III

Flange material = ASTM A 216-WCB

S = 17,800 psi 600°F

Analytical Approach:

The most highly stressed zone in both valve sizes during a seismic event is the inlet flange neck of the valve body casting. The valves are analyzed to determine the maximum significant stress intensity developed in the valve body inlet flange neck with the following occurring concurrently:

1. Simultaneously applied horizontal and vertical seismic loading.
2. Valve open and flowing at full relieving conditions.

The fundamental frequency of vibration for each valve size was calculated.

1. 6"x8" valve = 16 Hz
2. 6"x10" valve = 14.6 Hz

The corresponding accelerations for these frequencies were determined for the two plant areas in which the valves are located.

1. 6" x 8" Valve; Area 7 -1.8 g horizontal
   2.0 g vertical

   Area 8 -2.42 g horizontal
   0.65 g vertical

2. 6” x 10” Valve; Area 7 - 2.04 g horizontal
2.00 g vertical

Area 8 - 3.85 g horizontal
0.80 g vertical

The reaction forces which result during the opening of the valves were determined to be 14,979 lbs for the 6” x 8” valve and 21,618 lbs for the 6” x 10” valve.

The analysis considers the operating loads to be deadweight, internal pressure, the reaction and seismic forces. Using the determined accelerations and the operating forces, the vertical and horizontal forces are calculated. From these values the maximum stress is determined for the inlet flange of each valve size.

Seismic conditions as specified are acceptable for the valves as designed.

3.9.2.9.6 Containment Vessel Vacuum Relief Valves (No. 9)

The stress analysis conforms to ASME Section III, Class 2 requirements. All structural parts except the disc are analyzed for 150 pound pressure class. The disc is analyzed for operating conditions as stipulated in the design specification.

Body:

Stresses in the body are calculated per UG-27 of Reference 36.

Cover Plate:

The stress developed in the cover is determined, as specified by UG-34 of Reference 36.

Bolting:

The stress developed in the bolting is determined using UA-57 and UA-58 of Reference 36.

Body Flange:

The stress developed in the body flange is determined by UA-57 of Reference 36.

Disc:

The maximum stress developed at the center of the disc is determined by use of page 194, Case 1 of Reference 11.

Bushing housing:

The maximum stress developed in the bushing housing is determined per UG-27 of Reference 36.

Lower flange and bolting:

The stress in the lower flange and bolting is calculated per UA-57 of Reference 36.

Counter-balance assembly:

Calculations were made to determine the optimum position of the counter-balance. Two conditions were satisfied as follows:

1. The valve does not open under a seismic disturbance of 3.0g and zero differential pressure.

2. The valve does open normally under a 0.15 psi differential pressure and no seismic disturbance.

A vibration analysis of the counter-balance assembly was conducted, assuming the assembly to be rigidly fastened with the shaft which is further assumed to be fixed. The assembly is then approximated as a cantilever beam with the mass at the end. The natural frequency was determined to be 114 cps.

Minimum design life:

To conform to the minimum design life of 40 years as stipulated in the design specification, the number of design startup/shutdown and design operating cycles are evaluated on the basis of alternating stresses. The maximum alternating stresses are listed below.

a. Cover plate

Sa, alternating stress $\frac{15300}{2} = 7650$ psi

Fatigue strength reduction factor = 2.0

Maximum alternating stress = 2 x 7650
= 15300 psi

Corrected maximum alternating stress = $Sa \times \frac{E}{E'}$

$$= \frac{15300 \times 30 \times 10^6}{28 \times 10^6}$$

= 16390 psi

From Figure 1.9-1, Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components.

For $Sa^1 = 16390$, $Na > 10^6$ where Na is actual number of startup/shutdown cycles. Obviously $N>10^6$ where N is actual number of operating cycles.

b. Bolting

Sa, alternating stress $\frac{11400}{2}$ = 5700 psi

Fatigue strength reduction factor for bolting = 4.0

Maximum alternating stress = 4 x 5700 = 22800 psi

Corrected maximum alternating stress = 22652 psi

From Figure 1.9-1, for Sa^1 = 22800, Na = 8 x 10^4 where Na is actual number of start up/shut down cycles. Considering fluctuation factor of 0.5

Sa^1 = 0.5 x 22650
= 11400 psi

From Figure 1.9-4 for Sa = 11400 N>10^6 where N is actual number of operating cycles.

The actual number of operating cycles is greater than 10^6 and the actual number of startup/shutdown cycles is approximately 8x10^4. These cycles are considered adequate to meet the minimum design life criteria of 40 years.

Results:

Component	Material	Maximum Stress, psi	Allowable Stress, psi
Body	SA-216 Grade WCB	5980	17,500
Cover plate	SA-516 Grade 70	15,300	17,500
Bolting	SA-193 Grade B7	11,400	25,000
Body flange	SA-216 Grade WCB	5742	17,500
Disc	SA-240 Type 316	7580	18,400
Bushing housing	SA-516 Grade 70	720	17,500
Lower flange	SA-516 Grade 70	1775	17,500
Lower flange bolting	SA-193 Grade B7	1320	25,000

3.9.2.9.7 Service Water Pumps (No. 11)

Both horizontal and vertical seismic forces have been considered for two conditions: a Maximum Probable Earthquake (0.08g) and a Maximum Possible Earthquake (0.15g). Vertical accelerations are 2/3 of these values. The resulting stresses from the seismic forces are then added to the stresses occurring from the pump operation. The stresses are limited to code levels with the added requirement that the deflection cannot exceed 80% of the deflection that would cause loss of function during the maximum possible earthquake.

As the natural frequency of the pump in the vertical direction is 77.5 Hz., the vertical component of the seismic forces has been considered as a static load where applicable.

Computer programs have been used to obtain eigenvalues and eigenvectors describing the vibrational modes which have been converted into vibrational frequencies and mode shapes, respectively. Another program calculates the moment and stress at each of the mass points. The calculation of the natural frequencies of the pump assembly below the base takes into account the effect of the water surrounding the pump. This was calculated for both low and high water levels. Reference (10) presents data on the natural frequency of ships vibrating, wholly or partially submerged, in water. The information presented states that the free vibration frequency of a submerged body must be modified to account for the mass of liquid carried along with it as it moves. The mass to be added to a submerged cylinder is the volume of the cylinder times the density of the submerging liquid. If this mass is added to the cylinder its calculated natural frequency corresponds to observed values.

The flexibility of the base plate supporting the pump column assembly is accounted for as follows: The load at the attachment of the pump column to the discharge head consists of a pressure load, weight of the pump, and the moment due to vibration plus the base shear due to vibration. In order to calculate the spring constant in bending for use in the dynamic analysis, all but the moment on the lower portion of the discharge head was ignored. Considering a trunnion loading of a flat plate, per cases 5 and 10 of Reference (11), the spring constant in bending can be calculated by using the equation:

$$\theta = \frac{M}{\alpha E t^3}$$

Where α is an empirical function of the ratio ID/OD. Because of the rigidity provided by the shell of the discharge head, all the input loads concentrated at this point are considered. The base plate can be considered as simply supported at the inner edge of the foundation plate or having a fixed edge at the bolt circle. In actuality, the support is somewhere between these two extremes, and the constants for each have been averaged. The spring constant in inch pounds per radian is then used in the program to calculate the natural frequencies and deflections.

Having determined the frequencies for several modes, the input acceleration for each mode is used to calculate the resulting deflections and moments. The moments are added in quadrature, and the stresses for each mode are determined from the section properties at the different mass points. Stresses from the operation of the pump are then calculated, as are the stresses due to the vertical seismic forces. The resultant stress is calculated by combining in quadrature of direct addition as is appropriate in each individual case.

Deflections at each mass point are available from these calculations and can be compared with the allowable deflections for proper pump operation.

The calculations for the portion of the pump above the floor level are carried out using the same programs as used previously. Here, the components under consideration consist of the discharge head assembly, including the mounting plate, and the electric motor. Data obtained from the motor manufacturer gives the “Reed Critical Frequency” of the motor, the weight, and the center of gravity location. From this data, the deflection of the center of gravity and the equivalent moment of inertia can be calculated. Reference 12 outlines the general method used for these calculations and the method by which the reed critical frequency of the motor is obtained.

Resulting stresses in this portion of the pump assembly are obtained in the same manner as was used for the below base portion.

The shaft assembly between the motor and bowl assembly is supported by bearings at intervals along the column. The natural frequency of the shaft has been calculated by the method given in Reference 13, which takes into account the bearing spacing and the thrust load acting on the shaft. A similar derivation of this formula is also given in Reference 14.

In addition to the normal operating stresses, there are seismic loads associated with the natural frequency of the shafting between bearings and a bending moment due to the deflection of the column pipe. The inputs associated with the vertical frequencies at shutoff head and normal head are also added in determining the total shaft loads and stresses. Operating shaft stresses are calculated in accordance with the methods in Reference 15.

In all cases, operating stresses were calculated at shutoff head, which represents the highest pressure condition. An exception is the line shaft, where the highest stress occurs at normal head conditions.

The results of the calculations of the natural frequencies and stresses are presented in tabular form (Tables 3.9-4, 3.9-6). Stresses have been calculated at all points considered critical in the pump assembly.

The discharge head and motor assembly are designed to have a fundamental frequency below the operating speed of the pump. The design has a frequency of 11.96 Hz in the N-S direction and 12.47 Hz in the E-W direction. This is 63 percent of the operating frequency of 19.75 Hz which is considered to be a safe margin.

In the N-S direction, the frequency of the head assembly is 83.5 percent of the intake building frequency and in the E-W direction, 61.2 percent. The stresses associated with the inputs for these frequencies result in values that are well within the limits for the materials used. A damping factor of 5 percent was used in the calculations, as the assembly has three bolted flanged joints.

It is not advisable to reduce the frequency of this assembly, as the resulting deflections would approach the limit which would result in loss of function or damage to the packing box bearing. A study was made to determine the feasibility of raising the frequency well above the operating speed and building frequency, but it was determined that sufficient stiffness could not be built into the assembly to guarantee vibration-free operation during normal running of the pump.

The natural frequencies of the column assembly were determined for two different conditions: low and high water levels, and N-S direction. The highest stresses and moments occur during the high water condition and with the N-S earthquake. At this condition the fundamental frequency is 2.25 Hz, and the first harmonic is 16.64 Hz which falls on the peak of the response

curve at 1.94g. The maximum deflection in this mode is 0.022 inches, so no problem exists because of this coincidence of frequencies.

The vertical fundamental frequency of the line shaft is well above the peak seismic input. Considerable damping exists in this direction from the water in the pump impellers so that deflections cannot build up to a large value.

An examination of the stress tabulations shows that actual stresses are well within code and specification values for both postulated earthquake conditions. As most of these stresses were calculated on the assumption of the pump operating at shutoff head, or maximum pressure and thrust, they are less during the normal operation at rated head and capacity.

In addition to the stress limitations, a limitation is imposed on maximum deflection to a value of 80 percent of that which could cause loss of function. For the stationary parts of the pump, such as the column pipe and motor base, the stress limits would be reached first. For the rotating parts, such as shaft and bearings, excessive deflections could cause bearing seizure or high stresses in the rotating shaft. Stresses in the shaft due to the bending of the pump both above and below the baseplate have been included in the calculation of shaft stresses and add to the total stress.

The bearing at the packing box takes the largest load due to the pump deflections as the shaft is forced over an amount equal to the deflections of the housings. This load has been calculated to be about 42 pounds from the deflection of the column, and 87 pounds from the deflection of the motor base, or a total of 129 pounds. The bearing area is over 20 sq.in., so the added load is about 6.5 psi on the bearing which is well within the capability of the bearing at this point.

The calculations made indicate that the stress levels for all conditions of operation, when combined with the seismic stresses, are well within the limits set forth by the specifications and applicable codes. Although stresses were not calculated for all locations in the pump assembly, the stresses in locations not selected should be well below those calculated. A review of the drawings of the pump indicates that the design is one that should experience no difficulty in the event of an earthquake of the type and magnitude postulated. The stresses and deflections are such as to be satisfactory for the intended use as outlined by the design specifications.

3.9.2.9.8 Air Operated Valves

a. Butterfly valves (Nos. 37, 40 and 42)

Since the valve has been Code classified, the rules for design are those stated in Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components.

Since the valve is to be installed such that the distance from the body to the top of the actuator is extended, the length of this extension is the length of the appurtenance, and these parts--bracket, plate and actuator--are the appurtenances.

In accordance with the Code, the piping system, and not the valve, is limiting. In this case, the piping system must be considered the limit with respect to the valve.

On this basis, an analysis can be performed with respect to the effect of seismic loadings on:

1. Actuator yoke

2. Bracket-plate mounting connection

3. Body-bracket bolted flanged connection

The seismic force of 3.0g is considered as acting through the center of gravity of the extended masses. The vertical and horizontal forces and moments based on operational and seismic loads are calculated.

To determine the natural frequency, the actuator (and components) are analyzed as a cantilever beam. The lump parameter technique is followed in which the weight is considered as acting through the center of gravity and the vibrations are attributed to the elasticity of the component mounting.

The results of this analysis are as follows:

Component	Material	Actual Stress, psi	Allowable Stress, psi	Item No
Bracket	Gray C.I.	4640	10,000	37
Bracket bolts	A193 Gr.B7	17,147	32,600	37
Natural frequency = 178 cps				37
Bracket	AISI 1018	34.075	44,000	40
Bracket bolts	ASTM 325	24,900	30,700	40
Natural frequency = 55 cps				40
Bracket	Gray C.I.	4672	10,000	42
Bracket bolts	A193 Gr.B7	22,916	32,600	42
Natural frequency = 166 cps				42

b. Ball Valves (No. 18)

Since the valve has been Code classified, the rules for design are those stated in Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components. Since the valve is to be installed such that the distance from the body to the top of the actuator is extended, the length of this extension is the length of the appurtenance, and these parts-manual operator and actuator are the appurtenances. In accordance with the Code, the piping system, and not the valve, is limiting. In this case, the piping system must be considered the limit with respect to the valve. On this basis, an analysis can be performed with respect to the effect of seismic loadings on:

1. Actuator mount

2. Actuator to body neck connection

The seismic force of 3.0g is considered as acting through the center of gravity of the extended masses. The vertical and horizontal forces and moments based on operational and seismic

loads are calculated. The maximum stress occurs at a section of the valve body through the body neck.

To determine the natural frequency, the actuator is analyzed as a cantilever beam. The lump parameter technique is followed in which the weight is considered as acting through the center of gravity and the vibrations are attributed to the elasticity of the component mounting. The results of this analysis are as follows:

Component	Material	Actual Stress, psi	Allowable Stress, psi
Valve body	SA351-CF8M	13,260	25,870

Natural frequency = greater than 33 Hertz (minimum allowable frequency)

c. Butterfly Valve (No. 26)

Seismic Qualification for valves DH13A, DH13B, DH14A, and DH14B was performed by analysis. The natural frequency of the valve assembly was determined to be greater than 33 Hertz. Stresses for the valve design due to operation and seismic loads are within ASME B&PV Section III Code acceptance limits. The valve assembly is qualified in accordance with ASME B&PV Section III Code, 1971 Edition.

d. Globe valves (Nos. 30, 49, 56, 59, 63, 64, and 65)

The rules of design are those stated in Paragraph NC-3511 (No.30) and Paragraph ND-3511 (Nos. 49, 63, 64 and 65) of Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components.

The rules stated apply to the pressure boundary parts of the valve, in accordance with ANSI B 16-5; i.e., Minimum Wall Thickness and Pressure Temperature Ratings.

Therefore, the extended portions of the valve were analyzed to comply with Paragraph NA-3354, NB-3524, and Paragraph NB-3546 of the Code.

Since the valve is to be installed such that the distance from the body to the top of the actuator is extended, the length of this extension is the length of the appurtenance, and these parts--bonnet, flange, and actuator--are the appurtenances.

In accordance with NB-3524, the piping system and not the valve is limiting. In this case, the piping system must be considered the limit with respect to the valve.

On this basis, an analysis can be performed with respect to the effect of seismic loadings on:

1. Actuator yoke and clamp nut
2. Bonnet flange
3. Body

The seismic forces are considered as acting through the center of gravity of the extended masses. The vertical and horizontal forces and moments are determined using a seismic loading of 3.0g.

Yoke stresses:

This analysis of the columns in the actuator yoke is limited to those simple stresses created by the horizontal and vertical seismic loads and the normal operating load developed by the valve actuator. All axial and flexial loads are considered as acting simultaneously. The combined stresses are determined from these loads.

Clamp nut stress:

This analysis of the stresses in the clamp nut is limited to the tensile loads created by the axial loads of the valve actuator, and the bending moments transmitted through the yoke due to the horizontal seismic load.

To simplify the calculations, the effect of the flexial loads is grossly exaggerated. The load imposed by the bending moment of the yoke is considered as concentrated at a point. Also, in these calculations, only 50 percent (66 percent for Nos. 63, 64, and 65) of the thread engagement is used since the shearing stresses are more highly concentrated on the first few threads.

Bonnet flange bolting:

The seismic loading is combined with other live and dead loads and is applied as additional pressure to the flange. The formula from NB-3647-1 is used.

Operating condition loads are determined per Paragraphs VA-49(b) (1) and (2) and VA-49(c) from Reference 36.

Results -	No. 30	No. 49	No. 63	No. 64	No. 65
Actuator yoke					
Material	------------------ASTM A-395 Nodular iron---------------------—				
Stress intensity, psi	24,931	14,677	13,293	14,677	7,644
Allowable stress, psi			30,000		
Clamp nut					
Material			AISI 1020		
Stress intensity, psi	14,151	17,603	10,215	12,221	7,233
Allowable stress, psi	---------------------------------------18,750--------------------------------------				
Bolting					
Material	SA-564 - Type 630	------------------SA-193 Grade B7-----------------------------			
Stress intensity, psi	13,200	24,000	24,060	22,212	18,941
Allowable stress, psi	33,900	25,000	25,000	25,000	25,000
Natural frequency, cps	80.6	88.7	57.1	38.2	72.2

e. Diaphram valves (Nos. 43, 44, 45, 46, 47,and 60)

Loading Conditions:

The valves are designed to withstand normal operating loads plus a simultaneous seismic loading of 3.0g in any direction without exceeding code allowable stresses.

It is assumed that the valve is subjected to maximum pressure for its class rating. The valves are also assumed to be subjected to the maximum force from the actuator spring.

Design Data:

The diaphragm valve bodies and bonnets are constructed to ANSI B16.5 (1968), 150 lb. primary pressure rating minimum wall thickness. This satisfies the requirements of Subarticles NC-3500 and ND-3500 of Section III of The ASME Boiler and Pressure Vessel Code for Class 2 and Class 3 valves.

Methods and Assumptions:

This analysis assumes the valve assembly, from the bonnet flange up, to be a cantilever with transverse and axial loading.

The assembly was investigated at critical sections corresponding to points with minimum cross-sectional areas and section moduli. The loading conditions are then applied to the center of

gravity of the mass above the section under investigation, and the resultant stresses are found at the section. Due to their relatively small mass and short moment arm, the torsional contribution of nonsymmetrical parts is neglected.

This analysis employs classical strength of materials equations for the computations at all sections, except for those through the bonnet wall where stresses are induced by internal pressure. For these pressure stresses, the thick-wall cylinder equations are used (Reference 36, Article A-2000).

The bolted joints employed in the valve assemblies are of the soft or gasketed type of construction. The tensile load in the fasteners is due to preload plus the applied lead multiplied by a modifying stiffness coefficient, which is dependent on the gasket design. Bending loads are assumed distributed through the bolted joint proportionally.

Bending stress in the adaptor bushing is computed, using an effective section modulus. This effective section modulus is found by taking into account the stiffening effect of that portion of the adaptor bushing flange which is on the compressive side of the neutral axis.

Both the yoke and the bonnet have relatively constant cross sections and section moduli. These parts are evaluated at a point where the applied moment is the greatest.

Allowable Stress

The Code requirement concerning allowable stresses is applicable only to certain components of the assembly. Those components which are specifically not covered by the Code are listed in Subparagraphs NA-1130, NC-2121 (b), and ND-2121 (b). They include valve operators, shafts, bushings, gaskets, and non-pressure retaining accessories. Therefore, only pressure retaining materials of the body, bonnet, and body-bonnet fasteners are covered by the allowable stresses given in Table I-7.0 of the Code. All other components use allowable stresses consistent with good engineering practices.

Natural Frequency:

The natural frequency calculations are made assuming a single-degree-of-freedom system with the yoke being the only elastic member. For the purposes of both natural frequency and stress calculations, the yoke stiffness is taken as twice the stiffness of the individual arms.

It has been found that the yoke is more flexible in one plane than another. For the determining of the natural frequency, the section moment of inertia which yields the lowest natural frequency was used. The loading is the worst case which gives the greatest natural period. This case is the heaviest yoke-air motor combination that gives the greatest value of mass as all other variables are constant.

For all valves, the minimum natural frequency was calculated to be 74 cps.
Conclusions:

Component	Material	Yield Strength, psi	Allowable Stress, psi	Maximum Stress Intensity, psi
Yoke	Ductile Iron ASTM A-445 60-40-18	40,000	24,000	7462
Adapter bushing	AISI-12L14	60,000	36,000	18,744
Bonnet	SA-351 CF8M	30,000	17,500	11,946
Bolting	SA-453 GR 660	85,000	21,250	18,648

f. Gate Valves (No.55)

The analytical method is as indicated in Subsection 3.9.2.9.2a.

g. Globe Valves (No. 70)

The analytical method is as indicated in Subsection 3.9.2.9.2a.

h. Ball Valves (No. 14)

Since the valve has been code classified, the rules for design are those stated in Section III, ASME Boiler and Pressure Vessel Code, Nuclear Power Plant Components. The valves were purchased under the rules of Generic Letter 89-09.

Since the valve is to be installed such that the distance from the body to the top of the actuator is extended, the length of this extension is the length of the appurtenance, and these parts--manual operator and actuator--are the appurtenances.

The valve bracket and valve body were seismically analyzed using finite element analysis methods. Hand calculations were used to analyze the following internal components:

Actuator cylinder end cap screws
Actuator cylinder pipe
Actuator base cap nipple
Actuator piston stud
Actuator piston rod (shaft)
Actuator rod (shaft) bolt
Actuator bearing unit
Actuator connection arm
Actuator lever arm & stud
Actuator/bracket bolts
Bracket/body bolts

The seismic accelerations for each valve component were either determined by trial and error or were back solved by setting the actual stresses equal to the allowable stresses. Based on this analysis, it was determined that the limiting valve acceleration is based on the actuator piston rod (shaft). The allowable SSE accelerations for this component are 2.2g in the x and z directions and 1.9g in the vertical direction.

The controlling frequency of the assembly was determined to be in excess of 20 cps.

Based on the above, the actual valve accelerations are enveloped by the above valve accelerations and no stresses generated exceeded the material allowables.

3.9.2.9.9 Decay Heat Pumps (No. 25)

The pumps are analyzed for normal operating conditions, the Maximum Probable Earthquake, and the Maximum Possible Earthquake using the following criteria for acceptance:

a. The safety factor is equal to or greater than 1.5 based on yield strength for the Maximum Probable Earthquake or 1.1 for the Maximum Possible Earthquake.

b. The operating speed is not within 20 percent of the critical speed.

c. The shaft deflection at the coupling does not cause unacceptable misalignment.

d. The calculated stresses do not exceed the allowable stresses.

Results:

Component	Maximum Stress, psi	Allowable Stress, psi
Suction nozzle	2,184	30,000
Discharge nozzle	2,278	30,000
Foundation bolts	12,480	105,000
Anchor bolts (pump)	31,228	105,000
Anchor bolts (motor)	6,813	105,000

3.9.2.9.10 High Pressure Injection (HPI) Pumps (No.28)

The HPI pumps are analyzed for normal operating conditions, the Maximum Probable Earthquake, and the Maximum Possible Earthquake using the following criteria for acceptance:

a. The safety factor is equal to or greater than 1.5 based on yield strength for the Maximum Probable Earthquake or 1.1 for the Maximum Possible Earthquake.

b. The operating speed is not within 20 percent of the critical speed.

c. The shaft deflection at the coupling does not cause unacceptable misalignment.

d. The calculated stresses do not exceed the allowable stresses.

Results:

Component	Maximum Stress, Psi	Allowable Stress, psi
Suction nozzle	15,930	30,000
Discharge nozzle	16,820	30,000
Foundation bolts	21,107	105,000
Anchor bolts (pump)	37,319	105,000
Anchor bolts (motor)	7,657	105,000

3.9.2.9.11 Containment Spray Pumps (No. 32)

The seismic condition is the presence of horizontal and vertical ground accelerations over and above the steady state conditions. The resultant acceleration manifests itself in the pump as additional load on the parts. The pumps have been designed to prevent the following events:

a. Pump and driver misalignment due to momentum forces causing pump displacement.

b. Shaft failure or excessive displacement due to momentum forces.

c. Shaft natural frequency equal to seismic frequency.

Shaft:

Using the acceleration response spectrum curves at the appropriate floor level for the Maximum Possible Earthquake conditions, the resultant acceleration and hydraulic radial reaction are calculated. From this, the maximum shaft deflections are determined at the impeller centerline and seal races. The total shaft bending stress and shear stress due to the torsional moment are calculated. From this, the combined stress is determined.

The natural frequency of the shaft is also determined.

Hold down bolts:

The inertial force on the bolts is determined and compared with the force due to assembly torque. The frictional force between the base of the pump is calculated to assure that pump displacement does not occur. In addition, the natural frequency of all supports is calculated to assure that this value exceeds the frequency range of large seismic accelerations.

Results:

		Stress, psi	Deflection, inches	
			Maximum	Seal
Shaft 316SS	Actual	5541	0.006	0.003
	Allowable	30,000	0.0095	0.005

3.9.2.9.12 Component Cooling Pumps (No. 34)

Refer to Subsection 3.9.2.9.11 for analytical method and criteria. The results are as follows:

		Stress, psi	Deflection, inches	
			Maximum	Seal
Shaft SAE4340	Actual	2704	0.0047	0.0023
	Allowable	138,000	0.0115	0.005

3.9.2.9.13 Diesel Fuel Oil Day Tanks (No. 50)

The natural frequency for the tanks is calculated. From the acceleration response spectrum curves at the applicable floor level, the horizontal and vertical "g" forces are determined. References 37 and 38 were used for the stress calculations.

Saddles:

The longitudinal bending stress and tangential shear stresses are determined. The circumferential stress at the horn of the saddle and stress in stiffener in the head were also determined. In addition, the ring compression in the shell over the saddles was calculated.

Anchor Bolts:

Shear stress and maximum tensile stresses were calculated.

Results:

Saddles, SA-285, Grade C

Stress	Actual Stress, psi	Allowable Stress, psi
Longitudinal bending	1002	9596
Tangential shear	2788	10960
Circumferential stress at horn	26360	26438
Pressure in head	4745	17125
Ring compression	7762	14688

Anchor Bolts, A-307

Stress	Actual Stress, psi	Allowable Stress, psi
Shear	4315	4400
Tensile	2221	14670

3.9.2.10 Design Conditions for ASME Components

See Subsections 3.9.2.1, 3.9.2.2, and 3.9.2.3. No mechanical components are designed to Nuclear Class NE. According to Section III of the Code the operational cycles need only apply to the design of Nuclear Class NB only.

3.9.2.11 Analytical Methods and Criteria for ASME Components

See Subsection 3.9.2.9 for the analytical methods and criteria used to evaluate stresses in all pumps and valves. A summary of results is also given in the referenced Subsection. For all other components the following paragraphs apply (for component numbers refer to Table 3.9-2):

3.9.2.11.1 Auxiliary Feed Pump Turbines (No. 2)

See Subsection 3.9.3.

3.9.2.11.2 Service Water Strainers (No. 12)

A computer program based on the stiffness method of matrix structural analysis was utilized. Mode shapes and frequencies were obtained by matrix iteration.

Damping was assumed to be 1.0 percent of critical for the Maximum Probable Earthquake and 1.0 percent of critical for the Maximum Possible Earthquake. The E-W spectrum was used for the X direction. The N-S spectrum was used for the Y direction. For the Z direction, vertical acceleration was assumed to be 2/3 of the average of the two horizontal spectra.

The following assumptions were made: It is difficult to express loss of function of a strainer in terms of deflection. Therefore, it was assumed that if the strainer remained elastic there would be no loss of function. Assuming that at working stress levels there is a factor of safety of 1.67 against yield or buckling (e.g. AISC), then the loss of function stress is:

$$\sigma_{LOF} = 1.67\sigma a$$

Taking 0.8 of the loss of function stress gives

$$0.8\sigma_{LOF} = (0.8)(1.67)\sigma a$$

Thus for the maximum possible earthquake

$$\sigma MP/E = (1.34)\sigma_a$$

This criterion was used to evaluate tank shell and nozzle thickness requirements and to check the supports for the maximum possible earthquake conditions.

Tank Element:

The primary concern of this analysis is to determine the seismic effects on the tank shell and the nozzle connections to the tank shell, and to evaluate the adequacy of the supports. The internals are assumed to act as a unit with the tank shell. The tank geometric properties were taken as the sum of the tank and its internal components. This is a reasonable assumption based on the type of construction used for this configuration of strainer.

The mass of the tank and internals was lumped at five points along the vertical axis of the tank as well as at the nozzles for the inlet and outlet nozzles.

The nozzles were connected to the centerline of the tank with stiff elements having force-displacement characteristics equal to three times those of the tank along its own longitudinal axis. (The factor of three is arbitrary and is based on experience).

The points of attachment of the support legs to the tank are connected to the centerline of the tank by elements having force-displacement characteristics equal to ten times those of the tank along its longitudinal axis. This is essentially infinitely rigid.

A static analysis was carried out using pipe forces and moments. The structural model was the same as that used for the dynamic analysis with all elements west of the outlet nozzles deleted. The 20-inch pipe between the inlet and the pump was retained with the pump treated as an anchor.

The allowable stresses were taken as those for the Maximum Possible Earthquake.

Using σ_a = 21,500 psi (SA-515, Grade 70) and the combined axial and bending forces, the minimum nozzle wall thickness is determined.

Using σ_a = 17,500 psi (SA-53, Grade B) and the combined axial and bending forces, the minimum tank shell thickness is determined.

Support Element:

Supports are checked in accordance with the AISC Specification for design, fabrication, and erection of structural steel as contained in Reference (39).

3.9.2.11.3 Decay Heat Coolers (No. 24)

The Decay Heat Coolers were modeled with the attached piping system to obtain an accurate system response. A conservative equipment damping value of 1% was used for the Coolers. The piping damping values are in accordance with ASME Code Case N-411. The computer analysis utilized the appropriate floor response spectra for the frequencies of the system. Seismic forces were imposed in the three axis simultaneously and combined by the Square-Root-of-the-Sum-of-the-Square (SRSS). As a result of this analysis, nozzle loads, support loads, and Cooler accelerations were obtained. This information was then used in the qualification of the equipment.

The seismic qualification of the Coolers was performed in two separate analyses. The original manufacturer (Atlas Industrial Manufacturing) performed the qualification of the Cooler including the decay heat inlet/outlet and CCW inlet nozzle, the shell and the Cooler supports. The CCW outlet nozzle was qualified by Altran Corp. Three limiting loading conditions (upset and two faulted) were considered. They are: 1) Deadweight + normal operating thermal + OBE, 2) Dead weight + SSE, and 3) Deadweight + LOCA Thermal. Internal pressure loading was included with each of these loading conditions.

The CCW outlet nozzle as described below was analyzed utilizing the stress limits defined in Subsection ND-3321 of the 1977 ASME III Code. The remaining components were analyzed using the allowable stress intensity limits defined in the 1968 Edition of ASME III, Subsection A, Article 4. The results of these analyses are as follows:

DECAY HEATER COOLER

The Decay Heat inlet/outlet and CCW inlet nozzles, supports and shell were analyzed using the Welding Research Council Bulletin 107 methodology. Based on this analysis, the areas with the limiting stress intensities are:

Location	Material	Stress (psi) Intensity	Allowable Stress (psi) Intensity
Nozzle	SA-53, Grade B	23,693	28,050
Shell	SA-285, Grade C	50,403	54,000
Supports	SA-36	7,051	10,790

CCW OUTLET NOZZLE

An ANSYS finite element model of the CCW outlet nozzle juncture including the vessel shell, the nozzle, the flange and the shell reinforcing pad was performed. This analysis indicated the limiting stress are:

Location	Material	Stress (psi)	Allowable Stress (psi)
Nozzle	SA-53, Grade B	10,820	16,500
Shell	SA-285, Grade C	16,982	22,688

3.9.2.11.4 Component Cooling Heat Exchangers (No. 35)

The Component Cooling Water Heat Exchangers (Equipment no.E-22-1, 2 and 3) were qualified using a combination of hand calculations and finite element analysis using the ANSYS program. Altran Corporation performed this evaluation of the heat exchanger (reference Technical Report No. 97209-TR- 01, August 1998, and Toledo Edison calculation C-CSS-16.04-038).

The Component Cooling Heat Exchangers were modeled with the attached piping system to obtain an accurate system response. A conservative equipment damping value of 1% was used for heat exchangers. The piping damping values are in accordance with ASME Code Case N-411. The computer analysis utilized the appropriate floor response spectra for the frequencies of the system. Seismic forces were imposed in the three axis simultaneously and combined by SRSS. As a result of this analysis, nozzle loads, support loads, and heat exchanger's accelerations were obtained. This information was then used in the qualification of the equipment.

The qualification evaluated loads from internal pressure, deadweight, thermal, seismic and external nozzle loads. The heat exchangers were evaluated to the ASME Section III, Class 3 Vessel Code criteria and the supports were evaluated to the AISC Structural Code, 8th edition.

Based on the analyses performed, it was concluded that the OBE load combination was the controlling load case. The results of this analysis are as follows:

Location	Material	Stress (psi) Intensity	Allowable Stress (psi) Intensity
General Shell	ASTM A285, Gr. C	12,554	15,070
Shell at Nozzle	ASTM A285, Gr. C	21,994	22,605
Nozzle at Shell	ASTM A106, Gr. B	14,006	16,500
Shell at Supports	ASTM A285, Gr. C	16,973	22,605

3.9.2.11.5 Component Cooling Surge Tank (No. 36)

Considering the tank as a simple beam uniformly loaded, the maximum deflection at the center is calculated. Using Reference 20 the natural frequency was calculated to be 142.8Hz.

Shell:

The horizontal and vertical forces are calculated. The maximum longitudinal compressive and tensile stresses are calculated.

Saddle:

The saddle bearing stress, maximum compressive stress, and mold stress are determined.

Cross Bracing:

The maximum tensile and compressive stresses in the bracing are determined.

Results:

Component	Material	Actual Stress, psi	Allowable Stress, psi
Shell	A515, Grade 70	16,093	17,500
Saddle	A283, Grade C	601	15,000
Bracing	A-36	8473	20,000

3.9.2.11.6 Letdown Coolers (No. 48)

Linear elastic finite element analyses of the letdown coolers were made. Natural frequencies of the coolers were determined. For natural frequencies below 33 Hz, 2% damping SSE loads were imposed in three axes simultaneously. The resulting modal contributions were combined by the rules for closely spaced modes and the resulting stresses calculated. A static load case was also executed imposing the zero period acceleration plus 1.0 g dead weight in the vertical axis, along with pipe loads and design pressure loads.

Maximum cooler component stresses were determined to be:

Component	Material	Actual Stress, psi	Allowable Stress, psi
Base	A-36	9,507	27,000
Case	SA516 GR. B	3,242	42,000
End Plate	SA516 GR. B	6,851	42,000
Connections	SA182, 316L	22,276	37,440
	or		
	SA106 GR. B		36,000

Four Anchor Bolts were used to anchor each cooler. Maximum bolt tension and shear were determined to be no greater than 4,820 lbs. and 2,400 lbs., respectively. Allowable bolt tension and shear were based on ultimate strength provisions. The interaction ratio for shear plus tension is .67.

3.9.2.11.7 Hydrogen Dilution Blowers (No. 53)

A dynamic test was performed on the hydrogen dilution blowers and moisture separator assembly. The test was performed on a shaker system with base connections identical to the final installation configuration. The input acceleration for the shaker system is the maximum acceleration experienced by the structure at the appropriate floor elevation due to the time history method of analysis.

Pre-Shaker Table Testing:

A performance test and hydrostatic test were conducted prior to shaker tests.

Shaker Table Test Procedure:

The unit (pump and motor on base with coupling and coupling guard) was tested as follows in each of three mutually perpendicular directions (vertical, major horizontal-parallel to pump and motor shafts, and minor horizontal - perpendicular to pump and motor shafts). During tests, the pump was not running or operating in any way.

A frequency sweep from 1-35-1 Hz was made. Input level was 0.1g in the vertical mode and 0.184g in the major and minor horizontal modes. These values are based on acceleration curves @ 15 percent g at the applicable floor elevation.

Output graphs were analyzed for signs of resonance. A resonance can be determined from the 'Q' value.

$$Q = \frac{\text{Output response (g)}}{\text{Input response (g)}}$$

Once "Q" values were calculated, they were plotted as Q values vs. frequency. Each peak Q value was considered as a resonance.

The separator was subjected to vibration tests similar to but separate from the pump tests. During tests, the separator was filled with water to the point that the ball float assembly begins to drain. The drain was then plugged to simulate a normal operating level of water in the separator.

Dwell Testing:

All dwell testing was subjected to both horizontal and vertical motions occurring simultaneously. The horizontal input to the shaker table was 0.22g while the equipment subjected to the seismic test was mounted on 33.7° angle from horizontal. This results in a simultaneous input of 0.184g horizontal and 0.1226g vertical.

The dwell testing was performed in two axes (major horizontal, minor horizontal). The frequency, at which the dwell testing was conducted, was obtained from the Q value vs. frequency plot. Dwell testing was performed at each Q value peak, but not more than three dwells per axis. Also, dwell testing was performed at frequencies 6.0, 7.5, 18.0, and 21.0Hz.

The duration of the dwell test was thirty seconds.

Post-Shaker Table Testing:

The performance test was repeated.

Minor variations (±5 percent) in performance characteristics do not constitute failure of the units to demonstrate seismic adequacy.

Pump and separator (with ball float assembly plugged) were given the same hydrostatic test as in the pre-shaker table testing. Pump and motor bearings were replaced with new bearings following completion of the above tests.

The pump was given a final hydrostatic test with ASME Code inspector as witness.

Pump, motor, and separator were then given an operational check.

Qualification:

Successful completion of the above tests and acceptance by the vendor Quality Control and Engineering Departments, constituted qualification of this design and demonstration of seismic adequacy.

3.9.2.12 Storage Tank Design Conditions

The code, load combination, and stress limits for waste gas storage tanks which are required to prevent or mitigate the consequences of accidents and retain radioactive material are shown in Table 3.9-7. These tanks are not required for safe shutdown of the reactor.

3.9.3 Components Not Covered by ASME Code

3.9.3.1 Mechanical Components

The following safety-related mechanical components are not covered by the ASME Boiler and Pressure Vessel Code:

a. Containment air coolers

b. ECCS room cooling coils

c. Control room emergency condensing units

d. Emergency Diesels

e. Emergency Diesel heat exchangers

f. Borated Water Storage Tank

g. Auxiliary feed pump turbines

3.9.3.1.1 Stress and Dynamic Calculations

Non-ASME safety-related mechanical components are either covered by other applicable codes (such as AWWA, TEMA, AMCA, etc.), or the specifications for materials, construction, and

testing of the components are such that they are compatible with the associated ASME code components in the system. The design conditions for these components are based on the dynamic and seismic requirements for structural integrity and functional capability in the event of an accident.

A summary of the stress and dynamic calculations of experimental testing, performed to demonstrate that all design load combinations are sustained without impairment of structural integrity or functional capability, is as follows:

a. Containment air coolers, ECCS room cooling coils, and control room emergency condensing units

An analytical approach was used to check the structural integrity of the components. First, a static analysis of the structure was performed to obtain forces, moments, and displacements under gravity and peak pressure differential loads. For the earthquake-induced loads, a modal analysis was performed to determine the dynamic behavior of the system in the frequency range of interest. A response analysis, using specified floor response spectra, was then performed on the structure to find the total deformation of the structure and corresponding member forces and moments. The combined results from the static and dynamic analyses were then used to compute total member stresses.

The thermal environment is such that thermally induced stresses are negligible in comparison with stresses produced by operating loads and seismic loads. Therefore, they were not included in the analysis.

b. Emergency Diesels and heat exchangers:

A seismic analysis for the Emergency Diesel generator and its auxiliaries was performed to establish that all components function during and after a Maximum Probable (smaller) Earthquake of 0.08g and the Maximum Possible (larger) Earthquake of 0.15g. Dynamic tests were performed on the control panels using the shaker test table and established that these panels were operable during and after an earthquake.

Analysis of the diesel engine parts established that the equipment was rigid and the natural frequencies of the components were in the safe range. It was also proven that the fasteners securing all the components did not fail during any seismic event. The generator was analyzed for the rotor stresses and deflections, bearing loading, and generator mounting.

A comprehensive predelivery test was performed at the manufacturing plant before the unit was shipped to the job site. This included the prestart inspection of all components, various temperature rise tests, load tests, heat run tests, paralleling tests for both units, and starting air tests, etc. The units were also subjected to detailed pre-operational tests and periodic tests to ensure reliability.

c. Borated Water Storage Tank:

Refer to Subsections 3.7.1.6 and 3.7.2.3 for seismic analysis of tank structure. Loads imposed on the tank nozzles by piping are determined by an analysis of the operating and seismic forces and are maintained below the nozzle allowable forces and moments.

d. Auxiliary feed pump turbines:

The seismic loading used throughout the loading and design analysis is represented by equivalent static coefficients in the horizontal and vertical directions. Multiplying these coefficients by the weights of the affected components produces the equivalent static forces for the design and analysis of the components.

Since the subject turbines are relatively rigid, and are located in a low elevation of the auxiliary building, the coefficients used conservatively account for any amplification effects. A 2 percent damped acceleration floor response spectrum at the elevation of the floor supporting the turbine shows peaks of less than the 1.5g assumed. Therefore, it can be concluded that the horizontal and vertical coefficients used in the analysis contain significant margins of safety.

The unbalanced forces due to inlet-outlet arrangement are not considered, but are so small as to be relatively insignificant. Thermal and other start-up transients have also been neglected since they are of short duration and can readily be accommodated by the margins of safety that were found to exist. Employing conventional procedures, only the code allowable stresses were used in the nozzle calculations to determine primary stresses. Thermal and other start-up transients can be accommodated by the higher allowable stresses permitted for transient and secondary loadings.

The failure criterion used was predicated on the assumption, that the effective elastic limit of the material in question should not be exceeded, and that there would be no loss of function due to seismic loadings.

In order to evaluate the seismic capability of the turbine in the running mode, it is necessary to identify the critical operating components in the turbine. This identification is summarized in Table 3.9-8. The critical items have been segregated into three groups:

Group 1: Items which can be evaluated using analytical techniques.

Group 2: Items which, because of their complexity, can be evaluated by inference or analogy.

Group 3: Items which, because of their construction the nature of the earthquake motion, can be assumed adequate prima facie

Group 1 items listed in Table 3.9-8 have been analyzed for the specified seismic loadings and have been found to conform to the allowable criteria.

Group 2 items have not been specifically analyzed for seismic resistant capability. However, there is much evidence to indicate that these items would all perform satisfactorily during a seismic event. First, these items are subjected to considerable shock loadings during their handling, shipping, and installation. Second, prior to leaving the factory, the entire turbine assembly is tested for operational requirements. During this testing, but incidental to the testing, these items are subjected to considerable vibration and continue to perform as required. Third, all components are subjected to considerable shock loading when the turbine is shipped to the site. After installation at the site, the turbine is again tested for operational performance and Group 2 items still perform as required. Finally, a visual inspection of these items indicates that they are actually constructed to a way that appears to make them insensitive to vibratory motions. It is concluded that Group 2 items are capable of withstanding the specified seismic loadings and still continue to function during such an event.

Group 3 components which, because of their construction and the nature of the earthquake motion, can be assumed to be adequate prima facie; such as the control linkages which are light elements requiring only loose tolerances to function properly. Inertia forces due to large earthquake accelerations, therefore, are quite small. Furthermore, even should some nominal seismic displacements occur, the linkages are constructed so that they can accommodate large relative displacements and still function smoothly. Inertia forces associated with the stop valve switches are also small because of the small masses. Therefore, the item appears quite insensitive to seismic motion. The only other critical item that appears to fall into this category is the filter material in the oil filter which, by its very nature, is insensitive to seismic motion.

3.9.3.2 Fuel Rods

The fuel design data for the most current cycle is described in the applicable reload report and associated approved B&W Fuel Company topical reports. Commencing with Cycle 13, topical report BAW-10227P-A, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel” (Reference 85) approval by the NRC allowed use of M5 alloy for use in cladding and structural components (i.e. end caps, spacer grids and guide tubes). The following is a description of the original design.

The Zircaloy-4 cladding is designed to withstand strain resulting from the combined effects of reactor pressure, fission gas pressure, and both fuel thermal expansion and irradiation growth.

3.9.3.2.1 Clad Strain Resulting from Normal and Upset Conditions

a. Primary Stresses, which are not relieved by small material deformations, are limited so as not to exceed either the yield strength of the material or 75 percent of the stresses rupture life of the material. An example of such stress is the circumferential membrane stress in the clad due to internal or external pressure.

b. Secondary stresses, which are relieved by small material deformations, are permitted to exceed the yield strength. Strain limits for this stress condition are based on low-cycle fatigue techniques, not to exceed 90 percent of material fatigue life. Evaluation of cyclic loading is based on conservative estimates of the number of cycles to be expected. An example of this type of stress is the thermal stress resulting from thermal gradients across the clad thickness.

c. Combinations of these two types of stresses, in addition to the individual treatment outlined above, are evaluated for low-cycle fatigue. Clad circumferential plastic strain, due to diameter increases resulting from fuel swelling, thermal expansion, creep, and differential pressure, is limited to 1 percent.

3.9.3.2.2 Minimum Clad Collapse Pressure Margins

a. A ten-percent margin was used over system design pressure, on short- time collapse, at fuel rod end voids. End voids must not collapse (must be either free-standing or have adequate support) on a long-time basis.

b. A ten-percent margin was used over system operating pressure, on short-time collapse, at hot-spot average temperature of the clad wall. Clad must be free-standing at system design pressure on a short-time basis at the hot spot average temperature through the clad wall.

3.9.3.3 Fuel Assembly Components

The fuel design data for the most current cycle is described in the applicable reload report and associated approved B&W Fuel Company topical reports. Commencing with Cycle 13, topical report BAW- 10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (Reference 85) approval by the NRC allowed use of M5 alloy for use in cladding and structural components (i.e. end caps, spacer grids and guide tubes). The following is a description of the original design.

Primary stresses in fuel-supporting components, such as end fitting, spacer grids, guide tubes, and holddown components, are limited to 85 percent of the material yield strength for normal and upset operating conditions. These limits are determined by mathematical analyses and testing of the individual components.

3.9.3.4 Shock and Seismic Loadings

The fuel design data for the most current cycle is described in the applicable reload report and associated approved B&W Fuel Company topical reports. Commencing with Cycle 13, topical report BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (Reference 85) approval by the NRC allowed use of M5 alloy for use in cladding and structural components (i.e. end caps, spacer grids and guide tubes). The following is a description of the original design.

The following limits are not exceeded for the Maximum Probable Earthquake:

a. Loads on the fuel assembly spacer grid do not exceed the elastic load carrying capacity of the spacer grid as determined by production grid testing.

b. There is no permanent deformation of the fuel assembly spacer grid.

The following limits are not exceeded for the Maximum Possible Earthquake, Loss-Of-Coolant Accident (LOCA), and combined Maximum Possible Earthquake:

a. Loads on the fuel assembly spacer grid shall not exceed the loads which would permanently distort the guide tubes and prevent control rod insertion as determined by production spacer grids testing.

b. Loads on the control rod guide tubes and end spacer grid assembly do not exceed 85 percent of the Euler critical buckling load.

c. Loads on the connections of the end spacer grid skirt to the end fitting is limited to 85 percent of the load to produce connector shear failure as determined by production component testing. The design and method of analysis for seismic and LOCA conditions are provided in B&W Topical Report BAW-10041.

3.9.3.4.1 Fuel Assembly Seismic and LOCA Analyses

The fuel assembly analysis for seismic and LOCA conditions is based on the methods presented in B&W Topical Report BAW-10041. The criteria, methods, model, and limiting components as described are directly applicable to the Davis-Besse Nuclear Power Station (DBNPS) evaluation.

The loading conditions investigated are: (1) Maximum Probable Earthquake based on the acceleration response spectrum envelopes of 0.08g horizontal (N-S and E-W) and 0.053g vertical curves provided for the Davis-Besse Nuclear Power Station (DBNPS) site; (2) Maximum Possible Earthquake based on the acceleration response spectrum envelopes of 0.15g horizontal (N-S and E-W) and 0.10g vertical; (3) a LOCA based on an instantaneous reactor coolant pipe rupture; and (4) the worst case occurrence for a combined Maximum Possible Earthquake and LOCA.

The maximum loads or deflections occurring in the fuel assembly are determined for each loading condition using the model investigations, as presented in BAW-10041, for different combinations of reactor operating states. The resulting margins of safety are tabulated and demonstrate the ability of the fuel assemblies to withstand the postulated accident conditions without exceeding the respective allowable limits.

The results are presented in the two directional areas of investigation for both accident condition levels.

a. Horizontal - contact between fuel assembly spacer grids due to the assemblies motion in the horizontal plane with the contact occurring primarily at the mid-span spacer grids. The results presented are based on time history analysis with input spectra which exceed the Maximum Probable Earthquake and Maximum Possible Earthquake response spectra specified for the Toledo site. The combined Maximum Possible Earthquake + LOCA is based on the worst case LOCA (inlet nozzle rupture with the leak area restrained to 3 ft.^{2}) occurring simultaneously with the Maximum Possible Earthquake and timed to provide maximum load reinforcement. The results of the analysis described in BAW-10041 provide the safety margins for the accident conditions as follows:

Accident Condition	Safety Margin - %
Maximum Probable Earthquake	300%
Maximum Possible Earthquake	350%
Maximum Possible Earthquake + LOCA	175%

b. Vertical - contact between the fuel assemblies and the internals due to upward pressure and/or seismic motion with the primary contact occurring between the end fittings and grid plates. The results are based on the methods presented in BAW-10041 using time history analysis for input spectra which exceed the Maximum Probable Earthquake and Maximum Possible Earthquake response spectra specified for the Toledo site. The LOCA and combined LOCA + Maximum Possible Earthquake results are provided for the worst loading condition applicable to each area of investigation. The safety margins for the fuel assembly component loading are presented in Table 3.9-9.

These results demonstrate the ability of the Fuel Assemblies to meet the Toledo site accident conditions with a minimum margin of safety of 300% for the Maximum Probable Earthquake level and 140% for a Maximum Possible Earthquake level.

3.9.3.5 Control Rod Assemblies (CRAs) and Extended Life Control Rod Assemblies (ELCRAs)

Commencing with cycle 9, those CRAs that have approached design life will be replaced by ELCRAs.

The control rods are designed to withstand all operating loads, including those resulting from hydraulic force, thermal gradients, and reactor trip deceleration. Excessive stress in the CRA as well as the ELCRA components, is prevented by conservative design stress limits and by hydraulic snubbing in the drive mechanism. The ability of the control rod clad to resist collapse has been established in a test program on cold-worked stainless steel tubing. The ability of the ELCRA's inconel cladding to resist collapse has been assured by the initial pre-pressurization of the control rods to 465 psia. Because the Ag-In-Cd alloy poison does not yield a gaseous product under irradiation, internal pressure and swelling of the absorber material do not cause excessive stressing or stretching of the clad. Because of their length and the possible lack of straightness over the entire length of the rod, some contact between control rods and the fuel assembly guide tubes is expected. However, the parts involved, especially the control rods, are flexible, therefore only small friction drag loads result. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. Consequently, control rod assemblies do not encounter significant frictional resistance to their motion in the guide tubes. The insertability of the control rods is evaluated by analyzing the deformation of the fuel assembly as a result of seismic excitation and LOCA and presented in B&W Topical Report BAW-10041.

3.9.3.6 Gray Axial Power Shaping Rod Assembly (APSRA)

The Gray Axial Power Shaping Rod Assemblies (APSRAs) are designed to accommodate loading conditions for normal operation and operational transients, i.e., Condition I events; those that are expected to occur frequently or regularly, as defined in ANSI/ANS 57.5-1981. Also considered are fatigue stresses due to loadings expected during more severe events of moderate frequency that may result in a reactor shutdown, i.e., Condition II events. Design conditions in ANSI/ANS 57.5 are for fuel assemblies for light water-cooled reactors and are not used in Chapter 15 accident analysis.

The design criteria for the Gray APSRAs assure that the structures remain within operational tolerances under Condition I and II events. The coupling/spider assembly stress analyses consider the stresses during those events and verify they are within allowable stresses from the ASME Code (1983 Edition), Section III, Subsection NG. Gray APSRA design criteria ensure that the cladding integrity is maintained and that deformation of the Gray APSRAs does not prevent their insertion.

The individual rods of the Gray APSRAs contain an inconel neutron absorber encapsulated in a stainless steel cladding. The Gray APSRA rod was designed with an increased clad thickness (thereby reducing the absorber-to-cladding gap) to provide margin for creep collapse. This design change from the initial plant design for axial power shaping rod assemblies, containing Ag-In-Cd alloy as the neutron absorber, was possible since the in-reactor expansion of the inconel absorber is considerably less than that for an Ag-In-Cd absorber. Swelling of the inconel absorber, either thermally induced or due to irradiation is negligible.

Furthermore, to prevent cladding collapse, the rods are pressurized with helium. The helium also enhances conduction of the heat, from the gamma heating of the neutron absorber, so that

the absorber's maximum centerline temperature limit (i.e., below the melting temperature of inconel) is not reached during Condition I and II events.

3.9.3.7 Orifice Rod Assembly (ORA)

The orifice rods are designed to withstand all operating loads, including those resulting from hydraulic forces and thermal gradients. ORAs are no longer in use.

3.9.4 ASME Code Cases

Table 3.9-10 delineates ASME Section III Code Case interpretations applied to Nuclear Class 1, 2, and 3 components.

In addition to the code cases delineated in the Table 3.9-10, the Davis-Besse Nuclear Power Station (DBNPS) adopts other code cases that are listed in the following Regulatory Guides and complies with the Nuclear Regulatory's positions contained therein:

1. USNRC Regulatory Guide 1.84 - Design and Fabrication Code Case Acceptability ASME Section III Div. I

2. USNRC Regulatory Guide 1.85 - Materials Code Case Acceptability ASME Section III Div. 1

Prior to adopting such code cases, the DBNPS ensures that the code does not impact or violate other regulatory or jurisdictional requirements and that there are no conflicts with other code or design limitations resulting from code case usage.

Identification, justification and approval for the particular application of each code case is processed and documented in accordance with existing DBNPS Engineering and Procurement Procedures.

TABLE 3.9-1

Ratio of Stress to $2.4S_m$ and Fatigue Usage Factor[7]

Node Number[1] (see Figure 3.9-4)	Ratio of Calculated[2] Stress to $2.4S_m$	Ratioed Fatigue[3] Usage Factor (U/.1)
HOT LEG		
35	.431	.080
52	.518	.086
53	.438	.076
58	.440	.058
70	.520	.065
71	.924[4]	.780
87	.443	.065
89	.519	.026
104	.441	.066
107	.569*	.084
111 (Terminal End)	.501	.180
122	.592*	.035
125	.611[5]	.157
130	.522	.090
136	.555	.043
139 (Terminal End)		
UPPER COLD LEG	.460	.154
21	.397	.007
31	.614	.036
38	.475	.031
45	.640	.041
47	.669*	.049
51	.425	.034
56 (Terminal End)	.525	.065
69	.982*	.143
83 (Terminal End)	.495	.041
LOWER COLD LEG		
43 (Terminal End)	1.003	.217
55	.714[6]	.185
61	.794*	.139
79	.518	.058
94	.546*	.045
105 (Terminal End)	.775	.131

1. Location where loads and stresses are determined by calculation.
2. Stress calculated by Equation 13 per NB-3653.6 part b of the ASME Code (Section III) 1977 Edition, Stress Indices per USAS B31.7-1969.
3. Fatigue usage factor (U) per document USAS B31.7-1969 Nuclear Power Piping. U/.1<1.0
4. Location is at surge line nozzle, guillotine break is defined in branch pipe terminal end only.
5. Node is close enough to terminal end to be considered same location
6. Node is close enough to node 61 to be considered same location.
7. Original construction values, see appropriate calculation for the current values.

* Specified intermediate break location (guillotine).

TABLE 3.9-2

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
1	Main Steam	10.3-1	Isolation Valves FV100 &FV101	Yes	1971 ASME III 2	I	1050	600	Operating & Seismic	N-496	See Subsection 3.9.2.9.1
2	See Section 3.9.3										
3	Main Steam	10.3-1	Isolation Valves HV106, HV107, HV106A,HV107A	Yes	1971 ASME III 2	I	1050	600	Operating & Seismic	None	See Subsection 3.9.2.9.2a
4	Feedwater	10.4-12	Aux Feed Pumps P14-1 & P14-2	Yes	1971 ASME III 3	I	1500	300	Dynamic & Seismic	None	See Subsection 3.9.2.9.3
5	Main Steam	10.4-12	Atm Vent Valves PV1CS11A & PV1CS11B	Yes	1971 ASME III 2	I	1050	600	Operating & Seismic	None	See Subsection 3.9.2.9.4
6	Feedwater	10.4-12A	Isolation Valves HV608 & HV599	Yes	1971 ASME III 2	I	1350	300	Operating & Seismic	None	See Subsection 3.9.2.9.2a
7	Feedwater	10.4-12	Isolation Valves HV612 & HV601	Yes	1971 ASME III 2	I	1350	470	Operating & Seismic	None	See Subsection 3.9.2.9.2a
8	Main Steam	10.4-13	Safety PSVSP17A1 to PSVSP17A9 PSVSP17B1 to PSVSP17B9	Yes	Pump & Valve Code Class II	I	1050	650	Operating & Seismic	None	See Subsection 3.9.2.9.5
9	Containment Vessel Vacuum Relief	9.4-12	Vacuum Relief Valves PSV5042 to 5051	Yes	1971 ASME III 2	I	75	300	Operating & Seismic	None	See Subsection 3.9.2.9.6
10	Containment Vessel Vacuum Relief	9.4-12	Isolation Valves HV5070 to 5079	Yes	1971 & 1986 ASME III 2	I	45	264	Operating & Seismic	None	See Subsection 3.9.2.9.2b and Reference 77
11	Service Water	9.2-1	Service Water Pumps P3-1, P3-2, & P3-3	Yes	1971 ASME III 3	I	250	140	Dynamic & Seismic	None	See Subsection 3.9.2.9.7

TABLE 3.9-2 (Continued)

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
12	Service Water	9.2-1	Service Water Strainers F15-1, F15-2, & F15-3	Yes	1971 ASME III 3	I	150	140	Operating & Seismic	None	See Subsection 3.9.2.11.2
13	Service Water	9.2-1	Cooling Water Ht. Ex Auto Blocking Valves HV1395 & HV1399	Yes	1971 ASME III 3	I	150	140	Operating & Seismic	None	See Subsection 3.9.2.9.2b
14	Service Water	9.2-1	Comp Cooling Ht Ex Auto Outlet Valves TV1424, TV1429, & TV1434	Yes	1971 ASME III 3	I	150	175	Operating & Seismic	None	See Subsection 3.9.2.9.8h
15	Deleted										
16	Service Water	9.2-1	Aux Fd Pump Blocking Valves HV1382 & HV1383	Yes	1971 ASME III 3	I	150	140	Operating & Seismic	None	See Subsection 3.9.2.9.2b
17	Service Water	9.2-1	CAC Auto Inlet Valves HV1366, HV1367, & HV1368	No	1971 ASME III 2	I	150	150	Operating & Seismic	None	See Subsection 3.9.2.9.2f
18	Service Water	9.2-1	CAC Auto Outlet Valves TV1356, TV1357, & TV1358	Yes	1986 ASME III 2	I	150	264	Operating & Seismic	None	See Subsection 3.9.2.9.8b
19	Service Water	9.2-1	System Outlet Valves HV2929, HV2930 HV2931, HV2932	Yes No	1971 ASME III 3 3	I I	150 150	150 150	Operating & Seismic Operating & Seismic	None None	See Subsection 3.9.2.9.2b See Subsection 3.9.2.9.2b
20	Service Water	9.2-1	Cont Room Emerg Cond Unit Valves HV2927 & HV2928	Yes	1971 ASME III 3	I	150	150	Operating & Seismic	None	See Subsection 3.9.2.9.2a
21	Decay Heat Removal & ECCS	6.3-2A	Bor Water Stg. TK Outlet Valves HV DH 7A & B	Yes	Pump & Valve Code Class II	I	75	300	Operating & Seismic	None	See Subsection 3.9.2.9.2e
22	Decay Heat Removal & ECCS	6.3-2A	Emerg Sump Recirc Valves HV DH9 A & B	Yes	Pump & Valve Code Class II	I	75	300	Operating & Seismic	None	See Subsection 3.9.2.9.2e

TABLE 3.9-2 (Continued)

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
23	Decay Heat Removal & ECCS	6.3-2A	Valves HV 2733 & HV2734	No	1971 ASME III 2	I	300	350	Operating & Seismic	1388-1 1335-6	See Subsection 3.9.2.9.2a
24	Decay Heat Removal & ECCS E27-1 & E27-2	6.3-2A	Decay Heat Removal Coolers E 27-1 & E 27-2	----	ASME Sect III 1968 Class C	I	450 Tubes 150 Shell	350 Tubes 250 Shell	Operating & Seismic	None	See Subsection 3.9.2.11.3
25	Decay Heat Removal & ECCS	6.3-2A	Decay Heat Pumps P42-1 & P42-2	Yes	Pump & Valve Code Class II	I	450	350	Dynamic & Seismic	None	See Subsection 3.9.2.9.9
26	Decay Heat Removal & ECCS	6.3-2A	Valves HV DH-13 A & B HV DH-14 A& B	Yes	1971 ASME III 2	I	450	350	Operating & Seismic	None	See Subsection 3.9.2.9.8c
27	Decay Heat Removal & ECCS	6.3-2A	Valves HV DH 1 A & B	Yes	Pump & Valve Code Class I	I	2500	350	Operating & Seismic	None	See Subsection 3.9.2.9.2a
28	Decay Heat Removal & ECCS	6.3-2	High Pressure Inj Pumps P58-1 & P58-2	Yes	Pump & Valve Code Class II	I	2000	300	Dynamic & Seismic	None	See Subsection 3.9.2.9.10
29	Decay Heat Removal & ECCS	6.3-2	Valves HV HP2 A, B, C, & D	Yes	Pump & Valve Code Class I & 1971 ASME III 2	I	3050	300	Operating & Seismic	None	See Subsection 3.9.2.9.2d
30	Decay Heat Removal & ECCS	6.3-2	Valve HV 1556	No	1971 ASME III 2	I	3050	300	Operating & Seismic	None	See Subsection 3.9.2.9.8d
31	ECCS Cont Spray & Core Cooling	6.3-1A	Valves HV CF 1A & B	No	Pump & Valve Code Class II	I	2500	300	Operating & Seismic	None	See Subsection 3.9.2.9.2a
32	ECCS Cont Spray & Core Cooling	6.3-1	Cont Spray Pumps P56-1 & P56-2	Yes	Pump & Valve Code Class II	I	300	300	Dynamic & Seismic	None	See Subsection 3.9.2.9.11
33	ECCS Cont Spray & Core Cooling	6.3-1	Valves HV 1530 & 1531	Yes	1971 ASME III 2	I	300	300	Operating & Seismic	None	See Subsection 3.9.2.9.2c

TABLE 3.9-2 (Continued)

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
34	Comp Cooling	9.2-2	Comp Cooling Pumps P43-1, P43-2, & P43-3	Yes	Pump & Valve Code Class III	I	150	200	Dynamic & Seismic	None	See Subsection 3.9.2.9.12
35	Comp Cooling	9.2-2	Comp Cooling Heat Exchangers E22-1, E22-2, & E22-3	------	ASME Sect VIII 1968	I	150 Shell 120 Tubes	200	Operating & Seismic	None	See Subsection 3.9.2.11.4
36	Comp Cooling	9.2-2	Comp Cooling Surge Tank T12	-------	ASME Sect VIII 1968	I	100	200	Operating & Seismic	None	See Subsection 3.9.1.11.5
37	Comp Cooling	9.2-2	Valves HV1467 & HV1469	Yes	1971 ASME III 3	I	150	200	Operating & Seismic	None	See Subsection 3.9.2.9.8a
38	Comp Cooling	9.2-2	Valves HV2645, HV2649 HV5095, HV5096, HV5097, HV5098 HV1567 A & B	Yes Yes Yes	1971 ASME III 3 3 2	I I I	150 125 150	200 160 200	Operating & Seismic Operating & Seismic Operating & Seismic	None None None	See Subsection 3.9.2.9.2a See Subsection 3.9.2.9.2a See Subsection 3.9.2.9.2a
39	Comp Cooling	9.2-2	Valves HV1411 A & B HV1407 A & B	Yes	1971 ASME III 2	I	150	200	Operating & Seismic	None	See Subsection 3.9.2.9.2b
40	Comp Cooling	9.2-2	Valves HV1471 & HV1474	Yes	1971 ASME III 3	I	150	200	Operating & Seismic	None	See Subsection 3.9.2.9.8a
41	Comp Cooling	9.2-2	Valves HV1328 & HV1330	Yes	1971 ASME III 3	I	150	200	Operating & Seismic	None	See Subsection 3.9.2.9.2a
42	Comp Cooling	9.2-2	Valve HV1495	Yes	1971 ASME III 3	I	150	200	Operating & Seismic	None	See Subsection 3.9.2.9.8a
43	Gaseous Radwaste	11.2-4	Valve HV2853	Yes	1971 ASME III 3	I	16	300	Operating & Seismic	None	See Subsection 3.9.2.9.8e

TABLE 3.9-2 (Continued)

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
44	Gaseous Radwaste	11.2-4	Valves HV1823, HV1824, HV1825, HV1826, HV1827, & HV1828	Yes	1971 ASME III 2	I	150	150	Operating & Seismic	None	See Subsection 3.9.2.9.8e
45	Gaseous Radwaste	11.2-4	Valves HV1835, HV1836, HV1837, HV1838, HV1839, & HV1840	Yes	1971 ASME III 3	I	150	150	Operating & Seismic	None	See Subsection 3.9.2.9.8c
46	Gaseous Radwaste	11.2-4	Valves HV1810 HV1803	No Yes	1971 ASME III 3 3	I I	15 15	200 200	Operating & Seismic Operating & Seismic	None None	See Subsection 3.9.2.9.8c See Subsection 3.9.2.9.8c
47	Gaseous Radwaste	11.2-4	Valve HV2854	Yes	1971 ASME III 3	I	150	200	Operating & Seismic	None	See Subsection 3.9.2.9.8c
48	Makeup and Purification	9.3-16	Letdown Coolers E25-1 & E25-2	----	1977 ASME III 3	I	2500	600	Operating & Seismic	None	See Subsection 3.9.2.11.6
49	Component Cooling	9.2-2	Valve HV1460	Yes	1971 ASME III 3	I	150	200	Operating & Seismic	None	See Subsection 3.9.2.9.8d
50	Diesel Fuel Oil	9.5-8	Day Tanks T46-1 & T46-2	-----	1971 ASME III 3	I	15	110	Operating & Seismic	None	See Subsection 3.9.2.9.13
51	Hydrogen Dilution	9.4-11	Valves HV5065, HV5090, HV5038, & HV5037	Yes	1971 ASME III 2	I	40	265	Operating & Seismic	None	See Subsection 3.9.2.9.2b
52	Decay Heat Removal & ECCS	6.3-2A	Valves HV830 & HV831	Yes	1971 ASME III 2	I	350	280	Operating & Seismic	None	See Subsection 3.9.2.9.2a
53	Hydrogen Dilution	9.4-11	Blowers C62-1, C62-2	Yes	1971 ASME III 3	I	25	200	Seismic	None	See Subsection 3.9.2.11.7
54	Makeup & Purification	9.3-16	Valve HV MU2A HV MU2B, HV MU1A, & HV MU1B	Yes	Pump & Valve Code Class I	I	2500	600	Operating & Seismic	None	See Subsection 3.9.2.9.2a

TABLE 3.9-2 (Continued)

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
55	Makeup & Purification	9.3-16	Valve HV MU3	Yes	Pump & Valve Code Class II	I	2500	600	Operating & Seismic	None	See Subsection 3.9.2.9.8f
56	Makeup Water Treatment System	9.2-4	Valves HV6831A & HV6831B	Yes	1971 ASME III 2	I	150	150	Operating & Seismic	None	See Subsection 3.9.2.9.8d
57	Service Water	9.2-1	Valves HV1379, HV1380 & HV1381	Yes	1971 ASME III 2	I	150	140	Operating & Seismic	None	See Subsection 3.9.2.9.2a
58	Station Drainage	9.3-4	Valves HV2012A, HV2012B	Yes	1971 ASME III 2	I	45	264	Operating & Seismic	None	See Subsection 3.9.2.9.2a
59	Reactor Coolant System Details	5.1-2	Valves HV229 A & B	Yes	1971 ASME III 2	I	100	350	Operating & Seismic	None	See Subsection 3.9.2.9.8d
60	Reactor Coolant System Details	5.1-2	Valves HV1719 A & B HV1773 A & B	Yes Yes	1971 ASME III 2 2	I I	165 180	350 350	Operating & Seismic	None None	See Subsection 3.9.2.9.8e
61	Deleted										
62	Auxiliary Feedwater	10.4-12A	Valves HV3869, HV3870, HV3871 & HV3872	Yes	1971 ASME III 3	I	1350	300	Operating & Seismic	None	See Subsection 3.9.2.9.2e
63	Main Steam	10.3-1	Isolation Valves HV598 & HV607	Yes	1971 ASME III 3	I	1100	540	Operating & Seismic	None	See Subsection 3.9.2.9.8d
64	Station and Instrument Air System	9.3-1	Isolation Valve HV2010	Yes	1971 ASME III 3	I	150	150	Operating & Seismic	None	See Subsection 3.9.2.9.8d
65	Station and Instrument Air System	9.3-1	Isolation Valve HV2011	Yes	1971 ASME III 3	I	165	200	Operating & Seismic	None	See Subsection 3.9.2.9.8d
66	Containment Air Sample	9.4-11A	Isolation Valves HV5010E & HV5011E	Yes	1971 ASME III 3	I	50	300	Operating & Seismic	None	See Subsection 3.9.2.9.2g

TABLE 3.9-2 (Continued)

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
67	Auxiliary Feedwater	10.4-12A	Valves HV786 & HV790	Yes	B31.1	I	150	305	Operating & Seismic	None	See Subsection 3.9.2.9.2e
68	Decay Heat Removal & ECCS	6.3-2A	Valves HV DH11 & DH12	Yes	1971 ASME III 1	I	2500	650	Operating & Seismic	None	See Subsection 3.9.2.9.2a
69	Makeup & Purification	9.3-16	Valves HV MU59A, B, C, D	Yes	Pump & Valve Code Class II	I	2500	650	Operating & Seismic	None	See Subsection 3.9.2.9.2d
70	Makeup & Purification	9.3-16	Valves HV MU66A, B, C, D, and HV MU38	Yes	Pump & Valve Code Class II	I	3050/* 2500 *Top value refers to MU66 valves, bottom value to MU38	200/* 300	Operating & Seismic	None	See Subsection 3.9.2.9.8g
71	Deleted										
72	Makeup & Purification	9.3-16	Valve HV MU3971 HV MU6405	Yes	1971 ASME III 2	I	110	200	Operating & Seismic	None	See Subsection 3.9.2.9.2h
74	Component Cooling	9.2-2	Valves HV4100, HV4200, HV4300, & HV4400		1971 ASME III 2	I	3600	100		None	See Subsection 3.9.2.9.2.1
75	Core Flood	6.3-1A	Valves HVCF2A, HVCF2B, HVCF5A, & HVCF5B		Pump & Valve Code Cl 2	I	700	300		None	See Subsection 3.9.2.9.2.a
76	Containment Vessel Isolation	9.4-11A	Valves HV624B & HV645B		1971 ASME III 2	I	1235	100		None	See Subsection 3.9.2.9.2a
77	Containment Vessel Isolation	9.4-11A	Valves HV2000B, HV2001B, HV2002B, & HV2003B		1971 ASME III 2	I	1235	100		None	See Subsection 3.9.2.9.2a
78	Decay Heat Removal & ECCS	6.3-2A	Valves HV1517 & HV1518		1971 ASME III 2	I	300	350		None	See Subsection 3.9.2.9.2a
79	Decay Heat Removal & ECCS	6.3-2A	Valve HV2736		1971 ASME III 2	I	2240	800		None	See Subsection 3.9.2.9.2k
80	Decay Heat Removal & ECCS	6.3-2	Valves HVHP31 & HVHP 32		1971 ASME III 2	I	2240	800		None	See Subsection 3.9.2.9.2k

TABLE 3.9-2 (Continued)

ASME Code Class 2 and 3 Components and Active Components and Valves

No	System	Fig No	Component	Active	Code and Class	Seismic Class	Design Pressure psig	Design Temp °F	Loading Condition	ASME & ANSI Code Interpretations	Analytical Method or Test
81	Main Steam	10.3-1	Valves HV603 & HV611		1971 ASME III 2	I	915	536		None	See Subsection 3.9.2.9.2a
82	Makeup & Purification	9.3-16	Valves HV6408, HV6409, HV6420, HV6421, & HV6422		1971 ASME III 2	I	3050	200		None	See Subsection 3.9.2.9.2i
83	Makeup & Purification	9.3-16	Valve HV6419		1971 ASME III 2	I	3050	200		None	See Subsection 3.9.2.9.2j
84	Reactor Coolant System	5.1-2	Valve HV240B			I	3600	100			See Subsection 3.9.2.9.2a
85	Service Water Sys	9.2-1	Valve HV5067			I	1440	100			See Subsection 3.9.2.9.2a
86	Service Water Sys	9.2-1	Valve HV5068		1971 ASME III 2	I	1440	100			See Subsection 3.9.2.9.2a

TABLE 3.9-3

Load Combinations and Stress Limits for ASME Code Class 2 and 3 Components

Component	Item No (Table 3.9-2)	Load Combination	Stress Limits*
Piping	----	P_O + DW + OBE** P_O + DW + FV + OBE** P_O + DW + FV + OBE** P_O + DW + RVO + OBE** P_O + DW + DU + OBE**	$\sigma m + \sigma b \leq 1.2S$
		P_O + DW + SSE** + RVC P_O + DW + SSE** + FV P_O + DW + SSE** P_O + DW + DU + SSE**	$\sigma_m + \sigma_b \leq Sy$
Valves	All	----	$P_O \leq P_r$
	1	SSE + Operating Loads	$\sigma_m \leq Sy$ (flange, tie rods, body, and tubes)
			$\sigma_m \leq 2S_m$ (bolts)
	3,6,7,15,20,23,27,29, 31,38,41,52,54,55,57, 58,62,67,68,69,70,73, 74,75,76,77,78,81,84, 85,86	SSE + Operating Loads	$\sigma_m \leq 1.5S_m$ (bonnet, body, & yoke)
			$\sigma_m \leq S_m$ (bolts)
	10,13,16,19,39,51	SSE + Operating Loads	$\sigma_b + \sigma_m \leq Sy$ (body, bracket, bolts)
	30,33,49,56,59,63,64, 65	SSE + Operating Loads	$\sigma_b + \sigma_m \leq \frac{S}{2}t$ (yoke)
			$\sigma_m + \sigma_b \leq S$ (bolts)
			$\sigma_m + \sigma_b \leq Sy$ (body)

TABLE 3.9-3 (Continued)

Load Combinations and Stress Limits for ASME Code Class 2 and 3 Components

Component	Item No (Table 3.9-2)	Load Combination	Stress Limits*
Valves	21,22,26	SSE + Operating Loads	$\sigma_m + \sigma_b \leq S_m$ (bonnet, body)
			$\sigma_m + \sigma_b \leq .64Sy$ (yoke)
			$\sigma_m + \sigma_b \leq S_m$ (bolts)
	17,18,37,40,42	SSE + Operating Loads	$\sigma_m + \sigma_b \leq Sy$ (body)
			$\sigma_m \leq Sy$ (bolts)
	5,8	SSE + Operating Loads	$\sigma_m + \sigma_b \leq S$ (body, clamps, and yoke)
	9,43,44,45,46,47,60	SSE + Operating Loads	$\sigma_m + \sigma_b \leq S$ (body & bolts)
	14	SSE + Operating Loads	$\sigma_m + \sigma_b \leq Sy$ (body)
	79,80	SSE + Operating Loads	$\sigma_m + \sigma_b < .7Sy$ (body, yoke bushing, bolts)
	82,83	SSE + Operating Loads	$\sigma_m + \sigma_b < 1.5S$ (body neck)
			$\sigma_m + \sigma_b < .65Sy$ (yoke clamp)
			$\sigma_m + \sigma_b < .9Sy$ (yoke, actuator mounting flange)
			$\sigma_m + \sigma_b < .6Sy$ (actuator mounting bolting)
	72	SSE + Operating Loads	$\sigma_m + \sigma_b < 1.5Sa$ (body, yoke & bolts)

TABLE 3.9-3 (Continued)

Load Combination and Stress Limits for ASME Code Class 2 and 3 Components

Component	Item No (Table 3.9-2)	Load Combination	Stress Limits*
Valves	66	SSE + Operating Loads	$\sigma_m + \sigma_b < 1.0Sa$ (body)
			$\sigma_m + \sigma_b < 0.25St$ (yoke & bolts)
Pumps	11	½ SSE + Operating Loads	$\sigma_m + \sigma_b \leq S$
		SSE + Operating Loads	$\sigma_m + \sigma_b \leq .9Sy$
	25,28	½ SSE + Operating Loads	$\sigma_m \leq \frac{Sy}{1.5}$
		SSE + Operating Loads	$\sigma_m \leq \frac{Sy}{1.1}$
	32,34	SSE + Operating Loads	$\sigma_b + \sigma_m \leq Sy$
	4	½ SSE + Operating Loads	$\sigma_m \leq S$ (bolts)
			$\sigma_m \leq .8S$ (dowels)
			$\sigma_m + \sigma_b \leq Sy$ (nozzles)
		SSE + Operating Loads	$\sigma_m \leq .9Sy$ (bolts)
			$\sigma_m \leq .72Sy$ (dowels)
			$\sigma_m + \sigma_b \leq Sy$ (nozzles)
Vessels	12	SSE + Operating Loads	$\sigma_b + \sigma_m \leq 1.34S$
	24	Refer to Section 3.9.2.11.3	
	35	Refer to Section 3.9.2.11.4	
	36,48	SSE + Operating Loads	$\sigma_m \leq S$
	50	SSE + Operating Loads	$\sigma_m \leq 1.25S$

* The component stress limits indicated for the pumps, valves, and vessels are limiting.

** OBE and SSE loads determined per subsection 3.7.2.

TABLE 3.9-3 (Continued)

Load Combinations and Stress Limits for ASME Code Class 2 and 3 Components

P_O = Operating Pressure

DW = Dead Weight

RVC = Relief Valve, Closed System

RVO = Relief Valve, Open System

FV = Fast Valve

σ_b = Bending Stress

σ_m = General Membrane Stress

S = Material Allowable Stress (At Design Temperature)

S_y = Material Yield Stress

P_r = Primary Pressure Rating

S_m = Design Stress Intensity

S_t = Tensile Strength

DU = Any Other Primary Loads

OBE = Operating Basis Earthquake

SSE = Safe Shutdown Earthquake

TABLE 3.9-4

Natural Frequencies and Seismic Inputs for Service Water Pumps

		Mode				
		1	2	3	4	Vertical
Motor – Head Assembly E-W	f	12.47	224.98	732.99		35
	g1	0.48	0.0945	0.0945		0.063
	g2	0.78	0.162	0.162		0.108
Motor – Head Assembly N-S	f	11.96	213.5	711.27		35
	g1	1.26	0.1487	0.1487		0.10
	g2	1.94	0.2283	0.2283		0.152
Column Assembly N-S LWL	f	2.27	18.04	56.75	117.5	77.5
	g1	0.32	0.80	0.1487	0.1487	0.10
	g2	0.52	1.04	0.2283	0.2283	0.152
Column Assembly N-S NWL	f	2.25	16.64	51.82	108.0	77.5
	g1	0.32	1.26	0.1487	0.1487	0.10
	g2	0.52	1.94	0.2283	0.2283	0.152
Shaft Assembly N-S	f	70.37				81.5
	g1	0.1487				0.10
	g2	0.2283				0.152

f = Frequency in Hz

g1 = Input at 0.08 g earthquake

g2 = Input at 0.15 g earthquake

TABLE 3.9-5

Service Water Pumps Tabulation of Stresses for 0.08g Earthquake

Item	Material	Allowable Stress, psi	Actual Stress, psi	Margin %
Motor Base	A106-GRB	15,000	6,320	137
Discharge Head – Shell	A106-GRB	15,000	12,930	16
Discharge Head – Base	A515-GR70	17,500	14,210	23
Sub Base	A515-GR70	17,500	6,120	186
Column Pipe	A106-GRC	17,500	12,300	42
Column Flange	A515-GR70	17,500	12,480	40
Shaft	AISI-416	17,500	8,840	98
Shaft Coupling	AISI-416	17,500	1,090	1506
Top Bowl	A216-WCB	17,500	2,840	516
Bolting				
Motor to Base	SA-325	19,250	16,900	14
Base to Head	SA-325	19,250	16,350	18
Head to Sub Base	SA-325	19,250	11,750	64
-----	------	--------	--------	---------
Column Flange	SA-325	19,250	13,030	48
Column to Top Bowl	SA-325	19,250	2,150	795
Shaft Coupling Cap	SA-453-651	16,000	15,960	0.3

Note: Margin % = $\left[\frac{\text{Allowable Stress}}{\text{Actual Stress}} - 1\right]100$

Allowable stresses are as listed in Section III of ASME Boiler and Pressure Vessel Code.

TABLE 3.9-6

Service Water Pumps Tabulation of Stresses for 0.15g Earthquake

Item	Material	Allowable Stress, psi	Actual Stress, psi	Margin %
Motor Base	A106-GRB	31,500	8,780	258
Discharge Head – Shell	A106-GRB	31,500	16,440	91
Discharge Head – Base	A515-GR70	34,200	20,690	65
Sub Base	A515-GR70	34,200	9,600	256
Column Pipe	A106-GRC	36,000	16,700	116
Column Flange	A515-GR70	34,200	17,930	91
Shaft	AISI-416	36,000	9,170	293
Shaft Coupling	AISI-416	36,000	1,090	3203
Top Bowl	A216-WCB	32,400	2,970	1052
Bolting				
Motor to Base	SA-325	69,300	26,020	166
Base to Head	SA-325	69,300	25,180	175
Head to Sub Base	SA-325	69,300	18,430	276
-----	------	------	------	----
Column Flange	SA-325	69,300	20,090	245
Column to Top Bowl	SA-325	69,300	2,440	2740
Shaft Coupling Cap	SA-453-651	57,600	15,980	260

Note: Margin % = $\left[\frac{\text{Allowable Stress}}{\text{Actual Stress}} - 1\right]100$

Allowable stresses are 0.9 times yield as listed in Section III of ASME Boiler and Pressure Vessel Code.

TABLE 3.9-7

Design Parameters for Waste Gas Tanks

TANK	EQUIP NO	P&ID & FIG NO.	CODE	SEISMIC CLASS	DESIGN PRESS, psig	DESIGN TEMP °F	LOADING CONDITION	OPERATING CONDITION
Waste Gas Decay	T25-1	M-038 11.2-4	ASME III Class C 1968	I	150	200	Operating and seismic	Faulted
Waste Gas Decay	T25-2	M-038 11.2-4	ASME III Class C 1968	I	150	200	Operating and seismic	Faulted
Waste Gas Decay	T25-3	M-038 11.2-4	ASME III Class C 1968	I	150	200	Operating and seismic	Faulted
Waste Gas Surge	T24	M-038 11.2-4	ASME III Class C 1968	I	15	200	Operating and seismic	Faulted

Stress Limits

Component	Material	Maximum Stress, psi	Allowable Stress, psi*
Shell	304	8630	0.9 Py = 27,000
Legs	A-36	16,145	1.333 F_a = 25,790
Bracing	A-36	8312	1.333 F_{as} = 9,386

* AISC, Paragraph 1.5.6 and Appendix 5-67

Py = Material yield stress
F_a = Axial compressive stress permitted in absence of bending stress.
F_{as} = Axial compressive stress permitted in absence of bending stress, for bracing.

TABLE 3.9-8

Summary of Critical Items

Description	CLASSIFICATION Group 1 Analysis	Group 2 Analysis	Group 3 Prima Facie
Anchor bolts to foundation	X		
Base Plate	X		
Casing to pedestal connection, coupling end	X		
Casing to pedestal connection, governor end	X		
Control linkages			X
Control valve bolting	X		
Control valve assembly		X	
Journal bearings	X		
Mechanical overspeed trip device		X	
Oil cooler	X		
Oil filter material and internals			X
Pedestals	X		
Relief valve		X	
Remote trip solenoid valve		X	
Shaft driven oil pump		X	
Thrust bearings	X		
Turbine casing & related bolting	X		
Turbine shaft & wheels	X		
Governor complex		X	

TABLE 3.9-9

Vertical Contact Safety Margins

ACCIDENT CONDITION	VERTICAL MAXIMUM PROBABLE EARTHQUAKE	VERTICAL MAXIMUM POSSIBLE EARTHQUAKE	LOCA	COMBINED LOCA & MAXIMUM POSSIBLE EARTHQUAKE
Fuel Assembly Component Loading Condition	Margin of Safety %			
Guide Tube Buckling	320	680	310	310
Fuel Rod Buckling	1000	950	470	1000
End Grid Assembly – Buckling	880	190	190	190
End Grid Welds	400	190	190	190
End Grid Attachment Welds	325	140	140	140

TABLE 3.9-10

Code Cases*

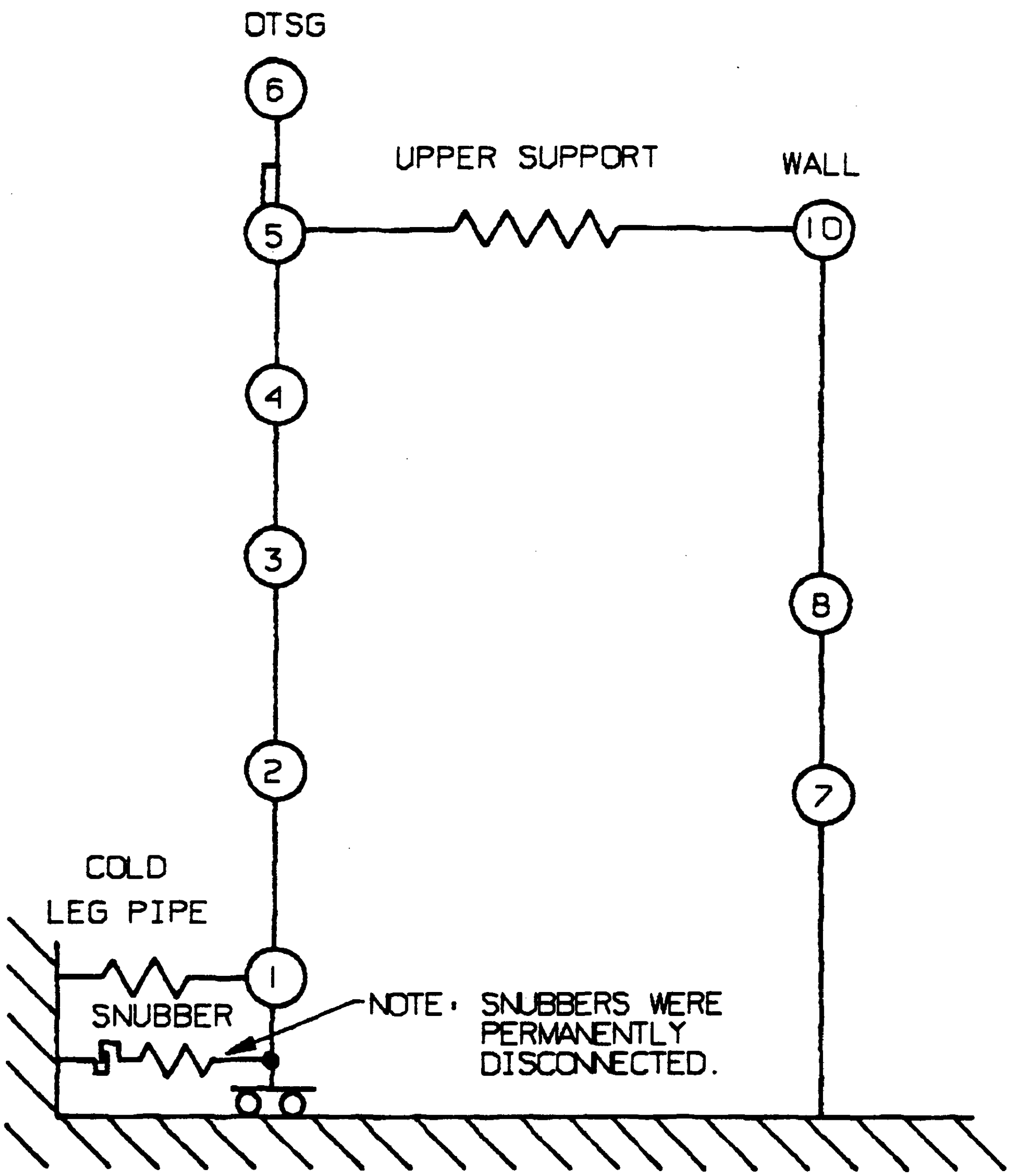
Code Case	Title
1177-7	Expansion Joints Section VIII Division 1
1330-3	Special Equipment Requirements Section III
1332-6	Requirements for Steel Forgings Section III and VIII Division 2
1335-6	Requirements for Bolting Materials Section III
1388-1	Requirements for Stainless Steel Precipitation Hardening Section III
1459-1	Welding Repairs to Base Metal of Classes 1, 2, and 3 Section III Components after Final Postweld Heat Treatment
1501	Use of SA-453 Bolts in Service Below 800°F Without Stress Rupture Tests Section III
1507	Weld Repairs of Castings Section III Class 3 Components
1519	Use of A105-71 in lieu of SA-105
1539	Metal Bellows and Metal Diaphragm Steam Sealed Valves Section III Class 1, 2, and 3
1540	Elastomer Diaphragm Valves Section III Class 2 and 3
1580-1	Butt Welded Alignment Tolerances and Acceptable Slopes for Concentric Centerlines for Section III Classes 1, 2, and 3 Construction
1644-2	Additional Materials for Component Supports, Section III, Subsection NF, Class 1, 2, 3, and MC Construction
1686	Furnace Brazing, Section III, Subsection NF, Component Supports
N-411	Alternative Damping Values for Seismic Analysis of Piping, an alternative to Table N-1230-1, Appendix N of ASME Section III, Division 1
N-474-1	Design Stress Intensity and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi, Class 1 Components Section III, Division 1
N-474-2	Design Stress Intensity and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi, Class 1 Components Section III, Division 1

TABLE 3.9-10 (Continued)

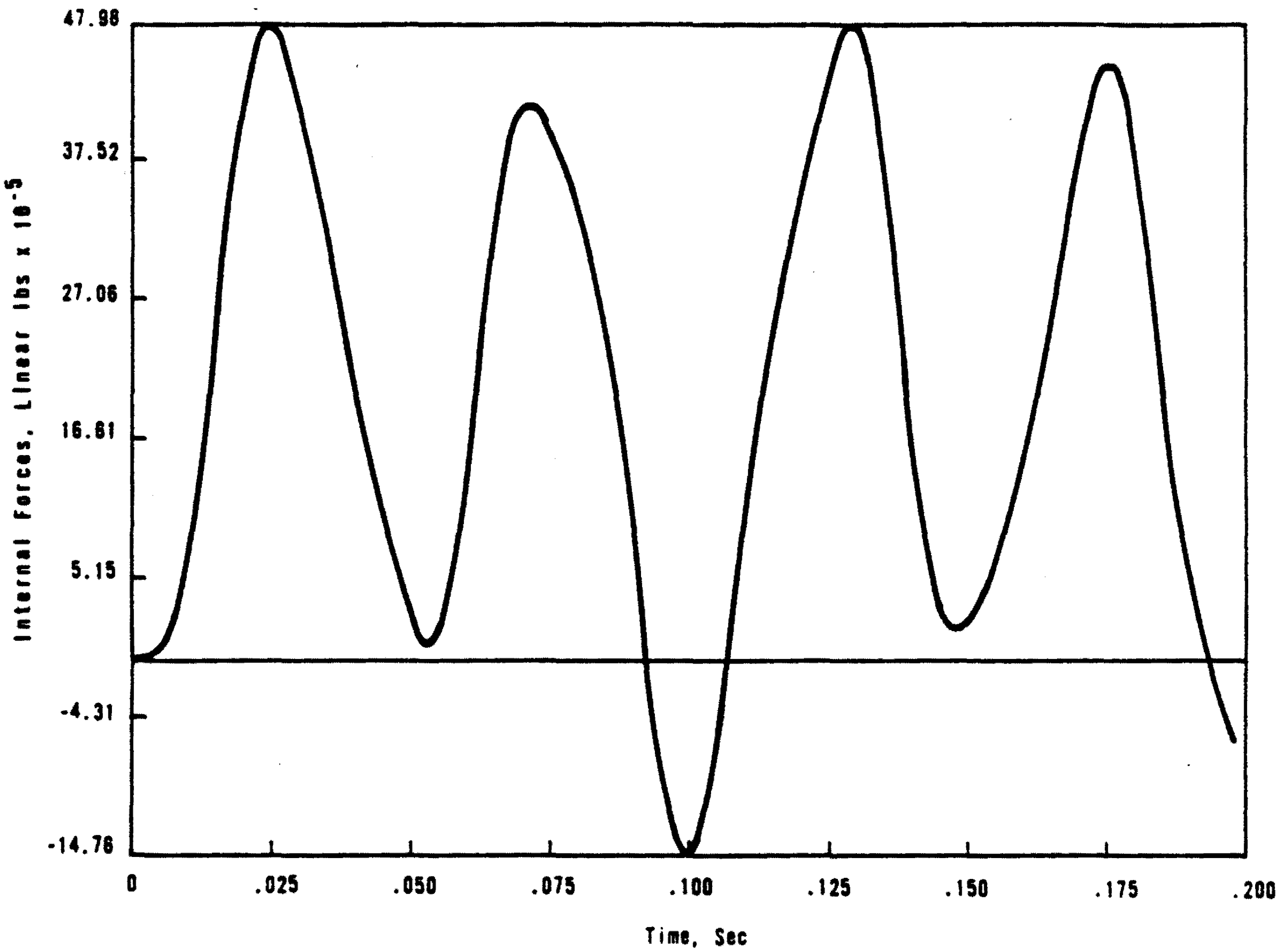
Code Cases*

Code Case	Title
N-496	Helical Coil Threaded Inserts, Section XI, Division 1
N-638-1	Similar & Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI Division 1

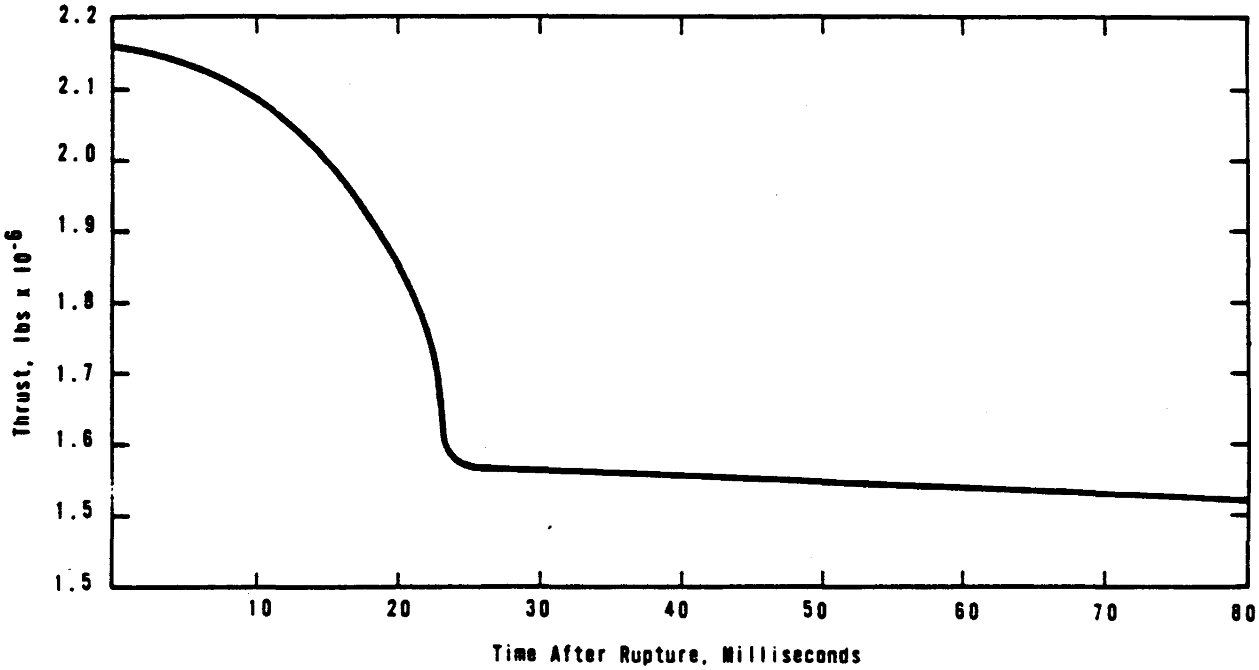
* Reference Section 3.9.4 for usage of other code cases.



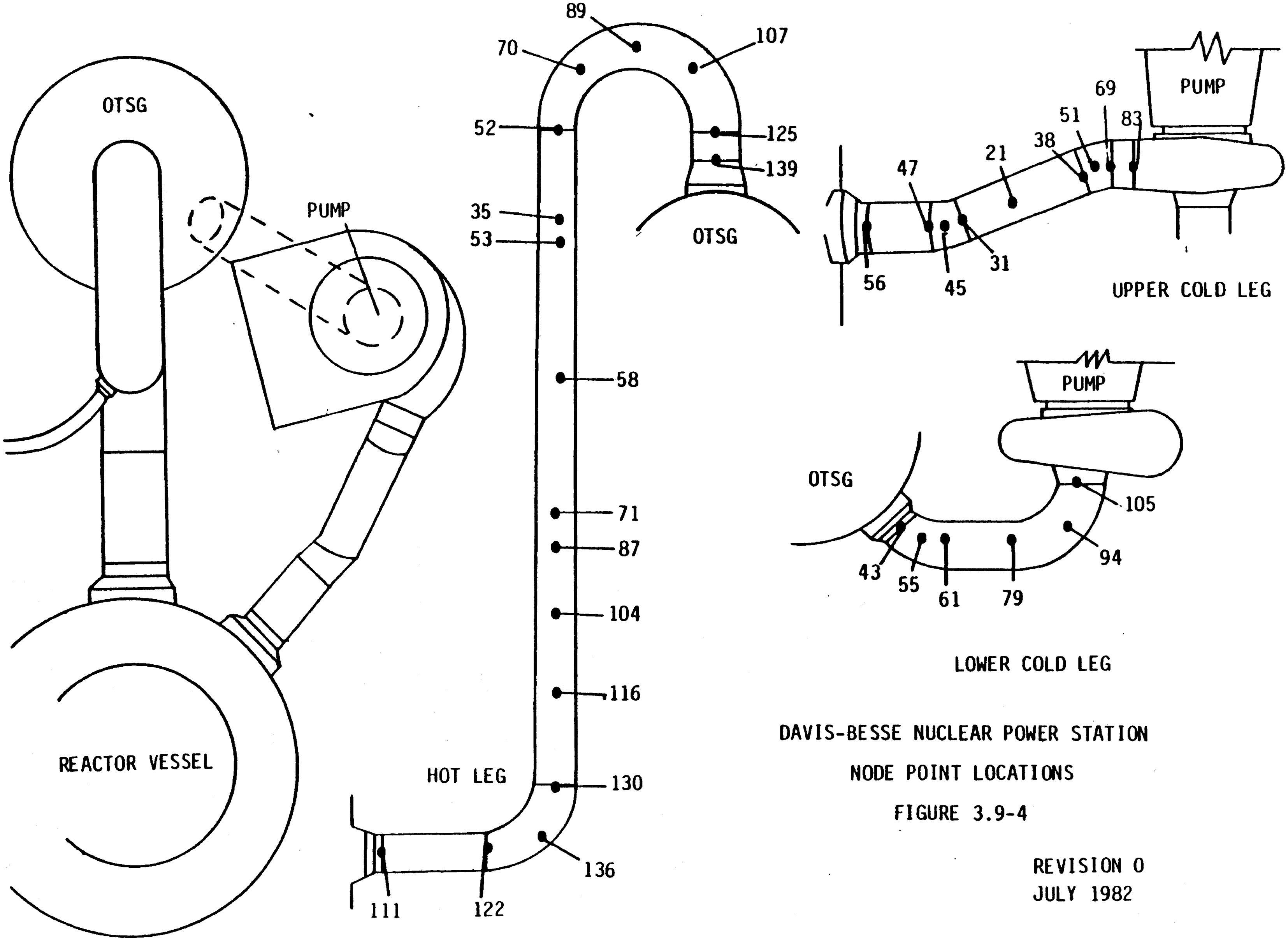
DAVIS-BESSE NUCLEAR POWER STATION
OTSG AND WALL DYNAMIC MODEL
FIGURE 3.9-1



DAVIS-BESSE NUCLEAR POWER STATION
LOCA LOADS ON UPPER SUPPORT OF
STEAM GENERATOR
FIGURE 3.9-2



DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR THRUST - 36''
PIPE DOUBLE-ENDED RUPTURE
14.14 FT^2 LEAK AREA
FIGURE 3.9-3

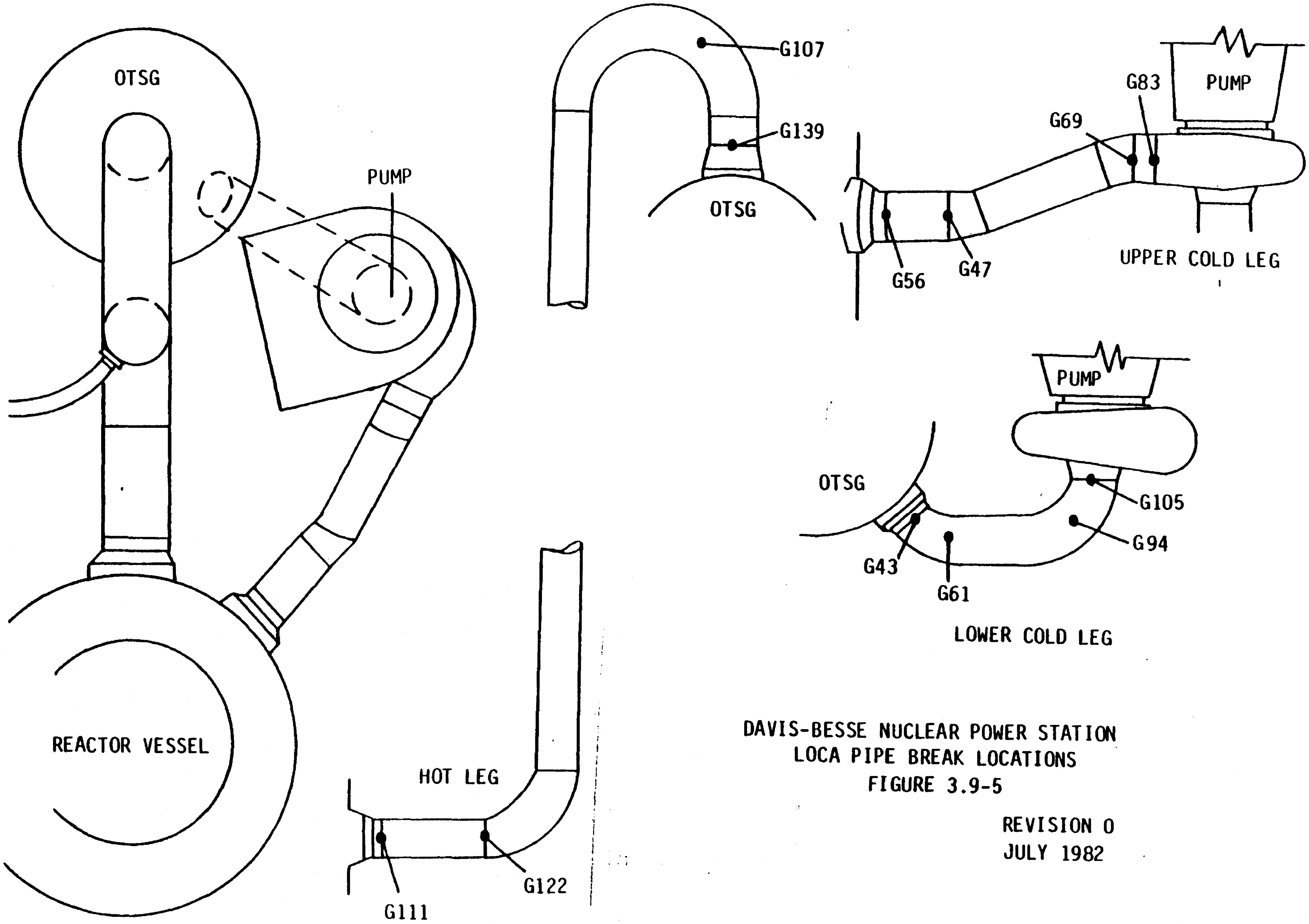


DAVIS-BESSE NUCLEAR POWER STATION

NODE POINT LOCATIONS

FIGURE 3.9-4

REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
LOCA PIPE BREAK LOCATIONS
FIGURE 3.9-5

REVISION 0
JULY 1982

3.10 SEISMIC DESIGN OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 General

The following electrical equipment have been designated as Class 1E and as such has been designed to withstand the Maximum Possible Earthquake conditions without loss of function.

a Emergency Diesel Generators 1 & 2

b. Essential 4.16 kV Switchgear, Buses C1 & D1

c. Essential Motor Control Centers

d. Essential Unit Substations

e. Class 1E Motors

f. Essential Motor Operated Valves (MOVs)

g. Safety Features Actuation Equipment

h. Electrical Penetrations

i. Batteries and Racks

j. Inverters and Chargers

3.10.2 Seismic Certification

Seismic certification on equipment (by testing or analysis) has been completed and is basically in accordance with IEEE 344-1971, as described below. All vital appurtenances have been tested and/or analyzed.

Replacement parts for this equipment will be qualified in accordance with the applicable standard (at least IEEE 344-1971) and to the acceleration levels required for the application. A description of the analysis or testing procedure will be contained in the applicable test report.

3.10.2.1 Emergency Diesel Generators and Class 1E Motors

The natural frequencies of the components of motors and generators have been calculated and found to be high enough that amplification as detailed in the floor response spectrum does not occur. On this basis, the equipment has been statically analyzed for rotor deflection, bearing loading, and holddown configuration.

The emergency diesel-generator was analyzed by calculation, using a model of the rotor as a single mass element. Only analysis by calculation was performed due to the impracticality of performing dynamic testing. A conservative 1 percent damping value for the 15 percent g earthquake was selected, giving a peak acceleration of 3.0g in the horizontal direction and 2.08g in the vertical direction. The analysis covered maximum rotor stress under magnetic pull and seismic conditions, rotor displacement, bearing load on shaft and frame, and mounting bolt loadings. All results were well within the design limitations.

The emergency diesel-generator excitation system has been replaced. The new system is seismically qualified based on a combination of testing and analyses. A seismic qualification report for the new system documents the testing and analyses performed to meet seismic qualification criteria in accordance with IEEE 344-1975. As part of the replacement, top bracing has been installed on the excitation system cabinets (C3617 and C3618) to connect them to the diesel generator room walls. This bracing restrains the motion, raises the natural frequency of the cabinet assembly and minimizes in-cabinet amplification of the floor response spectra. The cabinet internal amplification was analyzed using a finite-element model of the cabinets with the new equipment installed and top-bracing design. The top bracing is designed to withstand the loads imparted by seismically-induced movement of the cabinets with the new equipment installed. The seismically- sensitive components in the cabinet (i.e., relays and contactors) have all been tested to determine their seismic capacity. The capacity of each component exceeds the analyzed seismic demand to which each will be exposed during the safe shutdown earthquake or operating basis earthquake at the location where each is installed in the cabinet. For components in the cabinet that are not seismically sensitive (e.g., linear reactors, terminal blocks) data from generic tests as well as comparisons to originally installed components demonstrate the seismic ruggedness of the installed component.

Some original equipment was retained in C3617 and C3618. The seismic qualification report analyzes the effect of the replacement of the excitation system and installation of cabinet top bracing on this equipment and found that it is not adversely affected.

What follows is a description of the testing and redesign for the original equipment.

The originally installed emergency diesel-generator set exciter regulator board was seismically tested at the MTS Laboratory in Minneapolis, Minn. The equipment was mounted in a manner simulating intended service mounting, Testing was conducted in the vertical and horizontal axes simultaneously, then repeated with the equipment re-oriented 90 degrees to complete the tests for all three orthogonal axes. During testing, equipment was not energized, but vital contacts were monitored for proper functioning, and accelerometers were used to record maximum accelerations on the assembly.

The original exciter-regulator boards were sine-beat tested in the vertical and horizontal axes simultaneously using 0.25 g's at each resonant frequency and with 5 cycles per beat. Input was higher than necessary.

A natural frequency search was conducted from 1 to 33 Hz at 0.2g. The search was followed by a sinusoidal withstandability test conducted at several discrete frequencies and at any resonant frequency discovered during the search. The withstandability test consisted of an amplitude sweep from 0g to 0.15g for a minimum duration of four times the time constant, T, where:

$$T = \frac{1}{2\pi pf}$$

where:

f = frequency, Hz.
p = critical damping factor

In addition, the equipment was subjected to a random vibration test for 60 seconds and sine beat testing consisting of five cycles per beat, ten beats per frequency, at each of the resonant frequencies.

During the test, the voltage and frequency verification auxiliary relay, CR-3, developed contact chatter; therefore, the relay was remounted and the sub-support stiffened. Otherwise, the equipment was well within the design limits.

In the replacement excitation system this relay has been replaced with another model that has been separately subjected to shake-table testing. The testing showed the relay capacity exceeds the seismic demand at the location of the component on the regulator upper chassis.

The engine control panel is seismically the same as that supplied and tested for General Electric Atomic Power Energy Department (GEAPED) under their Specification No. 21A9236, which is the emergency diesel generator for high pressure core spray system.

The cabinets used are the same, and the equipment and devices used in the Davis-Besse panel were all tested on the GEAPED panel. The test was conducted at Acton Environmental Testing Corporation in Acton, Mass. A resonant search was made for each axis from 0.5 Hz, to 33 Hz at 0.2g. Resonant frequencies were found at 1 Hz, 17 Hz, and 24 Hz in the horizontal axis and at 11 Hz and 30 Hz in the vertical. The panel was then subjected to a beat frequency test at each resonant frequency, using a maximum of 3g in the horizontal, the 0.28g in the vertical. Each beat was for 5 cycles with a 30-cycle rest in between, repeated for 60 seconds. The equipment was energized during the test, and vital contacts were monitored for maloperation. Accelerometers were mounted in the assembly to record the actual accelerations observed. The recorded levels were in all cases in excess of the required levels for the Davis-Besse panel.

During the testing, relays R2, R1X3, and SDR worked loose from their socket, but a spring and bracket assembly was developed which prevented further occurrence. The Davis-Besse panel is acceptable on the basis of the above test.

Since the two emergency diesel generator control boards were identically built, one of the emergency diesel-generator relay and control boards was seismically tested at the Acton Environmental Testing Corporation Laboratories. Acceleration levels within the equipment were measured and recorded, and relay/switch contacts were monitored for false operation or control during the testing.

A continuous resonant frequency search was made from 1 to 33 Hz at 0.32g on both the front-to-back and side-to-side axes. For the vertical axis, a frequency search was made from 1 to 33 Hz at a level of 0.3g. The equipment was prooftested at the only resonant frequency of 56 Hz front-to-back and side-to-side. No resonant frequency was detected for the vertical axis. Proof tests were also run at 17 Hz front-to-back, 9 Hz side-to-side, and 15 Hz and 33 Hz vertically.

The test results were analyzed and the equipment proved suitable for service.

3.10.2.2 4.16 kV Switchgear

Prototype samples of the 4.16 kV essential metal-clad switchgear and its associated relaying and auxiliary components have been sine beat tested with input acceleration of 0.8g in each of the three axes. Single axis testing was performed in accordance with IEEE 344-1971.

The medium voltage switchgear was seismically qualified by Westinghouse Astronuclear Laboratories using a combination of testing and analysis to demonstrate suitability for nuclear power plant service under IEEE 344-1971.

A production model of the 50DHP350 switchgear was mounted on the table in a manner duplicating normal service and instrumented to measure the acceleration within the assembly and monitor relay contacts for maloperation.

The equipment was tested in each of the three orthogonal axes, each test consisting of a continuous sweep frequency search up to 25 Hz at 0.2g and sine beat testing at each of the natural frequencies found. The sine beat test consisted of five cycles per beat for five beats. During the test, the breaker was required to maintain position or to operate properly when commanded.

Natural frequencies were found at 7 Hz, side-to-side, and 9.5, 13, and 16 Hz, front-to-back. The lowest damping factor recorded in the switchgear was 5 percent, which was used to qualify the entire assembly. From the floor response spectrum, for a 5 percent damping factor an input excitation of 0.21g was required, whereas the equipment was tested at 0.8g without equipment failure.

The 50DHP250 used on Davis-Besse has the same basic cell design as the 50DHP350 with the only significant difference being that the 250 has a smaller arc chute. The arc chute was mathematically modeled and shown to impose lower stresses than on the 350 model.

Each C-1 and D-1 breaker (essential 4.16 Kv) has a seismic restraint installed in the front of the cubicle floor. This bracket will allow the breaker to be stored inside the cubicle when left in the test or withdrawn position for an extended amount of time without invalidating the existing seismic qualification report.

3.10.2.3 Unit Substations

Components of the essential substation transformers were designed to have natural frequencies above the amplification range shown by the floor response spectra. Analytic analysis was then performed in accordance with IEEE 344-1971. Prototypes of substation assemblies and components were vibration and shock tested to levels higher than required by the floor response spectra.

a. Transformers:

Due to their large size and their mechanical simplicity of design, the unit substation transformers were analyzed analytically. The components were modeled as one mass, one spring and the natural frequencies calculated were found to be in excess of 35 Hz.

Acceleration was derived from the spectrum curves for the equipment location using 0.15g, all frequencies above 35 Hz, and all values of damping, to give maximum values of 0.332g horizontal and 0.222g vertical.

Equivalent horizontal and vertical design earthquake inertia forces were applied simultaneously in addition to the normal design loads for idle and/or operating conditions in such a manner as to produce the most severe loading conditions.

The stresses were calculated. Component stresses will not exceed allowable working stress limits accepted as good practice in the appropriate design standards.

Deflections will not prevent normal operation of the equipment nor exceed 0.8 times the deflections which would cause loss of function of the equipment.

b. Substation assemblies:

The substation equipment has been successfully tested by General Electric for both vibration and shock conditions.

The vibration test consisted of mounting the AKD-5 switchgear assembly on the test bed with both AK-25 and AK-50 breakers installed and applying 0.5g in each of the three directions. Response accelerations were measured at each of the breaker positions. Breakers were then removed and tested separately to higher g levels. The test showed that at all resonant peaks, the ratio of response to input was no greater than 1.5, the basis for which a damping ratio of 0.3 was selected. The lowest resonant frequency discovered was 9 Hz.

The individual breakers were vibrated at 0.5g input in each of the three directions from 5 to 500 Hz. Were resonant peaks were found the breakers were vibration tested at those frequencies at higher g levels. The lowest frequency resonant peak for horizontal vibration for the AK-50 breaker was 29 Hz, and it was successfully vibrated at this frequency at 5.0g input. This would be a higher value at this frequency when referred to an input at the equipment base.

The lowest frequency resonant peak for horizontal vibration for the AK-25 breaker was 44 Hz, and it was successfully vibrated at this frequency at 3.0g input. This would be approximately equivalent to 2.0g when referred to an input at the equipment base.

The shock tests were made on a Navy-type medium weight shock machine with the equipment mounted at a 30-degree angle so that the equipment was subjected to simultaneous horizontal and vertical accelerations.

The summary of the tests and their results are as follows, shoving their acceptability.

1. AKD-5 switchgear has been vibration tested in each of the three directions at 0.5g over the frequency spectrum of 5 to 500 Hz.

2. AKD-5 switchgear has been shock tested to 40g on a Navy Medium Weight Shock Machine.

3. AK-50 and AK-25 breakers have been vibration tested in each of the three directions at 0.5g over the frequency spectrum of 5 to 500 Hz.

4. AK-50 and AK-25 breakers remain operative during shock at accelerations up to 15g on a Navy Medium Weight Shock Machine.

5. The AK-50 breaker has been vibration tested without loss of function during vibration at its lowest resonant frequency of 29 Hz at 5.0g input.

6. The AK-25 breaker has been vibration tested without loss of function during vibration at its lowest resonant frequency of 44 Hz at 3.0g input.

3.10.2.4 Motor Control Centers

Prototype samples of the essential motor control centers with ancillary equipment have been tested in accordance with IEEE 344-1971.

The Westinghouse Type W Motor Control Center was type tested in accordance with IEEE 344-1971.

Tests were performed at Wyle Laboratories, Huntsville, Alabama on the Wyle Three-Axes Simulator. Unidirectional accelerometers were employed with their response recorded on oscillographs. In addition, 13 recording channels for monitoring electrical continuity were used and recorded on both oscillographs and FM magnetic tape.

The Motor Control Center was a standard production model, four sections in length. Standard Westinghouse recommended boltdown methods were used. Tests were performed under both energized and de-energized conditions, using both single axes and simultaneous axes of excitation.

The Sine Beat Method of testing was employed. A sweep frequency search (independent horizontal and vertical) was performed to determine natural frequency(s). A sine beat was then performed at each of the natural frequencies determined. Test levels were based on maximum amplitude as determined by the floor response spectra.

It was demonstrated that the Type W Motor Control Center possessed sufficient integrity to sustain, and remain operational, without loss or compromise of structure or function for the worst seismic conditions within the station.

Westinghouse Five Star Motor Control Centers are also installed at Davis-Besse Nuclear Power Station. These motor control centers were tested in accordance with IEEE 344-1975. The seismic qualification of these Motor Control Centers is described in Westinghouse Seismic Certification Report 40226-SCR-0, dated February 28, 1983. This report demonstrates that the Five Star Motor Control Center possess sufficient integrity to sustain, and remain operational, without loss or compromise of structure or function for the worst seismic condition within the station.

3.10.2.5 Motor Operated Valves (MOV)

Prototype MOV’s have been sine beat tested to levels greater than given by the floor response spectra.

A model SMB-0-25 was seismically tested by Lockheed Electronics Company to demonstrate the suitability of Philadelphia Gear Corp. Limitorque valves for nuclear containment service. The limitorque operator was mounted on a test stand and had a threaded valve stem being driven to simulate opening and closing of the valve. The operation was electrically connected so as to stop at the full close position by means of the torque switch and to stop in the full open position by means of the geared limit switch. All contacts not being used for motor control were wired to electric indicating light as a remote panel.

An exploratory scan was conducted from 5 Hz to 35 Hz, and no resonant frequencies were found. The unit was then vibration tested in each axis for three cycles of two minutes on, one minute off, at a level of 5.3g. The unit was operated electrically to both the full open and full

closed position and all torque and limit switches functioned properly. All electrical and mechanical devices on the operator worked successfully.

3.10.2.6 Safety Features Actuation System (SFAS)

The SFAS equipment and cabinets have been vibration tested in accordance with IEEE 344-1971. The seismic test report for SFAS equipment is presented in Reference 52.

3.10.2.7 Batteries and Racks

Prototype cells, mounted in a battery rack, have been vibration tested in accordance with IEEE 344-1975.

The seismic test report for the NCX-21 and NCN-21 model battery is presented in Reference 92

3.10.2.8 Inverters and Chargers

The Solid State Inc. inverters have been seismically qualified in accordance with IEEE 344-1975. The seismic qualification of this equipment is described in Farwell and Hendricks, Inc. Qualification Report 20223, dated May 10, 1988. The report demonstrates that the Solid State Inc. inverter design is of a sufficient structural integrity to remain functional during and following a maximum seismic event at Davis-Besse.

Cyberex Battery Chargers

The chargers have been demonstrated as suitable for service in accordance with IEEE 344-1971.

The chargers were seismically tested at the Systems Control Corporation, Iron Mountain, Michigan of the Westinghouse Engineering Service Division. Single frequency sine-beat testing was used with 0.332g applied horizontally and 0.22g applied simultaneously in the vertical direction. The equipment was monitored and maximum accelerations recorded over a range of 20 frequencies, including all resonant frequencies, from 1 Hz to 35 Hz. Vital circuits were monitored to indicate any malfunctions, none of which occurred before, during or after testing. No failures occurred.

AMETEK SolidState Controls Battery Chargers

The AMETEK SolidState Controls battery chargers have been seismically qualified in accordance with IEEE 344-1975. The seismic qualification of this equipment is described in Curtiss Wright Flow Control Company, QualTech NP, Report Number Q1112.0 Revision 0, dated May 31, 2011. The report demonstrates that the AMETEK SolidState Controls battery charger design is of a sufficient structural integrity to remain functional during and following a maximum seismic event at Davis-Besse.

3.10.2.9 Electrical Penetrations

Prototype penetrations have been vibration tested in accordance with IEEE 344-1971.

Two prototype electrical penetrations were assembled by the Amphenol SAMS Division of Bunker Ramo Corp., California and subsequently seismically tested at the Applied Nucleonics Co., California.

An 18-inch and a 12-inch penetration assembly were each subjected to a resonant sinusoidal frequency search test in horizontal and vertical directions. No resonant frequencies were found in the range 0-33 Hz.

Resonance frequency sweeps were performed in all directions, and it was demonstrated that the equipment natural frequency was very much higher than 33 Hz. After the resonance frequency search, the equipment was prooftested at the building resonant frequencies of 5 Hz and 15 Hz using up to 1.0 g horizontally and 0.3g vertically as input motions, which are in excess of the required acceleration for an SSE. Proof testing was performed on both the vertical and horizontal axes simultaneously and no electrical or mechanical failure occurred.

3.10.2.10 Seismic Design of Reactor Protection System (RPS)

The RPS is capable of performing its intended functions during and after a Maximum Possible Earthquake. The seismic qualification program complies with the requirements of IEEE Standard 344-1971, "IEEE Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations."

The seismic test procedure verifies that (1) the cabinet is structurally capable of withstanding the seismic event, (2) that modules within the cabinets perform their protective functions, and (3) that remote equipment performs the necessary safety functions during the earthquake. The qualification testing of RPS equipment is contained in B&W Topical Report BAW-10003A, Rev. 4 "Qualification Testing of Protection System Instrumentation (January 1976)." Subsequent to the submittal and approval of BAW-10003A, Rev. 4, replacement RPS Reactor Trip Modules manufactured by Framatome have been approved for Installation in the RPS. These Reactor Trip Modules are seismically qualified to the same requirements as Identified in BAW-10003A, Rev. 4 in accordance with FTI Reactor Trip Module Qualification Test Report 51-5006947-00.

3.10.2.11 Class 1 Instrumentation

The following instrumentation has been successfully seismically tested as described in B&W Topical Report BAW-10003A, Revision 4. The first five are inputs to the SFAS or RPS.

a. Reactor coolant pressure (wide range pressure transmitter)

b. Reactor coolant flow (transmitters)

c. Reactor coolant outlet temperature (RTD)

d. Containment vessel pressure (switches)

e. Power range nuclear instrumentation (NI)

f. Steam generator level (startup range transmitters)

g. Steam generator level (operate range transmitters)

h. Pressurizer level (transmitters)

Replacement parts for this equipment will be qualified in accordance with the applicable standard (at least IEEE 344-1971) and to the acceleration levels required for the application. A description of the analysis or testing procedure will be contained in the applicable test report.

3.10.2.12 Electrical Cable Trays Carrying Class 1 Cable

Cable trays carrying Class 1 cable and their support systems have been mathematically modeled as an interdependent, idealized, single-degree-of-freedom system. The trays and their support systems were analyzed by the response spectrum technique for appropriate combinations of dead, live, Maximum Probable Earthquake, and Maximum Possible Earthquake loadings. The structural integrity of the trays and their support systems were thus confirmed for service conditions.

3.10.2.13. Seismic Analysis of Fuse Assemblies

A review of the available test data has shown that the structural integrity of the fuse/fuse reducer assembly is maintained during a Maximum Possible Earthquake. As for the functional integrity, these fuse reducers are designed to compensate for dimensional changes resulting from normal temperature variations of the fuse or fuse holder by means of spring tension. This assures that positive contact is not destroyed due to temperature variations. Because of the spring tension design feature, each reducer, once properly installed, maintains strong contact pressure. This pressure is adequate to assure functional integrity during a Maximum Possible Earthquake.

3.10.2.14 Seismic Qualification for Class 1 MCC, Item 99 MCC D1PA-structures 1 and 2. D1NA-structure 3

The Westinghouse Type W MCC was tested in accordance with IEEE Standard 344-1971 "Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations."

3.10.2.15 Class 1E Motors

Due to the physical size limitations and the impracticality of testing large electric motors while energized, seismic qualification was proven through analytic techniques as described below:

a. Component cooling pump motors:

The natural frequency of the rotor was computed to be 75 Hz. The motor, other than rotor, was considered to be a single mass model. Using a conservative one percent damping factor, the floor response spectrum indicates a maximum acceleration to be 4g horizontal and 3g vertical, which was used in the calculations. Analysis under the seismic loading showed that the rotor natural frequency, shaft deflection, shaft stresses, motor stress, and mounting device stresses are all within acceptable limits.

b. Containment spray pump motors:

The natural frequency of the rotor was found to be 109 Hz. The motor, other than the rotor, was considered to be a single massmodel. Using a conservative one percent damping factor, the floor response spectrum indicates a maximum acceleration of 4g horizontal and 3g vertical, which was used in the calculations. Analysis under the seismic loading shoved that the rotor

natural frequency, shaft deflection, shaft stresses, motor stresses, and mounting device stresses are all within acceptable limits.

c. Service water pump motors:

The natural frequency of the rotor was found to be 41.65 Hz. The motor, other than the rotor, was considered to be a single mass model. Using a conservative 1 percent damping factor, the floor response spectrum indicates a maximum acceleration of 4g horizontal and 3g vertical, which was used in the calculations. Analysis under the seismic loading showed that the rotor natural frequency, shaft stresses, shaft deflection, motor stresses, and mounting device stresses are all within acceptable limits.

d. High pressure injection pump motors and decay heat removal pump motors

Several analyses were performed, using both finite-element and lumped-mass techniques. The results of these analyses indicate that rate-as-mounted natural frequencies of auxiliary pumps and motors are greater than 33 Hz. Based on these results, it was concluded that in-situ tests for natural frequency tests were not required.

3.10.2.16 Main Control Boards - Auxiliary Shutdown Panel

a. Main control boards

The cabinets C-2 through 22 were analyzed by the modal response spectrum technique based on a three axis modal response spectrum. Using CALCOMP plots, models were developed shoving nodal coordinates, element connectivity, beam element properties, and nodal weights for the STARDYNE mathematical model, using beam, triangular and quadrilateral plate elements. The criterion for determining node locations was to select locations that correspond to bolt locations, major mass concentrations, and other structural discontinuities.

From the mathematical model, the free vibration results were determined to input to the seismic analysis section. The frequency and mode shapes were extracted from $(K - w_2 m)\phi = 0$, based on the “condensed” inertia matrix, by the Householder QR transformation technique.

The seismic analysis, including dead weight, was performed for the vertical-north south and vertical-east west directions. In addition, the analysis was performed for the Maximum Probable Earthquake Condition (8% g) and the Maximum Possible Earthquake (15% g). One percent damping was assumed and a cut off frequency of 35 Hz was used.

A static dead weight analysis was also performed using a 1.0g vertical load acting on the mathematical model.

As a result of this analysis, it was found that the highest panel acceleration, 5.51g’s, occurred on the front face of panel 22 with a north-south excitation. The largest axial and shear foundation bolt loads are 479 lbs. and 371 lbs. The highest plate stress intensity, s=34,769 psi, occurred on a triangular plate on panel 22 and was less than the material allowable stress of 42,000 psi. The largest beam stress intensity, s=4,587 psi, occurred also on panel 22 which was less than the material allowable stress of 42,000 psi.

b. Auxiliary shutdown panel

The suitability for service of the auxiliary shutdown panel was proven using analytic techniques. The scope of this analysis covered the adequacy of the auxiliary shutdown panel when subjected to the specified seismic criteria including the determination of the natural frequencies of vibration of the panel structure and various panel sections, dynamic amplification of seismic acceleration in the panel structure, stresses in the various structural and panel members under combined static and dynamic loadings with both horizontal and vertical affects included, buckling conditions in the panels, and stress conditions of anchor bolts and connections.

The auxiliary shutdown panel for the Davis-Besse Nuclear Power Station is structurally adequate to function properly when subjected to earthquake loadings as denoted in the specified seismic design criteria. The natural frequencies of vibration are greater than 33 Hz for the structural system and various panel sections including the instrument package panels.

Dynamic amplification of the flat spectra response of seismic acceleration is found to be a maximum of 0.3g in the structural system in the horizontal direction and 0.2g in the vertical direction. Use of these amplified dynamic loads as well as static loads with both horizontal and vertical affects show the bending, tensile, and shear stresses in the various structural members and connections to be much lower than the maximum allowable stresses. Stresses in the anchor bolts are quite low, and load capacity in the welds is quite high. Displacements in the instrument package panel sections are quite small and will not cause loss of function of the equipment. Loading of the panel plate sections has been found to be much less than that required for buckling. The analysis methods used herein are conservative; that is, structural elements, or the representations thereof, are subject to more critical conditions than would be encountered by the structural system in a prototype environment.

3.10.2.17 Class 1E Equipment Seismic Qualification Summary

The seismic qualification summary for the Class 1E equipment is presented in Table 3.10-1 (The seismic design basis is presented in Section 3.7).

TABLE 3.10-1

Class 1E Equipment Seismic Qualification Summary

Equipment	Location	Seismic Qualification Method	Test and/or Analysis Results
Emergency Diesel Generator	Aux.bldg – 585'-area 6	Analysis	Pass
Exciter regulator board	Aux.bldg – 585'-area 6	Analysis/Test	Pass
Engine control panel	Aux.bldg – 585'-area 6	Test	Pass
Relay and control boards	Aux.bldg – 585'-area 6	Test	Pass
Essential 4-16kV switchgear	Aux.bldg – 585'-area 6	Test	Pass
Essential unit substations	Aux.bldg – 603'-area 6	Test	Pass
Transformers	Aux.bldg – 603'-area 6	Analysis	Pass
Essential Motor Control Centers	Aux.bldg	Test	Pass
Class 1E Motor-Operated Valves	Aux.bldg/Cont Vessel	Test	Pass
Class 1E motors			
Component cooling pumps	Aux.bldg – 585'-area 7	Analysis	Pass
Containment spray pumps	Aux.bldg – 545'-areas 7 and 8	Analysis	Pass
Service water pumps	Intake Structure	Analysis	Pass
HP injection pumps	Aux.bldg – 545'-areas 7 and 8	Analysis	Pass
LP injection pumps	Aux.bldg – 545'-area 7	Analysis	Pass
Motors less than 200 hp	Aux.bldg/Cont Vessel	Analysis	Pass
Motor-operated valve motors	Aux.bldg/Cont Vessel	Test	Pass
Safety features actuation system	Aux.bldg – 623'-area 7	Test	Pass
Station batteries	Aux.bldg – 603'-area 6	Test	Pass
Battery chargers, inverters, rectifiers, and distribution panels	Aux.bldg – 603'-area 6	Test	Pass
Electrical penetrations	Aux.bldg/Cont Vessel	Test	Pass
Control Stations			
Main control panels	Aux.bldg – 623'-area 7	Analysis	Pass
Auxiliary shutdown panel	Aux.bldg – 585'-area 6	Analysis	Pass
Seismic cable tray supports	Aux.bldg/Cont Vessel	Analysis	Pass

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Safety related electrical equipment is capable of performing designated safety functions while exposed to applicable normal, test, accident, and post-accident environmental conditions. This section defines environmental parameters and describes the qualification process employed to demonstrate the required environmental capability.

Seismic qualification of safety-related mechanical and electrical equipment is presented in USAR Sections 3.9 and 3.10, respectively.

3.11.1 Equipment Qualification Program

The Davis-Besse Equipment Qualification Program falls under the scope of the Davis-Besse Nuclear Power Station's (DBNPS) 10 CFR 50 Appendix B Quality Assurance Program. Administrative procedures control the development and implementation of the equipment qualification (EQ) review, documentation control, equipment verification (walkdown), and implementation of EQ activities necessary to continue compliance to 10 CFR 50.49.

The qualification of safety related electrical equipment, located in a mild environment is ensured by compliance to the Quality Assurance Requirements of 10 CFR 50, Appendix B and Regulatory Guide 1.33 programmatic considerations.

Safety related electrical equipment, located in a harsh environment, is qualified to the requirements of 10 CFR 50.49 using the qualification program developed for the station.

The equipment qualification program also includes equipment that requires environmental qualification as a result of commitments to Regulatory Guide 1.97, Rev 3 - Post Accident Monitoring Instrumentation.

Implementation and maintenance of the Equipment Qualification Program ensures that:

1. Safety related electrical equipment will perform as intended during and following design basis events to ensure:

    a. The integrity of the reactor coolant pressure boundary.

    b. The capability to shut down the reactor and maintain it in a safe-shutdown condition.

    c. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR part 100 guidelines.

2. Non-safety related electric equipment, whose failure under postulated environmental conditions could prevent satisfactory accomplishment of the specified safety related electrical equipment required safety functions or mislead an operator, is qualified as required.

3. Post accident monitoring equipment, identified from the review of emergency procedures and those required by NRC Regulatory Guide 1.97, is qualified as required.

Safety related electrical equipment and components are qualified to meet their performance requirements under normal, test, and accident operating conditions during which they need to function. Environmental conditions have been developed for all safety-related areas (Section 3.11.1.2). Qualification is documented on the EQ Master List (Section 3.11.1.3) and in Equipment Qualification Packages (EQP) (Section 3.11.1.4) which describe the equipment functional capability during the described conditions and required timeframe. The EQP also demonstrates that the equipment will remain in a safe mode after its safety functions are performed.

3.11.1.1 Qualification Evaluation

Qualification evaluation is performed using checklists developed from NUREG-0588 and IE Bulletin 79-01B guidelines. Equipment procured prior to February 22, 1983 was evaluated to Division of Operating Reactors (DOR) guidelines, while equipment procured later than that date has been evaluated against 10 CFR 50.49 requirements. The following considerations comprised this evaluation process.

a. Classification - Equipment is classified into two categories.

   Equipment with a satisfactory 40 year design life to IEEE 323-1974 by type test, or combination of type test/analysis or,

   Equipment requiring an on-going qualification and/or replacement program to continue its life for the design life of the plant.

b. Operability - Equipment or component operability is defined as that condition which meets the requirements of performing its design basis safety function and the manufacturer’s technical specification.

c. Qualification Methods - Qualification can be by type testing or a combination of type test/analysis. Testing is the preferred method of qualification. Analysis has been used to verify or supplement test results.

   Service conditions simulated during the testing were reviewed to ensure that they enveloped the accident environments. The test duration and environmental parameters utilized in the test were reviewed to ensure that they equaled or exceeded specified values. When tests were found to be less severe than specified, a determination of the adequacy of the test is made on a case-by-case basis.

   The test specimen model, design, and construction material are reviewed against the equipment being qualified to verify applicability of test results.

   The selected test sequence is reviewed and, when determined not to be in accordance with the guidelines of IEEE 323-1974, is reevaluated for adequacy.

   Tests which are successful using components that had not been preaged are considered acceptable, provided the components do not contain materials known to be susceptible to significant degradation due to aging effects.

   Operational modes tested are reviewed to ensure that they are representative of the actual application requirements as defined in the procurement documents. The

length of time that each item of equipment is required to operate is reviewed against test data to ensure that the equipment's designated life is greater than the length of time the equipment is required to operate.

Failures identified during environmental qualification are reviewed and evaluated relative to their effect on the ability of the component to perform its required function. If a component fails at any time during the test, the applicability of the test with regard to demonstrating the ability of the component to function for the entire period prior to the failure is considered on a case-by-case basis.

Where seals are included as part of the component, the test results and conclusions include the seals. Materials used for terminating cables and similar components are addressed separately.

Qualification by a combination of methods (test, evaluation, analysis) is identified in the Equipment Qualification Package (EQP). A determination of the adequacy of the qualification methods used is made on a case-by-case basis.

d. Aging - A specified value of forty (40) years has been used for evaluation of all components. Considerations made for materials which are susceptible to thermal, vibration, electrical, mechanical, and/or radiation aging are addressed in the applicable System Component Evaluation Worksheet (SCEW) found in the equipment qualification package (EQP) for the equipment being evaluated. The order of application of the simulated aging conditions is reviewed against IEEE 323-1974 requirements to ensure that the most severe sequence is applied. Known synergistic effects, or those found during testing were investigated on a case-by-case basis. For those components not tested, the Arrhenius aging technique was used to perform end of life calculations.

e. Margin - Qualification type tests were reviewed to verify that adequate margin exists between the most severe specified service conditions of the plant and the conditions used in type testing.

f. Submergence - Equipment has been evaluated for submergence as a result of LOCA or HELB. Where submergence is possible it is identified on the SCEW sheet. All safety-related equipment located in areas with submergence potential is either qualified to operate submerged, or it is electrically isolated, and administratively controlled, or its location has been verified to be above the submergence level.

g. Exempt Equipment - Equipment which may be exempt from certain environmental qualification is evaluated against the following considerations: (e.g., equipment may be exempt from line break or submergence parameters but may require radiation qualification).

Equipment does not perform safety functions in the harsh environment and equipment failure in harsh environment will not impact safety related functions or mislead an operator.

Equipment performs its safety function before exposure to harsh environment and its subsequent failure as a result of harsh environment does not degrade other

safety functions or mislead the operator, and it has been determined that the equipment is not required post accident.

Safety-related functions can be accomplished by some other designated equipment that has been adequately qualified and satisfies the single-failure criterion.

A failure analysis is performed on the exempt equipment. Isolation devices and/or administrative controls are provided so that the fault in the exempt equipment does not travel back and impact other electrical equipment required during the design basis event.

h. Replacement Program - For equipment with a designated life less than design plant life, a maintenance/surveillance/replacement program based upon test data and analysis is utilized to evaluate the qualified life. The availability of new test data is used to modify the program as necessary. Replacement equipment will be installed if qualified life is being approached and cannot be extended to the plant design life.

3.11.1.2 Environmental Conditions

Environmental conditions have been developed for all safety-related areas of the plant. These environmental conditions are listed by a system of room numbers, each "room" defining a specific area or areas of the plant. The environmental design conditions have been developed utilizing the information presented in Section 3.6. A separate listing of these environmental conditions, by room, is maintained. The environmental parameters include temperature, pressure, relative humidity, chemical spray potential, submergence potential, accident duration, and gamma/beta (where applicable) radiation dose. Where applicable, these parameters are presented as tune-based profiles. USAR Section 3.11.2.2.1 describes the analyses and the respective computer programs used to develop HELB temperature pressure profiles.

Normal operating environmental conditions arc defined as conditions expected during routine plant operations and test operations. These environmental conditions represent the expected conditions that occur during routine plant and test operations.

Accident environmental conditions result from design basis accidents, described in Sections 3.6, 6.2, and Chapter 15. The following parameters have been evaluated:

a. Temperature and pressure-Time histories have been developed for the individual line breaks described in Section 3.6. Compartment pressurization, flow/vent paths, and duration of each event have been included in this evaluation.

b. Chemical Spray - has been reviewed for equipment inside containment. Plant specific worst case values of pH = 5.0 and concentration of 1800 ppm Boron were originally used. Later analyses performed for a 24 month cycle provided for a concentration of 2800 ppm Boron which was evaluated by Reference 83. If the plant values were not addressed in the vendor's test report, an engineering analysis justification was provided in the equipment qualification package.

c. Radiation - the post accident radiation doses to equipment are calculated using the following source terms (expressed in core release fractions).

Containment Air	100%	Noble Gases
	25%	Halogens
Pressurized Liquids	100%	Noble Gases
	50%	Halogens
	1%	Remainder
Depressurized Liquids	0%	Noble Gases
	50%	Halogens
	1%	Remainder

The exposure rate for specific equipment was calculated from the radioactive source and integrated over the one year period immediately following the accident. The credit for shielding afforded by the internal structures is considered on a selected basis. The total integrated dose (TID) used in qualifying the equipment consists of the normal operating dose added to the dose contribution during and following the course of the accident. Only the TID received up to the post-accident period in which the equipment is required to function is used for equipment qualification.

As a part of the 18 month cycle evaluation, the calculated equipment qualification doses using a 277 day equilibrium cycle were increased by 20 percent. The core activity for a 24 month fuel cycle is not significantly different than the core activities calculated for a 277 day fuel cycle, therefore the 20 percent increase previously applied bounds the 24 month cycle equipment qualification doses.

The radiation levels outside containment are calculated from the radiation levels for each room as a result of recirculated fluids, and containment shine.

3.11.1.3 Equipment Qualification Master List

The EQ Master Equipment List (EQML) consists of the equipment located in a harsh environment that are required to mitigate the consequences of a LOCA or HELB Design Basis Accident, restore the plant to a safe shutdown condition and allow sufficient post accident sampling, monitoring, and radiation monitoring. Components located in harsh environments are listed according to their plant ID numbers as shown on the P & ID drawings.

The EQML provides a list of equipment within the scope of the EQ program with sufficient detail to identify the equipment in the plant and to provide a link to the EQ files which establish the equipment environmental qualification. Typically this includes an appropriate combination of the Plant ID number, Description, EQ Package number, Manufacturer, Model, Location, but may contain more or less information provided the link between the plant equipment and the qualification documentation is maintained.

3.11.1.4 Equipment Qualification Packages

The Equipment Qualification Package (EQP) provides qualification documentation for safety related electrical equipment and draws information from other sources such as: EQ Master Equipment List, environmental conditions, and equipment test reports. The test report and other related equipment documentation is included within the EQP so that it serves as a single source reference for that equipment qualification record.

The review of available documentation is performed using 10 CFR 50.49 or DOR Guidelines. The review consists of the following:

Component Identification - Identify the model number, manufacturer, location interfacing/installation requirement, maintenance requirements.

Compliance with Regulatory Requirements-The review will identify open Technical Evaluation Report (TER) items and their resolution, IE Notices and Bulletins, review and resolution.

Operating conditions - The review will include system function, operating time, etc.

Environmental Conditions - DBA considered; i.e., Flood level, temperature, pressure, relative humidity, radiation, chemical spray, aging.

Equipment Performance - Response time, accuracy, voltage, pressure, etc.

After completion and review of each EQ Package, it is approved and issued.

3.11.2 Methodology for Development of LOCA and HELB Environments

3.11.2.1 Parameters

The results of the Containment Vessel's accident analysis were calculated to be:

	Initial Temperature	Maximum Vapor Temperature (F)	Maximum* Pressure (psig)	Humidity (%)
LOCA*	90	256.4	37.9	100
LOCA*	120	259.2	37.8	100
MSLB**	120	364.9	27.2	100
MSLB***	120	378.0	27.6	100

* Maximum Pressure includes initial pressure allowance (above atmospheric) of 0.9 psi per Technical Specifications
**Applicable to existing qualified equipment (Reference 103, Category II)
***Applicable to replacement equipment (Reference 103, Category I)

Separate temperature profiles are generated for a Main Steam Line Break (MSLB) and LOCA. Electrical equipment temperatures during a MSLB inside of the Containment Vessel (see Section 6.2.1.3.2 for predicted values) and in the Auxiliary Building (see Reference 67 for predicted values) are determined with a thermal lag model in accordance with NUREG-0588 (Reference 103). During a LOCA, electrical equipment temperatures are conservatively evaluated to be equal to the temperature of the Containment Vessel's atmosphere rather than computing equipment temperatures based on a thermal lag model.

As an additional conservatism, the peak Containment Vessel vapor temperature during a LOCA is set to 284 F which is the saturation temperature corresponding to the Containment Vessel's pressure limit of 38 psig. The peak temperature of 284 F is conservative because: (1) the large containment volume results in a lower predicted peak temperature and, (2) because no safety-related electrical equipment is located in the upper regions of containment, no additional margin is needed. The use of the saturation temperature corresponding to the peak containment pressure was discussed with the NRC. (Reference Serial Letter 750)

The peak pressure during a LOCA bounds the peak pressure during a MSLB. Therefore, the peak pressure during a LOCA is utilized by environmental analyses of electrical equipment

inside of the Containment Vessel for all accident scenarios. The primary reason for the low peak pressures inside of the Containment Vessel during postulated accidents is due to the containment's large volume (i.e., 2.834E06 ft^3).

The temperature profile utilized for equipment qualification should include margin to account for higher-than-average temperatures in the upper regions of the Containment Vessel that can exist due to stratification, especially following a MSLB. Since there is no safety-grade electrical equipment currently located in the upper regions of the Containment Vessel, this margin is not considered in the design analysis.

3.11.2.1.1 LOCA Analysis

A detailed discussion of Containment Vessel response modeling, including methods and assumptions, is provided in Section 6.2.

3.11.2.2 High Energy Line Break (HELB) Outside Containment

HELB analyses (Ref. 54, 60, 61, 62, 63, 67) were performed to predict temperature and pressure conditions that would result at essential equipment locations in the Auxiliary Building following pipe breaks in the following lines.

Main Steam to Auxiliary Feedwater Pump Turbine
Main Steam (MS)
Main Feedwater (MF)
Steam Generator Blowdown (SGBD)
Auxiliary Steam (AS)

A complete description of the methods used to identify pipe break locations and resultant environments is found in USAR Section 3.6.

A HELB analysis (Reference 100) was performed to predict temperature and pressure conditions at essential equipment located in structures adjoining the Turbine Building following breaks in:

Main Steam
Main Feedwater
Extraction Steam

A complete description of the analysis is provided in USAR Section 3.6.2.7.1.16.

3.11.2.2.1 HELB Computer Analysis

Either GOTHIC version 7.0 or PCFLUD methodology is used to perform High Energy Line Break (HELB) Compartment Pressurization Analysis outside of containment. PCFLUD is further described in Section 3.11.2.2.1.1. GOTHIC is further described in Section 3.11.2.2.1.2. The chosen methodology, either GOTHIC or PCFLUD must be applied in its entirety. There shall be no mixing of the attributes of one methodology with the attributes of another. Each methodology must also be applied in compliance with the restrictions and limitations specifically assigned to it.

3.11.2.2.1.1 PCFLUD HELB Computer Analysis

PCFLUD is a personal computer based code developed by Bechtel Power Corp, and is used to analyze HELBs in the Auxiliary Building.

PCFLUD considers a thermal and hydraulic system as a series of interconnecting user-defined control volumes. The program solves the mass and energy balances for volumes assumed to contain one-dimensional homogeneous fluid (water, air and steam) with the phases in thermodynamic equilibrium.

The main assumptions in PCFLUD are:

- One-dimensional fluid and heat conduction equations.
- Homogenous fluid equations with the phase in thermodynamic equilibrium.
- Steady-state empirical correlations to estimate heat transfer coefficients, heat fluxes, and critical mass fluxes.

PCFLUD is a proprietary program which is maintained by Bechtel Power Corp.

PCFLUD is based on an NRC approved code and meets the requirements of the NRC Standard Review Plan, Section 6.2.1.2, Subcompartment Analysis. The code is further described in Section 3.6.2.7.1.1.

The computer program (Reference 100) utilized in analyzing high energy line breaks in the Turbine Building allows one, two, or three dimensional models to be developed. It includes non-homogeneous flued distributions so that buoyancy driven mass and energy flow can be modeled. It also includes conservative heat transfer and condensation correlations for predicting environmental parameters. See section 3.6.2.7.1.1 for additional detail.

3.11.2.2.1.2 GOTHIC HELB Computer Analysis

GOTHIC 7.0 is a personal computer based code developed by Numerical Applications, Inc. (NAI) for the Electric Power Research Institute (EPRI), and is used to determine High Energy Line Break (HELB) Compartment Pressurization Analysis outside of containment.

GOTHIC 7.0 is a general purpose thermal hydraulics program that solves the conservation equations for mass, energy and momentum for multi-component, multi-phase flow for HELB response compartment analyses. Interface models between phases allow for thermal non-equilibrium and unequal phase velocities.

GOTHIC 7.0 contains the following NRC reviewed and approved features:

a) Inter-phase mass, energy and momentum transfer rates obtained through constitutive relations.
b) Separate mass equation solved for each fluid phase and gas component.
c) Separate energy equation solved for each fluid phase.
d) Compressible flow through all fluid phases.
e) Can model valve closure time.
f) Water in liquid phase can be accumulated at the bottom of a control volume.
g) Can model actual air properties. Air treated as an ideal gas for mixture calculations.
h) Can have relative humidity values other than zero or 100 percent.

i) Water in the vapor phase is dependent upon momentum, mass and energy equations.

GOTHIC 7.0 is an NRC approved code for use in HELB Compartment Pressurization Analysis and its use meets the requirements of the NRC SER (Reference 101), with the following limitations on its application:

a) Initial atmospheric conditions shall be selected to maximize the resultant differential pressure.

b) Nodalization shall be chosen such that there is no substantial pressure gradient within a node.

c) For “Duct Collapse” vent pathways that are not open prior to the pressurization, the collapse of the duct should be based on a dynamic analysis of the sub-compartment pressure response to the pipe rupture and this analysis should be supported by experimental data.

   In lieu of supporting experimental data, “Duct Collapse” can be modeled in a way that produces the most conservative results and with the “Duct Collapse” flow area equal to the duct cross-section flow area. This can be accomplished by modeling both with and without duct collapse, and using the most conservative case. With this type of application, no experimental data is necessary to support the application.

d) Homogenous equilibrium mode, with 100 percent water entrainment, shall be assumed except where this would yield non-conservative results (i.e., in the case of sub-cooled break flow). In this case (i.e., sub-cooled break flow), the GOTHIC “Drop Liquid Conversion” model will be enabled, with the default option of Forced Equilibrium disabled (i.e., enabling the thermal hydraulic non-equilibrium model). The use of the Drop Liquid Conversion Model is not allowed in any other application.

e) A Uchida condensation heat transfer correlation option shall be used in GOTHIC to calculate the heat transfer to structures.

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APPENDIX 3A

DESCRIPTIONS OF LOAD FACTORS FOR SHIELD BUILDING AND CONTAINMENT VESSEL INTERNAL STRUCTURE DESIGN

APPENDIX 3A

DESCRIPTIONS OF LOAD FACTORS FOR SHEILD BUILDING AND CONTAINMENT VESSEL INTERNAL STRUCTURE DESIGN

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APPENDIX 3A

DESCRIPTIONS OF LOAD FACTORS FOR SHIELD BUILDING AND CONTAINMENT VESSEL INTERNAL STRUCTURE DESIGN

The following are descriptions for load factors used in the design equations in Subsections 3.8.2.1.4, 3.8.2.2.4 and 3.8.2.3.4.

3A.1.0 Dead Load

The dead load factor for the shield building design is 1.0 in combination with the other factored loads. The reason for using this load factor instead of those suggested by ACI-318-63 Code is that the code is written for general types of construction. Since the dead load of the Shield Building can be accurately determined, a load factor of 1.0 is justified.

A factor of 1.0 is also justified for the internal structure since the most unpredictable dead loads such as the major equipment loads are excluded from the dead load, and considered separately. During operating and shutdown conditions the containment internal structures are investigated according to ACI-318-63 Code, and the dead load factor is 1.5.

3A.2.0 Live Load

The live load that would be present along with accident, seismic and wind loads would produce a very small portion of the stress at any point. Also, it is extremely unlikely that the full live load would be present over a large area at the time of an unusual occurrence. For these reasons, a load factor of 1.0 is felt to be justified.

3A.3.0 Earthquake Load

For the Maximum Probable (smaller) Earthquake of 0.08g ground acceleration, a factor of 1.25 is used in combination with the other factored loads in designing both the Shield Building and the containment internal structure. The selection of this load factor is in agreement with past and current practice of concrete containment design for nuclear power plants and also the ACI-318-63 Code. Under the condition of Maximum Possible (larger) Earthquake of 0.15g ground acceleration a factor of 1.0 is used both for the Shield Building and the containment internal structure.

The factor of unity is consistent with the loading condition which is to demonstrate no loss of function under a maximum possible (larger) loading condition.

3A.4.0 Wind Load

The Shield Building is designed to withstand the wind load and the associated pressures. The design load factor is 1.25 in combination with the other factored loads in accordance with the ACI-318-63 Code.

3A.5.0 Tornado Load

The Shield Building is designed to withstand the tornado load and the associated pressure differential without loss of function. A load factor of 1.0 is used and is consistent with the loading condition and demonstrates no loss of function under a maximum hypothetical condition.

3A.6.0 Temperature Load

The Shield Building and the containment internal structures are designed for thermal loads (temperature gradients) in combination with the other factored loads. Accurate extremes are determined in establishing the temperature gradient through walls, dome, and slabs. Therefore a thermal load factor of unity with variations of +5 percent and +10 percent is used for the Shield Building and containment internal structure, respectively. The variations in load factors are consistent with the degree of structural complexities in the relative structures, and are considered in the same category as dead loads.

3A.7.0 Loss-of-Coolant Accident Load

The steel containment vessel practically isolates the Shield Building from the Reactor Coolant Systems and therefore eliminates significant pressure and temperature loads on the Shield Building during an accident. However, small pressure buildups and temperature changes do occur in the annulus during the accident. These are taken into account in design. The load factor for LOCA is 1.0 in combination with the other factored loads. The Shield Building is also analyzed for one and one-half times the LOCA load for no loss of function.

3A.8.0 Equipment Load

The equipment loads in the containment internal structure are separated from the dead load so that more reasonable load factors can be assigned to the equipment dead weight. A factor of 1.5 is used for equipment load during operating and shutdown conditions in accordance with the ACI-318-63 Code. Otherwise, a factor of 1.25 is assigned to allow for inaccuracy resulting from weight calculations and load distributions. For the case of Maximum Possible (larger) Earthquake, a factor of unity is assigned to the equipment load.

3A.9.0 Miscellaneous Loads

During a loss of coolant accident, the containment internal structures are subjected to such accident loads as pressure, thermal, missile, jet force, and piping anchor loads. Load factors are assigned to different load sources in loading combinations as shown in Subsection 3.8.2. The load factors are assigned recognizing the degree of accuracy available in determining the loading and also the unlikely combination of simultaneous load occurrences.

APPENDIX 3B

SPLICING REINFORCING BAR USING THE CADWELD PROCESS IN CONTAINMENT VESSEL INTERNAL STRUCTURES

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APPENDIX 3B

SPLICING REINFORCING BAR USING THE CADWELD PROCESS IN CONTAINMENT VESSEL INTERNAL STRUCTURES

3B.1.0 Scope

This Appendix was written for construction phase activities. Section 3B.9.0 covers operational phase activities. These procedures cover the mechanical splicing of deformed concrete reinforcing bar for full tensile loading. The average tensile strength of the splices is equal to or greater than the specified minimum tensile strength of the rebar. The minimum acceptable tensile strength of any splices is 125 percent of the specified minimum yield strength for the particular bar size and ASTM specification.

3B.2.0 Records

Adequate records are maintained of all splices made by the Cadweld Process for “T” series connections. Records include splice location, splicing crew and material used.

3B.3.0 Qualifications of Operators

Prior to the production splicing of reinforcing bars, each operator or crew, including the foreman or supervisor for that crew, prepares and tests a joint for each of the positions used in production work. These splices are made and tested in strict accordance with this procedure. The completed splices must meet the acceptance standards of Paragraph 6.0 for workmanship, visual quality, and minimum tensile strength. A list containing the names of qualified operators and their qualification test results is maintained at the job site.

3B.4.0 Procedure

All joints are made in accordance with the manufacturer’s instruction sheets, “Rebar Instructions for Vertical Column Joints,” plus the following requirements:

a. A manufacturer’s representative, experienced in Cadweld splicing of reinforcing bar, was present at the job site at the outset of the work to demonstrate the equipment and techniques used for making quality splices. He was also present for at least the first 50 production splices and verified that the equipment was used correctly and that quality splices were obtained.

b. The splice sleeves, exothermic powder, and graphite molds are stored in a clean dry area with adequate protection from the elements to prevent absorption of moisture.

c. Each splice sleeve is visually examined immediately prior to use to insure the absence of rust and other foreign material on the inside diameter surface.

d. The graphite molds are preheated with an oxyacetylene or propane torch to drive off moisture at the beginning of each shift when the molds are cold or when a new mold is used.

e. Bar ends which are spliced are power brushed to remove rust, concrete and other foreign material. Prior to power brushing all water, grease and paint is removed by heating the bar ends with an oxyacetylene or propane torch.

f. A permanent line is marked 12 inches back from the end of each bar for a reference point to confirm that the bar ends are properly centered in the splice sleeve.

g. Immediately before the splice sleeve is placed into final position, the previously cleaned bar ends are preheated with an oxyacetylene or propane torch to insure complete absence of moisture.

h. Special attention is given to maintaining the alignment of the sleeve and guide tube to ensure a proper fill.

i. When the temperature is below freezing or the relative humidity is above 65 percent, the splice sleeve is externally preheated with an oxyacetylene or propane torch after all materials and equipment are in position.

j. Splice sleeves are wrapped in a special rust inhibiting paper. Sleeves are not unwrapped until they are used in the joining procedure.

3B.5.0 Testing

All completed splices are visually inspected at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. For purposes of quality control, production splices representing the work of each splicing crew are tensile tested for each position, bar size, and grade of bar. The number and frequency of tests for each splicing crew is as follows:

a. One out of the first lot of ten splices for each position, bar size, and grade of bar.

b. Two out of the next and subsequent lots of one hundred splices for each position, bar size, and grade of bar.

The first five tensile test specimens are made by cutting out randomly selected production splices. Thereafter; at least one of every twenty tensile tests is made from production splices. At least one tensile test specimen is cut out from the actual production splice for each position, bar size, and grade of bar. The remainder of the required tensile tests is made from three feet long test bars spliced in sequence with and in an otherwise identical manner as the production splices.

3B.6.0 Acceptance Standards

Sound, nonporous filler metal is visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed 1/4 inch from the end of the sleeve due to the packing material, and is not considered a poor fill.

Splices which contain slag or porous metal in the riser, tap hole, or at the ends of the sleeves (general porosity) are rejected. A single shrinkage bubble present below the riser is not detrimental and should be distinguished from general porosity as described above.

There is evidence of filler material between the sleeve and the bar for the full 360 degrees; however, the splice sleeves need not be exactly concentric or axially aligned with the bars.

The Cadweld splices, both horizontal and vertical, may contain voids at either or both ends of the Cadweld splice sleeve. At the end of the Cadweld splice sleeves, the acceptable size void for an 18S splice does not exceed three (3) square inches per end of splice sleeve. The area of the void is assumed to be the circumferential length as measured at the inside face of the sleeve times the maximum depth of wire probe minus 3/16".

The average tensile strength of the Cadweld joints is equal to or greater than the minimum tensile strength for the particular grade of reinforcing steel as specified in the appropriate ASTM standard. The minimum strength of the Cadweld joints is equal to or greater than 125 percent of the specified minimum yield strength for the particular bar.

3B.7.0 Repairs

Splices which do not meet the visual quality acceptance standards of Paragraph 6.0 are rejected and completely removed. The bars are then re-jointed with a new splice made in accordance with these procedures.

No failures of Cadweld splices below the required minimum tensile strength are expected; however, in the unlikely event that one should occur it is sent to an independent testing, laboratory for analysis of failure. Based on the Test Lab's report, additional samples are taken to ensure that there are no other defective welds.

3B.8.0 Margin of Safety

The average margin of reserve strength that exists over and above 125 percent of the specified minimum yield strength of a particular bar size has been determined to be 37 percent. This value was obtained from actual test data.

3B.9.0 Plant Operation Phase Requirements

Splicing reinforcing bars shall be performed in accordance with individual project specifications. Project Specifications shall include or reference manufacturer's instructions and comply with the applicable requirements of ANSI N45.2.5-1974. The Quality Assurance Program Manual (QAPM) identifies specific subarticles of ASME Section III Division 2-1995 edition that will be used in lieu of the corresponding requirements in ANSI N45.2.5-1974.

APPENDIX 3C

JUSTIFICATION FOR YIELD REDUCTION FACTORS (ø — FACTORS) USED IN DETERMINING YIELD STRENGTH OF SHIELD BUILDING AND CONTAINMENT VESSEL INTERNAL STRUCTURES

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JUSTIFICATION FOR YIELD REDUCTION FACTORS (ø — FACTORS) USED IN DETERMINING YIELD STRENGTH OF SHIELD BUILDING AND CONTAINMENT VESSEL INTERNAL STRUCTURES

The ø factors are provided to allow for variations in materials and workmanship. In the ACI Code 318-63, ø varies with the type of stress or member considered; that is, with flexure, bond or shear stress, or compression.

The ø factor is multiplied into the basic strength equation or, for shear, into the basic permissible unit shear to obtain the dependable strength. The basic strength equation gives the "ideal" strength assuming materials are as strong as specified, sizes are as shown on the drawings, the workmanship is excellent, and the strength equation itself is theoretically correct. The practical, dependable strength may be something less since all these factors vary.

The ACI Code provides for these variables by using these ø factors:

ø = 0.90 for flexure.
ø = 0.85 for diagonal tension, bond, and anchorage.
ø = 0.75 for spirally reinforced compression members.
ø = 0.70 for tied compression members.

The ø factors provide for the possibility that small adverse variations in material strengths, dimensions and workmanship occasionally may combine to result in under-capacity. The ø factors for columns are significantly lower since column failure may cause the entire structure to collapse.

The additional ø values used represent Bechtel's best judgment of how much understrength should be assigned to each material and condition not covered directly by the ACI Code. The additional ø factors are selected based on material quality in relation to the existing ø factors.

Conventional concrete design of beams requires that the design be controlled by yielding of the tensile reinforcing steel. This steel is generally spliced by lapping in an area of reduced tension. For members in flexure, ACI uses ø = 0.90. The same reasoning has been applied in assigning a value of ø = 0.90 to reinforcing steel in tension, which now includes axial tension. However, the code recognizes the possibility of reduced bond of bars at the laps by specifying an ø of 0.85. Mechanical and welded splices develop at least 125 percent of the yield strength of the reinforcing steel. Therefore, ø = 0.90 is recommended for this type of splice.

APPENDIX 3D

CONFORMANCE WITH THE NRC GENERAL DESIGN CRITERIA, SAFETY GUIDES, AND INFORMATION GUIDES

APPENDIX 3D

CONFORMANCE WITH THE NRC GENERAL DESIGN CRITERIA, SAFETY GUIDES, AND INFORMATION GUIDES

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APPENDIX 3D

CONFORMANCE WITH THE NRC GENERAL DESIGN CRITERIA, SAFETY GUIDES, AND INFORMATION GUIDES

3D.1.0 NRC General Design Criteria

The design of the Davis-Besse Nuclear Power Station meets the intent of Appendix A, 10CFR50, the General Design Criteria for Nuclear Power Plants as published in the Federal Register on February 20, 1971, and as amended in the Federal Register on July 7, 1971.

3D.1.1 Criterion 1 - Quality Standards and Records

Structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they are identified and evaluated to determine their applicability, adequacy, and sufficiency and are supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program is established and implemented in order to provide adequate assurance that these structures, systems, and components satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety are maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Quality standards relating to safety-related systems, structures, and components are generally contained in codes such as the ASME Boiler and Pressure Vessel Code. The applicability of these codes is identified throughout this report and is summarized in Chapter 3.

Chapter 17 describes the Project Quality Assurance Program established to provide adequate assurance that safety-related structures, systems, and components satisfactorily perform their safety functions. This section also describes the procedures for generating and maintaining appropriate design, fabrication, erection, test, operation, maintenance, and refueling records throughout the life of the units.

References: USAR Chapters 3 and 17

3D.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

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

The appropriate load combinations are provided in Subsections 3.8.1 and 3.8.2 for structures, systems, and components important to safety. The design bases for the internal and external missiles including tornado missiles are discussed in Section 3.5, while the tornadoes and

hurricanes are discussed in Section 3.3. The seismic criteria and associated design bases are discussed in Section 3.7.

Reference: USAR Chapter 3

3D.1.3 Criterion 3 - Fire Protection

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

The system arrangements and component designs are such that water damage to critical systems does not prevent the safe shutdown of the station.

Equipment and facilities for fire protection (Subsection 9.5.1), including detection, alarm, and extinguishment, are provided to protect both station and personnel from fire, explosion, and the resultant release of toxic vapors. Both wet and dry type fire-fighting equipment are provided.

Normal fire protection is provided by deluge systems, sprinklers, hose lines, and portable extinguishers.

The fire protection system is designed in accordance with the requirements of the American Nuclear Insurers and Nuclear Electric Insurance Limited as a guide and applicable codes and regulations of the State of Ohio.

The fire suppression system is provided with test hose valves that are accessible for periodic testing.

Reference: USAR Subsection 9.5.1

3D.1.4 Criterion 4 – Environmental and Missile Design Bases

Structures, systems and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Protective walls and slabs, local missile shielding, and restraining devices are provided to protect the containment vessel and engineered safety features within the containment vessel against damage.

The Reactor Coolant System is protected from internal missiles by the concrete shield wall enclosing it.

The Containment Vessel interior structure is designed to sustain loads which could result from failure of major equipment or piping such as jet thrusts, jet impingement or local pressure transients.

The Seismic Class I structures, systems and components other than the Containment Vessel are designed to sustain the impact from the missiles which are discussed in Section 3.5 and against any anticipated damages occurring as a result of pipe rupture, pipe whipping, and jet impingement. (See Section 3.6.)

Reference: USAR Chapter 3

3D.1.5 Criterion 5 - Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety are not shared between nuclear power units unless it can be shown that such sharing does not significantly impair their ability to perform their safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. Since Davis-Besse is a single unit station, this criterion is not applicable.

Reference: USAR Chapter 1

3D.1.6 Criterion 10 - Reactor Design

The reactor core and associated coolant, control, and protection systems 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 anticipated operational occurrences.

The integrity of the fuel cladding is assured under all normal and abnormal modes of anticipated operation by avoiding overstressing and overheating of the cladding. The core design, together with coolant system and necessary control systems, provides for this capability under all expected conditions and the transients defined in Section 15.2.

The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine power mismatch. Above approximately 28 percent (originally 15 percent) power, the reactor is operated at a constant average coolant temperature and has a negative power coefficient to damp the effects of power transients. The Integrated Control System (ICS) maintains the reactor operating parameters within preset limits, and the Reactor Protection System (RPS) shuts down the reactor if normal operating limits are exceeded by preset amounts.

The reactor coolant pumps have sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural-circulation coolant flow provides adequate core cooling after the pump energy has been dissipated.

References: USAR Chapters 4, 7, 10, and 15

3D.1.7 Criterion 11 - Reactor Inherent Protection

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

The overall power coefficient, which is the fractional change in neutron multiplication per unit change in core power level, is negative in the power operating range.

Reference: USAR Chapter 4

3D.1.8 Criterion 12 - Suppression Of Reactor Power Oscillations

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

Power oscillations resulting from variations of coolant temperature are minimized by constant average coolant temperature when the reactor is operated above approximately 28% (originally 15%) power. Power oscillations from spatial xenon effects are minimized by the large negative power coefficient and Axial Power Shaping Rod (APSR) assemblies.

Analysis has shown the reactor control and protection systems are adequate to control the oscillations resulting from variations in coolant temperature within the control system deadband and from spatial xenon oscillations. Variations in average coolant temperature provide negative feedback and enhance reactor stability during that portion of core life in which the moderator temperature coefficient is negative. When the moderator temperature coefficient is positive, rod motion will compensate for the positive feedback. The maximum rate of power change resulting from temperature oscillations within the control system deadband is less than 1%/minute. Since the unit is designed to follow ramp load changes of 5%/minute, this is well within the capability of the control system.

Reactor Protection System action to prevent excessive power peaking is provided by the power/imbalance/flow trip system in conjunction with the pressure/temperature and overpower trips. Reactor power peaking is not a directly observable plant variable; therefore, hot channel reactor power peaking limits are provided by placing limits on the reactor power imbalance. Power imbalance is defined as the power in the top half of the core minus the power in the bottom half of the core. The power imbalance limits provide both departure from nucleate boiling protection and fuel melt limit protection. The power/imbalance/flow trip system produces a power level trip which provides limits for the reactor power imbalance. The power level trip and imbalance limits are reduced in accordance with the reactor coolant system flow rate to account for less than four-pump operation.

Control flexibility, with respect to xenon transients, is provided by a combination of control rods and nuclear instrumentation. Axial, radial, or azimuthal neutron flux changes are detected by the nuclear instrumentation. Individual control rods or groups of control rods can be positioned to suppress and/or correct flux changes.

Reference: USAR Chapter 7

3D.1.9 Criterion 13 - Instrumentation and Control

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

containment and its associated systems. Appropriate controls are provided to maintain these variables and systems within prescribed operating ranges.

Adequate instrumentation and controls are provided to maintain operating variables within prescribed ranges for normal operation and to monitor accident conditions as appropriate to ensure adequate safety.

Instrumentation systems include the Nuclear Instrumentation System, which monitors the neutron flux level from the source range to 125 percent of rated power; the non-nuclear process instrumentation, which measures temperatures, pressures, flows, and levels in the Reactor Coolant System, steam system, and reactor auxiliary systems; and the incore instrumentation system, which measures neutron flux at specific locations within the reactor core.

Instrumentation is provided for monitoring containment vessel pressure and temperature during normal operation and accident conditions. The Containment Vessel Emergency Sump water level is also monitored.

Control is provided by two basic systems:

a. Protection Systems:

   The protection systems, which consist of the reactor protection system and the safety features actuation system, perform the most important control and safety functions. The protection systems extend from the sensing instruments to the final actuating devices, such as circuit breakers and pump or valve motor contactors. See response to Criterion 20.

b. Regulating Systems:

   The regulating systems which consist of the Integrated Control System, Control Rod Drive Control System, Nuclear Instrumentation System, and Non-Nuclear Instrumentation System monitor power output and regulate the output through movable control rods and soluble boron. The boron concentration in the reactor coolant is determined periodically by sampling and analysis. Reactivity can also be controlled with burnable poison rod assemblies or an Intergral Absorber, gadolinia, within the fuel rods. The regulating systems are designed to maintain the selected system variables and appropriate nuclear systems within prescribed operating ranges.

The Integrated Control System maintains constant average reactor coolant temperature and constant steam pressure at the turbine during steady-state and transient operation between 28 and 100 percent full power.

Reference: USAR Chapter 7

3D.1.10 Criterion 14 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary has been designed, fabricated, erected, and tested to ensure an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

This has been accomplished by the following procedures and programs:

a. Material selection, design, fabrication, testing, and certification in accordance with recognized codes, such as the ASME Code.

b. Manufacture and erection in accordance with procedures that reflect code requirements and are approved by the manufacturer, vendor, customer, and other parties having jurisdiction.

c. Selection of material properties with due consideration to the effects of neutron flux and general radiation.

d. System analysis to account for cyclic effects of thermal transients, and seismic loadings.

e. Quality assurance program as described in Chapter 17.

Reference: USAR Chapter 17

3D.1.11 Criterion 15 - Reactor Coolant System Design

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

An analysis and evaluation of all normal and abnormal operating conditions and transients is integrally related to all RCS and associated systems design. For all anticipated transients, plots of critical variables (e.g., temperature and pressure) are generated for critical components. Also, for each transient, the number of lifetime cycles is determined. All of the results of these analyses are invoked as functional requirements on the detailed design of the affected systems. Margins for uncertainties are included in (1) the basic analysis assumptions, (2) the assessment of lifetime cycles, and (3) the code-dictated procedures for stress analysis.

Reference: USAR Chapter 5

3D.1.12 Criterion 16 - Containment Design

Reactor containment and associated systems are provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The Containment Vessel is described in Subsection 3.8.2.1 and is the major barrier to uncontrolled release of radioactivity following the postulated accident. As outlined in Subsection 6.2.1 and the Technical Specification, this structure is periodically tested for leakage.

Reference: USAR Chapters 3 and 6

3D.1.13 Criterion 17 - Electric Power Systems

An onsite electric power system and an offsite electric power system are provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

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

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

The design of the electric power systems conforms to this criterion through provision of an offsite transmission system and onsite emergency diesel generators and batteries. Each of these systems has the capability and capacity to supply the necessary power to the engineered safety features during all modes of station operation and postulated accidents.

Four 345 kV transmission lines supply electric power to the station switchyard. The Bayshore, Lemoyne, and either Beaver or Hayes lines are independent of each other. The Hayes line and Beaver line share the same tower; therefore, these two sources cannot be identified as independent offsite sources with relation to each other. Three separate overhead 345 kV circuits are provided from the switchyard to the onsite distribution system, one independent circuit to each of the two startup transformers, and a third connecting the main transformer - unit auxiliary transformer to the switch yard. The unit auxiliary transformer is capable of carrying full station auxiliary loads. The startup transformer circuits are each capable of carrying full station auxiliary loads, as long as both bus tie transformers are functioning. In the event normal onsite ac power sources and offsite circuits are lost, the emergency diesel generators provide a source of electric power to essential loads to ensure that the specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. Following a loss-of-coolant accident, offsite power or the emergency diesel generators are available to ensure that core cooling, containment vessel integrity, and other essential safety functions are maintained.

The onsite electric distribution system arrangement minimizes the vulnerability of essential circuits to physical damage. Two onsite emergency diesel generators are provided as standby power sources. Each diesel engine is designed for an approximate starting time of 10 seconds from receipt of a starting signal to production of rated voltage and frequency. Normally, the 4160-volt essential buses are fed from the unit auxiliary transformer as the normal source. Upon loss of the normal and reserve (offsite) power sources, the two 4160-volt essential buses

are energized from their respective emergency diesel-generators. The essential buses are cleared of all ties prior to application of the emergency diesel generators. This protects each emergency diesel generator system from external faults.

Two independent and physically separated 250/125VDC batteries and dc motor control centers are provided and designed to supply dc power for control, instrumentation, emergency lighting, and dc motors during normal operation and emergency conditions. Each battery is sized to carry the expected dc emergency and safety loads for a minimum of one hour.

Provisions are included to minimize the probability of losing the remaining electric power sources as a result of, or coincident with, a loss of the main generating unit, the transmission network, or the onsite power sources. In the event the main generating unit is lost, station auxiliaries are transferred automatically by fast bus transfer schemes to the offsite power.

Reference: USAR Chapter 8

3D.1.14 Criterion 18 - Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connection, and switchboards, to assess the continuity of the systems and the condition of their components. The systems are designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Testing and inspection of electric systems is discussed in Subsections 8.3.1.1.9. and 8.3.2.1.9. Periodic testing requirements of emergency power sources are given in the Technical Specifications.

References: USAR Chapter 8 and the Technical Specifications

3D.1.15 Criterion 19 - Control Room

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

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

Following proven power station design philosophy, all control stations, switches, controllers, and indications necessary to startup, operate, and shutdown the nuclear unit and maintain safe

control of the facility under normal and accident conditions, including loss of coolant accidents, are located in one control room.

Safe occupancy of the control room during abnormal conditions is provided for in the design of the control room. The Control Room Emergency Ventilation System (CREVS) is provided with radiation detectors and appropriate alarms. When CREVS is operating, control room air is recirculated (with or without minimum makeup) through HEPA filters and charcoal adsorbers. When CREVS is operating with makeup air, a positive control room pressure is maintained to minimize in-leakage. Emergency lighting is provided to ensure visibility of all necessary indications and controls.

Instrumentation and controls located on the auxiliary panel remote from the main control room are available to effect a prompt hot shutdown of the reactor and maintain it in a safe hot shutdown condition. The potential capability exists for a subsequent cold shutdown of the reactor.

References: USAR Chapter 7, 9, and 12

3D.1.16 Criterion 20 - Protection System Functions

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

The reactor trip parameters and the respective trip setpoints have been selected based on a reactor safety analysis, the results of which are contained in Chapter 15. The trip and trip setpoints were selected so that no core design limits are exceeded as a consequence of any anticipated operational occurrences.

The RPS continuously monitors all system parameters that are used as trip inputs. In the event a parameter exceeds a limit dictated by safety analysis, the RPS trips; causing all control rods to be fully inserted. This action occurs automatically requiring no operator action, and no operator action can defeat it. The safety analysis conducted shows that the rod insertion caused by a reactor trip is sufficient to prevent core damage under all accident conditions.

The Safety Features Actuation System (SFAS) continuously monitors Reactor Coolant (RC) pressure and Containment Vessel (CV) pressure. When either parameter exceeds the trip setpoint, the SFAS initiates emergency core cooling, which consists of high-pressure and low-pressure injection. These systems prevent the core from exceeding design limits. When RC pressure and/or CV pressure exceeds their trip setpoint, the SFAS initiates the containment vessel isolation and starts the Emergency Ventilation System to control the containment vessel leakage to the environment. The Core Flooding System is self-actuating, so that no input from the SFAS is required.

The Containment Spray System is actuated by the SFAS as a result of high containment vessel pressure. This actuation pressure is higher than that for emergency core cooling (HPI & LPI systems). The Containment Air Cooling System is actuated at the same containment vessel pressure as the Emergency Core Cooling Systems. These systems ensure containment vessel integrity by reducing CV atmosphere pressure and temperature, and reduce the driving force for CV leakage.

The above systems are actuated by the SFAS automatically, requiring no operator action as described further in Section 7.3. Additional systems to be actuated by the SFAS which may require operator action are discussed in Sections 6.3 and 7.6.

3D.1.17 Criterion 21 - Protection System Reliability and Testability

The protection system is designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system are sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system is designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failure and loss of redundancy that may have occurred.

The Nuclear Instrumentation/Reactor Protection System employs four independent channels which have identical protection capabilities. The instrument strings necessary for protection are identical, from sensors to channel trip outputs, and have no components common to two channels. The reactor trips in the event two of the four channels trip.

A single failure can, in the worst case, prevent one channel from tripping since the channels are completely independent. This means that in the NI/RPS, two out of the three channels that are functioning properly must trip in order to trip the reactor. During testing, a channel may be bypassed so that a trip in that channel does not contribute to a coincidence trip of the reactor. A trip of two of the remaining three channels is necessary to trip the reactor so that a single failure requires the two remaining channels to trip. Thus, under no circumstances is the system prevented from initiating protective action during a single failure.

The Safety Features Actuation System (SFAS) employs four independent and identical sensing channels, which supply trip signals to four independent, identical logic channels which actuate the two independent, redundant safety actuation channels. In order to actuate the SFAS, two out of four sensing channels must trip. This causes all four logic channels to trip which then actuate both safety actuation channels.

Assuming a single failure does not allow a sensing channel to trip, two of the remaining three channels must trip to actuate both redundant safety actuation channels. Thus, a single failure that does not allow a sensing channel to trip, does not prevent the system from tripping.

Assuming a single failure does not allow a logic channel to trip, two of the associated remaining three logic channels must trip to actuate one of the redundant safety actuation channels. Thus, a single failure, that does not allow a logic channel to trip, does not prevent the other redundant safety actuation channels from tripping.

Since only one safety actuation channel is necessary to initiate protective action and these safety actuation channels are capable of tripping even when they are being tested, a single failure in a safety actuation channel cannot prevent protective action.

Removal of a module required for protection from an NI/RPS channel causes that channel to trip, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a trip bi-stable module required for protective action from a sensing channel of the SFAS causes that channel to trip, so that only one of the other three must trip to initiate protective action. Removal of a module required for protective action from a logic channel of the SFAS causes that channel to trip but does not result in protective action.*

The fact that any channel of the SFAS has been tripped is continuously monitored by the station annunciator and computer to the operator with the exception of the BWST Level Trip. The annunciation of the BWST Level Trip will be by logic modules which requires two bistable trips (two channel trips).

The testing schemes of both the NI/RPS and the SFAS enable complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

Reference: USAR Chapter 7

3D.1.18 Criterion 22 - Protection System Independence

The protection system is designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function or are demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, are used to the extent practical to prevent loss of the protection function.

The protection systems are designed so that no loss of function occurs under normal and accident conditions. Qualification testing has been performed to ensure that the systems are completely operable under all postulated conditions.

Other design features incorporated into the protection systems afford added reliability. There is complete independence between redundant portions of the systems. This not only includes electrical independence, but physical independence as well. Redundant portions of different channels (cabling, modules, sensors, etc.) are physically separated. Each channel is entirely self-contained with its own power supply so that internal communications of fire do not disrupt more than one channel. The testing and maintenance features incorporated into the systems enable the technician to completely test each channel independently. Maintenance can also be performed on a single channel without disrupting system operation.

Functional diversity is employed in the RPS design to the extent required by IEEE Standard 279-1968. The SFAS employs functional diversity for initiation of emergency core cooling by utilizing divergent trip parameters.

Reference: USAR Chapter 7

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*The removal of a logic channel module and the tripping of the associated logic channel module cause a safety actuation channel trip.

3D.1.19 Criterion 23 - Protection System Failure Modes

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

Protection systems are supplied primary power from battery-backed essential instrumentation distribution panels. In the unlikely event that primary power is lost to a channel or subsystem of a protection system, that fact is continuously annunciated to the operator.

Each NI/RPS channel is powered from a different primary source. If a channel experiences a loss of power, that channel trips, reducing the channel trip logic to one out of three coincidences. Loss of power to more than one channel causes the reactor to trip since the two-out-of-four channel trip coincidence has been satisfied.

Each SFAS channel is powered from a different battery-backed essential instrumentation distribution panel. The output relays are normally energized and must be de-energized to trip. A loss of power to a SFAS channel causes that channel to trip, but does not cause actuation of any SFAS actuated equipment. It changes the trip logic of the remaining channels from a two-out-of-four coincidence into a one-out-of-three mode. The loss of power of any further SFAS channel consequently actuates the SFAS equipment.

The protection systems are designed to perform their intended functions under adverse environmental conditions which could result from any design accident. Equipment used in the systems is tested to insure that they meet environmental specifications.

Reference: USAR Chapter 7

3D.1.20 Criterion 24 - Separation of Protection and Control Systems

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

Reference: USAR chapter 7

3D.1.21 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

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

The safety analysis demonstrates that the acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems. The safety analysis demonstrates that protection system inputs have been selected to provide the protection required to ensure that acceptable fuel design limits are not exceeded.

The reactor design meets this criterion by reactor trip provisions. The RPS limits reactor power

that might result from unexpected reactivity changes, and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits.

The reactor design meets the criterion under both normal operating conditions and accident conditions for any single malfunction of the reactivity control systems. The control rod drive system is capable of providing a shutdown margin of at least 1% $\Delta k/k$ with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot, zero-power condition.

Reactor subcritical margin is maintained during cooldown by changes in soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the reactivity change associated with the reactor cooldown rate of 50°F/hour. Thus, subcriticality is ensured during cooldown with the most reactive control rod totally unavailable.

A reactor trip protects against continuous withdrawal of a control rod.

Reference: USAR chapter 7

3D.1.22 Criterion 26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles are provided. The first of these utilizes control rods. The Control Rod Drive (CRD) System provides a positive means for inserting the rods. The total CRD system is capable of reliably controlling the rate of reactivity changes and ensures that, under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified fuel design limits are not exceeded.

The second reactivity control system is the Makeup and Purification System. By means of soluble poison (i.e., boron) this system is capable of controlling the rate of reactivity changes resulting from planned normal power changes (including xenon burnout) to ensure that acceptable fuel design limits are not exceeded.

The Makeup and Purification System also has the ability to initiate and maintain the cold shutdown condition in the reactor.

References: USAR Chapters 7 and 9

3D.1.23 Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control systems are designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The reactivity control system consists of control rods and soluble boron addition. The reactor is designed so that the control rods will provide a shutdown margin of at least 1% $\Delta k/k$ with the single most reactive control rod fully withdrawn.

The Borated Water Storage Tank (BWST) provides the source of injection water for the Emergency Core Cooling Systems. The borated water in the BWST is maintained at a minimum of 2600 ppm boron. When injected into the RC system this borated water in

combination with taking credit for 50% of the control rod's reactivity worth maintains a sub-critical reactor as well as a coolable geometry.

References: USAR Chapters 7 and 9

3D.1.24 Criterion 28 - Reactivity Limits

The reactivity control systems are designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

The reactor design meets this criterion with safety features that limit the maximum reactivity insertion rate. These include rod-group withdrawal interlock, soluble boron concentration reduction interlock, maximum rate of dilution water addition, and dilution-time cutoff. In addition, the rod drives and their controls have an inherent feature that limits overspeed in the event of malfunctions. Ejection of the control rod of maximum worth does not lead to further coolant boundary rupture or to internal damage that would interfere with emergency core cooling. Provisions have been included to isolate a ruptured steam line in a time and manner ensuring that the core remains intact for effective core cooling. The reactor system has been designed to avoid any postulated reactivity accidents that would result in a reactor coolant temperature or pressure change of sufficient magnitude to damage the reactor coolant pressure boundary. The design of the Reactor Cooling System ensures that no credible addition of cold water to the core would damage the system.

Reference: USAR Chapter 4

3D.1.25 Criterion 29 - Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems are designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences, and to carry out the intended functions of the systems under the most limiting anticipated situations. Many design features contribute to this capability.

The protection systems utilize redundancy as a means of minimizing the effects of failure. The Reactor Protection System (RPS) has four redundant channels, two of which must trip to cause a reactor trip. The Safety Features Actuation System (SFAS) contains four redundant analog channels which feed two redundant digital channels. To trip the safety features systems, two out of four analog channels must trip, causing both out of two digital channels to trip.

The redundancy is preserved by locating the various channels of the protection systems and the associated sensors so that an anticipated operational occurrence affecting one channel does not adversely affect another redundant channel. Where equipment is located close to redundant equipment, physical barriers are placed between the two.

On-line system tests are employed to ensure that the equipment performs its intended functions when called upon to do so. These tests uncover major failures as well as subtle equipment

failures, such as setpoint drift. Information readouts are provided to the operator so that he can determine the status of the systems as a measure of system operability.

In addition, the protection systems are ruggedly constructed so that they properly function under the design accident conditions. Extensive testing has been conducted to prove that the systems meet these requirements. The use of highly reliable components enhances overall system reliability.

The CRD system is designed so that when called for, the full insertion of rods causes the reactor to shut down. Completed analyses ensure that under all normal operating conditions, the control rods have sufficient negative reactivity to cause the reactor to go subcritical. Hot-to-cold reactivity effects, fuel burnup, and fission product buildup are compensated by removal or addition of soluble boron.

The design of the control rod drive mechanism ensures that the control rods drop when voltage is removed from the mechanism. A trip signal from the RPS causes a loss of power to the CRD mechanisms, resulting in a full insertion.

References: USAR Chapters 7 and 15

3D.1.26 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

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

The component specifications meet the detailed requirements of applicable codes with regard to material properties, cleanliness, fabrication techniques, finishing, inspection techniques, repairs, and acceptance criteria.

Vendors are required to submit design reports, material histories, and inspection results to document their compliance with criteria and codes. Quality Assurance is implemented through contract documents and in-plant inspection by representatives of the vendor and the customer.

Primary and secondary stresses and fatigue for the components are evaluated. Stress analysis includes all the anticipated effects of the temperature and pressure differentials and seismic disturbances. Theories have been confirmed by measuring actual strains under similar conditions. Also flanged fittings may be used on small piping which forms part of the RCS pressure boundary, such as instrument sensing lines and RCS high point vent piping.

To minimize leakage from the Reactor Coolant System, all components are interconnected by an all-welded piping system. Some components have flanged-gasket access openings.

In the unlikely event of system leakage during reactor operation, the leakage is detected by monitoring the level of the pressurizer or the makeup tank, by changes in the containment vessel normal sump level, or by indication of an increase in containment vessel activity.

Reference: USAR Chapter 5

3D.1.27 Criterion 31 - Fracture Prevention Of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary is designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

The reactor coolant pressure boundary design meets this criterion as follows:

a. The pressure boundary plate or forging material of the reactor vessel opposite the core is purchased so that a Charpy V-notch test result of 30 ft-lb or more can be obtained at an initial expected NDTT value of 40°F or less. And the material is tested to determine the actual NDTT value.

b. The service life of the RC system components has been established in accordance with ASME Section III involving the cyclic application of loads and thermal conditions for a 40-year design life. The cumulative fatigue usage factor is less than 1.0 for these design cycles.

c. Quality control procedures include permanent identification of materials and non-destructive testing.

d. Operating restrictions prevent failure toward the end of design life resulting from increase in the NDTT due to neutron irradiation as predicted by a material irradiation surveillance program.

The Reactor Vessel is the only component of the RC system pressure boundary exposed to a significant level of neutron irradiation; therefore, it is the only component subject to material irradiation damage. However, sufficient testing and analysis of ferritic materials in pressure boundary components are performed to ensure that the required NDT limits specified in the criterion are met. Unit operating procedures limit the operating pressure to 20% of the design pressure when the RC system temperature is below NDTT + 60°F throughout unit life. Analysis has revealed no potential reactivity-induced conditions that would result in an energy release to the primary system in the range expected to be absorbed by plastic deformation.

Reference: USAR Chapter 5

30.1.28 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

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

Consideration has been given to the inspectability of the Reactor Coolant System, in the design of the components, in the equipment layout, and in the support structures to permit access for inspection. Access to the various components of the Reactor Coolant System is explained in Subsection 5.2.3. In-Service Inspection Program.

Surveillance specimens of the reactor vessel weld, heat-affected zone, and base materials are encapsulated and installed in the reactor vessel in accordance with ASTM Specification E-185-70.

Reference: USAR Chapter 5

3D.1.29 Criterion 33 - Reactor Coolant Makeup

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

The Makeup and Purification System is designed to supply sufficient borated water makeup for small leaks in the reactor coolant pressure boundary so that an orderly shutdown may be achieved in accordance with the technical specification limits. The primary source of water for coolant injection is the Borated Water Storage Tank.

Either of the two independent makeup pumps is capable of supplying the required makeup water through the normal makeup line. Each of the pumps is powered by separate essential power sources.

Reference: USAR Chapter 9

3D.1.30 Criterion 34 - Residual Heat Removal

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

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

Fission product decay heat generated in the reactor at high reactor coolant temperatures is transferred through the steam generators to the main condenser. At low reactor coolant temperatures, however, heat can no longer be efficiently transferred from the steam generators. At these low temperatures, the decay heat removal system is used instead to transfer the decay heat to the component cooling water system.

Reactor decay heat is removed through the steam generators until the RC system is cooled to 280°F. Steam generated by decay heat can be vented to the atmosphere and/or bypassed to the condenser. Feedwater is supplied to the steam generators by the Main Feedwater system

or Auxiliary Feedwater system as described in Chapter 10.4.7 and Chapter 9.2.7, respectively. The Auxiliary Feedwater system is capable of functioning with either onsite or offsite power. As described in USAR Section 9.2.7, following a complete loss of offsite power, the Auxiliary Feedwater system provides water to the steam generators that is sufficient for decay heat removal. The reactor coolant pumps have sufficient inertia to maintain adequate flow to prevent fuel damage until natural circulation coolant flow is established.

Below 280°F, the decay heat removal system removes the decay heat until the RC system is cooled to the required temperature. The decay heat removal system is designed with sufficient redundancy in heat exchangers, pumps, piping, and valves to ensure that single failure does not prevent the system from fulfilling its function. If leakage occurs during system operation, provisions are made for isolation. The decay heat removal system serves as an engineered safety feature for emergency core cooling; consequently, it is capable of operation from either onsite or offsite power supplies. The two separate equipment strings, each consisting of a decay heat pump and a decay heat removal cooler, are powered by offsite or two independent onsite power sources.

Reference: USAR Chapter 9

3D.1.31 Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling is provided. The system safety function is to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

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

Abundant emergency core cooling is provided by the low-pressure injection (decay heat removal), high-pressure injection, and the core flooding systems. These three systems make up an Emergency Core Cooling System (ECCS) that maintains core cooling in the event of a loss-of- coolant accident (LOCA).

Redundancy of components, power supplies, and initiation logic and separation of functions are provided so that a single failure does not prevent the ECCS from fulfilling its function. The ECCS may be operated from either onsite or offsite power supplies.

The primary function of the Emergency Core Cooling System is to deliver cooling water to the reactor core in the event of a LOCA. The system provides protection for all potential break sizes in the reactor coolant system pressure boundary piping up to and including the double-ended rupture of the largest pipe. In addition, breaks in the High Pressure Injection line and the Core Flood Tank line are postulated.

The basic design criteria for loss-of-coolant accident evaluation are as follows:

a. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

b. The calculated total oxidation of the fuel cladding shall not exceed 17% of the total cladding before oxidation.

c. The amount of hydrogen generated from cladding metal-water reaction does not exceed 1% of the total amount of cladding in the reactor.

d. The core geometry is maintained in a state that is amenable to cooling.

e. The cladding temperature is reduced and maintained at an acceptably low value and decay heat is removed for extended periods of time.

For a rupture in the steam piping, the Emergency Core Cooling System adds shutdown reactivity, so that with minimum tripped rod worth and minimum ECCS operation, the reactor core does not return to criticality; thus, there is no core damage.

Reference: USAR Chapter 6 and 10 CFR 50.46 22

3D.1.32 Criterion 36 - Inspection of Emergency Core Cooling System

The Emergency Core Cooling System is designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.

Design provisions are made to the extent practical to facilitate access to the critical parts of the emergency core cooling system components, including valves and safety injection pumps for inspection for erosion, corrosion and vibration wear evidence, and for non-destructive test inspection where such techniques are desirable and appropriate.

Reference: USAR Chapter 6

3D.1.33 Criterion 37 - Testing of Emergency Core Cooling System

The Emergency Core Cooling System is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

A comprehensive program of station testing is established for all equipment, systems, and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer’s shop, integrated tests of the system as a whole, and periodic tests of the activation circuitry and mechanical components to ensure reliable performance, upon demand, throughout the station lifetime. During normal operation or during operational testing at shutdown, the Emergency Core Cooling System is visually inspected to ensure the structural and leaktight integrity of its components.

The design provides for periodic testing of active components of the Emergency Core Cooling System for operability and functional performance as detailed in Subsection 6.3.4, Tests and Inspections.

Power sources are arranged to permit individual actuation of each active component of the Emergency Core Cooling System.

The High-Pressure Injection pumps and the decay heat pumps can be tested periodically during station operation. The decay heat pumps are used whenever the decay heat removal loop is put into operation. Remote-operated valves can be exercised and actuation circuits can be tested periodically during plant operation or routine maintenance.

Capability is provided to test periodically the operability of the Emergency Core Cooling System up to a location as close to the core as is practical. Design provisions include instrumentation and piping lines to perform tests during station shutdown to demonstrate proper automatic operation of the Emergency Core Cooling System. An integrated system test can be performed when the station is cooled down and the decay heat removal loop is in operation. This test does not introduce flow into the reactor coolant system, but demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

The core flooding flow path may be tested during normal depressurization of the Reactor Coolant System for refueling or maintenance. Section 6.3.4 discusses core flood testing.

In addition to the periodic testing described above, the design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the Emergency Core Cooling System to demonstrate the state of readiness and capability of the system. The functional test is performed with the Reactor Coolant System initially cold and at low pressure. The Emergency Core Cooling System valving is set to simulate initially the system alignment for station power operation.

Reference: USAR Chapter 6

3D.1.34 Criterion 38 - Containment Heat Removal

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

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

The Containment Spray System, consisting of two pumps and their associated spray nozzle headers and the containment air cooling system, consisting of two of three fan coolers, function as Containment Vessel heat removal systems.

All equipment, piping, valves, and instrumentation associated with these systems' safety functions are designed to withstand the temperature and pressure transient conditions resulting from a LOCA and the seismic forces resulting from the applicable earthquake.

The failure of the normal and reserve electrical power supplies will automatically connect the units to the essential electric power supply from the Emergency Diesel Generators.

References: USAR Chapter 6 and the Technical Specifications

3D.1.35 Criterion 39 - Inspection of Containment Heat Removal System

The Containment Heat Removal System is designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

The equipment, piping, valves and instrumentation are arranged so that all items can be visually inspected. The containment air cooler units and associated piping are located outside the secondary concrete shield walls around the reactor coolant system loops, thus permitting personnel to enter this area of the containment vessel during station operation for emergency inspection and maintenance of this equipment. The service water piping and valves outside the shield building are inspectable at all times.

Reference: USAR Chapter 6

3D.1.36 Criterion 40 - Testing of Containment Heat Removal System

The Containment Heat Removal System is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components. (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Testing requirements of the containment vessel heat removal systems are given in the Technical Specifications.

Reference: Technical Specifications

3D.1.37 Criterion 41 - Containment Atmosphere Cleanup

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

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

Fission products, oxygen, hydrogen and other substances which may be released within the Containment Vessel are confined there until removed by one of several atmosphere cleanup systems. The systems are:

a. Containment Air Cooling System and Containment Spray System

b. Containment Emergency Ventilation System

c. Containment Vessel Combustible Gas Control System

The design and operation of these systems is discussed in detail in Subsections 6.2.2, 6.2.3 and 6.2.5, respectively.

A brief summary is given here. The Emergency Ventilation System, consisting of two (2) full capacity redundant fan-filter assemblies can function as a Containment Atmosphere Cleanup System. The Emergency Ventilation System, as described in Subsection 6.2.3 is designed to provide a negative pressure within the Shield Building and Penetration Rooms following a loss-of-coolant accident and to reduce airborne fission product leakage to the environment by filtration prior to release of air through the station vent. The Containment Vessel Purge System is used to completely replace the containment vessel atmosphere. The Hydrogen Dilution System, consisting of two (2) rotary positive displacement blowers, and the Hydrogen Purge System are designed to control the concentration of hydrogen and other substances in the containment vessel atmosphere. The hydrogen purge filter train provides high efficiency particulate filtration and charcoal filtration prior to discharging to the station vent. Natural convection currents along with the turbulence created by the combined action of Containment Spray and the Containment Air Coolers provide adequate air circulation within the containment vessel to avoid any possibilities of the creation of hydrogen pockets. The Containment Spray System serves to cool the containment vessel atmosphere as well as to remove a portion of the iodine released from the reactor coolant.

All equipment and instrumentation associated with these systems is designed to withstand the seismic forces resulting from the applicable earthquake.

The failure of the normal and auxiliary electrical power automatically connects the units to the emergency diesel generators.

Reference: USAR Chapter 6

3D.1.38 <u>Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems</u>

The Containment Atmosphere Cleanup Systems are designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

All equipment described for Criterion 41 is arranged to provide visual inspection at any time for visible indications of damage or potential failure.

Reference: USAR Chapter 6

3D.1.39 <u>Criterion 43 - Testing Of Containment Atmosphere Cleanup Systems</u>

The Containment Atmosphere Cleanup Systems are designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole

and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Testing requirements of the Emergency Ventilation System are given in Subsection 6.2.3.4 and the Technical Specifications.

References: USAR Chapter 6 and the Technical Specifications

3D.1.40 Criterion 44 - Cooling Water

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

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

The station Ultimate Heat Sink meets the specifications of Safety Guide 27 and is discussed in Subsection 9.2.5.

One Component Cooling Water pump supplies sufficient cooling water to one train of the Engineered Safety Features components during normal operation as well as a loss-of-coolant accident. The Component Cooling Water System is fully redundant, consisting of three pumps and two headers. The pumps, piping and components serving emergency functions are designed to the ASME Code, Section III, Nuclear Class 3 and Seismic Class I. The system design is sufficiently redundant such that portions of the system may be isolated while other portions contrive to supply adequate cooling water for the system loads. Leak detection capability is provided to ensure safe shutdown can be accomplished under all postulated conditions. See Subsection 9.2.2.

Heat from the Component Cooling Water System is rejected to the Service Water System which, in turn, rejects heat to the cooling tower via the Circulating Water System during normal operation, or to the lake during a LOCA. The Service Water System is fully redundant, including three pumps and two headers. The portion of the system required for station safety is designed to ASM.E Code, Section III, Nuclear Class 3 and Seismic Class I. See Subsection 9.2.1.

Reference: USAR Chapter 9

3D.1.41 Criterion 45 - Inspection of Cooling Water System

The Cooling-Water System is designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

All important components of the station Component Cooling and Service Water Systems are accessible for appropriate periodic inspection. See Section 9.2.

Reference: USAR Chapter 9

3D.1.42 Criterion 46 - Testing of Cooling Water System

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

The station Component Cooling and Service Water Systems are designed to permit testing to verify system integrity and operability. Portions of the systems are used during refueling and continuously during normal operation. The balance of the systems is tested periodically. See Section 9.2.

Reference: USAR Chapter 9

3D.1.43 Criterion 50 - Containment Design Basis

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

The Containment Vessel and engineered safety features system have been evaluated for various combinations of energy releases. The analyses took into account system energy and decay heat. The emergency injection system is designed so that no single failure could result in significant metal-water reaction. The cooling capacity of either the Containment Air Cooling System or the Containment Spray System is adequate to prevent overpressurization of the Containment Vessel and to return the Containment Vessel to near atmospheric pressure. The details of this analysis are contained in Subsection 6.2.1.3. Structural design of the Containment Vessel and Shield Building are described in Subsections 3.8.2.1 and 3.8.2.2, respectively.

References: USAR Chapters 3 and 6

3D.1.44 Criterion 51 - Fracture Prevention Of Containment Pressure Boundary

The reactor containment boundary is designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated

accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (2) size of flaws.

The uncertainties in determining material properties, residual stress, transient stress and steady state stress have been considered by the ASME Code. A detailed description has been made in Appendix II, “Basis for Establishing Design Stress Intensity Values,” ASME Code, Section III.

Considering that the fracture strain is approximately two hundred times the yield strain and considering further the conservative approach taken in the ASME Code in limiting stress intensity and flaw size, the possibility of both creating and propagating a fracture is highly unlikely.

The Containment Vessel has been designed for the following temperatures:

Maximum operating internal temperature	120°F
Maximum operating ambient temperature	120°F
Design internal temperature	264°F
Accident internal temperature	264°F
Pneumatic test temperature	68°F (nominal)
Lowest service metal temperature	30°F

Ferritic materials used at the containment boundary are SA-299 and SA-516 (Grade 60 and 70). The lowest service metal temperature of these materials is specified to be 30°F in the contract specifications.

In reviewing design temperatures, it is apparent that the containment temperature is never lower than 30°F below the temperature at which a brittle fracture might occur. However, Charpy V-notch impact tests (ASTM A-370 Type A) have been specified for both SA-299 and SA-516 materials at 0°F in order to ensure that adequate material ductility is provided in accordance with the acceptable requirements of Paragraph N-332 of the ASME Code. Operational, test, and postulated accident temperatures are combined with appropriate pressures and other loads in the load-combination equations of Subsection 3.8.2. The codes cited provide adequate safety margins to cover variations in properties, residual stresses, and sizes of flaws, while the QA/QC program discussed in Chapter 17 provides the means of control and surveillance for such variations.

References: USAR Chapters 3 and 17

3D.1.45 Criterion 52 - Capability for Containment Leakage Rate Testing

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

The Containment Vessel is designed to permit periodic leakage rate testing at design pressure. Subsection 6.2.1.4, Testing and Inspection, provides preoperational and in-service testing procedures. The Containment Leakage Rate Testing Program, which has been established in

accordance with Technical Specifications, requires that the containment be periodically tested at design pressure to determine the actual leakage rates.

References: USAR Chapter 6 and the Technical Specifications

3D.1.46 Criterion 53 - Provisions for Containment Testing and Inspection

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

The Containment Leakage Rate Testing Program, which has been established in accordance with Technical Specifications, discusses test and surveillance programs and criteria. Subsection 6.2.1.4 describes the leak-test capability of the containment vessel, penetrations, and components.

References: USAR Chapters 6 and the Technical Specifications

3D.1.47 Criterion 54 - Piping Systems Penetrating Containment

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

All piping penetrations, with the exception of a few that are part of the Safety Actuation System, are provided with double barrier isolation systems, so that no single credible failure or malfunction of one active component can result in loss of isolation capability or intolerable leakage. The installed double barriers take the form of closed piping systems inside and outside of the Containment Vessel and various types of isolation valves. A detailed description of the containment isolation system is found in Subsection 6.2.4.

The containment and all piping penetrations are designed such that after installation and while in service, all isolation systems can be tested independently for leakage. A detailed description of initial and periodic leak testing and other inspections can be found in Subsection 6.2.4.4.

Reference: USAR Chapter 6

3D.1.48 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

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

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

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

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

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve is not used as the automatic isolation valve outside containment.

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

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

The reactor coolant pressure boundary is defined as those piping systems or components which contain reactor coolant at design pressure and temperature. With the exception of the reactor coolant sampling line, the reactor coolant pressure boundary, as defined above, is located entirely within the containment vessel. The sampling line is provided with remotely operated valves for isolation in the event of a failure. This line is normally isolated and is used only during actual sampling operations. All other lines and components which penetrate the containment and may contain reactor coolant are at low temperatures such that any leakage would be collected by the contaminated drain system. No significant environmental dose would arise from these sources. (See Subsection 6.2.4.2.)

Reference: USAR Chapter 6

3D.1.49 Criterion 56 - Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment is provided with containment isolation valves as follows; unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

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

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

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

(4) One automatic isolation valve inside and one automatic isolation valve outside

containment. A simple check valve is not used as the automatic isolation valve outside containment.

Isolation valves outside the containment are located as close to the containment as practical and, upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety. A detailed description of this type of penetration can be found in Subsection 6.2.4.2.

Reference: USAR Chapter 6

3D.1.50 Criterion 57 - Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere has at least one containment isolation valve which is either automatic, or locked closed, or capable of remote manual operation. This valve is outside containment and located as close to the containment as practical. A simple check valve is not used as the automatic isolation valve. A detailed description of this type of penetration can be found in Subsection 6.2.4.2.

Reference: USAR Chapter 6

3D.1.51 Criterion 60 - Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design includes means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity is provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

As discussed in Chapter 11, radioactive wastes are processed in accordance with 10 CFR 50 requirements.

Solid wastes are processed in a batch manner for off-site disposal. Processed liquid wastes and gaseous wastes released to the environment are monitored and discharged with sufficient dilution to ensure tolerable activity levels on the site and at the site boundary. Ample holdup storage capacity for liquid and gaseous waste is provided. Wastes are sampled to establish release rates consistent with environmental conditions. In-line monitors provide a continuous check on the level of radioactivity released.

Under accident conditions, radioactive gaseous effluents which may be released into enclosed areas are collected by the ventilation systems and discharged to the station vent (Subsection 6.2.3). Permanently installed area monitors and the station vent detectors are used to monitor the activity levels of discharges to the environment. In addition, portable monitors are available onsite for supplemental surveys.

References: USAR Chapters 6 and 11

3D.1.52 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity are designed to assure adequate safety under normal and postulated accident

conditions. These systems are designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection. (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

The Spent Fuel Cooling System and Decay Heat Removal System are designed to prevent damage to fuel during handling and storage and to prevent undue risk to the station operating areas and the public environs. The systems have full inspection and testing capability and are described in Chapter 9.

The shielding provided in the spent fuel and waste storage area is sufficient to provide access as delineated in Chapter 12.

The spent fuel storage pool (Chapter 9) is located within the fuel handling and storage area of the Auxiliary Building. The liquid waste processing equipment and the gaseous waste storage and disposal equipment is located within a separate area of the same building. Both of these areas provide confinement capability in the event of an accidental release of radioactive materials; and both are ventilated by a system with monitored discharges. Analysis has indicated that the accidental release of the maximum activity content of the waste gas decay tanks does not result in excessive doses to persons at the site boundary.

Radioactive liquid effluent leakage into the component cooling water system is determined by radiation monitors. Any accidental leakage from liquid waste processing equipment is collected in a sump and transferred to tanks to prevent releases to the environment.

The Fuel Handling System is designed and constructed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release.

The Radioactive Waste System is designed to provide controlled handling and disposal of liquid, gaseous and solid wastes from the Davis-Besse Nuclear Power Station.

References: USAR Chapter 9, 11, and 12

3D.1.53 Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system is prevented by physical systems or processes by using geometrically safe configurations.

New fuel assemblies are stored in a protected dry area, in racks in parallel rows having a center-to-center distance of 21 inches in both directions. This spacing, combined with the practice of keeping rows C and F vacant, is adequate to maintain a k_{eff} of less than 0.95 (for an enrichment of 5.0 percent) even if flooded with unborated water, and is adequate to maintain a k_{eff} of less than 0.98 even if immersed in a hydrogenous “mist” that produces optimum moderation.

The spent fuel storage pool, a reinforced-concrete pool lined with stainless steel, is sized to store 1624 irradiated fuel assemblies. The spent fuel pool rack cells are located in parallel rows with a center to center spacing of a nominal 9.22 inches. These racks use BORAL neutron

absorber between adjacent fuel assemblies to maintain a k_{eff} of less than 0.95 for fuel assemblies with a maximum nominal initial enrichment of 5.05 weight percent of uranium-235 assuming the storage racks are flooded with unborated water. (Fuel stored in the spent fuel pool is limited to a maximum nominal initial enrichment of 5.0 wt% U^{235} . This limit allows compliance with sub-section b of 10 CFR 50.68, Criticality Accident Requirements.) The Technical Specifications define the enrichment/burnup restrictions for these racks. The thermal hydraulic and shielding considerations are specified in the USAR Technical Requirements Manual.

Reference: USAR Chapter 9

3D.1.54 Criterion 63 - Monitoring Fuel And Waste Storage

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

The fuel handling and storage systems are comprehensively monitored to readily detect any unusual conditions and to initiate the appropriate alarms (Subsection 12.1.4). The spent fuel pool is instrumented to alarm on high or low water level, high temperature, or high skimmer filter Δp. The Refueling Canal is instrumented to alarm on high or low water level or high skimmer filter Δp. The spent fuel pool cooling loop alarms on low flow, and the purification loop alarms on high temperature, high or low flow, high resin Δp or high filter Δp.

The systems are also periodically sampled and analyzed for effectiveness. The fuel handling and storage areas and radioactive waste areas are monitored for reactivity releases by the following types of monitors:

Area Radiation Monitors:

New fuel receiving area

New fuel storage area

Detergent waste drain tank room

Process Radioactivity Monitors:

Miscellaneous radioactive waste discharge

Clean waste discharge

Fuel handling area exhaust vent

Radwaste area exhaust vent

References: USAR Chapters 9, 11, and 12

3D.1.55 Criterion 64 - Monitoring Radioactivity Releases

Means are provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the

plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

There are numerous radiation and radioactivity monitoring systems throughout the station which are designed to allow early detection and isolation of radioactivity releases and radiation sources. These systems are described in Section 11.4 and Subsection 12.1.4. They include the Airborne Radioactivity Monitoring System, the Waterborne Radioactivity Monitoring System, and the Area Radiation Monitoring System. All of these systems are designed to monitor their respective media under normal, accident and post-LOCA conditions.

References: USAR Chapters 11 and 12

3D.2.0 NRC Safety Guides

The design of the Davis-Besse Nuclear Power Station takes into consideration the NRC Safety Guides as applicable in the following paragraphs.

3D.2.1 Safety Guide 1- "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps" (November 1970)

The Net Positive Suction Head (NPSH) for the Low Pressure Injection (decay heat removal) pumps and the Containment Spray pumps, during the recirculation phase following a LOCA was calculated in accordance with the guidelines of the Safety Guide. No credit has been taken for the containment pressure. It was assumed that the vapor pressure is equal to the containment pressure thus only a static head (difference in elevations between the pump centerline and emergency sump) was available. In addition, for added conservatism only a minimum water level inside containment was used. The calculational method for available NPSH is described in Subsection 6.3.2.14.

3D.2.2 Safety Guide 2- "Thermal Shock to Reactor Pressure Vessels" (November 1970)

The B&W Topical Report BAW-10018, Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock, has been incorporated into this license application. This report demonstrates that the reactor vessel does not lose its integrity due to crack propagation as a result of thermal shock caused by actuation of the ECCS following a LOCA. In addition, the design of the reactor vessel does not preclude the use of annealing if required to assure recovery of the fracture toughness properties of the vessel material.

Subsequent to issuance of the above report, BAW-2325 Revision 1 (January 1999), "Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity" was issued to address revised Reactor Vessel pressurized thermal shock criteria. BAW-10018 has been withdrawn from NRC review. Further reference to BAW-2325 Revision 1 is located in Subsection 5.2.3.8.

3D.2.3 Safety Guide 3 - "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors"

This Safety Guide is not applicable to the Davis-Besse Nuclear Power Station as it refers to a boiling water reactor.

3D.2.4 Safety Guide 4 - "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" (November 1970)

The radiological consequences of a Maximum Hypothetical Accident (MHA) as discussed in Subsection 15.4.6.4 have been evaluated based on the assumptions predicated in this safety guide and Regulating Guide 1.4 iodine composition.

3D.2.5 Safety Guide 5 - "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors"

This Safety Guide is not applicable to the Davis-Besse Nuclear Power Station as it refers to a boiling water reactor.

3D.2.6 Safety Guide 6 - "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems" (March 1971)

This system electrical design as described in Chapter 8, Subsection 8.3.1 incorporates the requirements of this Safety Guide.

3D.2.7 Safety Guide 7 - Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident" (March 1971)

The combustible gas control systems are described in Subsections 6.2.5 and 9.3.2. The parameter values specified in this guide were used to calculate the total quantity of hydrogen gaseration. Davis-Besse design incorporates the following:

a. Hydrogen sampling

b. Mixing of the containment atmosphere

c. Controlling of combustible gas concentrations

d. Purging of containment atmosphere through charcoal filters

The Davis-Besse design of the combustible gas control system complies with the guidelines established in the Safety Guide 7.

3D.2.8 Safety Guide 8 - "Personnel Selection and Training" (March 1971)

See Chapter 17.

3D.2.9 Safety Guide 9 - "Selection of Diesel Generator Set Capacity for Standby Power Supplies" (March 1971)

This system electrical design as described in Subsection 8.3.1.1.4.1 incorporates the requirements of this Safety Guide.

3D.2.10 Safety Guide 10 - "Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments"

This Safety Guide is not applicable to the Davis-Besse Nuclear Power Station.

3D.2.11 Safety Guide 11 - "Instrument Lines Penetrating Primary Reactor Containment" (March 1971)

Instrument lines that penetrate the Containment Vessel are equipped with isolation valves as described in Subsection 6.2.4. The only exceptions are the four instrument lines that provide containment pressure indication to the Reactor Protection System (RPS) and the Safety Features Actuation System (SFAS). The pressure switches and transmitters on these lines serve to maintain containment integrity in accordance with GDC 55 and 56. A detailed discussion is presented in Subsection 6.2.4.

3D.2.12 Safety Guide 12 - "Instrumentation for Earthquakes" (March 1971)

Instrumentation which includes four strong motion triaxial accelerographs, peak accelerographs and a seismic trigger, has been provided to monitor and record seismic disturbances at the station. A detailed discussion is presented in Subsection 3.7.4.

3D.2.13 Safety Guide 13 - "Fuel Storage Facility Design Basis" (March 1971)

The design of the fuel storage facility is in general conformance with the guidelines of this Safety Guide. The design of the various systems and facilities of the fuel handling area are discussed in Subsections:

9.1.1 and 9.1.2 for New and Spent Fuel Storage
9.1.3 Spent Fuel Pool Cooling and Cleanup System
9.4.2.2 Fuel Handling Area Ventilation

The exceptions taken to the guide are as follows:

a. During normal operation, the Fuel Handling Area Ventilation System is lined up so that, in the event of a fuel handling accident, the normal Fuel Handling Area Ventilation System is automatically isolated and the unit Emergency Ventilation System (EVS) is automatically started to maintain a negative pressure in the Fuel Building.

b. The Spent Fuel Pool Cooling and Cleanup System is not entirely designed as Seismic Class I. Only that portion of the piping which extends from the Spent Fuel Pool to the isolation valves on the suction and discharge sides of the pool is Seismic Class I. However, the backup cooling system, the Decay Heat Removal System, is designed as Seismic Class I.

The 130 ton cask crane in the Auxiliary Building is prevented from traveling over the spent fuel pool by providing electronic safety interlocks that require an administratively controlled bypass key.

A reinforced concrete roof cast of metal deck is provided over the pool to protect the fuel storage facility from any external missiles. For further detail, refer to Section 10.2.5.4.

The spent fuel pool, storage fuel racks and equipment, are all designed to withstand the Maximum Probable Earthquake and Maximum Possible Earthquake for Seismic Class I requirements.

3D.2.14 Safety Guide 14 - "Reactor Coolant Pump Flywheel Integrity" (October 1971)

Information demonstrating adequacy of the RC pump flywheel is contained in Appendix 5A.

3D.2.15 Safety Guide 15 - "Testing of Reinforcing Bars for Concrete Structures" (October 1971)

Reinforcing bars used for Category 1 and Category 2 concrete structures that are important to safety are tested to verify their yield and ultimate tensile strengths. The method of testing conforms to the ASTM A-615 requirements.

An independent testing laboratory located at the nuclear station site performs a "user's tests" for the purpose of making chemical analysis checks and physical property tests for the random samples of the steel delivered. User's tests are performed as follows:

1. No. 11 bar size and smaller: One random diameter size sample from each 50 tons of bar delivered for tension and bend tests.

2. No. 14 and No. 18 bar sizes: One sample for each bar size from each 100 tons of bar delivered for tension test only.

If the test sample bar does not meet the minimum strength requirements as defined in ASTM A-615-68 and project specification, the reinforcing bar is rejected.

The splice length required for bonding and maintaining the tensile strength of reinforcing bar is in accordance with the Ultimate Strength Design Handbook, Special Publication No. 17, ACI C-318-63.

Mechanical splicing of deformed concrete reinforcing bars is performed in accordance with the project specification.

3D.2.16 Safety Guide 16 - "Reporting of Operating Information" (October 1971)

Reports are submitted in accordance with Technical Specifications and other regulations per plant programs and procedures.

3D.2.17 Safety Guide 17 - "Protection Against Industrial Sabotage" (October 1971)

This Safety Guide has been superseded by NRC Order EA-03-086, dated April 29, 2003.

3D.2.18 Safety Guide 18 - "Structural Acceptance Test for Concrete Primary Reactor Containments"

This Safety Guide is not applicable to the Davis-Besse Nuclear Power Station.

3D.2.19 Safety Guide 19 - "Nondestructive Examination of Primary Containment Linear Welds (Revision 1)"

This Safety Guide is not applicable to the Davis-Besse Nuclear Power Station.

3D.2.20 Safety Guide 20 - "Vibration Measurements On Reactor Internals" (December 1971)

A vibration analysis and test program for prototype reactor internals similar to Davis-Besse was developed and submitted in Topical Report BAW-10038, Rev. 1 "Prototype Vibration Measurement Program for Reactor Internals." The Davis-Besse program conforms to the regulatory position in Safety Guide 20 for reactor internals similar to the prototype design.

3D.2.21 Safety Guide 21 - "Measuring And Reporting of Effluents From Nuclear Power Plants" (December 1971)

Subsection 11.4.2 describes the systems available for monitoring and measuring station effluents which may contain radionuclides. Station Technical Specifications define reporting requirements relative to these station effluents. Both measuring and reporting requirements incorporate the requirements of Safety Guide 21.

3D.2.22 Safety Guide 22 - "Periodic Testing of Protection System Actuation Functions" (February 1972)

The system electrical design as described in Chapter 8, Subsection 8.3.1.1.9 incorporates the requirements of this Safety Guide.

3D.2.23 Safety Guide 23- "Onsite Meteorological Programs" (February 1972)

Information regarding this subject is provided in Subsection 2.3.3, "Onsite Meteorological Measurements Programs." This section meets all requirements of Safety Guide 23, "Onsite Meteorological Programs."

3D.2.24 Safety Guide 24 - "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure" (March 1972)

The radiological consequences of a gas decay tank rupture are discussed in Subsection 15.4.1. The assumptions postulated in this Safety Guide were used in this evaluation.

3D.2.25 Safety Guide 25 - "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (March 1971)

The radiological consequences of a fuel handling accident are discussed in Subsection 15.4.7. This evaluation was based on the postulated assumptions set forth in this Safety Guide.

3D.2.26 Safety Guide 26 - "Quality Group Classifications and Standards" (March 1972)

Quality group classification for the systems and components important to safety is shown in Table 3.2-2. Davis-Besse design in general complies with the intent of this safety guide. However, as this safety guide only references the new ASME Section III Code and many of the components for the Davis-Besse Station were purchased to other than ASME Section III Code depending on the date of purchase, a blanket statement of complete compliance with their guide cannot be made.

3D.2.27 Safety Guide 27 - "Ultimate Heat Sink" (March 1972)

The ultimate heat sink for this station is Lake Erie, which is the source of cooling water for the Service Water System. While this is the single source for the sink, an analysis given in Section 9.2.5.1 demonstrates that the most severe natural phenomenon which can occur does not prevent a safe shutdown of the reactor. The Seismic Class I portion of the intake structure forebay provides adequate storage that is capable of providing sufficient cooling for at least 30 days. Procedures for ensuring a continued capability after this time are available. A detailed comparison of the degree of compliance of this design with Safety Guide 27 is given below:

a. Sink Availability For At Least 30 Days:

The heat sink for this station provides adequate cooling for at least 30 days.

Procedures for maintaining continued capability after this period are available.

b. The Most Severe Phenomena Expected Taken Individually:

An earthquake, which may result in loss of the source of lake water to the intake forebay, is the most severe event. This occurrence does not cause loss of the heat sink safety functions.

c. Site-Related Events:

The occurrence of extremely low lake level, which reduces the quantity of available water in the forebay, in conjunction with loss of the canal, was considered. The lowest Technical Specification level was assumed for the analysis, and this condition does not preclude sink safety functions.

d. At Least Two Sources of Water for The Ultimate Heat Sink:

This station has a single source of water for the heat sink. The collapse of the intake pipe or complete closure of the canal was postulated for the analysis. It is demonstrated that additional sources of water are not required since the stored water in the forebay is adequate for safe shutdown.

With regards to the amount of conservatism available for dissipating heat loads, the design of the Davis-Besse Ultimate Heat Sink is also consistent with the recommendations of Regulatory Guide 1.27, Revision 1, March 1974, as discussed in Section 9.2.5.1.

3D.2.28 Safety Guide 28 - "Quality Assurance Program Requirements (Design and Construction)" (June 1972)

See Chapter 17.

3D.2.29 Safety Guide 29 - "Seismic Design Classification" (June 1972)

The nuclear station is designed in accordance with the requirements of general design Criterion 2 of Appendix A to 10CFR50 "General Design Criteria for Nuclear Power Plants," and proposed Appendix A to 10CFR100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." The structures, systems, and components that are important to safety, are designed to withstand the

effects of earthquakes. They are designed such that, if the maximum possible earthquake occurs, all structures, systems, components and equipments important to safety remain functional.

The definitions and a classification list of the Seismic Class I and Class II structures, systems components and equipment are given in Subsection 3.2.1. The seismic design of the Seismic Class I structures, systems, Components and equipment is outlined in Section 3.7.

The types of internal and external missiles and their analyses for the Seismic Class I buildings, systems, and equipment are discussed in Subsections 3.5.1, 3.5.2 and 3.5.3.

The design methods for Seismic Class I structures are described in Section 3.8.

The dynamic analysis methods and testing procedures of the mechanical systems and components are provided in Section 3.9.

3D.2.30 Safety Guide 30 – "Quality Assurance Requirements for Installation, Inspection, and Testing of Instrumentation and Electrical Equipment" (August 1972)

See Chapter 17.

3D.2.31 Safety Guide 31 – "Control of Stainless Steel Welding" (August 1972)

The Safety Guide requires that the delta-ferrite content of weld deposits in austenitic stainless steel structures and components be in the range 5 to 12-15%, and that a magnetic measuring instrument be used to ensure that the ferrite content of such welds is kept within this range.

> Control of the delta-ferrite content of welds in austenitic stainless steels is accomplished by controlling the ferrite content of welding materials to be within the 8-25 percent range. This may result in the ferrite content of the weld deposits exceeding the 15 percent maximum allowed by the Safety Guide. High delta-ferrite content of the finished weld is only a concern in high temperature service (800°F and higher), and for the present service (600°F and lower) the 15 percent maximum is not a necessity.
>
> It is felt that the accuracy of presently available magnetic ferrite measuring devices is questionable, particularly when used in the root and lower passes of thicker joints.
>
> This method of measuring ferrite content of partially or fully completed welds has therefore not been used.

The Safety Guide also requires that macroscopic examination of transverse side bend test specimens be performed to assess the presence or absence of fissures in the weld metal.

> Such examination is performed as a matter of course during the welding procedure and welder performance qualification test.

The Safety Guide also requires that controls on welding parameters (current, voltage, speed of travel, interpass temperature) be specified.

> Such controls have been maintained by specifying a maximum interpass temperature (350°F) and limiting the size of individual weld passes made by the submerged-arc

process to a maximum of 3/8 inch in thickness. It is felt that these controls adequately keep the heat input down.

Except for the above mentioned non-conformances, all other requirements of Safety Guide 31 are met.

3D.2.32 Safety Guide 32 "Use of IEEE Standard 308-1971, 'Criteria for Class IE Electric Systems for Nuclear Power Generating Stations'" (August 1972)

The system electrical design as described in Subsection 8.1.5 incorporates the requirements of this Safety Guide.

3D.3.0 NRC Information Guides

The design of the Davis-Besse Nuclear Power Station takes into consideration the NRC Information Guides as indicated in the following paragraphs:

3D.3.1 Information Guide 1 - "Primary Reactor Containment Systems - (Steel Construction)"

1. The limits of containment pressure boundary are shown on Figure 3.2-1.

2. The design conditions that provide a basis for the containment system design have been described in Subsection 3.8.2.1.2. The basis upon which the design conditions have been established were discussed in Subsection 6.2.1.3.

   The combination of design loadings have been listed in Subsection 3.8.2.1.4. Their categories with respect to the ASME Section III Code are as follows:

a.	Pressure Test	Normal Conditions
b.	Leak Test	Normal Conditions
c.	Construction	Not Specified
d.	Normal Operating	Normal Conditions
e.	Loss of Coolant Accident (LOCA) Condition	Faulted Conditions*
f.	Pipe Rupture Other Than LOCA	Emergency Conditions

*For simplicity, the more conservative allowable stress for Emergency Condition is used.

The design stress intensity limits that are selected for the design of the containment system have been specified in Subsection 3.8.2.1.6.

3. The material fracture toughness criteria that has been applied to the pressure-containing components of the containment system under both service and periodic leakage rate testing conditions during the service lifetime of containment has been described in detail in Subsection 3D.1.44.

   The lowest service temperature that the containment steel may experience has been specified as 30°F. The temperature in the annulus is maintained above 30°F by the Containment Purge Ventilation System or the Containment Annulus Heaters.

   The material fracture toughness criteria have been described in detail in Subsections 3.8.2.1.7 and 3D.1.51.

4. The seismic design criteria which include the analytical model; the method of analyses and the seismic input have been described thoroughly in Section 3.7.

5. The bases of design criteria and loading combinations applied with respect to the buckling of the containment vessel has been specified in Subsections 3.8.2.1.6 and 3.8.2.1.4. The meridional buckling stress of the cylindrical wall is limited to (-)3485 psi, the circumferential buckling stress of the cylindrical wall is limited to (-)260 psi, and the combined meridional and circumferential buckling stresses of the spherical head are limited to (-)1875 psi. Under the loading condition of Dead Load plus 0.25 psi external pressure, Safe Shutdown Earthquake, Polar Crane, and Live Load, the maximum compressive stress in meridional direction was found to be (-)2740 psi which provides a minimum safety factor of 1.27 against buckling.

6 For each piping system, a static and seismic analysis are generally performed by using Bechtel's computer program ME101 (Bechtel Computer Program ME632 was used previously), the reaction of the piping system at the containment vessel is then used in the design of the penetration. In the event of a postulated pipe rupture, the penetration is designed to withstand the loading conditions which have been described in Subsection 3.8.2.1.10.

   Piping systems which could impose significant reactions at the containment penetration have been isolated from the containment vessel by using flued heads with expansion bellows. For a detailed description, refer to Subsection 3.8.2.1.10.

   The systems of restraints for piping systems under operating condition have been described in Subsection 3.6.2.6. The systems of restraints for pipe rupture have been described in detail in Section 3.6.

7. The design provisions that are included in the construction of the Containment Vessel and its penetrations for the purpose of periodic leakage rate testing during the containment service lifetime have been described in Subsections 6.2.1.4, 3D.1.45, 3D.1.46, and 3.8.2.1.6. The design provisions that were included in the construction of the containment vessel and its penetrations for the purpose of periodic leakage rate testing during the containment service life time have been described in Subsection 6.2.1.4.

8. The Containment Vessel and its penetrations have been designed for the test pressure as described in Subsection 3.8.2.1.4.

3D.3.2 Information Guide 2 - "Instrumentation and Electrical Systems"

Chapter 7.0 includes all of the instrumentation items suggested in this information guide.