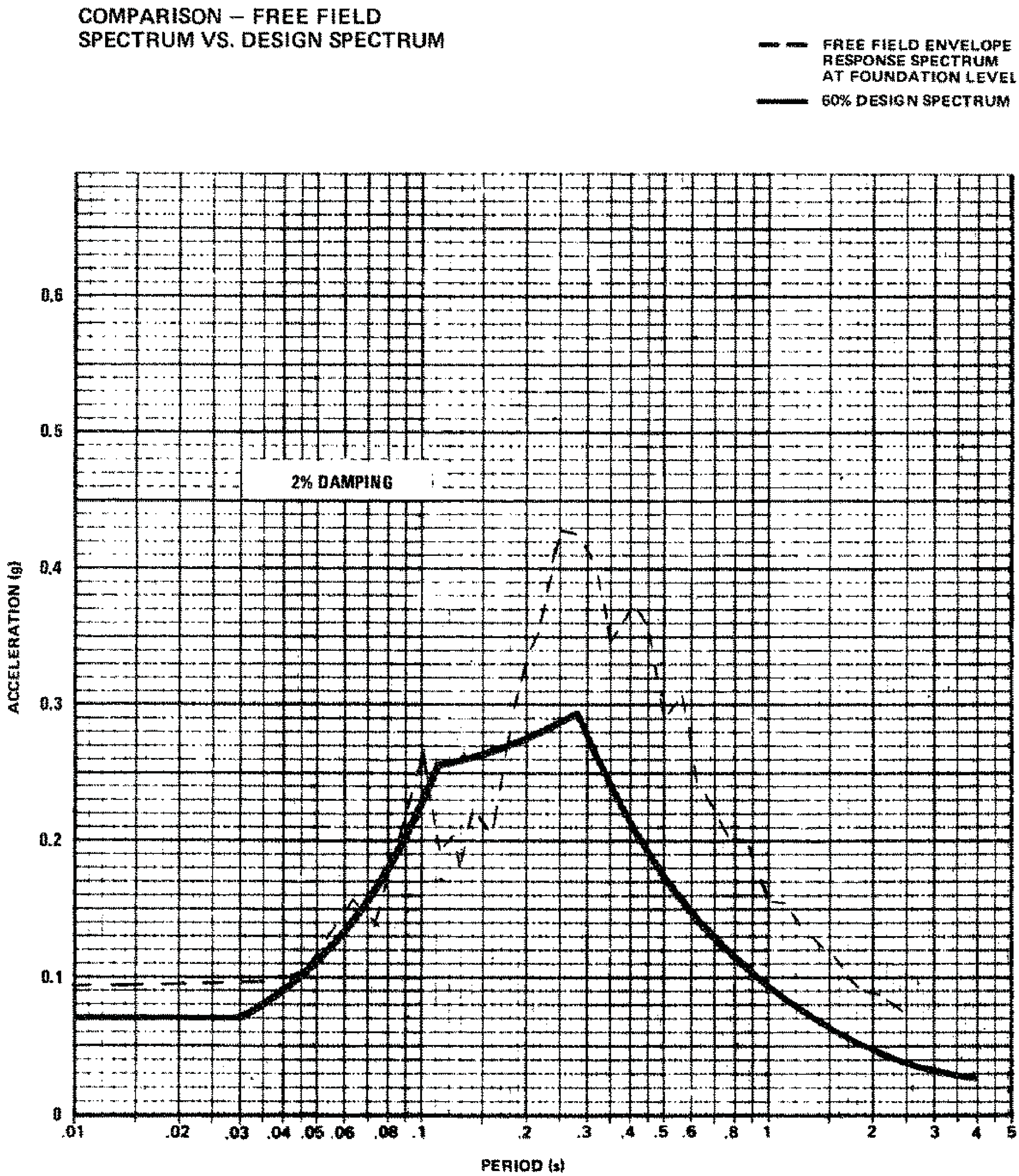
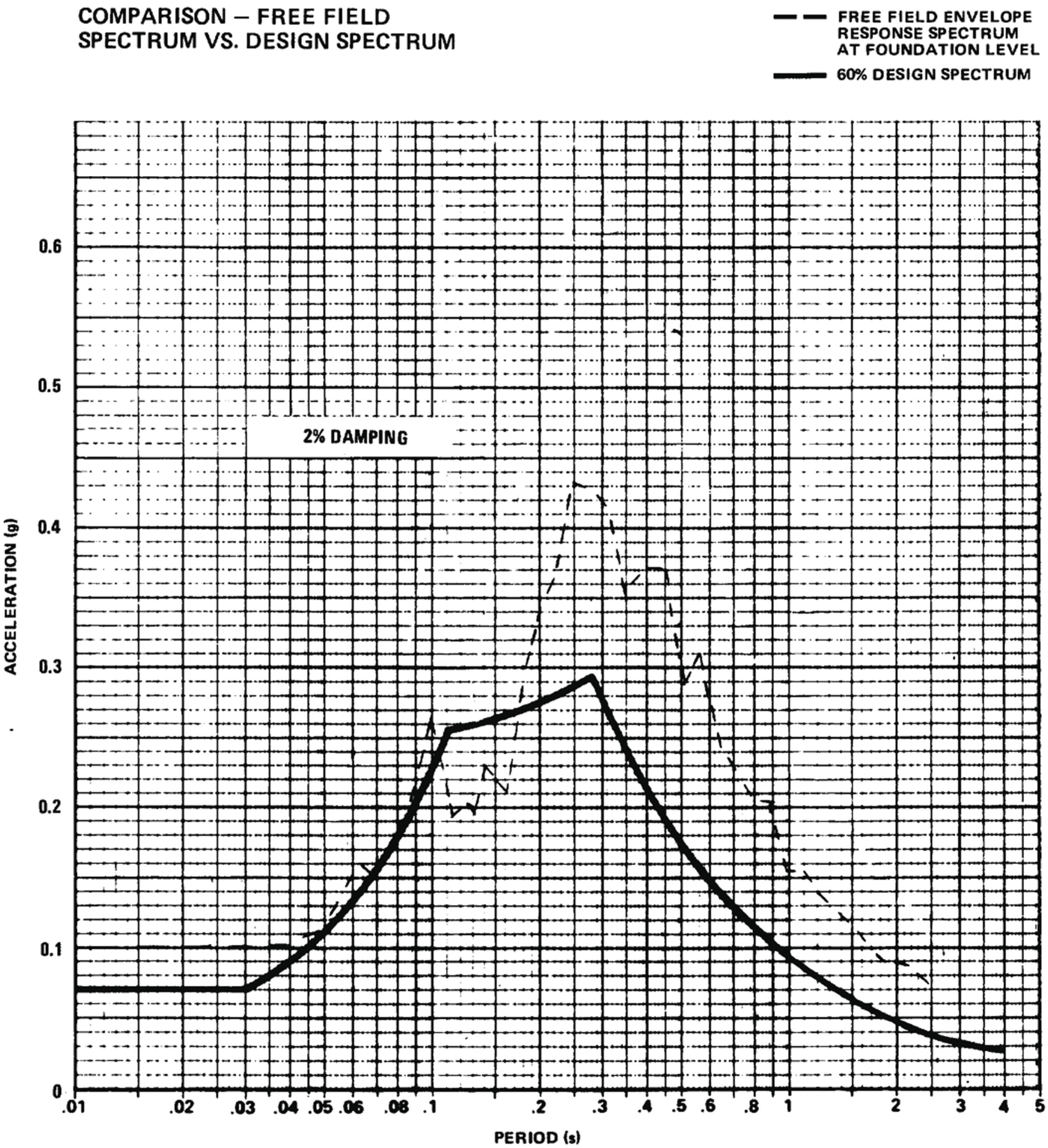


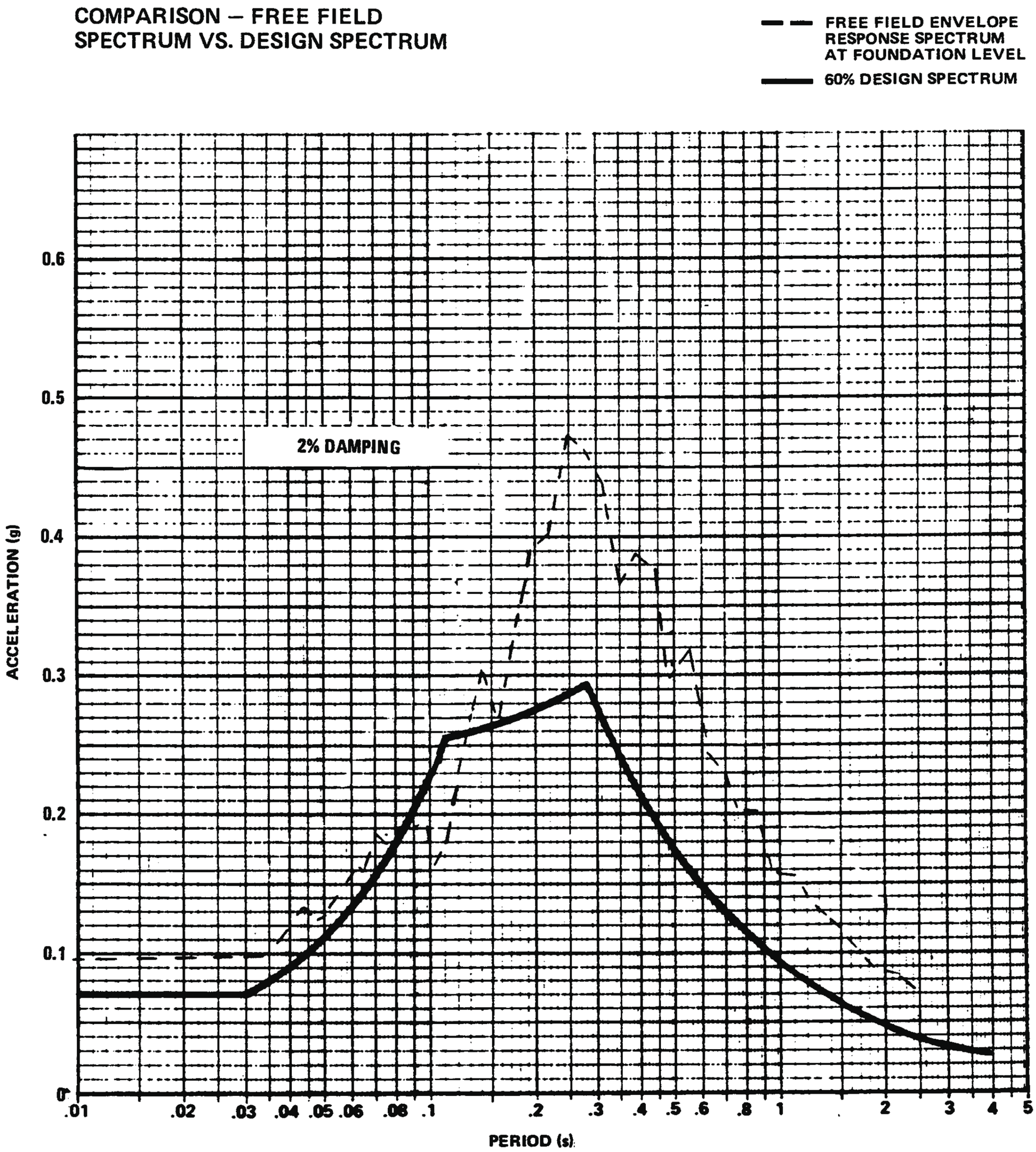
REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CONTAINMENT BUILDING el 144 ft 0 in., OBE VERTICAL
	FIGURE 3.7.B.2–25



REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	FUEL HANDLING BUILDING el 152 ft 0 in., OBE VERTICAL
	FIGURE 3.7.B.2–26



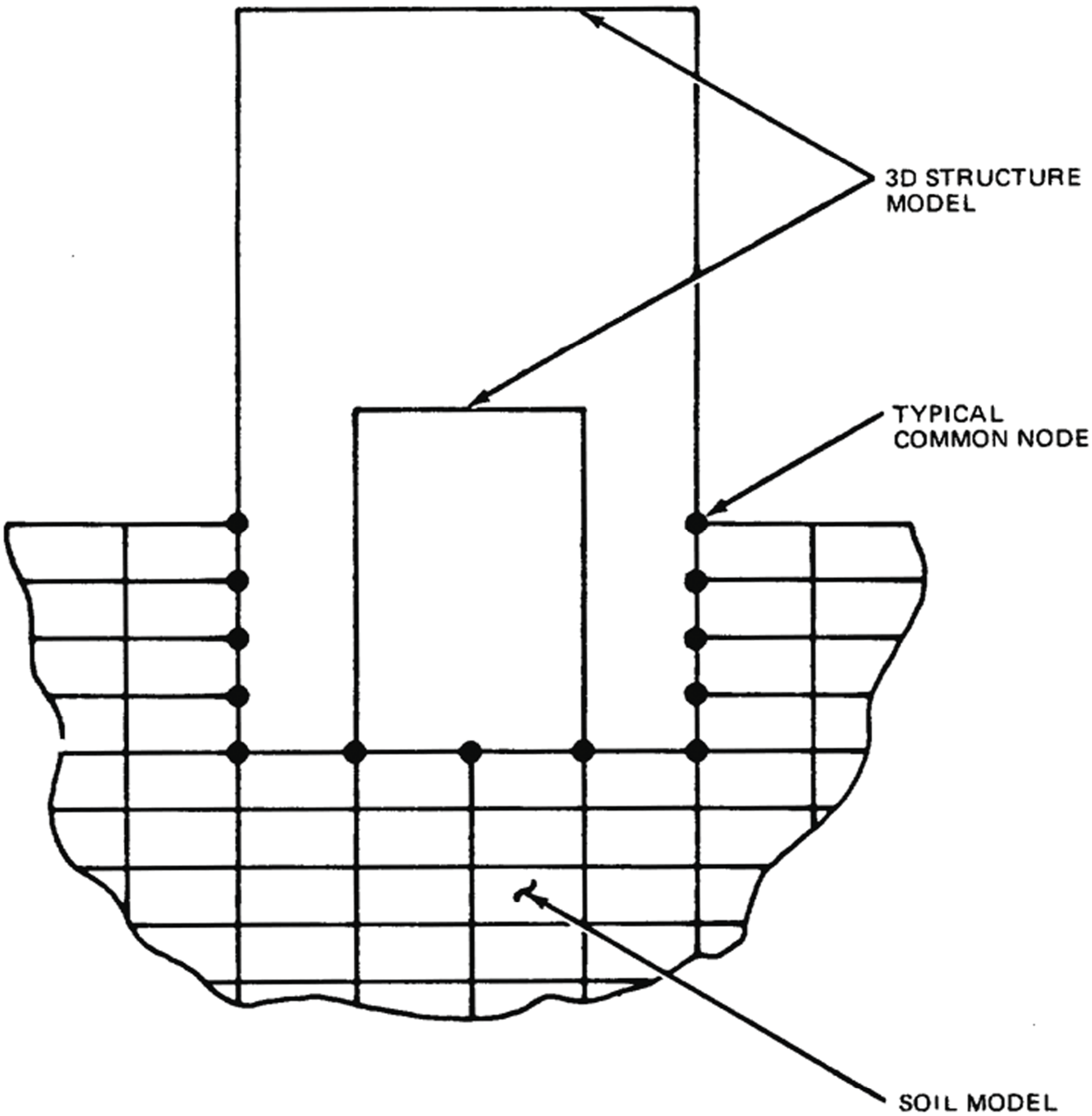
REV 14 10/07

SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CONTROL BUILDING
el 173 ft 0 in., OBE VERTICAL

FIGURE 3.7.B.2–27

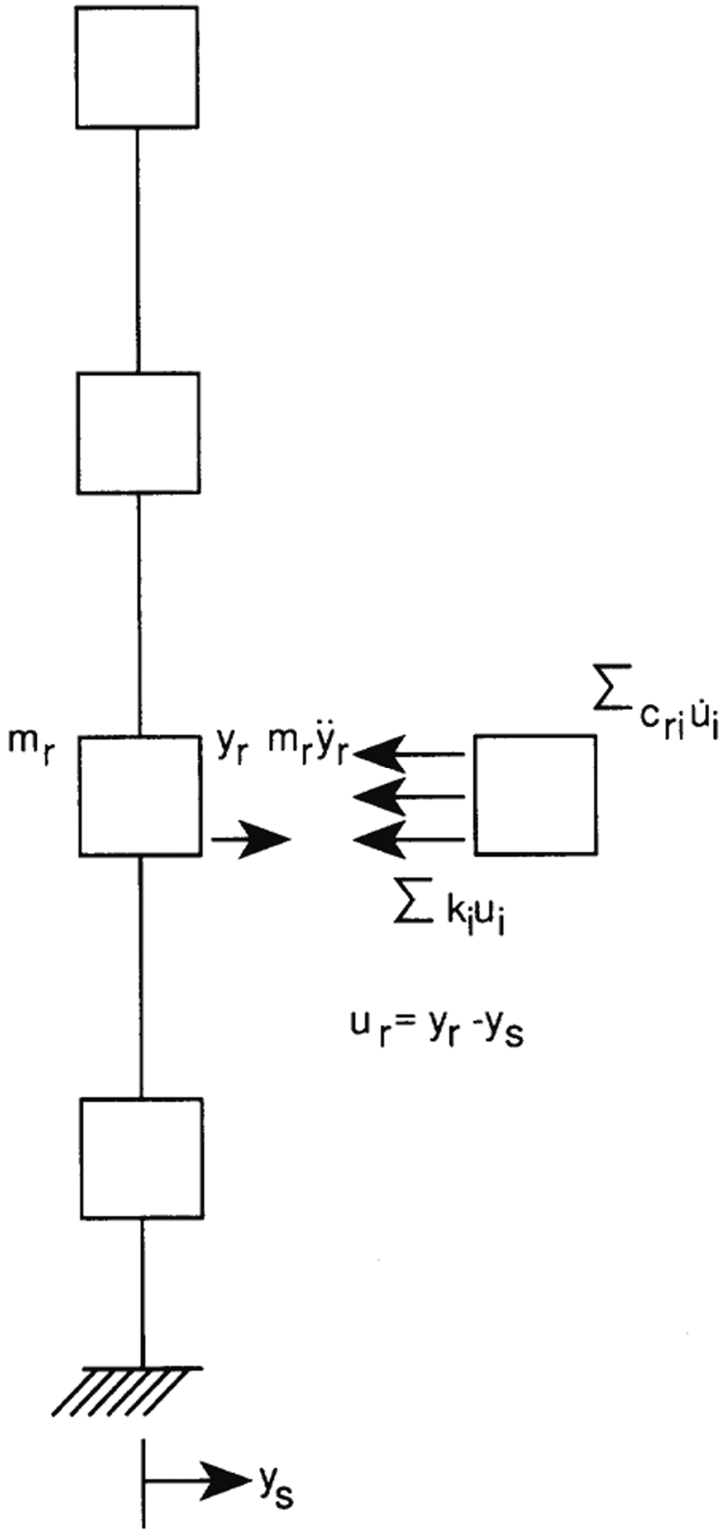


ILLUSTRATIVE FIGURE

REV 14 10/07

SOIL–STRUCTURE INTERACTION
MODEL COMMON NODES

FIGURE 3.7.B.2–28



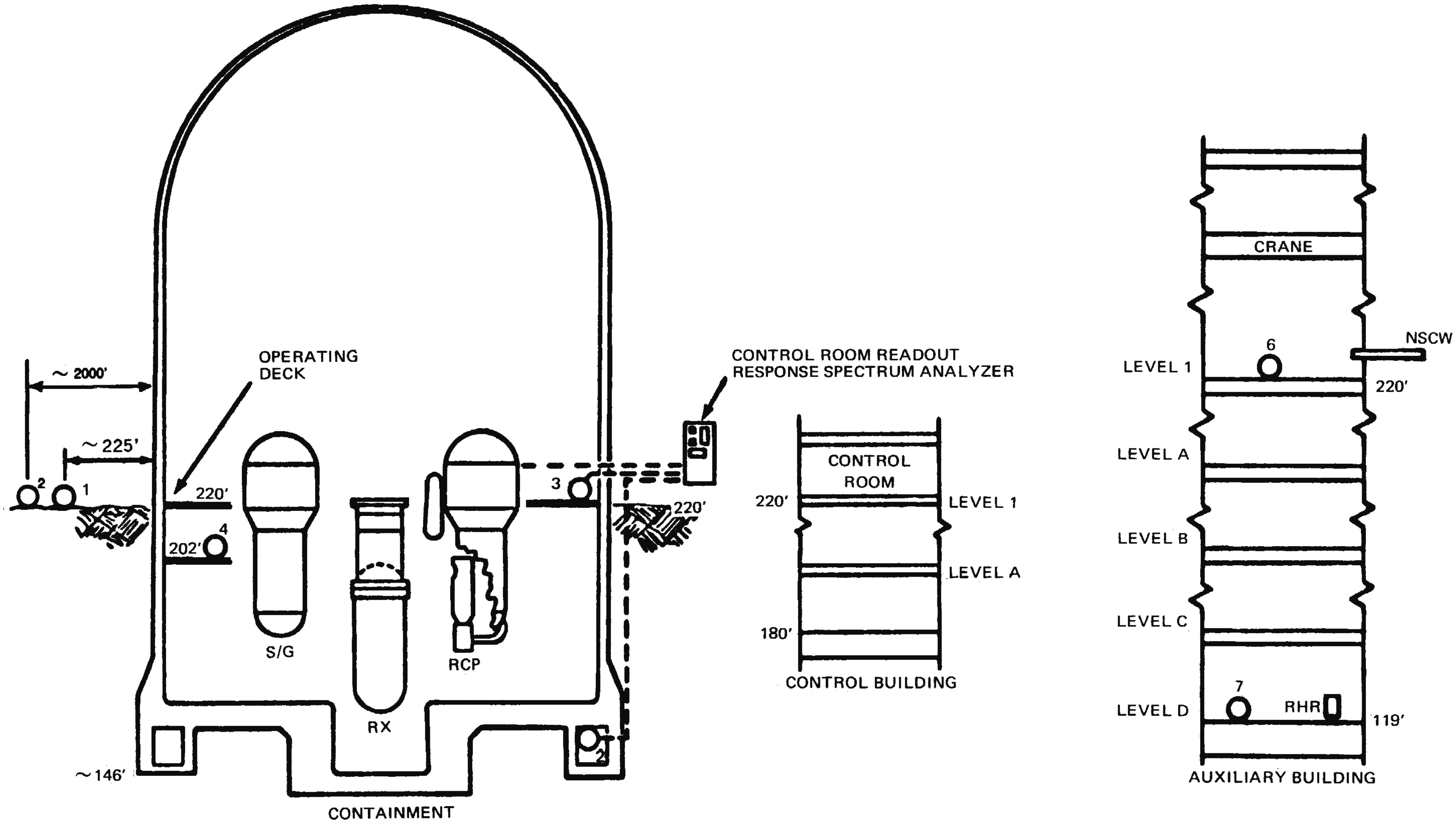
REV 14 10/07

SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MILTIDEGREE–OF–FREEDOM SYSTEM

FIGURE 3.7.N.2–1



NOTE: NUMBERS DEFINED IN TABLE 3.7.4-1

REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	SEISMIC MONITORING INSTRUMENT LOCATIONS FIGURE 3.7.4–1

3.8 DESIGN OF CATEGORY 1 STRUCTURES

This section provides information concerning the containments and their internal structures and other Category 1 structures and their foundations and supports.

3.8.1 CONCRETE CONTAINMENT

This section provides the following information on the concrete containment:

A. The physical description.

B. The applicable design codes, standards, and specifications.

C. The loading criteria, including loads and load combinations.

D. The design and analysis procedures.

E. The structural acceptance criteria.

F. The materials, quality control programs, and special construction techniques.

G. The testing and inservice inspection programs.

3.8.1.1 Description of the Containment

The containment (drawing 1X2D01A001) consists of a prestressed reinforced concrete cylinder and hemispherical dome supported on a flat, conventionally reinforced concrete basemat with a central cavity and instrumentation tunnel to house the reactor vessel. The inside face of the containment is lined with steel plates welded together to form a leaktight barrier. The liner is typically 1/4 in. thick and is thickened locally around penetrations, basemat anchorages, and large brackets. The liner plate, including the thickened plate areas, is anchored to the concrete. Leak chase channels are provided at seam welds which are inaccessible after construction. The liner plate system is shown in drawing 1X2D01J015.

3.8.1.1.1 Containment Foundation

The foundation consists of a circular basemat which is 154 ft 6 in. in diameter and 10 ft 6 in. thick, with a minimum thickness of the concrete basemat of 8 ft. 3 in. The minimum thickness of the concrete instrumentation cavity is 8 ft.

Typical basemat reinforcing is shown in drawings 1P01C016, 1P01C018, and 1P01C017.

Attached to the underside of the basemat, at its periphery, is a tendon gallery, 10 ft wide by approximately 9 ft 6 in. high. The gallery provides access to the vertical tendons for installation, tensioning, and inservice inspection.

Vertical shafts surround each of the shell buttresses to provide access for installation, tensioning, and inservice inspection of the hoop tendons. Access shaft No. 1 is attached to the basemat and tendon gallery; it provides access to the tendon gallery. Above the basemat, this access shaft is structurally isolated from the containment shell. Access shafts Nos. 2 and 3 are integral portions of adjoining buildings and are also structurally isolated from the containment.

A steel liner plate with a leak chase system covers the top of the basemat. A structural concrete fill slab, 2 ft 9 in. thick, is placed on top of the liner plate to protect the liner from damage during erection and maintenance.

Anchorage of interior structures (drawings 1X2D48E003, 1X2D48E004, 1X2D48E005, 1X2D48E008, and 1X2D48E007) through the floor liner into the basemat is accomplished by welding cadweld connector sleeves to opposite sides of thickened sections of liner plate. Further discussion is provided in subsection 3.8.3.

3.8.1.1.2 Containment Shell (Cylinder and Dome)

The cylinder and dome, both 3 ft 9 in. thick, are provided with reinforcing steel for load resistance and crack control. Typical shell reinforcing, including reinforcing around piping penetrations, the equipment hatch, and at the buttresses, is shown in drawings 1P01W077, 1P01W081, 1P01W072, 1P01W058, and 1P01W059. A haunch, varying in thickness from 3 ft 9 in. to 4 ft 9 in., is provided at the junction of the cylinder and basemat. (See drawing 1X2D01A001.) At the equipment hatch penetration, the concrete shell is thickened locally on the inside face. (See drawings 1X2D01J017 and 1X2D01J018.) The shell liner plate and its anchorage and stiffening system are shown in drawing 1X2D01J015.

A two-way ungrouted prestressing system is used; it consists of vertical inverted U-shaped tendons and circumferential hoop tendons. The tendon arrangement is shown on drawings 1X2D01K002, 1X2D01K003, 1X2D01K005, 1X2D01K006, 1X2D01K007, and 1X2D01K008, and the anchorage is schematically shown on drawing AX6DD313. Each tendon consists of 55 strands with end anchor components, consisting of anchor heads, wedges, and retainer plates, utilizing the VSL Corporation anchorage and post-tensioning techniques. The tendons are installed in metal sheaths which form ducts through the concrete. The sheaths are filled with grease to protect the tendons from corrosion.

Attachments to the shell, such as brackets for the support of the polar crane, electrical conduit or cable tray, spray piping, and ventilation, are anchored into the shell concrete. For details of the polar crane bracket, see drawing 1X2D01J022.

Steel components of the containment that resist pressure and are not backed by concrete are discussed in subsection 3.8.2.

3.8.1.2 <u>Applicable Codes, Standards, and Specifications</u>

The following documents are applicable to the design, materials, fabrication, construction, inspection, testing, and inservice surveillance of the containment structure.

3.8.1.2.1 Codes and Standards

The following code is applicable to containment design:

- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter called ASME Code), Section III, Division 2, 1975 Edition through the Winter 1975 Addenda, Article CC-3000 only.

3.8.1.2.2 Regulations

The following regulations apply to containment design:

- 10 CFR 50, Domestic Licensing of Production and Utilization Facilities.

3.8.1.2.3 General Design Criteria (GDC)

The following GDC apply:

- GDC 1, 2, 4, 16, and 50 of Appendix A to 10 CFR 50.

3.8.1.2.4 Regulatory Guides

Applicable dates and revisions are provided in section 1.9.

- 1.10, Mechanical (Cadweld) Splices in Reinforcing Bars of Category 1 Concrete Structures.
- 1.15, Testing of Reinforcing Bars for Category 1 Concrete Structures.
- 1.18, Structural Acceptance Test for Concrete Primary Reactor Containments.
- 1.19, Nondestructive Examination of Primary Containment Liner Welds.
- 1.35, Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures.
- 1.55, Concrete Placement in Category 1 Structures.
- 1.94, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.
- 1.103, Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments.
- 1.136, Materials, Construction, and Testing of Concrete Containments.

3.8.1.2.5 Industry Standards

Nationally recognized industry standards, such as American Society of Testing Materials (ASTM), American Concrete Institute (ACI), and American Iron and Steel Institute (AISI), are used to specify material properties, testing procedures, fabrication, and construction methods.

3.8.1.3 Loads and Load Combinations

The loads and load combinations used in the analysis and design of the containment are in conformance with the requirements of ASME Code, Section III, Division 2, Subarticle CC-3200, except as discussed below. The load symbol definitions and load combinations are shown in table 3.8.1-1.

A. The word "or" has been inserted between E_o and W in the third abnormal/severe environmental combination to correct an error in the ASME Code. This error has also been recognized in section 3.8.1 of the Standard Review Plan (July 1981).

B. W loads have been included in the construction combination.

C. R_o loads have been included in the extreme environmental load combinations.

D. An additional combination accounting for the effects of the blast load has been included in the extreme environmental factored load combination.

The blast load is conservatively taken as a peak positive incident over pressure, as discussed in subsection 2.2.3. This is considered in the design as an equivalent static pressure of 2 psi.

The determination of wind and tornado loads, flood design loads, and seismic loads is discussed in sections 3.3, 3.4, and 3.7, respectively. Missile effects and pipe rupture effects are discussed in sections 3.5 and 3.6, respectively.

The relationship between the level of prestressing and the containment design pressure is discussed in subsection 6.2.1 of BC-TOP-5-A.

Pressure transients resulting from the design basis accident (DBA) and other lesser accidents serve as the design basis for the containment design pressure of 52 psig.

Temperature gradients through the containment shell for loss-of-coolant accident (LOCA) and main steam line break conditions are shown in figure 3.8.1-1. The variation of temperature with time and the expansion of the liner plate with temperature are considered in determining thermal effects.

Post-LOCA flooding of the containment is postulated to reach el 181 ft 2 in. and is considered as a hydrostatic load.

As described in sections 3.4 and 3.5, there are no credible loads induced by floods, potential aircraft crashes, missiles generated from nearby military installations, or turbine missiles.

3.8.1.4 Design and Analysis Procedures

The design and analysis procedures for the containment generally conform to the requirements of ASME Code, Section III, Division 2, Article CC-3000. Exceptions to this article of the code are as follows:

A. Changes to the load combination table are noted in paragraph 3.8.1.3.

B. In subparagraph CC-3421.4.1(b), the second sentence is taken to read "If M' is negative, use equation (4)." (This conforms to an error correction made by the Winter 1976 Addenda.)

C. In subparagraph CC-3421.4.2, the second sentence is taken to read "For $\rho >$ 0.003, $K = 1.75 - (0.036/\eta\rho) + 4.0\ \eta\rho$, but not less than 0.6 inch." (This conforms to an error correction made by the Summer 1977 Addenda.)

D. The increase in allowable stress permitted for combinations which include W or Eo loads (Table CC-3431-1) is not used.

The guidelines for design presented in BC-TOP-5-A, sections 6.2 and 6.3, are used.

The design of tendon end anchorage zones and reinforcement for the buttresses are as discussed in BC-TOP-7, BC-TOP-8, and BC-TOP-5-A, section 6.6.

3.8.1.4.1 Analysis Procedures

The containment shell and basemat are analyzed employing two three-dimensional finite element computer models. One model is used to obtain results for use in the shell design. The other model is used to obtain results for use in the basemat design. Both models represent the cylinder, dome, basemat, and internal structures. For the model used to obtain results for shell design, the basemat is modeled as a flat disc. The cylinder, dome, and basemat are modeled using shell elements. The mesh near the basemat junction and near the springline is made finer than the mesh for the remainder of the structure to obtain accurate results at these discontinuity locations. The model used to obtain results for the basemat design is discussed in paragraph 3.8.5.4.1. A separate finite element model is used for the thickened shell area around the equipment hatch.

Classical theory and methods are used in the analysis of other local areas such as the personnel and escape locks and small penetrations.

3.8.1.4.2 Boundary Conditions and Load Treatment

Both containment models are supported on translational linear soil springs which simulate the VEGP foundation condition. The model of the local area around the equipment hatch uses results from the analysis of the overall structure as boundary conditions. All loads, axisymmetric and nonaxisymmetric, are applied to the three-dimensional models as nodal loads, element loads, or accelerations.

3.8.1.4.3 Creep, Shrinkage, and Cracking Considerations

In the design of the containment post-tensioning system, conservative values for creep and shrinkage are used, based on past experience. The values used are verified by the evaluation of tests performed on samples of the concrete used in the shell. Concrete crack control is accomplished by following the provisions of paragraph CC-3535.

3.8.1.4.4 Computer Programs

The BSAP, BSAP-POST, and TENDON computer codes are used in the containment design and analysis. Descriptions of these computer codes are provided in appendix 3B. The seismic-related codes employed for dynamic analyses are discussed in subsections 3.7.B.2 and 3.7.N.2.

3.8.1.4.5 Tangential Shear

The ASME Code has no provision for allowable stresses for tangential shear in prestressed concrete. The design procedure use for the VEGP containment shell provides a conservative and viable technique for accounting for the effects of tangential shear. Reinforcement for tangential shear is provided by defining "equivalent" membrane forces:

For seismic loads:

$$N_{ht} = \left(N_h{}^2 + V_u{}^2\right)^{1/2}$$

$$N_{vt} = \left(N_v{}^2 + V_u{}^2\right)^{1/2}$$

For wind, tornado, or blast loads:

N_{ht} = $N_h + V_u$

N_{vt} = $N_v + V_u$

where:

N_{ht}, N_{vt} = "equivalent" membrane forces in the horizontal and vertical directions, respectively.

N_h, N_v = membrane forces, due to the same loading causing the tangential shear, in the horizontal and vertical directions, respectively.

Vu = tangential shear force due to seismic, wind, tornado, or blast loads.

3.8.1.4.6 Variation in Physical Material Properties

In the design and analysis of the containment, consideration is given to the effects of possible variations in the physical properties of materials on the analytical results. The variations in physical properties are accounted for by using allowable stress levels, below ultimate strength, for the design of the structure under full service and factored load conditions.

3.8.1.4.7 Liner Plate and Anchorage System

The details which depict the typical liner plate system with anchorages are shown in drawing 1X2D01J015. The design and analysis methods of the liner and its anchorage system are covered in BC-TOP-1. The relative strength of the liner plate against buckling as compared with its anchor and anchor welds is discussed in BC-TOP-1. BC-TOP-1 provides sample calculations that demonstrate that the strength of the anchor and the anchor welds is sufficient to preclude any possibility of overall buckling failure as a result of anchor pullout.

The elastic and plastic solutions are used to analyze the stresses in the anchors and concrete resulting from a buckled panel of the liner. The panel and anchors are modeled as a series of springs. The stresses in the anchors and concrete are obtained by using the strains from all loads being considered, equilibrium of forces, and compatibility of deformations. BC-TOP-1 provides details and further discussions.

3.8.1.4.8 Ultimate Capacity of the Containment

The containment ultimate capacity was calculated and presented in the containment design report submitted under separate cover on October 31, 1984, by GPC Letter GN-433, from Foster to Director NRR.

3.8.1.5 Structural Acceptance Criteria

The fundamental acceptance criterion for the completed containment building is successful completion of the structural integrity test where measured responses are required to be within the limits predicted by analyses. The limits are based on test load combinations and code values for stress, strain, or gross deformation for the range of material properties and construction tolerances specified, as described in paragraph 3.8.1.6.

The limits for allowable stresses and strains are given in ASME Code, Subarticle CC-3400, and are compatible with nationally recognized codes of practice. In this way, the margins of safety associated with the design and construction of the containment building are, as a minimum, the accepted margins associated with nationally recognized codes of practice.

The structural integrity test is planned to yield information on both the overall response of the containment and the response of localized areas. This information, together with the test information documented in BC-TOP-7 and BC-TOP-8, provides direct experimental evidence that the containment structure can withstand the design internal pressure.

The design and analysis methods, as well as the type of construction and construction materials, are chosen to allow assessment of the capability of the structure throughout its service life[a]. Additionally, surveillance testing provides further assurances of the continuing ability of the structure to meet its design functions.

3.8.1.5.1 Factors of Safety and Margin of Safety

Design of the containment is based on ASME Code, Section III, Division 2, Article CC-3000; therefore, adequate margins of safety exist when the containment is subject to the load combinations prescribed by the code.

3.8.1.5.2 Allowable Stresses

The allowable stresses for service load and factored load conditions in concrete, reinforcing steel, and the tendon system are as specified in ASME Code, Section III, Division 2, Subarticle CC-3400. The design for tangential shear in prestressed concrete is discussed in paragraph 3.8.1.4.5.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

This section contains information relating to the materials, quality control program, and special construction techniques used in the fabrication and construction of the containment.

3.8.1.6.1 Concrete

The compressive strength of concrete used for the containment is as follows:

---

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. The containment tendon prestress analysis was evaluated as a time-limited aging analysis (TLAA) for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in subsection 19.4.4.

Basemat, basemat fill slab, tendon gallery, and access shaft No. 1, f'_C = 5000 psi.

Cylinder and dome f'_C = 6000 psi.

The test age of concrete using pozzolan in the concrete mix is designated as 91 days. The test age of concrete without pozzolan in the concrete mix is designated as the normal 28 days. These strength designations are in accordance with ACI 318-71.

Structural concrete is batched and placed in accordance with the Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete (ACI 304) and Building Code Requirements for Reinforced Concrete (ACI 318-71).

A. Cement

Cement is Type II conforming to the Specification for Portland Cement (ASTM C 150). The cement contains no more than 0.60 percent by weight of alkalis calculated as sodium oxide plus 0.658 potassium oxide. Certified copies of material test reports showing the chemical composition and physical properties are obtained for each load of cement delivered.

B. Aggregates

All aggregates conform to standard specifications for concrete aggregates (ASTM C 33). In addition to the specified gradation, the fine aggregate (sand) has a fineness modulus of not less than 2.3 nor more than 3.1 during normal operations. The average of five successive tests is required to be between 2.5 and 3.0. At least four of five successive test samples should not vary in fineness modulus more than 0.20 from the moving average established by the last five tests. Coarse aggregates may be rejected if the loss, when subjected to the Los Angeles Abrasion Test, using grading A, exceeds 40 percent by weight at 500 revolutions.

Acceptance of source and aggregates is based on the following tests:

ASTM No.	Name
C 131	Los Angeles Abrasion Test
C 142	Clay Lumps and Friable Particles
C 117	Material Finer Than No. 200 Sieve
C 123	Lightweight Pieces
C 40	Organic Impurities
C 235	Scratch Hardness
C 289	Potential Reactivity (Chemical)
D 1411	Water Soluble Chlorides
C 125	Definition of Terms Relating to Concrete and Concrete Aggregates

ASTM No.	Name
C 127	Specific Gravity and Absorption of Coarse Aggregates
C 128	Specific Gravity and Absorption of Fine Aggregates
C 136	Sieve Analysis
C 88	Soundness
C 295	Petrographic Examination

C. Water

Water and ice for mixing shall be clean, with a total solids content of not more than 2000 ppm as measured by ASTM D 1888. The mixing water, including that contained as free water in aggregate, shall contain not more than 250 ppm of chlorides as Cl^- as determined by ASTM D 512. Chloride ions contained in the aggregate shall be included in calculating the total chloride ion content of the mixing water. The chloride content contributed by the aggregate shall be determined in accordance with ASTM D 1411.

D. Admixtures

The concrete may also contain an air-entraining admixture and/or a water-reducing and retarding admixture. The air-entraining admixture is in accordance with the Specification of Air Entraining Admixtures for Concrete (ASTM C 260). It is capable of entraining 3 to 6 percent air, is completely water soluble, and is completely dissolved when it enters the batch. Superplasticizers, entraining from 1.5 to 4.5 percent air, may be used in concrete mixes ($f' = 5000$ psi, maximum) for congested areas to improve workability and prevent the formation of voids around reinforcement. The water-reducing admixture conforms to standard specification for Chemical Admixtures for Concrete (ASTM C 494), Types A and D. Type A is used when average ambient temperature for the daylight period is below 70°F. Type D is used when average ambient air temperature for the daylight period is 70°F and above. Pozzolans, if used, conform to Specifications for Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete (ASTM C 618) except that the loss on ignition shall be limited to 6 percent. Admixtures containing more than 1 percent by weight chloride ions are not used.

E. Concrete Mix Design and Testing

Concrete mixes are designed in accordance with ACI 211.1, Recommended Practice for Selecting Proportions for Normal and Heavy Weight Concrete, using materials qualified and accepted for this work.

An independent testing laboratory at the site designs and tests the concrete mixes. Only mixes meeting the design requirements specified for concrete are used.

3.8.1.6.2 Reinforcing Steel

Reinforcing bars for concrete are deformed bars meeting requirements of Specification for Deformed and Plain Billet Steel Bars for Concrete Reinforcement (ASTM A 615), Grade 60.

Mill test reports, in accordance with ASTM A 615, are obtained from the reinforcing steel supplier to substantiate specification requirements.

In addition, tonnage of reinforcing steel of each size and grade for user tests on full-diameter specimens is in accordance with Regulatory Guide 1.15.

The test procedures are in accordance with ASTM A 370, and acceptance standards are in accordance with ASTM A 615.

3.8.1.6.3 Prestressing System

A description of the prestressing system is given in paragraph 3.8.1.1.2. Additional material properties for each component of the prestressing system are described as follows:

A. Prestressing Strands

Strands are nominal 1/2-in. diameter, of the seven-wire low-relaxation type, having a guaranteed minimum ultimate tensile strength, f_{pu}, of 270,000 psi or greater based on the nominal steel area of the strand. The minimum yield stress is .90 f_{pu}. All strands conform to ASTM A 416 for weldless grade.

B. Other Prestressing System Components

Materials for the other components required for the prestressing system conform with the following steel specifications:

Bearing plates	ASTM A 537 Class 1 carbon-manganese-silicon steel
Trumpets	ASTM A 569
Trumpet extensions	ASTM A 513
Anchor heads	AISI 1026
Shims	ASTM A 572
Wedges	DIN 17210 and AISI 8620
Tendon sheathing (semirigid corrugated tubing)	ASTM A 527

3.8.1.6.4 Liner Plate

A. The containment structure is lined with 1/4-in.-thick welded steel plate, except in limited areas where thickened plate is utilized, and conforms with the requirements of ASME SA-285, Low and Intermediate Tensile Strength Carbon Steel Plates for Pressure Vessels, Grade A, to ensure a leaktight membrane.

This steel has a minimum yield strength of 24,000 psi and a minimum elongation in an 8-in. specimen of 27 percent.

The materials, fabrication, and erection of the containment reactor cavity liner plate system (el 142 ft 6 in. to 169 ft 0 in.) and thickened basemat liner plates, cadwelds, and embeds at el 169 ft 0 in. are in accordance with 1971 ASME Code, through Summer 1973 Addenda, Section III, Division 1; Section V; Section VIII, Division 1; and Section IX.

The materials, fabrication, and erection of the containment structure 1/4-in. basemat liner plates at the basemat level, el 169 ft 0 in., and all of the cylinder wall and dome liner plate system, support brackets, and penetrations integral with the liner plate are in accordance with 1974 ASME Code, through Summer 1975 Addenda, Section III, Division 1, Subsection NE; Section V; Section VIII, Division 1; and Section IX.

B. Liner plate greater than 1/4 in. thick conforms to the requirements of ASME SA-516, Grade 70, or SA-36 and satisfies the impact requirements specified in ASME Section III, Division 1, Subsubarticle NE-2320, for materials to be impact tested at a maximum temperature of 30°F below the lowest service metal temperature.

C. Materials for the equipment hatch and personnel airlocks, process pipe penetration assemblies, electrical penetration assemblies, and fuel transfer tube housing bellows assemblies are listed in paragraph 3.8.2.6.

3.8.1.6.5 Liner Plate Anchors and Associated Hardware

The following parts wherever used conform to the material requirements of the standards listed below:

A. Anchor bolts and machine bolts - ASTM A 307.

B. High-strength bolts - ASTM A 325.

C. High-strength anchor bolts and studs - ASME SA-540, CL1 and CL4.

D. Nelson concrete anchor studs Type H4 - ASTM A 108.

E. Cadweld connectors - AISI 1018 or AISI 1026.

F. Unistruts - ASTM A 570, Grade C.

G. Heavy hex nuts for high-strength bolts - ASTM A 194, Grade 2H.

H. Hardened washers ASTM A 325, Rockwell Hardness C 38, or as indicated on design drawings.

I. Long hex couplings - ASTM A 307, Grade A, may be substituted for ASTM A 36.

J. Leak chase test couplings - carbon steel ASME SA-234.

3.8.1.6.6 Structural Steel

A. The following parts conform to the requirements of ASTM A 36:

1. Liner plate angle, flat bar, tee, and channel stiffeners.

2. Support brackets not penetrating the liner plate. (ASME SA-516, Grade 70 may be substituted for ASTM A 36).
3. Framing, bracing, and tie rods.
4. Embedded floor beams.
5. Backing strips.
6. Leak chase system channels and angles.

B. Polar crane support bracket plates conform to ASME SA-516, Grade 70. Material for parts of the bracket that penetrate the thickened liner and exceed 5/8 in. in thickness are impact tested at a maximum temperature of -13°F, per ASME Code, Section III, Division 1, Subsection NE, Subparagraph NE-2321.2.

C. Nuclear steam supply system (NSSS) embeds located below top of basemat and penetrating the pressure boundary conform to ASTM A 588, F = 50.0 ksi minimum.

D. Penetration sleeve material conforms to the requirements of the standard specifications listed below. The lowest service metal temperature is 17°F, and the maximum test temperature is -13°F.

1. Penetration sleeves - ASME SA-333, Grade 1 or 6; or ASME SA-516, Grade 70.
2. Penetration sleeve reinforcing (thickened liner plate) - ASME SA-516, Grade 70.
3. Anchor rings, stiffener rings, and gussets - ASME SA-516, Grade 70.

3.8.1.6.7 Tendon Corrosion-Retarding Compound

The tendon corrosion-retarding compound used for sheathing filler material is Visconorust 2090P-4.

3.8.1.6.8 Quality Control

The quality control program for fabrication, construction, testing, and examination is discussed in this paragraph. Conformance to Regulatory Guide 1.94 is as described in paragraph 3.8.3.6.2.C.

Quality control procedures are implemented during construction and inspection. The quality control procedures are specified in the specifications covering the fabrication, furnishing, and installation of each structural component and provide inspection and documentation to assure that the codes and construction practices are met.

The quality control program includes an organizational scheme which functions in accordance with policies delineated in the Bechtel Quality Assurance Manual, Vogtle Field Procedure Manual, and Bechtel Procurement Supplier Quality Manual.

A. Tests for Concrete

Concrete for the containment structure is tested in accordance with ACI 318-71. Concrete placement is accomplished in accordance with Regulatory Guide 1.55 as discussed in section 1.9.

B. Tests for Reinforcing Steel

Reinforcing steel is tested in accordance with Regulatory Guide 1.15.

C. Tests for Mechanical Splices

Control of cadweld mechanical splices for reinforcement utilizing filler metal and enclosing sleeve is in accordance with Regulatory Guide 1.10.

D. Test Procedures for the Liner Plate

The nondestructive examination of the liner plate welds is in accordance with Regulatory Guide 1.19 with exceptions noted in section 1.9.

1. Liner Plate and Thickened Liner Plate

   The liner plate is 1/4 in. thick. Thickened liner plates are used as penetration sleeve reinforcing plates and as a part of the bracket and attachment assemblies. Thickened floor liner plate is ultrasonically inspected in accordance with ASME SA-435, except that inspection covers 100 percent of the plate area. Thickened plates over 5/8 in. in thickness require Charpy V-notch impact tests in accordance with ASTM A 593-72. Testing is as specified in paragraph 3.8.1.6.4.B. Structural steel members and electrical ground rods are not classified as thickened liner plate.

2. Penetration Sleeves

   Sleeves over 5/8 in. in thickness require the same Charpy V-notch impact tests as for thickened liner plates in paragraph 3.8.1.6.4.B, except for minimum requirements of ASME SA-333, Grades 1 and 6. See paragraph 3.8.3.6.

3. The quality control procedures that ensure the suitability of the steel plate material are discussed as follows:

   a. For brackets and attachments that are not continuous through the liner plate:

      (1) The liner plate greater than 5/8 in. in thickness is ultrasonically examined for delaminations.

      (2) The strength in the through-the-thickness direction is taken as one-half of that in the transverse direction unless tests are performed to justify higher values.

   b. Ultrasonic examination is required for the floor liner plate greater than 1/4 in. in thickness.

E. Control Procedures for the Containment Liner Plate Attachments

The quality control procedures associated with penetrations, attachments, and hardware are incorporated in the quality control procedures for the liner plate.

F. Control Procedures for Structural and Miscellaneous Steel

Quality control procedures for structural and miscellaneous steel and the associated structural welding conform to the requirements specified in paragraph 3.8.1.6.6.

G. Control Tests and Inspection of Prestressing System

The following quality control procedures are used:

1. Prestressing Strands

   a. Each tendon is individually identified and traceable to the heat numbers of the wire utilized in its buildup. Physical test reports supporting the integrity of each heat of material are reviewed as a condition of acceptance.

   b. Specimens are cut from each reel of strand and tension tested to rupture, to assure compliance to specifications.

   c. Strands are examined for workmanship and quality prior to fabrication of the tendon.

2. Bearing Plates and Transitions

   a. Verification is made that the bearing plate material complies with that specified on the drawings. Compliance is evidenced by mill test reports traceable to the heat number by a serial number permanently marked on each bearing plate.

   b. Plates are examined for workmanship and quality. Cracks, burrs, corrosion, and other defects are not acceptable.

3. Anchor Head

   Raw material is accompanied by mill certificates and subjected to receiving inspection.

4. System Performance Tests

   a. Static Tensile Test

      Typical anchorage and tendon details are tested to show that the anchorage develops the minimum guaranteed ultimate strength of the tendon.

   b. Dynamic Tensile Test

      High and low cycle dynamic tensile tests are performed on specimens having at least 10 percent of the steel area of the full-size production tendons. The high cycle test has 500,000 cycles of stress variation, from 60 percent to 66 percent of the minimum ultimate tensile strength. The low cycle test has 50 cycles of stress variation, from 40 percent to 80 percent of the minimum ultimate tensile strength. These tests demonstrate that the prestressing system is able to withstand cyclic loadings without failure.

c. Low Temperature Test

Documentary evidence of certified testing at temperatures below the lowest anticipated service temperature substantiates that the anchorage assembly, including the bearing plate, is capable of transmitting the ultimate load of the tendon into the structure without brittle fracture. BC-TOP-8 presents an example of accepted documentation for the cold environment test.

H. Liner Plate Erection Tolerances

Cylinder and dome liner plates are used as forms and erection precedes the concrete placements. Tolerances for erection of the liner plate and the penetration sleeve assemblies are specified as follows:

1. Tolerances for Liner Plate (Including Floor, Cylinder, and Dome)

The following tolerances apply to the fabrication of the liner plate:

a. The radial location of any point on the cylindrical liner shell does not vary from the design radius by more than ±3 in. Measurement is made at 30° spacings for each 10 ft of rise. For the hemispherical dome roof, the actual radius does not exceed +8 in. or -12 in. from the design radius of 70 ft 0 in. measured from the springline el 327 ft 9 in.

b. Misalignment of butt-welded joints is not more than 1/16 in. for 1/4-in. plates. On plates thicker than 1/4 in., the alignment tolerance is as specified in the ASME code.

c. A 15-ft-long template curved to the required radius does not show deviations of erected liner plate from template of more than 1 in. when placed against the completed surface of the shell within a single plate section and not closer than 12 in. to a weld seam. When the template is placed across one or more weld seams, the deviation does not exceed 1/2 in. The effect of change in plate thickness or of weld reinforcement is excluded when determining deviations.

d. A 15-in.-long template curved to the required radius does not show deviations of erected liner plate from template of more than 1/8 in. inward and 3/8 in. outward when placed against the completed surface of the shell within a single plate section and not closer than 12 in. to a weld seam.

e. The deviation in the liner plate between liner plate stiffeners does not exceed ±1/8 in. when referenced to the theoretical surface.

f. Sharp bends are not permitted unless provision has been made for them in the design. A sharp bend is defined as any local bend that deviates from the design radius or a vertical straight edge by an offset of more than 1/2 in. in 1 ft. The template used to measure the local deviations is 1 to 2 ft longer than the area of the deviation itself.

g. The slope of the cylindrical wall liner plate, referred to true vertical, does not exceed 1/180 within any 10-ft ring. The overall vertical out-of-plumb of the cylindrical liner plate at springline el 327 ft 9 in. does not exceed 3 in.

h. An individual wall liner plate section does not deviate from a vertical straight template by more than ±3/4 in. in 10 ft.

i. Dimensions of all liner plates as cut are within ±1/8 in. of the design dimensions.

j. Liner plate sections are fabricated in the lengths necessary for the most efficient methods of handling and erection.

2. Tolerances for Penetration Sleeve Assemblies

The following requirements and those of items 1.b, c, g, and i immediately above apply for penetration sleeve assemblies:

a. A 30-in.-long template curved to the required radius does not show deviations of more than 3/4 in. when placed against the completed surface of the liner plate within a single plate section.

b. Alignment of the axis of cylindrical liner plate penetrations as erected does not vary by more than 1° for pipe diameters greater than 12 in. or by more than 2° for pipe diameters less than or equal to 12 in. Individual penetrations and penetrations other than the main steam and feedwater penetrations in thickened liner plates having multiple penetrations are located within ±1 in. of their design elevations and circumferential locations in the cylindrical liner plate. The location of the main steam and feedwater penetrations in the shell are within ±1/2 in. of the design location.

c. The location of penetrations in a thickened liner plate having multiple penetrations is within ±1/4 in. of the dimensions shown on the design drawings, relative to the thickened liner plate edges.

3. Miscellaneous Tolerances

The following tolerances apply to various items comprising the liner plate system:

a. The location of the crane support brackets with respect to azimuth and elevation are within ±1/2 in. of the design location. All other brackets located in the cylindrical shell are located within ±1 in. of the design location.

b. The liner plate stiffeners are placed so that the relative locations with respect to each other are within ±1/2 in. of the dimensions shown.

c. The radius of curvature of the channels, angles, etc., that maintain the shell curvature are within ±5 percent of the design radius shown.

d. The field placement of the embedded floor beams and thickened floor liner plate is within ±1/2 in. of the theoretical location in plan and are level within ±1/4 in. overall and ±1/8 in. within any 20-ft arc.

e. The locations of cadwelds and threaded sleeves on the thickened floor liner plate are within ±1/8 in. of the theoretical location. The thickened plate dimensions are within ±1/8 in. of dimensions given on the engineering design drawings.

I. Reinforcing Steel Placement Tolerances

The tolerances for placing reinforcing steel conform to the following:

1. In concrete members where the effective depth, d, is less than 24 in., the maximum vertical cover does not exceed the minimum cover by more than 1 in.
2. In concrete members where the effective depth, d, is 24 in. or more, the maximum vertical cover does not exceed the minimum cover by more than 2 in.
3. For bars placed crosswise of members, the total number of bars is maintained and spaced evenly within 2 in.
4. Longitudinal location of bends and ends of bars is ±3 in., provided the minimum cover is maintained, except at discontinuous ends of members, where the tolerance shall be ±1/2 in.

J. Erection Tolerances for the Prestressing System

1. Alignment of tendon sheathing is maintained within ± 3/4-in., except as noted on the design drawings.
2. Bearing plate assemblies are positioned within ± 1/2 in., unless otherwise shown on erection drawings.

3.8.1.6.9 Special Construction Techniques

Because of their more efficient usage on cylindrical surfaces, 10-ft-high steel jump forms are employed on the exterior surfaces of the containment cylinder. This construction technique does not affect the structural integrity of the completed structures.

3.8.1.7 Testing and Inservice Inspection Requirements

3.8.1.7.1 Structural Integrity Pressure Test

Following construction, each containment is proof tested at 60 psi, which is 115 percent of the design pressure. During this test, deflection measurements and concrete crack inspections are made to determine that the actual structural response is within the limits predicted by the design analyses.

The VEGP containments are considered as nonprototype structures and need not be strain measured. The San Onofre Nuclear Generating Station (SONGS) Unit 2 containment is chosen as the prototype structure, based on the following comparison:

		VEGP Units 1 and 2	SONGS Unit 2
A.	Shape	Vertical cylinder with hemispherical dome	Vertical cylinder with hemispherical dome
B.	Size:		
	Inside diameter	140 ft 0 in.	150 ft 0 in.
	Cylinder height (basemat to springline)	158 ft 9 in.	97 ft 0 in.
C.	Design pressure	52 psi	60 psi
D.	Basemat thickness	10 ft 6 in.	9 ft 0 in.
E.	Number of wall buttresses	3	3
F.	Number of dome buttresses	3	3
G.	Tendon style	55 each 1/2-in. 7-wire strands	55 each 1/2-in. 7-wire strands

The test procedure conforms to the requirements of Regulatory Guide 1.18 except as noted in section 1.9. Section 9 of BC-TOP-5-A also describes test results obtained using a typical procedure as well as those obtained from early tests where a substantial amount of strain information was collected.

3.8.1.7.2 Long-Term Surveillance

The long-term surveillance program consists of evaluating the general condition of the post-tensioning system. Data on wire corrosion level and tendon lift-off forces are obtained and analyzed. The surveillance program conforms to ASME Code, Section XI, Subsection IWL and applicable addenda as required by 10 CFR 50.55a except where an exemption, relief, or an alternative has been authorized by the NRC.

This surveillance program provides assurances of the continuing ability of the structure to meet the design functions as stated in paragraph 3.8.1.5.

Only the Unit 1 containment is subject to the complete surveillance program; every tendon has provision for removal with detensionable anchorage assemblies.

The Unit 2 containment has permanent anchorage assemblies (nondetensionable). Refer to drawing AX6DD313.

The ASME Section XI Inservice Inspector Program's IWE and IWL inspections are credited as license renewal aging management activities (see subsections 19.2.30 and 19.2.31).

3.8.1.8 Standard Review Plan Evaluation

The Standard Review Plan specifies the ASME Section III, Division 2 Code for concrete vessels and containments as the acceptable code. Only Article CC-3000 of this code is considered for the containment design. VEGP takes some alternative positions to Regulatory Guides 1.18, 1.19, 1.55, 1.94, and 1.136.

The ASME Section III, Division 2 Code was not in effect at the time the VEGP construction permit was issued. As directed by the Nuclear Regulatory Commission (Preliminary Safety Analysis Report paragraph 3.8.1.5.2) and consistent with subsection 3.8.1 of the Safety Evaluation Report, the design of the containment is based on Article CC-3000 of the ASME Code. The design procedures and construction practices delineated in the FSAR ensure that the containment structure will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

Refer to section 1.9 for VEGP positions on regulatory guides.

3.8.2 STEEL CONTAINMENT

The VEGP does not utilize a steel containment. However, the concrete containment penetration sleeve assemblies, American Society of Mechanical Engineers (ASME) Code Class MC components and parts, and ASME Code Class 2 piping penetrations are discussed in this subsection.

3.8.2.1 Description of Penetration Sleeve Assemblies, Class MC Components and Parts, and Class 2 Piping Penetrations

A penetration sleeve assembly is a steel weldment composed of a cylindrical steel sleeve (welded or seamless), stiffening ring plate(s) with or without gusset(s), anchors, and a portion of the thickened liner plate.

The sleeve assemblies are field-welded to the steel liner plate and are embedded in the concrete containment wall. Generally, several inches of sleeve extends into the containment, past the surface of the liner plate backed by concrete and acts as a nozzle for later component, part, or piping attachment.

Sleeve assemblies installed as spares for future use are capped with shop-welded closure plates. (See Type V, drawings 1X4DL4A14 and 2X4DL4A14.) The sleeve assemblies (capped or uncapped) are considered to be parts of the containment liner plate system (paragraph 3.8.1.4) and are not subject to ASME Section III, Division 1, Class MC requirements.

Class MC components or parts of the containment structure are those code-stamped items that function as part of the pressure boundary and are not backed by concrete, and which are attached to the penetration sleeve assemblies during construction of the containment vessel.

The components include the personnel and escape lock assemblies. The parts consist of the equipment hatch assembly, fuel transfer tube housing and bellows assemblies, electrical penetration assemblies, and the isolation valve encapsulation vessel assemblies.

Process piping assemblies are field welded to the penetration sleeve assemblies and are not deemed components or parts of the containment. All piping attached to the containment wall penetration sleeves is ASME III Code Class 2.

3.8.2.1.1 Equipment Hatch and Personnel Lock Assemblies

A 20-ft circular equipment hatch, a 9-ft 10-in. circular personnel lock, and a 5-ft 9-in. circular personnel lock penetrate the containment wall and are welded to the penetration sleeves. Hatch and lock doors are provided with concentric double-sealing gaskets, with provision for leak testing as described in paragraph 6.2.6.2. The equipment hatch and personnel lock penetration details are shown on drawings 1X2D01J017 and 1X2D01J018.

The smaller of the personnel locks is for emergency escape purposes. Each personnel lock has a door at each end and is an ASME Code Class MC component. For the two locks, that portion of the penetration sleeve assembly between the liner plate and the ASME III Code Class MC jurisdictional boundary is considered part of the liner plate and is not subject to ASME Class MC requirements.

During plant operation, the two doors of each personnel lock are interlocked to prevent both being opened simultaneously. Quick-acting equalizing valves connect the personnel lock with the interior and exterior of the containment to equalize pressures. Provision is made to bypass the interlock system during plant cold shutdown to facilitate ingress and egress.

3.8.2.1.2 Fuel Transfer Tube and Housing Bellows Assembly

One fuel transfer tube penetration per containment is provided for refueling. An ASME III Code Class MC one-piece inner pipe acts as the transfer tube, through which both new and spent fuel rod assemblies are passed. The scheme is shown in drawing 1X2D48J006. The tube is fitted with a double-gasketed blind flange at the containment refueling canal end and a gate valve at the fuel transfer canal end within the fuel handling building. This arrangement prevents leakage through the tube.

Three separate penetration sleeve assemblies, joined by four bellows assemblies, act as a transfer tube housing. The penetration sleeve assemblies permit the transfer tube to penetrate the refueling canal wall, the containment shell, and the exterior wall of the fuel handling building, while maintaining a pressure-retaining boundary at each wall. The sleeve assemblies are considered part of the liner plate and are not subject to ASME, Section III, Code Class MC requirements.

The transfer tube is supported by the two outboard bellows assemblies. The refueling canal end employs a gimbals yoke (swivel support) mechanism, while the fuel handling building end uses a pin-connected, sliding support bearing.

The ASME III Code Class MC bellows assemblies allow thermal expansion of the transfer tube and housing and permit differential movement between buildings.

3.8.2.1.3 Electrical Penetration Assemblies

Shop-welded weld neck flanges with caged nuts are provided on the outer ends of all penetration sleeves designed to accept the ASME III Code Class MC bolt-on electrical penetration assemblies. In addition, field-welded slip-on ring plates are installed as flanges on the inner ends of the penetration sleeves to receive bolt-on electrical enclosures. See drawings 1X2D01J019, 1X2D01J020, 1X4DL4A14, 2X2D01J019, and 2X4DL4A14.

Canister-type penetrations are used for medium-voltage power conduction through the containment wall. The stainless steel canisters are factory-sealed units, each consisting of a hollow cylinder with a header plate at each end, through which the conductors are run. One header plate is bolted to the exterior penetration sleeve flange. The header plate mating the flange is provided with two concentric O-ring gaskets, and the annular space between the O-rings is continuously pressurized during plant operation. The canister reposes within the penetration sleeve and is capable of being pressurized through a test connection. All other electrical penetrations are of the modular type, consisting of one header plate, which is bolted to the exterior penetration sleeve flange. Two concentric O-rings gasket the mating pieces, and the annular space is also pressurized during plant operation. Feed-through modules carry the electrical conductors (including fiber optics). The feed-through conductors are mated to the modules by double high-temperature thermoplastic seals with interspace pressure connections. These connections are manifolded so that all feed-through modules for a single penetration may be pressurized at the same time and throughout plant operation. The modules pass through the header plate and are sealed with metal compression fittings.

3.8.2.1.4 Isolation Valve Encapsulation Vessel Assemblies

Two containment spray isolation valve encapsulation vessels and two residual heat removal isolation valve encapsulation vessels are provided for each containment. Although one pair of vessels (one containment spray and one residual heat removal) is present in the auxiliary building and one pair of vessels is present in the fuel handling building, respectively, these vessels are leaktight housings. As such, these vessels and their bellows expansion joints are subject to ASME III Class MC requirements and leaktight tests. However, the encapsulation vessel design is such that they do not communicate directly with the containment atmosphere. The containment spray isolation valve encapsulation vessel is a steel tank, with an inside diameter of 4 ft and a height of approximately 10 ft 8 in. (See drawing 1X4AH04-25.) The residual heat removal isolation valve encapsulation vessel is a similar steel tank, with an inside diameter of 5 ft 6 in. and a height of approximately 11 ft.

3.8.2.1.5 Process Pipe Penetration Assemblies

The closure for process piping to the liner plate is made with a special flued head welded into the piping system and to the penetration sleeve assembly. In certain cases, the piping assemblies include an intermediate sleeve which is inserted between the penetration sleeve assembly and the flued head. In the case of piping carrying hot fluid, the pipe is insulated and the flued head/penetration is designed to prevent excessive concrete temperatures and to prevent excessive heat losses from the fluid. The main high-temperature piping consists of four penetrations for feedwater and four penetrations for main steam, with a maximum operating temperature range between 445°F and 557°F. Thermal insulation is provided on the outside diameter of each line. The penetration assembly is designed to limit the concrete maximum temperature to 200°F.

For typical details of process pipe penetration assemblies used through the containment wall, refer to types I, II, III, IV, and VI on drawings 1X4DL4A14 and 2X4DL4A14. The type IV arrangement allows the piping to be welded directly to the penetration sleeve. Such a juncture is used for the containment purge lines.

3.8.2.2 Applicable Codes, Standards, and Specifications

The following are applicable codes and Regulatory Guides. See section 1.9 for additional discussion of Regulatory Guides.

- ASME Code, Section III, Division 1, Subsections NC, NE, NF. (See paragraph 3.8.2.4 for dates.)
- Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.
- 10 CFR 50, Appendix J.

3.8.2.3 Loads and Load Combinations

The loads and load combinations for the Class MC items conform with Article NE-3000 of the ASME Code and Regulatory Guide1.57. The component supports associated with the fuel transfer tube housing bellows assemblies conform with ASME Code Article NF-3000.

ASME Code, Article NC-3000, Section III, Division 1, Subsection NC applies to Class 2 piping.

3.8.2.4 Design and Analysis Procedures

The procedure for each component, part, or piping assembly is given in the following paragraphs. Since the penetration sleeve assemblies are considered portions of the containment liner plate system, their design and analysis are discussed in paragraph 3.8.1.4. Closure plates for spare penetration sleeves are designed and fabricated to the applicable rules of ASME Code, Section III, Division 1, Code Class MC, without stamping.

3.8.2.4.1 Equipment Hatch and Personnel Lock Assemblies

As Class MC items, these assemblies are designed and analyzed in accordance with Article NE-3000 of the 1974 ASME Code, Subsection NE, through Summer 1975 Addenda.

3.8.2.4.2 Fuel Transfer Tube and Housing Bellows Assemblies

As Class MC parts, these assemblies are designed and analyzed in accordance with Article NE-3000 of the 1974 ASME Code, through Winter 1976 Addenda. The tube supports come under the jurisdiction of ASME Code Subsection NF (same addenda) since these are designated as component supports.

3.8.2.4.3 Electrical Penetration Assemblies

As Class MC parts, these assemblies are designed and analyzed in accordance with Article NE-3000 of the 1977 ASME Code, through Summer 1979 Addenda.

3.8.2.4.4 Isolation Valve Encapsulation Vessel Assemblies

As Class MC parts, these assemblies are designed and analyzed in accordance with Article NE-3000 of the 1974 ASME Code, through Summer 1976 Addenda.

3.8.2.4.5 Process Pipe Penetration Assemblies [a]

Flued heads (forgings) and the basic Type IV arrangement used in penetration assemblies as Class 2 items are designed and analyzed in accordance with Article NC-3000 of the 1977 ASME Code, through Summer 1978 Addenda. The entire emergency sump line penetration assemblies are also Class 2.

3.8.2.5 Structural Acceptance Criteria

The fundamental acceptance criterion for the completed containment is successful completion of the structural integrity test.

The structural acceptance criteria for steel items other than the liner plate and sleeve assemblies include allowable stress values, deformation limits, and factors of safety and are established in accordance with ASME Section III, Division 1, Subsections NC (Class 2 piping) and NE (Class MC components or parts) as applicable. No permanent deformations are allowed under any loading condition.

The steel items, which are an integral part of the containment pressure boundary, are designed to meet minimum leakage rate requirements. The leakage rate shall not exceed the acceptable value indicated in the applicable technical specification.

The allowable peak stresses, both primary and secondary, and buckling criteria, are provided in the NE Subsection.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

A. The materials of the Class MC items listed in paragraph 3.8.2.1 comply with the requirements of ASME Code, Section III, Division 1, Subsection NE, Article NE-2000. The Class 2 piping assemblies comply with Subsection NC, Article NC-2000.

B. The quality control program involving welding procedures, erection tolerances, and nondestructive examination of both shop- and field-fabricated welds is in conformance with Articles NE-4000 and NE-5000 or NC-4000 and NC-5000 of the ASME Code, as applicable.

C. There are no special construction techniques used on the Class MC items or Class 2 piping assemblies.

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[a] Fatigue of the pipe penetration assemblies was evaluated as a TLAA for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in subsection 19.4.5.

3.8.2.7 Testing and Inservice Inspection Requirements

Testing of the Class MC items and Class 2 piping assemblies forming the pressure boundary within the containment conforms with Articles NE-6000 and NC-6000 of the ASME Code, respectively, with exception. Class 2 pneumatic requirements do not satisfy Article NC-6000, as the containment structural integrity test uses 1.15 times the design pressure, instead of an ASME Code value of 1.25. Regulatory Guide 1.18 is discussed in section 1.9.

Periodic testing of those Class MC items or Class 2 piping assemblies listed in paragraph 3.8.2.1 is planned in accordance with 10 CFR 50, Appendix J.

ASME III, Class MC components will be examined in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE except where relief has been authorized by the NRC. ASME III, Class 2 items are examined in accordance with section 6.6.

The 10 CFR 50 Appendix J testing and the ASME Section XI Inservice Inspection Program's IWE inspections are credited as license renewal aging management activities (see subsections 19.2.29 and 19.2.30).

3.8.2.8 Standard Review Plan Evaluation

The loading combinations specified in the design of Class MC components or parts differ from those in the Standard Review Plan.

The specifications for Class MC items require compliance with Article NE-3000 of the ASME Code. Even though the design calculations for some of the Class MC items do not specifically address the Standard Review Plan load combinations, the loads and load combinations conform with Article NE-3000 of the ASME Code and Regulatory Guide 1.57. Refer to section 1.9 for the VEGP position on Regulatory Guide 1.57.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF CONCRETE CONTAINMENT

3.8.3.1 Description of the Internal Structures

The internal structures are those concrete or steel structures inside of (not part of) the containment pressure boundary which support the reactor coolant system (RCS) components and related piping systems and equipment. The concrete structures also provide radiation shielding. The internal structures consist of the primary shield wall, secondary shield and pressurizer compartment walls, refueling canal walls, operating floor, intermediate slabs and platforms, and the polar crane runway girders. The polar crane brackets and equipment hatch hoist brackets are considered liner plate components, as described in paragraphs 3.8.1.1.2 and 3.8.1.6.6.B.

Component supports are those steel members designed to transmit loads from the RCS to the load-carrying building structures.

The RCS component supports are on drawing AX6DD312. Component supports are further described in subsection 5.4.14. A description of each major structure is provided below.

3.8.3.1.1 Reactor Pressure Vessel (RPV) Support System

The RPV is supported by four seats under two hot leg and two cold leg nozzles which are spaced approximately 90° apart in the primary shield wall. Reactor pressure vessel support is designed in such a way as to provide for radial thermal growth of the RCS, including the RPV, but so as to prevent the vessel from lateral and torsional movement during a loss-of-coolant accident (LOCA). The vertical loads are carried by the support seats to the embedded steel weldments under each support, while the radial and tangential loads are carried by the embedded steel weldments in the primary shield wall placed radially and tangentially to the wall. Reactor pressure vessel support seats and the associated embedded weldments are shown in drawings 1X2D48A018 and 1X2D48A019. A description of the primary shield wall and reactor cavity is provided in paragraph 3.8.3.1.4.

3.8.3.1.2 Steam Generator Support System

The steam generator support system is shown on drawing AX6DD312. The steam generator is vertically supported by four steel columns, pinned at both ends and bolted to support pads on the vessel and basemat embeds. (See drawings 1X2D01J008, 1X2D48E003, 1X2D48E004, and 1X2D48E005.) A pipe restraint is provided on the hot leg of the steam generator inlet pipe to prevent the formation of a plastic hinge at the primary shield wall and to limit the break area for a steam generator inlet nozzle break. A lower lateral component support is supplied by bearing blocks and a steel beam which spans the inside of the compartment walls. (See drawing AX6DD312.) The upper lateral component support consists of a bearing ring located near the center of gravity of the steam generator. The bearing ring is in turn restrained by a combination of hydraulic snubbers and a hard stop in the direction of thermal growth and by hard stops in the perpendicular direction. The steam generator is supported such that a main steam line or feedwater line break does not result in a break in the RCS or vice versa.

3.8.3.1.3 Reactor Coolant Pump Support System

The reactor coolant pump component supports consist of three steel columns, pinned at both ends and bolted to support pads on the pump and basemat embeds. Drawing AX6DD312 shows the general arrangement and design features. Horizontal steel tie rods, anchored to the primary and secondary shield walls, are provided for lateral support.

3.8.3.1.4 Primary Shield Wall and Reactor Cavity

The primary shield wall, a quasi-cylindrical, reinforced concrete structure extending from el 169 ft 0 in. to el 194 ft 1 3/4 in., provides a support system for the RPV. The primary shield wall and reactor cavity are illustrated on drawings AX6DD312, AX6DD314, 1X2D48A018, and 1X2D48A019. The primary shield wall is anchored into the basemat of the containment with reinforcing steel. Continuity of reinforcement across the pressure boundary is achieved with B-series cadweld splices welded to both sides of the thickened liner plate. (See drawing AX6DD314.)

Only a small vent area around each RPV nozzle has been provided to limit the flow of steam/water into the void space between the outside of the RPV and the primary shield wall, which therefore limits the differential pressure loadings on the RPV. An annular inspection gallery with eight entry ports is provided in the primary shield wall as an access to nozzle welds for inservice inspection. Coolant pipe penetrations through the primary shield wall are provided with special restraints to limit the postulated break area.

3.8.3.1.5 Secondary Shield Walls

The secondary shield is a series of reinforced concrete walls which, together with the refueling canal walls and primary shield wall, enclose the steam generators. The secondary shield walls are anchored into the basemat of the containment in a manner similar to the primary shield walls to allow for load transfer to the basemat. (See drawing AX6DD314.) Each of the four secondary shield compartments provides support and houses a steam generator, a reactor coolant pump, and nuclear steam supply piping.

3.8.3.1.6 Pressurizer Supports

The pressurizer is supported on a steel ring bearing plate bolted to the flange of the pressurizer support skirt. This ring, in turn, rests on a structural steel frame which is attached to steel embeds in the pressurizer compartment walls. (See drawing AX6DD312.) The pressurizer compartment is a rectangular, reinforced concrete structure built integrally with the secondary shield wall on the outside of the No. 4 loop steam generator compartment. (See drawing AX6DD312.) The pressurizer is also supported laterally at an upper level by four stops projecting from embeds within the pressurizer compartment walls.

3.8.3.1.7 Refueling Canal

The refueling canal is a reinforced concrete structure extending above and to the side of the primary shield wall between el 194 ft 1 3/4 in. (el 179 ft 9 3/4 in. at its extreme low point) and el 220 ft 0 in. It provides for the underwater transfer of fuel assemblies, and for the storage of the reactor internals after removal from the reactor. The entire refueling canal is lined with 1/4-in.-thick stainless steel plate. The refueling canal is shown on drawings AX6DD312, 1X2D48E004, and 1X2D48E005.

3.8.3.1.8 Operating Floor

The operating floor, shown in drawing 1X2D48E005, consists of concrete slabs and steel beams with steel grating. The grating covers the space between the secondary shield wall and the containment shell and covers the reactor coolant pumps hatches. The concrete slabs are supported by the refueling canal walls and the secondary shield walls and by steel columns originating in the fill slab (el 171 ft 9 in.).

3.8.3.1.9 Internal Steel Framing

There are no intermediate floors, but a number of steel access platforms and concrete equipment support areas exist above and below the operating floor. (See drawings 1X2D48E008 and 1X2D48E007.) Steel framing exists at four main lower intermediate elevations to support piping; cable trays; heating, ventilation, and air-conditioning (HVAC) ducts; grating floors; and miscellaneous equipment. Steel framing is also provided at two elevations above the operating floor to support the containment coolers, containment auxiliary coolers, preaccess filtration units, and hydrogen recombiners.

3.8.3.1.10 Reactor Missile Shield

The missile shield is a circular steel plate, approximately 15 ft in diameter, mounted within the integrated head of the reactor vessel. The missile shield is used to prevent any postulated missiles from the reactor vessel head appendages from penetrating other RCS pressure boundaries or the containment structure. In addition to this function, the missile shield also transfers the reactor vessel head load to the lifting rig. The missile shield also provides seismic support for the control rod drive mechanisms (CRDMs). During plant operation, the missile shield can be easily detached from the lifting ring assembly to provide access to the CRDMs.

3.8.3.1.11 Polar Crane Supporting Elements

The polar crane runway box girders, which support the crane rails, are chords traversing the containment circumference and are supported by the crane brackets. (See drawing 1X2D01J022.)

3.8.3.2 <u>Applicable Codes, Standards, and Specifications</u>

The following documents are applicable to the design, materials, fabrication, construction, inspection, and testing of the containment internal structures.

3.8.3.2.1 Codes and Standards

The following codes and standards apply:

- American Concrete Institute (ACI), Building Code Requirements for Reinforced Concrete, Standard 318-71 including the 1974 supplement.
- American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, adopted February 12, 1969, and including supplements 1, 2, and 3.
- American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (hereinafter called the ASME Code), Section III, Division 1, Subsection NF (Component Supports), 1977 ASME Code including the Winter 1977 Addenda.
- American National Standards Institute (ANSI), ANSI N45.2.5-1974, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.

3.8.3.2.2 Regulations

Regulations applicable to the internal structures are included in those listed in paragraph 3.8.1.2.2.

3.8.3.2.3 General Design Criteria

The design of the containment internal structures complies with General Design Criteria 1, 2, 4, 5, and 50 of Appendix A to 10 CFR 50 and 10 CFR 50.55a.

3.8.3.2.4 Nuclear Regulatory Commission (NRC) Regulatory Guides

The following regulatory guides apply:

- Regulatory Guide 1.10, Mechanical (Cadweld) Splices in Reinforcing Bars of Category 1 Concrete Structures.
- Regulatory Guide 1.15, Testing of Reinforcing Bars for Category 1 Concrete Structures.
- Regulatory Guide 1.55, Concrete Placement in Category 1 Structures.
- Regulatory Guide 1.94, Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.
- Regulatory Guide 1.142, Safety-Related Concrete Structures for Nuclear Power Plants.

The conformance to these regulatory guides is discussed in section 1.9.

3.8.3.2.5 Industry Standards

Nationally recognized industry standards, such as American Society of Testing Materials (ASTM), ACI, American Iron and Steel Institute (AISI), and American Welding Society (AWS), are used to specify material properties, testing procedures, fabrication, and construction methods.

3.8.3.3 <u>Loads and Load Combinations</u>

The loads and load combinations are the same as for other Category 1 structures described in paragraph 3.8.4.3 and the associated tables, except for the following modifications:

> Wind loads (W), tornado loads (W_t), blast loads (B), and precipitation loads (N) are not applicable to the design of the internal structures because of the protection provided by the containment shell; therefore, these loading terms have been excluded in the load combinations for the internal structure.

The RCS component supports are in conformance with the ASME Code, Section III, Division 1, Subsection NF, Component Supports.

3.8.3.4 <u>Design and Analysis Procedures</u>

The internal concrete structure is designed by the strength method as specified in the ACI Building Code Requirements for Reinforced Concrete, Standard 318-71, including the 1974 supplement.

The internal structural steel is designed in accordance with the AISC 1969 Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, including supplements 1, 2, and 3. Specific components as described in paragraphs 3.8.3.4.1, 3.8.3.4.2, 3.8.3.4.3, and 3.8.3.4.6 are governed by the ASME Code, Subsection NF, instead of the AISC specification.

Two three-dimensional finite element mathematical models are created for computer analysis of the major concrete internal structures. One model comprises only the primary shield wall. The second encompasses the secondary shield walls, pressurizer compartment walls, and refueling canal walls.

The procedures described in section 3.7 are employed to obtain the seismic accelerations at various levels of the internal structures. The generation of the magnitudes of the accident loads due to pipe break is discussed in Containment Functional Design, subsection 6.2.1. The design for the effects of postulated pipe breaks is performed as specified in subsection 3.6.2.

Nuclear steam supply system (NSSS) support loads from the steam generator and reactor coolant pump columns and the crossover leg horizontal run restraint are transmitted into the basemat by embedded steel weldments. Anchor bolt sleeves which cross the pressure boundary form the continuity. (See drawing 1X2D01J008.) Models and methods of analysis for the RCS component supports are described in paragraph 3.9.N.1.4.4.

3.8.3.4.1 Reactor Pressure Vessel Support System [a]

The reactor pressure vessel supports embedded within the primary shield wall are procured in accordance with ASME Code, Section III, Division 1, Subsection NF; however, since they are outside the ASME jurisdictional boundary, their design follows AISC specifications.

3.8.3.4.2 Steam Generator Support System

The embeds for the steam generator supports are designed and procured in the manner described in paragraph 3.8.3.4.1. Lateral support loads on the concrete walls are discussed in paragraph 3.8.3.4.5.

3.8.3.4.3 Reactor Coolant Pump Support System

The embeds for the reactor coolant pumps are designed and procured in the manner described in paragraph 3.8.3.4.1.

3.8.3.4.4 Primary Shield Wall and Reactor Cavity

The primary shield wall is modeled separately using eight-node brick finite elements and is analyzed using the BSAP computer program described in appendix 3B. Seismic loads are derived from the acceleration values obtained using the procedures specified in section 3.7; accident loads are calculated from data described in subsection 6.2.1. The load combinations are discussed in paragraph 3.8.3.3. The loads are applied to a finite element model similar to that described in paragraph 3.8.3.4.5. During normal plant operation, a thermal gradient across the wall is generated by the heat of the reactor and the attenuation of gamma and neutron radiation originating from the reactor core. An insulation and cooling system is provided to reduce the severity of this gradient by limiting the temperature at the inside concrete face of the

[a] Fatigue of the reactor vessel supports was evaluated as a TLAA for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in paragraph 19.4.2.4.

wall to 150°F. Localized areas, such as around a penetration, are allowed to have increased operating temperatures (not to exceed 200°F).

Should a LOCA occur, the steady-state temperature distribution across the primary shield wall, acting in conjunction with the accident pressure differential, is considered in the primary shield wall design. Design bases for differential pressure analyses within the interior subcompartments are discussed in paragraph 6.2.1.2. The peak pressure differential is of short duration, since equalization immediately begins to take place through its passages into the steam generator compartments and the free volume of the containment. The low thermal conductivity of the primary shield wall prevents occurrence of rapid changes in the temperature profile. Changes in the temperature distribution start from the steady-state temperature profile and proceed towards a more uniform distribution, due to heating on both wall faces. As such, the initial temperature effects on the primary shield wall due to a LOCA are considered negligible and the operating condition temperature differential is used for design.

3.8.3.4.5 Secondary Shield Walls

The secondary shield wall, in conjunction with the pressurizer compartment and the refueling canal, is modeled using both triangular and rectangular plate finite elements and beam, brick, and boundary finite elements and is analyzed using the BSAP computer program. Forces on the finite elements are introduced in the finite element model either as uniformly distributed or nodal loads. Seismic load input is a constant acceleration. Jet impingement and accident pressures occur as uniformly distributed surface loads while pipe reaction and pipe whip accident loads are nodal loads.

The secondary shield walls are mainly subjected to the lateral loadings of the steam generators, reactor coolant pumps, pressurizer, and compartment pressures due to accident conditions.

3.8.3.4.6 Pressurizer Support

The steel frame and its anchorage into the concrete walls are analyzed by conventional methods and designed in accordance with the AISC specification.

The bolts attaching the pressurizer base to the steel frame conform to the ASME Code, Section III, Division 1, Subsection NF, for material, design, and fabrication.

The finite element model for the pressurizer compartment walls is incorporated in the secondary shield wall finite element model.

3.8.3.4.7 Refueling Canal

For the refueling condition, the walls are designed for the hydrostatic head due to the maximum depth of water in the canal during refueling, including the hydrodynamic pressure effects due to the postulated seismic events, as well as a 140°F maximum water temperature. The steam generator compartment pressure loads due to postulated pipe rupture and hydrostatic head are not considered to occur simultaneously.

The NRC publication TID 7024(1) is used for computing hydro-dynamic loads imposed on the refueling canal walls by sloshing water during seismic events.

3.8.3.4.8 Operating Floor

Concrete floor slabs are analyzed by conventional beam and slab methods and designed in accordance with the ACI Code. For analysis and design of the operating floor structural steel, see paragraph 3.8.3.4.9.

3.8.3.4.9 Intermediate Slabs and Platforms

Structural steel in the containment internals is modeled using beam and truss finite elements and is analyzed using the BSAP computer program mentioned in appendix 3B. The steel is designed in accordance with the AISC specification.

Internal slabs are analyzed by conventional methods and designed in accordance with the ACI Code.

3.8.3.4.10 Polar Crane Runway Girders

The runway box girders are designed to AISC specifications. The runway brackets are described in paragraph 3.8.1.6.6.B.

The polar crane seismic loading is based on the seismic response of the containment at the crane support level.

The design of the polar crane is described in subsection 9.1.5.

3.8.3.5 Structural Acceptance Criteria

The structural acceptance criteria for the concrete and steel internal structures are described in paragraph 3.8.4.5.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Materials

The following basic materials are used in the construction of the internal structures:

A. Concrete f'_c = 5000 psi (91-day strength with pozzolans; 28-day strength without pozzolans)

B. Concrete ingredients, reinforcing steel, cadweld splices, structural and miscellaneous steel, bolts, anchors, unistruts, and shear studs all follow those applicable specifications listed in paragraph 3.8.1.6.

   Superplasticizer admixtures are added to the concrete mixes used in congested areas to facilitate placement and prevent the formation of voids around reinforcement. These admixtures do not affect concrete quality and require an air content range of 1.5 to 4.5 percent. Other air-entrained concrete has a 3- to 6-percent air content range.

C. The refueling canal is lined with welded 1/4-in.-thick stainless steel plate. The liner plate, attachments thereto (exposed on the water side), and exposed insert plates are stainless steel material conforming to ASTM A 240, Type 304L. Exposed mechanical fasteners are also fabricated from equivalent corrosion-resistant stainless steel materials.

D. NSSS embedded items above the basemat level conform to SA-537, Class 1.

E. SA-516, Grade 70, unless otherwise noted on the design drawings, is the material used to fabricate the crossover leg horizontal run restraint steel weldments.

F. The material and quality control requirements for concrete comply with ACI 318-71. Structural steel is in conformance with the 1969 AISC specification.

G. The steel linear supports of the RCS are in compliance with Subsection NF of the ASME Code, Section III, Division 1.

H. The quality control for the internal structures is per Regulatory Guide 1.55, Concrete Placement in Category 1 Structures.

I. Structural steel is A36 material. A325 or A490 bolts are used for structural column or beam connections; A36 or A193 material is used for column anchor bolts.

3.8.3.6.2 Quality Control

A. Seam and plug welds of the stainless steel liner plate are tested by the vacuum box method or the helium mass spectrometer method. Where this is not practical, the liquid penetrant test is employed. Other nondestructive examination is in accordance with that described in paragraph 3.8.1.6.8.

B. Construction tolerances for the internal structures and reinforcing steel tolerances are discussed in paragraph 3.8.1.6.8.

Tolerances applicable to the refueling canal are as follows:

1. Location of embedded backing bars: ±1/2 in. of horizontal and vertical design locations.
2. Flatness of concrete wall surfaces: 3/8 in. measured in 5 ft in any direction.
3. Plumbness of concrete walls: 1/4 in. in 10 ft.
4. The extent of compliance with ANSI N45.2.5-1974 is as described in paragraph 1.9.94.2.

3.8.3.6.3 Special Construction Techniques

There are no special construction techniques used in the construction of the internal structures.

3.8.3.7 Testing and Inservice Inspection Requirements

A formal program of testing and inservice surveillance is not required for the internal structures. Tests and inspections for the RCS component supports are discussed in subsection 5.4.14.

For the period of extended operation, periodic inspections of the Category 1 containment internal structures by the Structural Monitoring Program are required license renewal aging management program activities (see subsection 19.2.32).

3.8.3.8 Standard Review Plan Evaluation

The Standard Review Plan specifies ACI 349, augmented by Regulatory Guide 1.142, as the acceptable code for design of concrete structures. The Standard Review Plan also specifies the load combinations that would result from the use of ACI 349, as modified by Regulatory Guide 1.142. VEGP design is based on ACI 318-71 and is in conformance with the load combinations specified in the Standard Review Plan. VEGP takes some alternatives to Regulatory Guides 1.55, 1.94, and 1.142.

Code ACI 349 was not in effect at the time the VEGP construction permit was issued. In accordance with the Preliminary Safety Analysis Report commitment, ACI 318-71 was used. The differences between ACI 349 and ACI 318 are minor except for the load combination equations which, in the case of VEGP, are in conformance with the Standard Review Plan. Thus, the design procedures and construction practices delineated in the FSAR ensure that the structure will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

Refer to section 1.9 for VEGP positions on regulatory guides.

3.8.3.9 Reference

1. "Nuclear Reactors and Earthquakes," NRC Report TID-7024, August 1963.

3.8.4 OTHER SEISMIC CATEGORY 1 STRUCTURES

Category 1 structures, other than the two containments and their internal structures, include the auxiliary building, fuel handling building, control building, two diesel generator buildings, two auxiliary feedwater pumphouses, four nuclear service cooling water (NSCW) towers and valve houses, two refueling water storage tanks, two reactor makeup water storage tanks, four condensate storage tanks, two diesel fuel storage tank pumphouses, and Category 1 tunnels.

3.8.4.1 Description of the Structures

3.8.4.1.1 Equipment Building

The equipment building is composed of portions of the control building and fuel handling building, as shown in drawings 1X4DE313, 1X4DE314, 1X4DE320, and 1X4DE322.

3.8.4.1.2 Auxiliary Building

The auxiliary building is a seven-story reinforced concrete building common to the two-unit plant. It is located south of the fuel handling building. Three stories are above grade; four are subterranean. There are two penetration areas, one on the south side of each containment. This building houses the chemical and volume control system (CVCS); emergency core cooling system (ECCS); residual heat removal (RHR) system; heating, ventilation, and air-conditioning (HVAC) facilities; and other associated equipment. There is a railroad access at grade and crane facilities for shipping and receiving of new and spent fuel. Tendon access shafts No. 2, which provide access to the containment tendon galleries and buttresses for each unit, are formed by portions of the auxiliary building and fuel handling building. Drawings 1X4DE315, 1X4DE318, 1X4DE321, 1X4DE323, 1X4DE324, and 1X4DE325 show the general layout and geometric description of the building. Directly south of each containment is a main steam isolation valve (MSIV) room.

3.8.4.1.3 Fuel Handling Building

The fuel handling building is a five-story, boxlike, reinforced concrete structure common to the two-unit plant. It is completely surrounded by other Category 1 buildings. It is south of the control building, north of the auxiliary building, and between the two containment structures.

The building has a center section which consists of two spent fuel storage pools, a new fuel storage area, cask loading pit, fuel transfer canals, and cask washdown area. The height of this section extends approximately 69 ft above grade and 40 ft below grade. On the east and west sides of the center section are penetration areas which provide access to the containment structures. These areas extend from 60 ft below grade to grade level.

The spent fuel storage pools have concrete walls and floors and are lined on the inside surfaces with 1/4-in.-thick stainless steel plate for leak prevention.

The new fuel storage area is a separate reinforced concrete pit providing temporary dry storage for the new fuel assemblies. An equipment and cask cleaning area is located adjacent to the spent fuel pools and new fuel pit.

The fuel transfer canal system is provided for transport of the new and spent fuel assemblies between the fuel handling building and the two containment buildings. A flexible transfer tube is provided for transfer of the fuel assemblies between the fuel handling building and each of the containment buildings. A mechanical fuel transfer system has been provided which is capable of moving a single fuel assembly in a horizontal orientation along the transfer canal and through the transfer tube.

The fuel handling building contains a bridge crane at el 220 ft which travels the length of the building (east-west), for handling of the spent fuel assemblies. A bridge crane runs north-south in the center of the building for lifting and transporting spent fuel casks. These cranes are discussed at length in section 9.1.

Tendon access shafts No. 2, which provide access to the containment building tendon galleries and buttresses for each unit, are formed by portions of the auxiliary and fuel handling buildings.

Drawings 1X4DE312, 1X4DE313, 1X4DE314, 1X4DE317, 1X4DE320, and 1X4DE322 show the general layout of the fuel handling building.

3.8.4.1.4 Control Building

The control building is a six-story, deeply embedded, reinforced concrete structure common to the two-unit plant. It is situated north of and adjacent to the fuel handling building and the two containment buildings. It is supported on a mat foundation 40 ft below grade. The boxlike center section has three upper levels extending to 60 ft above grade. A partial fourth level extends an additional 20 ft. Penetration areas east and west of the center section provide access to the two containment buildings. These are the primary areas for routing of electrical and control systems cable into the containment. Directly north of each containment building is an MSIV room which extends 40 ft above grade.

The floor at grade is principally occupied by the control room and technical support center. The floors immediately above and below grade house the cable spreading rooms. The lowest level houses switchgear and HVAC equipment. The third and fourth floors mainly contain HVAC equipment, while the fourth floor is primarily occupied by nonsafety-related components.

Access shafts No. 3, providing access to the containment tendon gallery and one buttress for each unit, are formed by portions of the control building.

Drawings 1X4DE312, 1X4DE313, 1X4DE314, 1X4DE317, 1X4DE320, and 1X4DE322 show the general layout of the control building.

3.8.4.1.5 Diesel Generator Buildings

The diesel generator buildings (one for each unit) are two-story rectangular, reinforced concrete structures approximately 92 ft by 114 ft in plan. They are approximately 60 ft in height and are supported on basemat foundations at grade. Each building is divided into two isolated bays. The first floor of each bay houses a diesel generator and a 5-ton bridge crane for equipment removal. The second floor houses air handling, exhaust, and silencing equipment. Drawing 1X4DE327 shows the general layout of the diesel generator buildings.

3.8.4.1.6 Auxiliary Feedwater Pumphouses

The auxiliary feedwater pumphouses (one for each unit) are one- story, rectangular, reinforced concrete structures approximately 40 ft by 74 ft in plan. They extend 22 ft above grade and are supported on basemat foundations 4 ft below grade. Four interior walls provide separation for the steam and electric driven pumps. Roof hatches are present for pump access. Drawing 1X4DE316 shows the general layout of the pumphouses.

3.8.4.1.7 NSCW Towers and Valve Houses

The NSCW cooling towers are mechanical draft cylindrical towers. They function as water storage reservoirs as well as a mechanical heat removal system. There are four NSCW cooling towers (two per unit). They are deeply embedded and are identical. They are separated by a minimum of 77 ft at the closest point. The NSCW cooling towers are reinforced concrete structures. Each consists of a 112-ft-high cylindrical shell, a 9-ft-thick mat foundation, and a 2-ft-thick flat roof deck. The shell wall is 3 ft thick above el 155 ft 5 1/2 in. and 5 ft thick below. The elevation of the top of the basemat is 137 ft 0 in. The top of the roof deck is at el 250 ft 11 in. Grade elevation is approximately 218 ft 0 in. The interior diameter of the cylindrical shell is 88 ft. Each tower has a nominal storage capacity of 3,670,000 gal and is normally filled with water to el 217 ft 9 in.

Large rectangular openings in the cylindrical shell are provided for air intake. The openings are 12 ft high with a bottom el of 218 ft 3 in. The openings are spaced around the entire circumference of each tower. The width varies, with an average of about 8 ft.

Inside the upper part of each tower, two perpendicular cross walls divide the towers into four separate cells. The cross walls are 2 ft 3 in. thick and extend from el 209 ft 9 in. to the roof. The cross walls support the roof deck and are designed as deep beams. At the top of each of the four cells, a large circular opening has been provided in the roof slab for air discharge.

Two levels of concrete beam grids are provided within each of the four cells. They are supported by the cross walls and cylindrical shell and are located at el 235 ft 8 in. and 242 ft 11 in. They support fill, spray distribution piping, eliminators, and other internals necessary to distribute and cool incoming water and to reduce vapor loss from the towers.

Surrounding each of the four roof slab openings on each tower and extending 14 ft above the roof deck are concrete fan stacks. The fan blades are centered in the fan stacks. The diameter of the fan stacks varies in order to enhance fan and airflow performance. The fan motors are mounted on the roof deck outside of the fan stacks and are protected with concrete housing structures.

A concrete splash ring (slab and wall) surrounds each of the towers at grade. The wall is located 12 ft from the exterior of the cylindrical shell and extends to a minimum of 12 in. above the top of the air intake openings.

A concrete buttress is provided on one side of each tower. It extends 10 ft 10 1/2 in. out from the 3-ft cylinder wall and is 38 ft 2 in. wide. It runs from el 218 ft 3 in. down to the basemat. The buttress encloses the four pump wells and supports the pumps.

Drawing 1X2D05E001 shows the general layout of the NSCW towers.

Each NSCW tower has an adjoining one-story, reinforced concrete, boxlike valve house. The valve house extends 30 ft above grade and is founded on a 6-ft-thick basemat foundation 14 ft below grade. The valve house also consists of two reinforced concrete missile shield slabs at el 248 ft 6 in. and el 241 ft 6 in. which protect piping entering the cooling tower from the valve house and the NSCW pumps mounted on the tower buttress. In plan, the wall patterns are irregular and follow the curvature of the cooling tower peripheries. The floor is split-level, allowing for pits to house process piping, valves, and equipment. Drawing 1X2D05E001 shows the general layout of the NSCW towers and the valve houses.

3.8.4.1.8 Category 1 Water Storage Tanks

The storage tanks for refueling water (one for each unit), reactor makeup water (one for each unit), and condensate water (two for each unit) are cylindrical, reinforced concrete shells. They are each covered with concrete roofs. They are supported on basemat foundations at grade. The condensate storage tanks are paired and share a common basemat. The tank capacities and dimensions are as follows:

Tank	Nominal Capacity	Inside Diameter	Height
Refueling water	715,500 gal	48 ft	60 ft
Reactor makeup water	165,000 gal	33 ft	40 ft
Condensate water	480,000 gal	44 ft	54 ft

The water side of the wall and floor of each of the tanks is lined with continuous stainless steel liner plate to ensure leaktight integrity. The refueling water, reactor makeup water and the condensate storage tanks have perimeter dikes for retention of spilled water. Each pair of condensate water tanks have a common reinforced concrete valve house which provides missile protection for piping and equipment. The reactor makeup water and condensate water storage tanks are fitted with floating diaphragms which minimize oxygen absorption. Drawings 1X2D28A001, 1X2D29A001, and 1X2D30A001 show the general layout of the Category 1 tanks.

3.8.4.1.9 Diesel Fuel Storage Tank Pumphouses

The diesel fuel storage tank pumphouses (two for each unit) shelter pumps and valves for the buried diesel fuel storage tanks. The reinforced concrete pumphouses straddle the tanks and extend 3 ft above grade except for a common entry between each pair of pumphouses, which extends 14 ft above grade. Each pumphouse foundation consists of wall strip footings. The pumphouses are boxlike with work space levels above the top of the tanks. Drawings 1X4DE327 and 1X4DE330 show the general layout.

3.8.4.1.10 Category 1 Tunnels

The Category 1 tunnels are boxlike, reinforced concrete structures, either completely buried or with their roofs exposed at, or near, grade level. The tunnels house piping and electrical trays. Drawings 1X2D44A005 and AX2D11A048 show the partial layout and sections of the tunnels.

3.8.4.1.11 Masonry Walls

There are a few nonseismic masonry walls in Category 1 structures at VEGP. Three-hour-rated fire barriers (nonseismic) are located in seismic category 1 areas. Certain access openings are sealed with concrete units for radiation shielding and maintenance purposes. These concrete units and fire barriers are held captively in place by structural elements such as steel angles, concrete reinforcement bars, or steel beams. Seismic category 1 components are not supported from, on, or by these barriers.

3.8.4.2 Applicable Codes, Standards, and Specifications

The following documents are applicable to the design, materials, fabrication, construction, inspection, and testing.

3.8.4.2.1 Codes and Standards

The following codes and standards apply:

- American Concrete Institute (ACI), Building Code Requirements for Reinforced Concrete, ACI 318-71, including 1974 Supplement.
- American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, adopted February 12, 1969, and Supplements Nos. 1, 2, and 3.

3.8.4.2.2 Regulations

The following regulation applies:

- 10 CFR 50.

3.8.4.2.3 General Design Criteria (GDC)

The following criteria apply:

- GDC 1, 2, 4, and 5 of Appendix A, 10 CFR 50.

3.8.4.2.4 Nuclear Regulatory Commission (NRC) Regulatory Guides

Conformance to the following Regulatory Guides is discussed in section 1.9:

- 1.10 Mechanical (Cadweld) Splices in Reinforced Bars of Category 1 Concrete Containment Structures.
- 1.15 Testing of Reinforcing Bars for Category 1 Concrete Structures.
- 1.55 Concrete Placement in Category 1 Structures.
- 1.91 Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants.
- 1.194 Quality Assurance Requirements for Installation, Inspection, and Structural Steel During the Construction Phase of Nuclear Power Plants.
- 1.142 Safety-Related Concrete Structures for Nuclear Power Plants Other than Reactor Vessels and Containments.
- 1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.

3.8.4.2.5 Industry Standards

Nationally recognized industry standards, such as American Society for Testing and Materials, American Concrete Institute, and American Iron and Steel Institute, are used to specify material properties, testing procedures, fabrication, and construction methods.

3.8.4.3 <u>Loads and Load Combinations</u>

3.8.4.3.1 Loads

The loads considered are normal loads, severe environmental loads, extreme environmental loads, abnormal loads, and potential site proximity loads.

3.8.4.3.1.1 Normal Loads. Normal loads are those loads to be encountered, as specified, during initial load construction stages, during test conditions, and later, during normal plant operation and shutdown. They include the following:

D Dead loads or their related internal moments and forces, including any permanent loads except prestressing forces, including hydrostatic loads.

L Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence, such as: floor area occupancy loads, moveable equipment loads, equipment laydown loads, nuclear fuel casks and fuel cask equipment loads, vehicular traffic loads, railroad equipment loads, and lateral soil pressures. Live load intensity varies depending upon the status of plant operation (i.e, prefuel load condition versus post-fuel load condition), the level of functional activity for a specific area, and the type of structural element. The prefuel load maximum design live loads for floor area occupancy include such things as temporary storage during construction of bulk materials and equipment, and maximum occupancy due to workers, tool boxes, scaffolding, and localized cribbing. The post-fuel load maximum anticipated live loads for floor area occupancy, pertaining to loading combinations including earthquake generated loads, are 100 percent of the live load occupancy levels expected to be present during plant operation, and are considered to be not less than 25 percent of the prefuel load maximum design live load for floor area occupancy.

T_o Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

R_o Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state conditions.

3.8.4.3.1.2 Severe Environmental Loads. Severe environmental loads are those loads to be infrequently encountered during plant life. Included in this category are:

E Loads generated by the operating basis earthquake (OBE). These include the associated hydrodynamic and dynamic incremental soil pressures.

W Loads generated by the design wind specified for the plant.

3.8.4.3.1.3 Extreme Environmental Loads. Extreme environmental loads are those loads which are credible but are highly improbable. They include:

E' Loads generated by the safe shutdown earthquake (SSE). These include the associated hydrodynamic and dynamic incremental soil pressures.

W_t Loads generated by the design tornado specified for the plant. Tornado load effects consist of tornado wind load (W_w), tornado differential pressure load (W_p), and tornado missile load (W_m). In general, the controlling load combination for tornado load effects in the design of structures or elements is:

$W_t = W_w + 0.5\ W_p + W_m$

N Loads generated by the probable maximum precipitation.

B Loads generated by postulated blast along transportation routes.

3.8.4.3.1.4 Abnormal Loads. Abnormal loads are those loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

P_a Pressure load within or across a compartment and/or building, generated by the postulated break.

T_a Thermal loads generated by the postulated break and including T_o.

R_a Pipe and equipment reactions under thermal conditions generated by the postulated break and including R_o.

Y_r Load on a structure generated by the reaction on a ruptured high-energy pipe during the postulated event.

Y_j Load on a structure generated by the jet impingement from a ruptured high-energy pipe during the postulated break.

Y_m Load on a structure or pipe restraint resulting from the impact of a ruptured high-energy pipe during the postulated event.

3.8.4.3.1.5 Site Proximity Loads. Potential loads induced by floods, explosions, and aircraft hazards have been discussed in paragraphs 3.4.1.1, 3.5.1.5, and 3.5.1.6. Loads due to projectiles and missiles from nearby missile bases, in addition to those discussed, are not considered credible in the design of Category 1 structures.

3.8.4.3.1.6 Dynamic Effects of Loads. The dynamic effects from the impulsive and impactive loads caused by P_a, T_a, R_a, Y_r, Y_j, Y_m, B, and tornado missiles are considered by one of the following methods:

- Applying an appropriate dynamic load factor to the peak value of the transient load.
- Using impulse, momentum, and energy balance techniques.
- Performing a time-history dynamic analysis.

Elastoplastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Dynamic increase factors appropriate for the strain rates involved may be applied to static material strengths of steel and concrete for purposes for determining section strength.

3.8.4.3.2 Load Combinations

3.8.4.3.2.1 Steel Structures. The steel structures and components are designed in accordance with elastic working stress design methods of Part 1 of the AISC specification, using the load combinations specifications in table 3.8.4-1.

If plastic design methods are used, the design is performed in accordance with the plastic design methods of Part 2 of the AISC specification, using the load combinations specified in table 3.8.4-2.

3.8.4.3.2.2 Concrete Structures. The concrete structures and components are designed in accordance with the strength design methods of ACI 318, using the load combinations specified in table 3.8.4-3.

3.8.4.4 Design and Analysis Procedures

The procedures used for the analysis and design of Category 1 structures (other than the containment and its internal structures) are described in this paragraph. The bases of design for the tornado, pipe break, and seismic effects are discussed in sections 3.5, 3.6, and 3.7, respectively. Evaluations for the postulated explosions are discussed in subsection 2.2.3. The foundation design is described in subsection 3.8.5. Each of these structures is founded on its own foundation, separated by a seismic gap to prevent interaction between the structures under seismic events. Computer programs used in the analysis and design of structures are described in appendix 3B.

The seismic design of the radwaste transfer building, radwaste transfer tunnel, and radwaste solidification building, for which Regulatory Guide 1.143 is applicable, are described in section 3.7.

3.8.4.4.1 Basic Structural Systems

Category 1 structures can be classified as three basic types of structures:

A. Shear Wall Structures

Included in this group are the auxiliary building, fuel handling building, control building, diesel generator buildings, auxiliary feedwater pumphouses, and NSCW valve houses.

B. Shell Structures

These include the NSCW towers, refueling water storage tanks (RWSTs), reactor makeup water tanks, and condensate storage tanks.

C. Rigid Box Structures

This system is used for buried structures such as the tunnels and diesel fuel oil storage tank pumphouses.

3.8.4.4.1.1 Shear Wall Structures. Concrete shear wall structures consist of vertical shear/bearing walls and horizontal slabs. Lateral loads are distributed to shear walls based on principles of relative rigidity; the more rigid shear walls resist proportionally more lateral load. Shear wall rigidities may be determined by hand calculations using traditional pair analysis assuming fixed-fixed or pinned-fixed boundary conditions, or shear distribution may be determined by a finite element model analysis. Vertical axial loading of the shear walls is determined based on slab tributary area carried by the wall. Additionally, out-of-plane bending and shear loads, such as seismic, lateral earth pressure, hydrostatic, hydrodynamic, and wind pressure, are evaluated and considered in the shear wall design. Slabs are designed for vertical and in-plane horizontal diaphragm loading.

The load distribution analysis and design of shear walls and slabs are in accordance with ACI 318 code.

3.8.4.4.1.2 Shell Structures. Shells are curved plates which, in circular structures, resist out-of-plane loading through axial (circumferential and meridional) forces. Bending stresses are important primarily at boundaries and discontinuities. The Category 1 tanks and NSCW towers fit well into this type of analysis.

The three Category 1 water tanks are structurally similar. Each has a foundation basemat, cylindrical tank walls, and a relatively flat circular roof. Taking advantage of symmetry, the tanks are analyzed, by hand, as simple axisymmetric cylindrical shells. Bending stresses are analyzed at the wall- to-roof and wall-to-basemat boundaries. Concrete design is done in accordance with the ACI 318 Code. Hydrodynamic forces are analyzed as outlined in reference 1.

The primary structural components of the NSCW towers are the basemat, cylindrical shell, fan deck, and cross walls. Axisymmetric shell analysis is not realistic due to the many discontinuities and irregularities in the structure (large openings, intermediate beam-grid systems, cross walls, varying cylinder wall thickness, and irregular fan deck geometry). The NSCW towers are therefore analyzed by three-dimensional finite element modeling. The BSAP computer program is used with beam, brick, and shell elements. A static analysis is performed using seismic loads developed per section 3.7, hydrodynamic loads calculated as outlined in reference 1, lateral earth pressure loads, and other applicable loads. The concrete design is performed using the computer program OPTCON.

A second, more detailed, finite element model is used to analyze the fan deck and fan stacks. The BSAP computer program is used with beam and LCCT9 plate elements. Static and dynamic analyses are performed. The fan deck is treated as a structural subsystem of the NSCW towers. The floor response spectra at the fan deck level are used as the seismic input for the dynamic analysis. All concrete design is done manually, ACI Code 381.

3.8.4.4.1.3 Rigid Box Structures. Buried structures such as the tunnels and diesel fuel storage tank pumphouses are rigid box systems. They are primarily designed to resist bending and shearing stresses resulting from out-of-plane forces. Static and dynamic soil pressures comprise a large portion of the loading. Analysis is done manually, using conventional moment distribution techniques. Concrete design is performed in accordance with ACI 318 Code.

3.8.4.5 Structural Acceptance Criteria

The analysis and design of concrete structures are in accordance with ACI 318-71 (with 1974 Supplement). ACI 349-76 is not employed, except for limited cases after the construction phase, since it was not published at the VEGP construction permit stage (June 1974). The structural steel design is based on the AISC 1969 specification and the three subsequent supplements. Welding of structural steel, miscellaneous steel, raceway supports, and HVAC duct supports is performed in accordance with the Structural Welding Code, AWS D1.1, 1975 or later edition. The AWS D1.1 Code edition invoked is the edition in effect at the purchase order date for material specifications and the initial issue date for construction specifications, unless otherwise stated in the specifications. Changes in the code edition by construction or the supplier require engineering approval. The VEGP acceptance criteria for visual inspection of structural welds is described in section 1.9.94.2. The load combinations and the strength limits are described in paragraph 3.8.4.3 and the associated tables.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Materials, quality control, and construction techniques follow those described in paragraph 3.8.3.6.

The minimum design compressive strengths of the Category 1 reinforced concrete structures, other than the containment and its internal structures, are provided below. These are 91-day strengths when pozzolans are used and 28-day strengths without pozzolans.

Structure	f_C'
Auxiliary building	5000 psi
All others	4000 psi

Superplasticizer admixtures are added to the concrete mixes used in congested areas to improve flow and prevent the formation of voids around reinforcement. These admixtures do not affect concrete quality; they require an air content range of 1.5 to 4.5 percent. Other air-entrained concrete has a 3- to 6-percent air content range.

In selected areas (e.g., exhaust silencer pads, exhaust stack missile barriers, and roof slabs in locations of barriers, located in diesel generator building), heat resistance concrete is used. The compressive strength for this concrete is 6000 psi at 24 h.

3.8.4.7 Testing and Inservice Inspection Requirements

There are no testing or inservice inspection requirements for the Category 1 structures except for the containment structure.

For the Category 1 structures required to be maintained as Category 1, periodic inspections performed under the Structural Monitoring Program are required license renewal aging management program activities for the period of extended operation (see subsection 19.2.32). The noncivil features of the outdoor tanks (e.g., fluid-retaining) are age-managed separately from these structural inspection programs as part of the associated fluid systems.

3.8.4.8 Standard Review Plan Evaluation

The Standard Review Plan specifies ACI 349, augmented by Regulatory Guide 1.142, as the acceptable code for design of concrete structures. The Standard Review Plan also specifies the load combinations that would result from the use of ACI 349, as modified by Regulatory Guide 1.142. VEGP design is based on ACI 318-71 and is in conformance with the load combinations specified in the Standard Review Plan. VEGP also takes some alternative positions to Regulatory Guide 1.143.

Code ACI 349 was not in effect at the time the VEGP construction permit was issued. In accordance with the Preliminary Safety Analysis Report commitment, ACI 318-71 was used. The differences between ACI 349 and ACI 318 are minor except for the load combination equations which, in the case of VEGP, are in conformance with the Standard Review Plan. Thus, the design procedures and construction practices delineated in the FSAR ensure that the structure will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

Refer to section 1.9 for VEGP positions on regulatory guides.

3.8.4.9 Reference

1. "Nuclear Reactors and Earthquakes," NRC Report TID-7024, August 1963.

3.8.5 FOUNDATIONS

3.8.5.1 Description of the Foundations

All Category 1 structures have reinforced concrete foundations. The majority employ thick basemats. Only the diesel fuel storage tank pumphouse and Category 1 tunnels utilize different footing types.

Major adjoining buildings are structurally separated from neighboring buildings by a seismic separation gap which typically is 5.5 in. This separation provides ample space to prevent interaction between the buildings during a seismic event. A typical seismic separation joint incorporating a waterstop is shown in drawing AX2D94V031.

Resistance to sliding of the massive concrete basemat structures is provided by passive soil pressure and soil friction. This is adequate to provide the required factor of safety against lateral movement under the most stringent loading conditions.

For ease of construction, Category 1 structures are built on mud mats. The mud mats are of lean, nonstructural concrete of approximately 6-in. thickness and rest upon the load-bearing soil. Waterproofing treatment is employed as described in paragraph 3.4.1.1.1. The waterproofing material is applied directly to the mud mats prior to placing the foundations. Vertical surfaces are waterproofed on the earth side prior to backfilling.

The computer programs used in foundation analyses, either BSAP or STRUDL-II, are described in appendix 3B.

3.8.5.1.1 Containment

The containment foundation is a mat and is described in subsection 3.8.1.

3.8.5.1.2 Containment Internal Structures

The containment internal structures are supported by the containment basemat foundation, which is described in subsection 3.8.1.

3.8.5.1.3 Auxiliary Building

This structure is founded on a 10-ft-thick mat, continuous over the plan of the building.

3.8.5.1.4 Fuel Handling Building

The fuel handling building foundation consists of three mats, each 6 ft. thick. The two wing area mats are placed at lower elevations than the central mat. The central mat, at its top, is at el 179 ft 0.5 in. Portions of the Category 1 tunnels are attached to the underside of the central mat.

3.8.5.1.5 Control Building

The control building foundation mat is 7 ft thick and is similar to the auxiliary building basemat. Portions of the Category 1 tunnels are attached to the underside of the basemat.

3.8.5.1.6 Diesel Generator Building

This structure is supported by a 9-ft-thick mat. The diesel generators rest on pads which are integral with the basemat.

3.8.5.1.7 Auxiliary Feedwater Pumphouse

This structure is supported on a 3-ft-thick mat foundation with a pit provided for valves and piping.

3.8.5.1.8 Nuclear Service Cooling Water Tower and Valve House

The foundation for the nuclear service cooling water (NSCW) tower consists of a 9-ft-thick circular mat, 100 ft in diameter. The NSCW valve house is supported on a 6-ft-thick mat foundation.

3.8.5.1.9 Category 1 Water Tanks

Each reactor makeup water storage tank and each refueling water storage tank (RWST) is supported by a separate foundation mat.

Each pair of condensate water storage tanks is supported by a combined foundation mat.

The approximate mat dimensions are:

- Refueling water tank, 62 ft square x 4 ft thick.
- Reactor makeup water tank, 51 ft square x 4 ft thick.
- Condensate storage tanks, 63 ft x 115 ft x 4 ft thick.

The dikes for Category 1 storage tanks are constructed of reinforced concrete and are integral portions of the basemats.

3.8.5.1.10 Diesel Fuel Oil Storage Tank Pumphouse

This building is supported on wall strip footings with the depth of footings extending approximately 10 ft into the supporting soil. The interior excavations are occupied by the fuel oil storage tanks and then backfilled. Resistance to horizontal loads is sustained by passive pressure of the soil.

3.8.5.1.11 Category 1 Tunnels

The Category 1 tunnels are box-shaped culverts with moment-resistant joints. Sliding resistance is afforded by passive soil resistance.

3.8.5.2 Applicable Codes, Standards, and Specifications

For the foundations of the containments and internal structures, refer to paragraph 3.8.1.2. For the other Category 1 structures, refer to paragraph 3.8.4.2.

3.8.5.3 Loads and Load Combinations

Foundation loads and load combinations for the containment and its internal structures are described in paragraph 3.8.1.3.

The foundation loads and load combinations for the other Category 1 structures are described in paragraph 3.8.4.3.

3.8.5.4 Design and Analysis Procedures

The basemat is analyzed either using a finite element model or conventional methods, such as treating the mat as a plate on an elastic foundation. In the finite element model, the soil underneath the basemat is represented by discrete soil springs attached to the nodes of the model. The spring constants are functions of the tributary areas of the finite elements and the coefficients of subgrade reaction.

The heavy mat footings preclude significant differential settlements within a particular structure.

The effects of differential settlements on the VEGP structures have been shown to be negligible (reference response to Q241.18). The maximum total predicted settlements of the power block structures are given in figure 2.5.4-1.

In the shear wall structures, diaphragm action of the horizontal slabs carries lateral forces (causing overturning moments) to the shear walls, which in turn distribute loads to the basemats. The following shear wall structures exhibit this action:

- Auxiliary building.
- Fuel handling building.
- Control building.
- Diesel generator buildings.
- Auxiliary feedwater pumphouses.
- NSCW valve houses.

The diesel fuel oil storage tank pumphouse is a box-like structure which transfers the lateral forces through the shear walls to its wall strip footings.

In the tank structures, including the NSCW towers, the concrete shells sustain both the vertical and lateral loads and transfer them to the mat foundations. The containment shells likewise deliver vertical and lateral forces to their basemats.

The rigid frame actions of the Category 1 tunnels transfer lateral forces to the supporting soil.

The three-dimensional nature of earthquakes, as described in paragraph 3.7.B.2.14, is used as seismic input when considering the effects of overturning and sliding.

3.8.5.4.1 Containment

For the basemat analysis, the basemat and reactor cavity are modeled with brick elements. The cylinder and dome are modeled with plate elements.

The tendon gallery is designed as an inverted rigid frame, supported with fixed ends at the underside of the containment basemat. An access shaft is attached to the basemat and tendon gallery and is designed as a retaining well.

3.8.5.4.2 Containment Internal Structures

The containment internal concrete structures are modeled with plate elements.

3.8.5.4.3 Auxiliary Building

The basemat is analyzed using a finite element analysis, with the basemat modeled with plate elements.

3.8.5.4.4 Fuel Handling Building

Plate finite elements are used in the model of the fuel handling building.

3.8.5.4.5 Control Building

The control building foundation is modeled with plate elements.

3.8.5.4.6 Diesel Generator Building

The foundation is designed by hand calculations and a finite element model using beam elements.

3.8.5.4.7 Auxiliary Feedwater Pumphouse

The basemat is designed using hand calculations.

3.8.5.4.8 Nuclear Service Cooling Water Tower and Valve House

The cooling tower and valve house basemats are modeled with plate elements.

3.8.5.4.9 Category 1 Water Tanks

The condensate storage tank basemat is modeled using plate elements. The basemats of refueling and reactor makeup storage tanks are designed by conventional methods using hand calculations.

3.8.5.4.10 Diesel Fuel Storage Tank Pumphouses

The wall strip footings are designed by conventional methods using hand calculations.

3.8.5.4.11 Category 1 Tunnels

The tunnels are designed as either box culverts using hand methods or as inverted rigid frames if attached to the underside of basemats.

3.8.5.5 <u>Structural Acceptance Criteria</u>

The structural acceptance criteria for the containment foundation follow those given for the buildings in paragraph 3.8.1.5. Refer to paragraph 3.8.4.5 for the foundations, also use the same structural acceptance criteria as their superstructures. The limiting conditions for the foundation medium together with a comparison of actual capacity and estimated structure loads are found in subsection 2.5.4. The minimum factors of safety against overturning, sliding, and flotation for Category 1 buildings are given in table 3.8.5-1.

3.8.5.6 <u>Materials, Quality Control, and Special Construction Techniques</u>

The materials and quality control program for the foundations are the same as those listed for the structures. The foundations for the containments and internal structures follow paragraph

3.8.1.6, while the other Category 1 structure foundations conform with those shown in paragraph 3.8.4.6.

There are no special construction techniques employed in construction of the foundations.

3.8.5.7 Testing and Inservice Inspection Requirements

Testing and inservice surveillance other than obtaining and recording foundation settlement data is not required or planned for the structural foundations. The containment basemat, being a portion of the pressure boundary is, however, subject to the requirements specified in paragraph 3.8.1.7.

Settlement monitoring is achieved by use of over 100 markers placed in the basemats and building columns of the Category 1 structures and the turbine building. Two permanent benchmarks, founded in the marl, act as reference points for settlement observations.

For the period of extended operation, periodic inspections of the Category 1 structures required to be maintained as Category 1 (see paragraph 3.8.4.7) are required license renewal aging management program activities to manage aging of foundations (see subsection 19.2.32).

3.8.5.8 Standard Review Plan Evaluation

The Standard Review Plan specifies American Society of Mechanical Engineers (ASME) Section III, Division 2 Code for the containment foundation and American Concrete Institute (ACI) 349 for the other Seismic Category 1 foundations as the applicable code. The design of the VEGP containment foundation is based on Article CC-3000 of the ASME Code. The design of other Seismic Category 1 foundations is based on ACI 318-71. The Standard Review Plan specifies the codes, standards, and guides listed in its sections 3.8.1 and 3.8.3 as acceptable for the containment foundation and other Seismic Category 1 foundations, respectively.

Code ACI 349 was not in effect at the time the VEGP construction permit was issued. In accordance with the Preliminary Safety Analysis Report commitment, ACI 318-71 was used. The differences between ACI 349 and ACI 318 are minor except for the load combination equations which, in the case of VEGP, are in conformance with the Standard Review Plan. Thus, the design procedures and construction practices delineated in the FSAR ensure that the structure will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

Refer to section 1.9 for VEGP positions on regulatory guides.

3.8.5.9 Reference

1. Bechtel Power Corporation, Inc., Report on Foundation Investigations, Vol 1, VEGP, San Francisco, California, July 1974.

TABLE 3.8.1-1 (SHEET 1 OF 2)

CONTAINMENT LOAD COMBINATIONS AND LOAD FACTORS

Category	D[a]	L[b]	F	Pt	Pa	Tt	To	Ta	Eo	Ess	W	Wt	Ro	Ra	Rr	Pv	B
Service																	
Test	1.0	1.0	1.0	1.0	--	1.0	--	--	--	--	--	--	--	--	--	--	--
Construction	1.0	1.0	1.0	--	--	--	1.0	--	--	--	1.0	--	--	--	--	--	--
Normal	1.0	1.0	1.0	--	--	--	1.0	--	--	--	--	--	1.0	--	--	1.0	--
Severe environmental	1.0	1.0	1.0	--	--	--	1.0	--	1.0	--	--	--	1.0	--	--	1.0	--
	1.0	1.0	1.0	--	--	--	1.0	--	--	--	1.0	--	1.0	--	--	1.0	--
Factored																	
Severe environmental	1.0	1.3	1.0	--	--	--	1.0	--	1.5	--	--	--	1.0	--	--	1.0	--
	1.0	1.3	1.0	--	--	--	1.0	--	--	--	1.5	--	1.0	--	--	1.0	--
Extreme environmental	1.0	1.0	1.0	--	--	--	1.0	--	--	1.0	--	--	1.0	--	--	1.0	--
	1.0	1.0	1.0	--	--	--	1.0	--	--	--	--	1.0	1.0	--	--	1.0	--
	1.0	1.0	1.0	--	--	--	1.0	--	--	--	--	--	1.0	--	--	1.0	1.0
Abnormal	1.0	1.0	1.0	--	1.5	--	--	1.0	--	--	--	--	--	1.0	--	--	--
	1.0	1.0	1.0	--	1.0	--	--	1.0	--	--	--	--	--	1.25	--	--	--
Abnormal/severe environmental	1.0	1.0	1.0	--	1.25	--	--	1.0	1.25	--	--	--	--	1.0	--	--	--
	1.0	1.0	1.0	--	1.25	--	--	1.0	--	--	1.25	--	--	1.0	--	--	--
	1.0	1.0	1.0	--	--	--	1.0	--	1.0	or	1.0	--	--	--	--	--	--
Abnormal/extreme environmental	1.0	1.0	1.0	--	1.0	--	--	1.0	--	1.0	--	--	--	1.0	1.0	--	--

a. Symbols:
- D = dead loads.
- F = prestressing force loads.
- E_o = operating basis earthquake.
- E_{ss} = safe shutdown earthquake.
- L = live load.
- P_v = pressure load due to pressure variation during normal operating conditions.
- P_t = test pressure load.
- P_a = accident/incident maximum pressure.
- R_o = piping loads during operating conditions.

TABLE 3.8.1-1 (SHEET 2 OF 2)

R_a	=	piping loads due to increased temperature resulting from design accident.
T_a	=	accident/incident temperature.
T_o	=	operating temperature.
T_t	=	test temperature.
W	=	wind loads.
W_t	=	tornado loads.
R_r	=	the local effects due to pipe rupture. (See CC-3220, ASME Section III, Division 2.)
B	=	blast load due to postulated site-proximity explosion.

b. Includes all temporary construction loading during and after construction of containment.

TABLE 3.8.4-1[(A)]

OTHER CATEGORY 1 STRUCTURES
STEEL DESIGN LOAD COMBINATIONS
ELASTIC METHOD

	EQN	D	L	P_a	T_o	T_a	E	E'	W	W_t	R_o	R_a	Y_j	Y_r	Y_m	N	B	Strength Limit(f_s)
Service Load Conditions																		
	1	1.0	1.0															1.0
	2	1.0	1.0				1.0											1.0
	3	1.0	1.0						1.0									1.0
	4	1.0	1.0		1.0						1.0							1.5
	5	1.0	1.0		1.0		1.0				1.0							1.5
	6	1.0	1.0		1.0				1.0		1.0							1.5
Factored Load																		
	7	1.0	1.0		1.0			1.0			1.0							1.6
(See note b.)	8	1.0	1.0		1.0					1.0	1.0							1.6
	9	1.0	1.0	1.0		1.0						1.0						1.6
(See notes c and d.)	10	1.0	1.0	1.0		1.0	1.0					1.0	1.0	1.0	1.0			1.6
(See notes c and d.)	11	1.0	1.0	1.0		1.0		1.0				1.0	1.0	1.0	1.0			1.7
	12	1.0	1.0		1.0						1.0						1.0	1.6
	13	1.0	1.0		1.0						1.0					1.0		1.6

a. See paragraph 3.8.4.3 for definition of load symbols including the definition of L during prefuel load and post-fuel load conditions. f_s is the allowable stress for the elastic design method defined in Part 1 of the AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings." The one-third increase in allowable stresses permitted for seismic or wind loadings is not considered.

b. When considering tornado missile load, local section strength may be exceeded provided there will be no loss of function of any safety-related system. In such cases, this load combination without the tornado missile load is also to be considered.

c. When considering Y , Y_r, and Y_m loads, local section strength may be exceeded provided there will be no loss of function of any safety-related system. In such cases, this load combination without Y_j, Y_r, and Y_m is also to be considered.

d. For this load combination, in computing the required section strength, the plastic section modulus of steel shapes, except for those which do not meet the AISC criteria for compact sections, may be used.

TABLE 3.8.4-2[a]

OTHER CATEGORY 1 STRUCTURES
STEEL DESIGN LOAD COMBINATIONS
PLASTIC METHOD

	EQN	D	L	P_a	T_o	T_a	E	E'	W	W_t	R_o	R_a	Y_j	Y_r	Y_m	N	B	Strength Limit(Y)
Service Load Conditions																		
	1	1.7	1.7															1.0
	2	1.7	1.7				1.7											1.0
	3	1.7	1.7						1.7									1.0
	4	1.3	1.3		1.3						1.3							1.0
	5	1.3	1.3		1.3		1.3				1.3							1.0
	6	1.3	1.3		1.3				1.3		1.3							1.0
Factored Load Conditions																		
	7	1.0	1.0		1.0			1.0			1.0							1.0
(See note b.)	8	1.0	1.0		1.0					1.0	1.0							1.0
	9	1.0	1.0	1.5		1.0						1.0						1.0
(See note c.)	10	1.0	1.0	1.25		1.0	1.25					1.0	1.0	1.0	1.0			1.0
(See note c.)	11	1.0	1.0	1.0		1.0		1.0				1.0	1.0	1.0	1.0			1.0
	12	1.0	1.0		1.0							1.0					1.0	1.0
	13	1.0	1.0		1.0							1.0				1.0		1.0

a. See paragraph 3.8.4.3 for definition of load symbols including the definition of L during prefuel load and post-fuel load conditions. Y is the required section strength based on plastic design methods described in Part 2 of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

b. When considering tornado missile load, local section strength may be exceeded provided there will be no loss of function of any safety-related system. In such cases, this load combination without the tornado missile load is also to be considered.

c. When considering Y_j, Y_r, and Y_m loads, local section strength may be exceeded provided there will be no loss of function of any safety-related system. In such cases, this load combination without Y_j, Y_r, and Y_m is also to be considered.

TABLE 3.8.4-3[a]

OTHER CATEGORY 1 STRUCTURES
CONCRETE DESIGN LOAD COMBINATIONS
STRENGTH METHOD

	EQN	D	L	P_a	T_o	T_a	E	E'	W	W_t	R_o	R_a	Y_j	Y_r	Y_m	N	B	Strength Limit
Service Load Conditions																		
	1	1.4	1.7															U
(See note b.)	2	1.4	1.7						1.7									U
(See note c.)	3	1.4	1.7				1.9											U
	4	1.05	1.275		1.275							1.275				U		
	5	1.05	1.275		1.275					1.275		1.275			U			
	6	1.05	1.275		1.275		1.425				1.275				U			
Factored Load Conditions																		
	7	1.0	1.0		1.0			1.0			1.0							U
(See note d.)	8	1.0	1.0		1.0					1.0	1.0							U
	9	1.0	1.0	1.5		1.0						1.0						U
(See note e.)	10	1.0	1.0	1.25		1.0	1.25					1.0	1.0	1.0	1.0			U
(See note f.)	11	1.0	1.0	1.0		1.0		1.0				1.0	1.0	1.0	1.0			U
	12	1.0	1.0		1.0						1.0						1.0	U
	13	1.0	1.0		1.0						1.0					1.0		U

a. See paragraph 3.8.4.3 for definition of load symbols including the definition of L during prefuel load and post-fuel load conditions. U is the required strength based on strength method per ACI 318-71.

b. Unless this equation is more severe, the load combination 1.2D+1.7W is also to be considered.

c. Unless this equation is more severe, the load combination 1.2D+1.9E is also to be considered.

d. When considering tornado missile load, local section strength may be exceeded provided there will be no loss of function of any safety-related system. In such cases, this load combination without the tornado missile load is also to be considered.

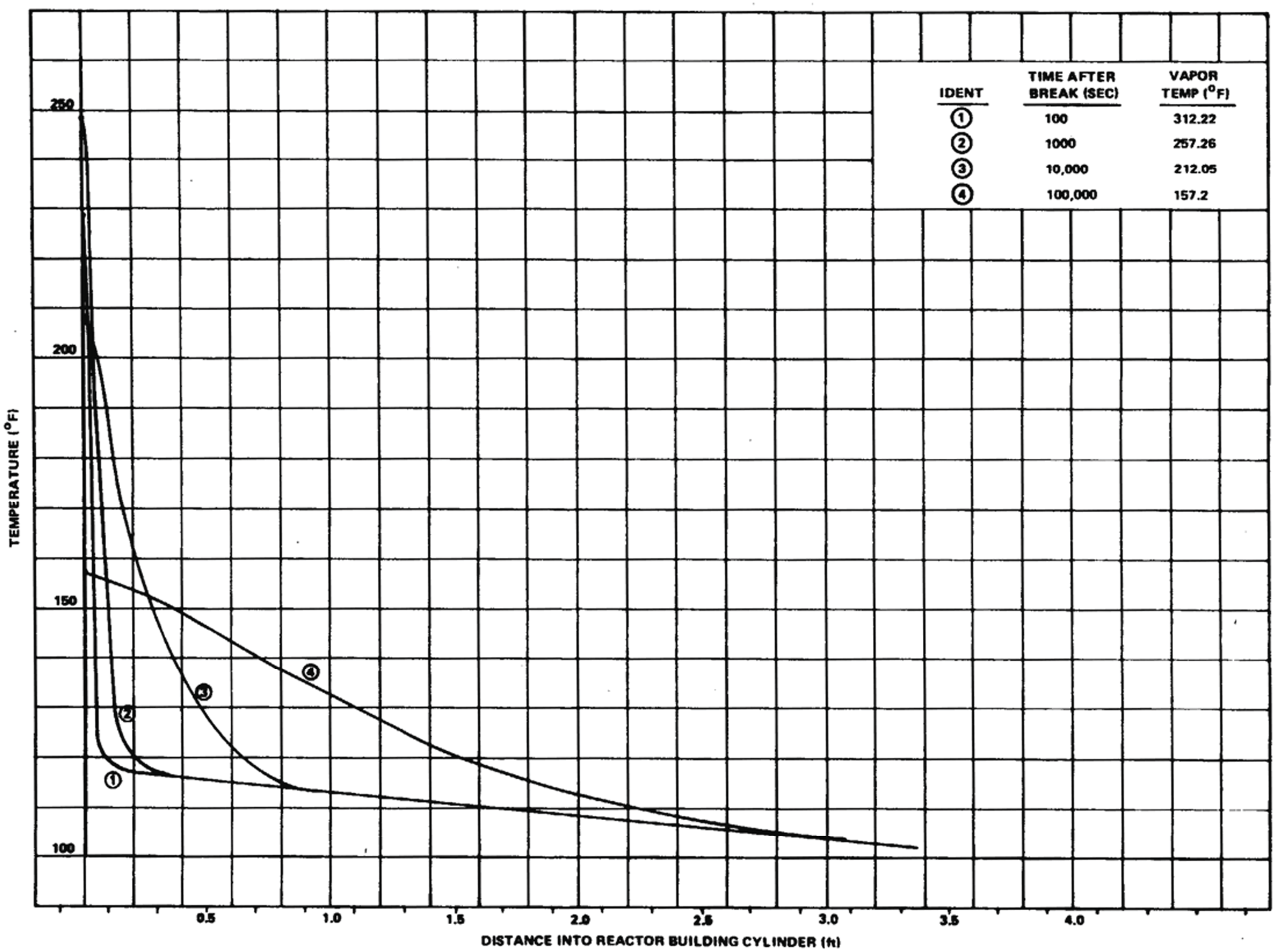
e. When considering Y_j, Y_r, and Y_m loads, local section strength may be exceeded provided there will be no loss of function of any safety-related system. In such cases, this load combination without Y_j, Y_r, and Y_m is also to be considered.

TABLE 3.8.5-1

MINIMUM DESIGN FACTORS OF SAFETY

For Combination[a]	Overturning	Sliding	Flotation
D+H+E	1.5	1.5	-
D+H+W	1.5	1.5	-
D+H+E'	1.1	1.1	-
D+H+W	1.1	1.1	-
D+F'	-		1.5

a. H = Lateral earth pressure.
F' = Flood loads.
D = Dead loads.
E = Loads generated by the operating basis earthquake (OBE).
E' = Loads generated by the safe shutdown earthquake (SSE).
W = Wind loads.
W_t = Tornado loads.



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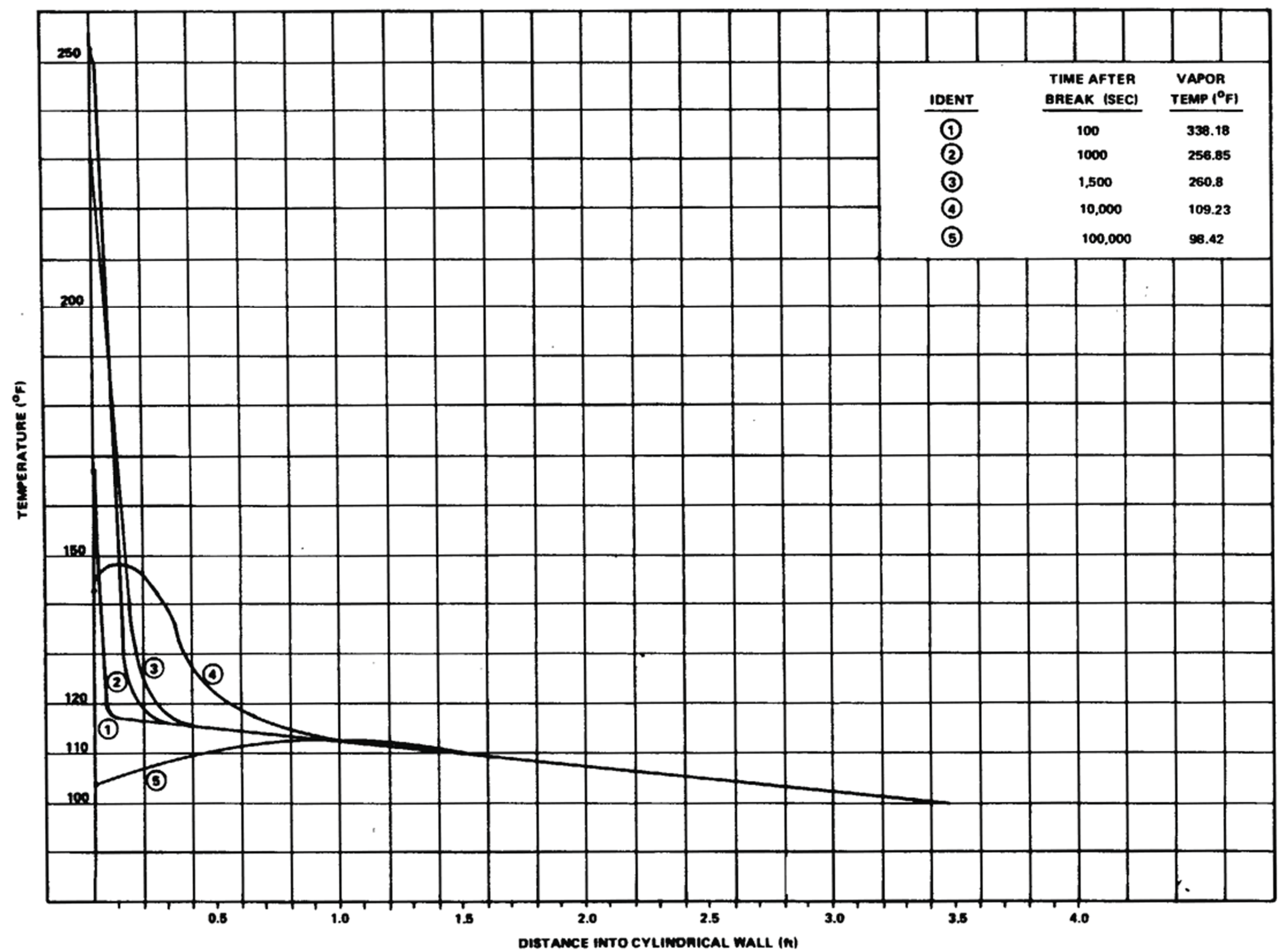
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEMPERATURE GRADIENTS THROUGH CONTAINMENT WALL FOR LOCA

FIGURE 3.8.1–1 (SHEET 1 OF 2)



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SOUTHERN COMPANY
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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEMPERATURE GRADIENTS THROUGH
CONTAINMENT WALL FOR
MAIN STEAM LINE BREAK (MSLB)

FIGURE 3.8.1–1 (SHEET 2 OF 2)

3.9.B MECHANICAL SYSTEMS AND COMPONENTS

3.9.B.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.B.1.1 Design Transients

Refer to paragraph 3.9.N.1.1 for a description of the operating conditions considered in the design of the reactor coolant system (RCS), RCS component supports, and reactor internals. Class 1 piping systems are designed and analyzed using the design transients described in paragraph 3.9.N.1.1.

Class 2 and 3 piping systems and components do not require thermal transient analysis. Class 2 and 3 piping systems and components are designed and analyzed for dynamic transients, as listed in subsection 3.9.B.2.

3.9.B.1.2 Computer Programs Used in Analysis

For nuclear steam supply systems (NSSSs) and Class 1 branch lines, refer to paragraph 3.9.N.1.2.

3.9.B.1.2.1 Seismic Category 1 Items Other Than NSSS Scope

Analysis of piping systems and supports not included in the NSSS scope is performed by use of the following proprietary computer programs:

- ME-101.
- ME-210.
- BSAP (CE800, CE212, and CE217).
- GENERAL FRAME ANALYSIS.
- ANSYS.
- BISEPS (ME-140 and ME-240).
- BASE PLATE PROGRAM (ME-035)
- ME-105
- FAPPS (ME-150)
- PIPESTRESS

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of balance of plant components. These programs are described and verified in reference 2.

- WESTDYN and its associated pre- and post-processors - static and dynamic analyses of redundant piping systems.

- ITCHVALVE - transient hydraulic analyses for a piping system.
- FORFUN - calculation of unbalanced hydraulic forces between two piping elbows as a function of time.
- PIPSAN - finite beam elements to model, analyze, and evaluate three-dimensional linear elastic structures.
- STRUDL - linear elastic two- and three-dimensional analysis of frame or truss structures.
- WECAN - performs finite element structural analyses.
- WESTAT - equivalent static analyses of redundant piping systems.

These proprietary computer programs conform to the requirements of 10 CFR 50, Appendix B, Section III, and are verified as described below.

3.9.B.1.2.1.1 ME-101 Program, Linear Elastic Analysis of Piping System. The ME-101 program is used to determine stresses and loads in the piping systems due to restrained thermal expansion, deadweight, seismic inertia and anchor movements, externally applied loads such as jet-loads, and transient forcing functions such as created by fast relief valve opening and closing, fast check valve closure after pipe breaks in main feedwater line, fast valve closure in main steam line, etc. ME-101 analyzes piping systems in accordance with ANSI and ASME codes.

The ME-101 program is a finite element computer program which performs linear elastic analysis of piping systems using the stiffness method of finite element analysis; the displacements of the joints of a given structure are considered basic unknowns. The dynamic analysis by the modal synthesis method utilizes known maximum accelerations produced in a single degree of freedom model of a certain frequency. The principal program assumptions are as follows:

A. It is a linearly elastic structure.

B. Simultaneous displacement of all supports is described by a single time-dependent function.

C. Lumped mass model satisfactorily replaces the continuous structure.

D. Modal synthesis is applicable.

E. Rotational inertia of the masses has negligible effect.

The results obtained from pipe stress program ME-101 have been compared with the following:

A. ME-632, computer program, seismic analysis of piping systems, VERB MOD8, Bechtel International Corporation, San Francisco, California, 1976.

B. ASME Benchmark problem results, Pressure Vessel and Piping 1972 computer programs verification, American Society of Mechanical Engineers.

C. Longhand calculations--ME-101 is compatible with NRC Regulatory Guide 1.92. A synthesis of closely spaced modes is provided based on equation 4 of Regulatory Guide 1.92.

The verification report is on file at Bechtel.

3.9.B.1.2.1.2 ME-210 Program, Local Stress in Cylindrical Shells Due to External Loading. The ME-210 computes the local stresses in cylindrical shells that results from external loadings. The program is based on Welding Research Council Bulletin 107, August 1965. The program has been verified based upon hand calculations.

3.9.B.1.2.1.3 BSAP. This program is used in calculating moments, forces, displacements, and stiffness of the support components of Seismic Category 1 piping. A description and verification of this program is included in appendix 3B.

3.9.B.1.2.1.4 General Frame Analysis. This program uses a matrix method for solutions. The program solves two-dimensional structural frames used for pipe support and supporting structures.

3.9.B.1.2.1.5 ANSYS. The ANSYS program is a general purpose computer program for solution of several classes of engineering problems. ANSYS includes capabilities for structural analysis, including static elastic, plastic, creep, dynamic and dynamic plastic analyses, large deflection and stability analyses, one-dimensional fluid flow analyses, and heat transfer analysis including conduction and convection. The ANSYS has been developed and verified by Swanson Analysis System, Inc.

3.9.B.1.2.1.6 BISEPS Program, Bechtel Interactive System for Engineering Pipe Support. The BISEPS program is an integrated computer system used in the design of pipe supports. This program is divided into two subprograms:

A. BISEPS STAND (ME-140) performs the sizing of support hardwares, standard steel configurations, and welds in accordance with the standard support design specification. BISEPS STAND also generates a design "hard copy" or sketch showing the bill of materials, location plan and elevation, and other information used in the design, such as loads, movements, etc.

B. BISEPS FRAME (ME-240) performs only the sizing of nonstandard or skew steel configurations or any configuration not scoped within the BISEPS STAND program.

The program verification report is on file with Bechtel data processing.

3.9.B.1.2.1.7 ME-105 Program, OPTIPIPE. The ME-105 is a preprocessing program which provides automated data collection and data entry for the ME-101 program. The verification report is on file at Bechtel.

3.9.B.1.2.1.8 BASE PLATE (ME-035). The BASE PLATE program is a finite element program used to design and analyze pipe support flexible base plates on geometrically nonlinear foundations. The program performs geometry calculations to generate the finite element model and creates data sets for output report tables and the determination of the deformed and/or undeformed geometry configuration. The verification report is on file at Bechtel.

The Base Plate code was verified by comparing test problem results with the results of CDC Base Plate II code runs for the same problems. The code capabilities investigated include:

(1) Calculation of maximum bolt pullout, bolt SRSS shear force, and bolt safety factor

(2) Maximum principal stress of plates

(3) Weld shear stresses

(4) Pressure on concrete

In each case the results of the Base Plate code were consistent with the CDC benchmark.

3.9.B.1.2.1.9 ME-150 Program, FAPPS. The ME-150 optimizes frame member sizes, welds, base plates, and embedments based upon various user-specified design limitation. The verification report is on file at Bechtel.

The FAPPS code was verified by comparing test problem results with the results of STRUDL code runs for the same problems. The code capabilities compared included stiffness matrix formulation and stress calculation techniques common to both codes. In other applications where FAPPS calculations are not within STRUDL capabilities, hand calculations were performed to ensure proper operation of the code. In each case, the FAPPS results were consistent with the STRUDL results or the hand calculations.

3.9.B.1.2.1.10 PIPESTRESS Program, Linear Elastic Analysis of Piping System. The PIPESTRESS is an interrelated finite element computer program which performs linear elastic analysis of piping systems using the stiffness method of finite element analysis; the displacements of the joints of a given structure are considered basic unknowns. The dynamic analysis by the modal synthesis method utilizes known maximum accelerations produced in a single degree of freedom model of a certain frequency. The principal program assumptions are as follows:

It is a linearly elastic structure. Simultaneous displacement of all supports is described by a single time-dependent function. Lumped mass model satisfactorily replaces the continuous structure. Modal synthesis is applicable. Rotational inertia of the masses has negligible effect.

PIPESTRESS has advanced static and dynamic analysis capabilities including detailed uniform and multilevel response spectrum analyses, time history and fatigue calculations, and multiple load cases and load combinations. Stresses due to internal pressure are calculated according to the code. PIPESTRESS solves static problems by constructing a linear finite element model of the piping system using the load-deflection relationships based on the displacement method. Dynamic analysis calculates bound solutions or time history solutions for dynamic loads, which may be described by response spectra or time history data. The dynamic analysis methods used by PIPESTRESS are Modal Extraction, Single or Multilevel Response Analysis, Multimodal/Multilevel Response Analysis, Generalized Response Analysis, Selective Time History Analysis, Left-Out-Force Method, and Primary and Secondary Terms involved in Multilevel Response Analysis. Thermal transient analysis can be performed using a finite difference approximation to find thermal gradients in the pipe walls, thereby determining the maximum value during the transient analysis of the various stress terms. Fluid properties are calculated as functions of instantaneous transient fluid temperatures and pressures.

PIPESTRESS was benchmarked against all seven test problems in NUREG/CR-1677, BNL-NUREG-51267, Vol. I, and against three test problems in NUREG/CR-1677, BNL-NUREG-

51267, Vol. II. These permitted the verification of the analysis methods implemented in PIPESTRESS. The program was verified under a nuclear quality assurance program established in accordance with 10 CFR 50, Appendix B and 10 CFR 21. The verification was directed by personnel competent in the design of ASME Section III, Nuclear Power Plant Components, Class 1, 2, and 3 nuclear power plant piping under ASME nuclear quality assurance procedures. The program performs calculations in accordance with the requirements and intention of Subarticles NC/ND-3600 of ASME, Section III. PIPESTRESS has been developed and verified by DST Computer Services S. A.

3.9.B.1.3 Experimental Stress Analysis

3.9.B.1.3.1 NSSS and Class 1 Branch Lines

Refer to paragraph 3.9.N.1.3.

3.9.B.1.3.2 Seismic Category 1 Items Other Than NSSSs and Class 1 Branch Lines

Experimental stress analysis methods are not used in the design of code or noncode components for the faulted condition. For code components, the stresses do not exceed the limits of Section III of the ASME Boiler and Pressure Vessel Code.[1]

3.9.B.1.4 Considerations for the Evaluation of the Faulted Condition

3.9.B.1.4.1 Seismic Category 1 Items in the NSSS and Class 1 Branch Lines

Refer to paragraph 3.9.N.1.4.

3.9.B.1.4.2 Seismic Category 1 Items Other Than the NSSS

For statically applied loads, the stress allowables of Appendix F of ASME Section III, Division 1, are used for code components. For noncode components, allowables are based on tests or accepted standards consistent with those in the 1974 edition of Appendix F of ASME Section III, Division 1.

Dynamic loads for components loaded in the elastic range are calculated using dynamic load factors, time-history analysis, or any other method that assumes elastic behavior of the component. A component is assumed to be in the elastic range if yielding across a section does not occur. The limits of the elastic range are defined in Paragraph F-1323 of Appendix F of ASME Section III, Division 1, for code components. Local yielding due to stress concentration is assumed not to affect the validity of the assumptions of elastic behavior. The stress allowables of Appendix F of ASME Section III, Division 1, for elastically analyzed components are used for code components. For noncode components, allowables are based on tests or accepted material standards consistent with those in Appendix F for linear elastically analyzed components.

Analysis concerning the rupture of high-energy piping is addressed in section 3.6.

3.9.B.1.5 Weld Inspection Criteria

3.9.B.1.5.1 ASME Boiler and Pressure Vessel Code Section III/ANSI B31.1 Welding

The acceptance criteria for visual inspection of welds meets, or exceeds, the requirements of ASME Code, Section III, 1977 Edition through Winter 1977 Addenda, or ANSI B31.1 Code, 1977 Edition, through Winter 1977 Addenda, as applicable to the weld joint being evaluated.

3.9.B.1.5.2 AWS D1.1 Welding

The acceptance criteria for visual inspection of welds in pipe support supplementary steel structures which are completed in accordance with AISC [a], 7th Edition, and AWS D1.1-75 conform to the requirements of these codes except as noted in paragraph 1.9.94.

3.9.B.1.6 References

1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1974 edition including Addenda through Summer 1975. For small bore piping analysis only, subartical NC-3600 of the Winter 1981 Addenda to the 1980 Edition for Class 2 piping, and subartical ND-3600 of the Summer 1984 Addenda to the 1983 Edition for Class 3 piping.
2. The description and verification of the computer codes are in compliance with quality assurance requirements (WCAP-9550, WCAP-9565, and WCAP-9805) and are maintained in the applicable central file.

3.9.N MECHANICAL SYSTEMS AND COMPONENTS

3.9.N.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.N.1.1 Design Transients

The following five operating conditions are considered in the design of the reactor coolant system (RCS), RCS component supports, and reactor internals.

A. Normal Conditions

Any conditions in the course of startup, operation in the design power range, hot standby, and system shutdown, other than upset, emergency, faulted, or testing conditions.

B. Upset Conditions (Incidents of Moderate Frequency)

Any deviations from normal conditions anticipated to occur often enough so that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients that result from any

[a] Steel construction manual.

single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition is included in the design specifications.

C. Emergency Conditions (Infrequent Incidents)

Those deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an S_a value greater than that for 10^6 cycles from the applicable fatigue design curves of the American Society of Mechanical Engineers (ASME) Code Section III.

D. Faulted Conditions (Limiting Faults)

Those combinations of conditions associated with extremely low probability; postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

E. Testing Conditions

Testing conditions are those pressure overload tests, including hydrostatic tests, and pneumatic tests. Other types of tests are classified as normal conditions.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients which may occur during plant operation. To a large extent, the specific transients to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are sufficiently severe or frequent enough to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant. These transients are described by pertinent variations in pressure, fluid temperature, and fluid flow. The Fatigue Monitoring Program, as described in subsection 19.3.2, will be used to monitor plant transients that are significant contributors to the fatigue cumulative usage factor to ensure that the design limit on fatigue usage is not exceeded during the period of extended operation.

The design transients and the number of cycles of each that is normally used for fatigue evaluations are presented in table 3.9.N.1-1. In accordance with ASME III, emergency and faulted conditions are not included in fatigue evaluations. The cyclic or transient limits for the components listed in table 3.9.N.1-2 shall be monitored in accordance with the Component Cyclic or Transient Limit Program.

The transients and components that are monitored in the Component Cyclic or Transient Limit Program (tables 3.9.N.1-2 and 3.9.N.1-3) are based on the following methodology:

- Class 1 components monitored are determined by comparing both design fatigue usage and projected fatigue usage to a screening value of design cumulative usage fatigue

(CUF) $\leq$ 0.1. Metal fatigue, including the effect of environmentally assisted fatigue, was evaluated as a time-limited aging analysis in accordance with 10 CFR 54.21. The results for the period of extended operation are summarized in subsection 19.4.2.

- Plant cycles monitored are determined by evaluation of the contribution to fatigue usage of lifetime projected plant cycles for any Class 1 component, as well as a screening level of approximately 10% of design allowable cycles applied to the 60-year cycle projections.
- Fatigue monitoring of the limiting component(s) affected by a cycle may be used to show that the ASME Code acceptance criteria of CUF ≤ 1.0 remains valid even if the assumed number of cycles has been exceeded.
- The screening levels were selected to accommodate the maximum anticipated effect of reactor water environmental factors for a projected 60-year operating period.

A feedwater bypass system, minimizing thermal stratification conditions at the main feedwater nozzle, is a feature of the VEGP steam generators. This system reroutes all feedwater through the auxiliary feedwater nozzle when the steam generators are operating below 12% of full feedwater flow, except as otherwise noted.

For any period during a transient in which total feedwater flowrate drops below about 12% of nominal flow, the total feedwater flow enters through the auxiliary feedwater nozzle. For transients which involve total feedwater flowrates greater than about 12% of nominal feedwater flow, the flow through the auxiliary nozzle is approximately 8% of the total flow in effect at the time; i.e., 92% of total flow is through the main feedwater nozzle, and 8% of total flow is through the auxiliary feedwater nozzle.

3.9.N.1.1.1 Normal Conditions

The following primary system transients are considered normal conditions:

- Reactor coolant pump (RCP) startup and shutdown.
- Plant heatup and cooldown.
- Unit loading and unloading between 0 and 15% of full power.
- Unit loading and unloading at 5% of full power/min.
- Reduced temperature return to power.
- Step-load increase and decrease of 10% of full power.
- Large step-load decrease with steam dump.
- Steady-state fluctuations.
- Boron concentration equalization.
- Feedwater cycling.
- Loop out of service.
- Refueling.
- Turbine roll test.
- Primary side leakage test.

- Secondary side leakage test.

3.9.N.1.1.1.1 RCP Startup and Shutdown. The RCPs are started and stopped during routine operations such as RCS venting, plant heatup and cooldown, and in connection with recovery from certain transients such as loop out of service and loss of power. Other (undefined) circumstances may also require pump starting and stopping.

Of the spectrum of RCS pressure and temperature conditions under which these operations may occur, three conditions have been selected for defining transients:

• Cold condition - 70°F and 400 psig	1000 occurrences.
• Pump restart condition - 100°F and 400 psig	500 occurrences.
• Hot condition - 557°F and 2235 psig	2500 occurrences.

For RCP starting and stopping operations, it is assumed that variations in RCS primary side temperature and in pressurizer pressure and temperature are negligible and that the steam generator secondary side is completely unaffected. The only significant variables are the primary system flow and the pressure changes resulting from the pump operations.

- Minimum pressure required for RCP operation may be as low as 300 psig. Four-hundred psig is considered a conservative value for design purposes.
- These conditions are included to take care of situations requiring stopping and restarting the pumps after plant heatup has commenced and venting of the RCS prior to starting a heatup.

The following cases are considered:

Case 1 - First Pump Startup (Last Pump Shutdown)

This case involves variations in reactor coolant loop (RCL) flow which accompany startup of the first pump, both in the loop containing the pump being started and in the other loops (loops in which the pumps remain idle). This case also involves a higher dynamic pressure loss in the loop containing the pump being started, but the magnitude of the flow change is less than in case 2. For the last pump shutdown case, the transient is the reverse of the first pump startup transient.

Case 2 - Last Pump Startup (First Pump Shutdown)

This case conservatively represents the variations in RCL flow accompanying startup of the second, third, and fourth pumps as applicable. Initially, flow exists through these loops in the reverse direction as the result of starting the first pump. The remaining pumps are then started in sequence and a new equilibrium flow is established. The magnitude of flow reversal is the largest in the loop containing the last pump to be started. For the first pump shutdown case, the transient is the reverse of the last pump startup transient.

The 4000 occurrences listed in table 3.9.N.1-1 include RCP startups and shutdowns associated with RCS heatup and cooldown.

3.9.N.1.1.1.2 Heatup and Cooldown. For purposes of designing the major RCS components, the plant heatup and cooldown operations are conservatively represented by continuous 100°F/h ramp temperature changes, between the shutdown temperature of 120°F[(1)] and the no-load temperature of 557°F (for the pressurizer vessel, the design cooldown rate is 200°F/h). The number of plant heatup and cooldown operations is defined as 200 each, which corresponds to 5 occurrences per year for the 40-year plant design life.[a] In practice these operations occur more slowly. Some factors which contribute to the lower rates are as follows:

- Material ductility considerations which limit temperature rates of change as functions of RCS pressure and temperature.
- Slower heatup rates when using pump energy only.
- Interruptions due to factors such as pressurizer steam bubble formation, control rod withdrawal, sampling, oxygen scavenging, and other reactor coolant chemistry adjustments.

RCS temperature can be as low as 70°F during the shutdown period. Between 70°F and 120°F the temperature is assumed to change very slowly, without causing any significant thermal transient effects.

3.9.N.1.1.1.3 Unit Loading and Unloading Between 0 and 15% of Full Power. The unit loading and unloading cases between the 0- and 15% power levels are represented by continuous and uniform ramp power changes, requiring 30 min for loading and 5 min for unloading. During loading, reactor coolant temperatures are increased from the no-load value to the normal load program temperatures at the 15% power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot standby, with 32°F feedwater cycling. Loading commences and the feedwater temperature is assumed to increase from 32°F to the 15% power value. During each loading cycle, the auxiliary feedwater nozzle experiences two cycles of cold (32°F) feedwater addition. In addition, the main feedwater nozzle experiences a step change from 32°F to a final temperature determined for each loading cycle. The duration of the cold feedwater is 30 to 60 s and is picked to maximize the stresses on the steam generator and nozzles. Subsequent to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal (RHR) requirements, and feedwater temperature decreases from the 15% power value to 32°F. RCS pressure and pressurizer pressure are assumed to remain constant at the normal operating values during these operations.

The number of these loading and unloading transients is assumed to be 500 each during the 40-year plant design life.[b]

3.9.N.1.1.1.4 Unit Loading and Unloading at 5% of Full Power/min. The unit loading and unloading operations are conservatively represented by continuous and uniform ramp power

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. Metal fatigue, including the effect of the extended operating term on the number of transient cycles or occurrences, was evaluated as a time-limited aging analysis (TLAA) for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in subsection 19.4.2.

[b] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. Metal fatigue, including the effect of the extended operating term on the number of transient cycles or occurrences, was evaluated as a TLAA for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in subsection 19.4.2.

changes of 5%/min between the 15- and 100% power levels. This load swing is the maximum possible consistent with operation under automatic reactor control. The reactor temperature varies with load as prescribed by the reactor control system. The unit loading is accomplished by manual rod control because the automatic control rod withdrawal capability has been disabled.

The number of unloading operations is defined as 13,200, based on one swing per day during the 40-year design life of the plant[a] and assuming a 90% availability factor.

It is also possible that as many as 2000 of the loading operations may be conducted in accordance with the "reduced temperature return to power" transient discussed in the following section. Both of these transients must be evaluated to determine which is more severe for a particular component design. If the reduced temperature mode is the more severe, then 2000 occurrences of that transient should replace 2000 occurrences of the 5% of full power/min loading operation, reducing the number of 5% of full power/min loadings to 11,200.

3.9.N.1.1.1.5 Reduced Temperature Return to Power. The reduced temperature return to power operation is designed to improve the spinning reserve capabilities of the plant during load-follow operations. The transient normally begins at the ebb (50%) of a load-follow cycle and proceeds at a rapid positive rate (typically 5%/min) until the abilities of the control rods and the coolant temperature reduction (negative moderator coefficient) to supply reactivity are exhausted. At that point, further power increases are limited to approximately 1%/min, by the ability of the boron system to dilute the reactor coolant. The reduction in primary coolant temperature is limited by the protection system to about 20°F below the programmed value. The plant loading during load follow operations is accomplished by manual rod control because the automatic control rod withdrawal capability has been disabled.

The reduced temperature return to power operation is not intended for daily use. It is designed to supply additional plant capabilities when required because of network fault or upset condition. Hence, this mode of operation is not expected to be used more than once a week in practice (2000 times in 40 years[a]).

3.9.N.1.1.1.6 Step-Load Increase and Decrease of 10% of Full Power. The ±10% step change in load demand is a transient which is assured to be a change in turbine control valve position due to disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a 10% step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 and 100% of full load, the power range for automatic reactor control. In effect, during load change conditions, the RCS attempts to match turbine and reactor outputs such that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease. Manual rod control may be needed following a load increase transient because the automatic control rod withdrawal capability has been disabled.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase because the decrease in nuclear power lags the decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also increase, but this change lags slightly behind the secondary side change. Because of the coolant temperature increase and the power mismatch between turbine and reactor, the control system automatically inserts the control rods to reduce core power. The reactor coolant temperature then decreases from its peak value to a value below its initial equilibrium value. The change in reactor coolant average temperature setpoint is made as a function of turbine-generator load, as determined by first-stage turbine pressure measurement. Pressurizer pressure also decreases from its peak value and follows the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash; this reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs; i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. Manual rod control is used to withdraw the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature is raised to a value above its initial equilibrium value.

The number of each operation is specified at 2000 times, or 50 times per year for the 40-year plant design life[a].

3.9.N.1.1.1.7 Large Step-Load Decrease with Steam Dump. This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature automatically initiates a secondary side steam dump that prevents both reactor trip and lifting of steam generator safety valves. The VEGP Units 1 and 2 are designed to accept a step decrease of 50% from full power.

Subsequent to the large step-load decrease, reactor power is reduced at a controlled rate, resulting in lower flow through the steam dump system. Another consequence of this event is turbine overspeed to as high as 110% of nominal (controlled overspeed just below the turbine overspeed trip setpoint). This results in proportional increases in generator bus frequency, RCP speed, and reactor coolant flowrate.

The number of occurrences of this transient is specified at 200 times (5 per year) for the 40-year plant design life. [b]

3.9.N.1.1.1.8 Steady-State Fluctuations. It is assumed that reactor coolant pressure and temperature can vary around the nominal (steady-state) values during power operation. These variations can occur at many frequencies, but for design purposes two cases are considered:

- Initial Fluctuations

  Initial fluctuations are due to control rod cycling during the first 20 full-power months of reactor operation. Reactor coolant temperatures are assumed to vary by ±3°F and pressure by ±25 psi once during each 2-min period. The total number of occurrences is

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[b] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. Metal fatigue, including the effect of the extended operating term on the number of transient cycles or occurrences, was evaluated as a TLAA for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in subsection 19.4.2.

limited to 1.5 x 10^5. These fluctuations are assumed to occur consecutively, but not simultaneously, with random fluctuations.

- Random Fluctuations

  Reactor coolant temperature is assumed to vary by ±0.5°F and pressure by ±6 psi, once during each 6-min period. The total number of occurrences during the plant design life[a] does not exceed 3.0 x 10^6.

3.9.N.1.1.1.9 Boron Concentration Equalization. Following any large change in boron concentration in the RCS, the pressurizer spray is operated to equalize concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heaters, thus causing a pressure increase and initiation of spray at a pressurizer pressure of approximately 2275 psia. The pressure increases to approximately 2281 psia before being returned to 2250 psia by the proportional spray. The pressure is then maintained at 2250 psia by spray operation, matching the heat input from the backup heaters until the concentration is equalized.

For design purposes, it is assumed that this operation is performed once after each load change (increase and decrease) in the design load follow cycle. With two load changes per day and a 90% plant availability factor over the 40-year design life[a], the total number of occurrences is 26,400.

3.9.N.1.1.1.10 Feedwater Cycling. This transient can occur when the plant is being maintained at hot standby or no-load conditions. It is assumed that the low steam generation rate is made up by intermittent (slug) feeding of 32°F feedwater into the steam generator.

For design purposes, 2000 occurrences are assumed over the life of the plant. [a] Feedwater additions required during plant heatup and cooldown operations are also assumed to be covered by the feedwater cycling transient but with no increase in the total number of cycles.

3.9.N.1.1.1.11 Loop out of Service. The plant may be operated at a reduced power level with a single loop out of service for limited periods of time. This is accomplished by reducing reactor power and tripping a single RCP. Flow increases in the loops which remain in service (active loops), and reverse flow is established in the loop with the idle pump (inactive loop). Flow through the reactor is reduced.

For design purposes, loop shutdown is assumed to occur twice per year or 80 times during the life of the plant[a].

Returning an inactive loop to service involves reducing reactor power to approximately 10% and stabilizing conditions.

Then the inactive RCP is started up and conditions are again stabilized at the same power level. Subsequent return to full power is then conducted in accordance with a normal loading operation.

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Seventy occurrences of loop startup are defined for design purposes. This number is based on the assumption that the inactive pump is inadvertently started up at maximum allowable power level 10 times during the life of the plant.[a] (This transient is covered under upset conditions.)

3.9.N.1.1.1.12 Refueling. At the beginning of the refueling operation, the RCS is assumed to have been cooled down to 140°F. The vessel head is removed, and the refueling canal is filled. This is done by transferring water from the refueling water storage tank, which is outdoors and conservatively assumed to be at 32°F, into the loops by means of the RHR pumps. The refueling water flows directly into the reactor vessel via the accumulator connections and cold legs.

This operation is assumed to occur twice per year or 80 times over the life of the plant[a].

3.9.N.1.1.1.13 Turbine Roll Test. This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. RCP power is used to heat the reactor coolant to operating temperature (no-load conditions), and the steam generated is used to perform a turbine roll test. However, the plant cooldown during this test exceeds the 100°F/h design rate.

Twenty such test cycles are specified to be performed at the beginning of plant operating life prior to reactor operation. This transient occurs before plant startup, and the number of cycles is therefore independent of other operating transients.

3.9.N.1.1.1.14 Primary Side Leakage Test. A leakage test is performed after each opening of the primary system. During this test, the primary system pressure is raised (for design purposes) to 2500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

In actual practice, the primary system is pressurized, in accordance with ASME Section XI IWA-5211(a) and IWB-5221(a), as measured at the pressurizer, to prevent the pressurizer safety valves from lifting during the leakage test. In addition, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This is accomplished with the steam, feedwater, and blowdown lines closed off.

For design purposes, it is assumed that 200 cycles of this test occur during the 40-year design life of the plant.[a]

3.9.N.1.1.1.15 Secondary Side Leakage Test. During the life of the plant it may be necessary to check the secondary side of the steam generator, particularly the manway closure, for leakage. For design purposes, it is assumed that the generator secondary side is pressurized to just below its design pressure to prevent the safety valves from lifting. In order not to exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized. The primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements, that is, between 120°F and 250°F.

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It is assumed that this test is performed 80 times during the life of the plant.[a]

3.9.N.1.1.2 Upset Conditions

The following primary system transients are considered upset conditions:

- Loss of load without immediate reactor trip.
- Loss of power.
- Partial loss of flow.
- Reactor trip from full power:
  - With no cooldown.
  - With cooldown and no safety injection (SI).
  - With cooldown and SI.
- Inadvertent RCS depressurization.
- Inadvertent startup of an inactive loop.
- Control rod drop.
- Inadvertent SI actuation.
- Excessive feedwater flow.
- Operating basis earthquake (OBE).
- Excessive bypass feedwater flow transient.
- Reactor Coolant System Cold Overpressurization.

3.9.N.1.1.2.1 Loss of Load Without Immediate Reactor Trip. This transient involves a step decrease in turbine load from full power (turbine trip) without immediate automatic reactor trip. These conditions produce the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the reactor protection system. Since redundant means for tripping the reactor are provided by the reactor protection system, a transient of this nature is not expected, but is included to ensure a conservative design.

The number of occurrences of this transient is specified at 80 times, or twice per year for the 40-year plant design life.[a]

3.9.N.1.1.2.2 Loss of Power. This transient applies to a loss of outside electrical power to the plant, which is assumed to be operating initially at 100% power, followed by reactor and turbine trips. The RCPs are deenergized, as are electrical loads connected to the turbine-generator bus, including the main feedwater and condensate pumps. As the RCPs coast down,

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RCS flow reaches an equilibrium value under natural circulation. This condition permits removal of core residual heat through the steam generators which by this time are receiving feedwater, assumed to be at 32°F, from the auxiliary feedwater system. For equipment design purposes, it is conservatively assumed that the auxiliary feedwater pumps operate within 1 min following the loss of offsite power. Later in the transient, the auxiliary feedwater pumps are operated under manual control to obtain stable plant conditions. Steam is removed for reactor cooldown through atmospheric power-operated steam relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times, or once per year for the 40-year plant design life.[a]

3.9.N.1.1.2.3 Partial Loss of Flow. This transient applies to a partial loss of flow from full power in which an RCP is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor trip on low reactor coolant flow, followed by turbine trip and automatic opening of the steam dump system. Flow reversal occurs in the affected loop, which causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooled water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified as 80, or twice per year for the 40-year plant design life.[a]

3.9.N.1.1.2.4 Reactor Trip from Full Power. Reactor trips from full power, which may occur for a variety of reasons, cause temperature and pressure transients in the RCS. Transients also occur in the secondary side of the steam generator due to continued heat transfer from the reactor coolant through the steam generators. These transients continue until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. Continuation of feedwater flow and controlled steam dump remove the core residual heat and prevent the actuation of steam generator safety valves. The reactor coolant temperature and pressure undergo rapid decreases from full-power values as the reactor protection system causes the control rods to move into the core. For design purposes, reactor trip is assumed to occur a total of 400 times, or 10 times per year over the life of the plant.[a]

The severity of the cooldown transient following a reactor trip depends on the extent of steam generator secondary side cooling. Three basic cooldown cases are considered:

Case 1 - Reactor Trip with No Cooldown

Steam and feedwater flow are both controlled to bring the plant back to the no-load conditions and maintain it at no-load. For design purposes, 230 occurrences of this transient are specified.

Case 2 - Reactor Trip with Cooldown and No SI

For this case, it is assumed that normal feedwater flow continues for approximately 1 min after the reactor trip, maintaining a high heat transfer rate through the steam generator which

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continues to drive the primary side pressure and temperature down. The RCS pressure decreases to just above the SI setpoint. After the feedwater flow is terminated, the plant is brought back to no-load conditions. For design purposes, 160 occurrences of this transient are specified.

Case 3 - Reactor Trip with Cooldown and SI

This transient is similar to case 2, but it is assumed that the protection system setpoints are such that the RCS pressure decreases to just below the SI setpoint. The high-head safety injection system (SIS) is actuated; its operation lowers the RCS temperature and raises the RCS pressure. After approximately 1 min, main feedwater flow is terminated while auxiliary feedwater flow (actuated on the SI signal) is continued. The plant is brought back to the no-load condition after SI is manually terminated. For design purposes, 10 occurrences of this transient are specified.

It is assumed that the turbine control system operates as designed in 95% of the 400 reactor trip cases. For the remaining 5%, or 20 occurrences, it is conservatively assumed that this system fails, resulting in an emergency turbine overspeed. This situation could be initiated with malfunction of the turbine control system following a large step-load decrease with steam dump, resulting in turbine speed increase past the overspeed trip setpoint. It is assumed that the reactor then trips and that the turbine speed increases to 120% of nominal, with accompanying proportional increases in generator bus frequency, RCP speed, and reactor coolant flowrate.

For design purposes, it is assumed that the emergency turbine overspeed constitutes a special case of the reactor trip with no cooldown transient. Thus, for 20 of the 230 occurrences, the effects of the reactor coolant flow variation are to be considered in addition to the basic pressure and temperature variations.

3.9.N.1.1.2.5 Inadvertent RCS Depressurization. Several events can be postulated as occurring during normal plant operation which cause rapid depressurization of the RCS. These include:

- Actuation of a single pressurizer safety valve.
- Inadvertent opening of one pressurizer power-operated relief valve due to equipment malfunction or operator error.
- Malfunction of a single pressurizer pressure controller causing one power-operated relief valve and two pressurizer spray valves to open.
- Inadvertent opening of one pressurizer spray valve due either to equipment malfunction or operator error.
- Inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe depressurization transient and is used as a conservative case.

When a pressurizer safety valve opens and remains open, the system rapidly depressurizes, the reactor trips, and the SIS is actuated. The passive accumulators of the SIS are actuated when RCS pressure decreases by approximately 1600 psi, about 5 min after the depressurization begins. The RCS reaches an equilibrium condition where the water release rate through the open pressurizer safety valve is equivalent to the SI flow. The RCS is also cooled down by the flow through the safety valve, the SI flow, and auxiliary feedwater flow. It is assumed that auxiliary feedwater flow is terminated by the operator 10 min after the depressurization begins.

Eventually, the plant must be taken to a cold shutdown condition, as the operator can take no immediate action to stop the transient and bring the plant to hot standby if the safety valve remains open.

For design purposes, 20 occurrences of this transient are specified.

Although inadvertent auxiliary spray actuations are included among the depressurization transient events covered above, the pressurizer safety valve actuation case selected to represent all the depressurization transients does not involve spray operation. Therefore, for the previous case it is assumed that pressurizer spray is not actuated and that no temperature transients due to flow occur at the spray nozzle.

However, should auxiliary spray flow be initiated inadvertently, it could cause severe thermal shock at the pressurizer spray nozzle and on the pressurizer vessel. Therefore, to ensure a conservative design for these components, an "inadvertent auxiliary spray" transient is defined.

The inadvertent auxiliary spray transient occurs when the auxiliary spray valve is opened during normal plant operation due to failure of a control component or operator error. This introduces cold water into the pressurizer resulting in a sharp pressure decrease and eventually in a low pressure reactor trip. The temperature of the auxiliary spray flow is dependent upon the performance of the regenerative heat exchanger. The most conservative case assumes that the letdown stream is shut off and that unheated charging fluid enters the 653°F pressurizer. It is assumed that the temperature of the spray water is 70°F, and the spray flowrate is equal to the normal charging rate. It is also assumed that auxiliary spray flow continued for 5 min before termination.

The total number of occurrences of this transient during the 40-year design life of the plant[a] is specified as 10.

3.9.N.1.1.2.6 Inadvertent Startup of an Inactive Loop. This transient can occur when a loop is out of service. With the plant operating at maximum allowable power level, the RCP in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. All feedwater flow for the inactive loop is through the auxiliary feedwater nozzle.

For design purposes, the inadvertent loop startup transient is assumed to occur 10 times during the life of the plant.[a]

3.9.N.1.1.2.7 Control Rod Drop. This transient occurs if a bank of control rods (worth 1-% reactivity) drops into the fully inserted position due to a single component failure. The reactor is tripped on low pressurizer pressure, depending on time in core life and magnitude of the reactivity insertion. It is assumed that this transient occurs 80 times over the life of the plant.[a]

3.9.N.1.1.2.8 Inadvertent SI Actuation. A spurious SI signal results in an immediate reactor trip followed by actuation of the high-head centrifugal charging pumps. These pumps deliver the contents of the boron injection tank to the RCS cold legs. The initial portion of this transient is similar to the reactor trip from full power with no cooldown. Controlled steam dump and

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auxiliary feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decrease as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase to the pressurizer power-operated relief valve setpoint and the primary and secondary temperatures to decrease gradually. The transient continues until the operator stops the charging pumps. It is assumed that the plant is then returned to no-load conditions, with pressure and temperature changes controlled within normal limits.

For design purposes, this transient is assumed to occur 60 times during the 40-year design life of the plant.[a]

3.9.N.1.1.2.9 Excessive Feedwater Flow. An excessive feedwater flow transient is defined as a conservative case to cover occurrence of several events of the same general nature. The postulated transient results from inadvertent opening of a feedwater regulating valve while the plant is at the hot standby or no-load condition, with the feedwater, condensate, and heater drain systems in operation.

It is assumed that the stem of a feedwater regulating valve fails, and the valve immediately reaches the full-open position. In the steam generator, directly affected by the malfunctioning valve (failed loop), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of all operating feedwater pumps. Steamflow is assumed to remain at zero and the temperature of the feedwater entering the steam generator is conservatively assumed to be 32°F. Feedwater flow is isolated on a reactor coolant low T_{avg} signal; a low pressurizer pressure signal actuates the SIS. Auxiliary feedwater flow, initiated by the SI signal, is assumed to continue with all pumps discharging into the affected steam generator via the auxiliary feedwater nozzle. It is also assumed, for conservatism in the secondary side analysis, that auxiliary feedwater flows to the steam generators not affected by the malfunctioned valve, in the "unfailed loops." Plant conditions stabilize at the values reached in 600 s, at which time auxiliary feedwater flow is terminated. The plant is then either taken to cold shutdown or returned to the no-load condition at a normal heatup rate with the auxiliary feedwater system under manual control.

For design purposes, this transient is assumed to occur 30 times during the life of the plant.[a]

3.9.N.1.1.2.10 Operating Basis Earthquake. The OBE is that earthquake which can reasonably be expected to occur during the plant life.[a] The number of occurrences for fatigue evaluation is assumed to be 5 earthquakes of 10 cycles each (50 cycles total).

3.9.N.1.1.2.11 Excessive Bypass Feedwater Flow Transient. This transient results from a postulated feedwater bypass valve failure and provides 75% of nominal feedwater flow through the auxiliary feedwater nozzle. Thirty occurrences of this transient are assumed.

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3.9.N.1.1.2.12 RCS Cold Overpressurization. RCS cold overpressurization occurs during startup and shutdown conditions at low temperature, with or without existence of a steam bubble in the pressurizer, and is especially severe when the reactor coolant system is in a water-solid configuration. The event is inadvertent, and usually generated by any one of a variety of malfunctions or operator errors. All events which have occurred to date may be categorized as belonging to either events resulting in the addition of mass (mass input transient) or events resulting in the addition of heat (heat input transient). All these possible transients are represented by composite "umbrella" design transients, referred to here as RCS cold overpressurization.

3.9.N.1.1.3 Emergency Conditions

The following primary system transients are considered emergency conditions:

- Small loss-of-coolant accident (LOCA).
- Small steam line break.
- Complete loss of flow.

3.9.N.1.1.3.1 Small LOCA. For design transient purposes the small LOCA is defined as a break equivalent to the severance of a 1-in. inside diameter (ID) branch connection. (Breaks smaller than 0.375 in. ID can be handled by the normal makeup system and produce no significant fluid systems transients). Breaks which are much larger than 1 in. cause accumulator injection soon after the accident and are regarded as faulted conditions. It is assumed that the SIS is actuated immediately after the break occurs and delivers water at a minimum temperature of 32°F to the RCS.

For design purposes, it is assumed that this transient occurs five times during the life of the plant.[a]

3.9.N.1.1.3.2 Small Steam Line Break. For design transient purposes, a small steam line break is defined as a break equivalent in effect to a steam generator safety valve opening and remaining open.

The following conservative assumptions are made:

- The reactor is initially in a hot, zero-power condition.
- The small steam line break results in immediate reactor trip and SI actuation.
- A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

Operation of the high-head SI/charging pumps repressurizes the RCS within a relatively short time to the actuation pressure of the pressurizer power-operated relief valves. These valves

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open, and an equilibrium condition is established in which the flow through the valves matches the SI flow input to the RCS.

This transient is assumed to occur five times during the life of the plant.[a]

3.9.N.1.1.3.3 Complete Loss of Flow. This accident involves a complete loss of flow from full power resulting from simultaneous loss of power to all RCPs. The consequences are a reactor trip and turbine trip on undervoltage followed by automatic opening of the steam dump system.

For design purposes, this transient is assumed to occur five times during the plant lifetime.[a]

3.9.N.1.1.4 Faulted Conditions

The following primary system transients are considered faulted conditions, which are evaluated for one occurrence:

- Reactor coolant pipe break (large LOCA).
- Large steam line break.
- Feedwater line break.
- RCP locked rotor.
- Control rod ejection.
- Steam generator tube rupture.
- Simultaneous steam line-feedwater line break.
- Safe shutdown earthquake (SSE).

3.9.N.1.1.4.1 Reactor Coolant Pipe Break (Large LOCA). Following rupture of a reactor coolant pipe resulting in a large loss of coolant, the primary system pressure decreases rapidly causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will remain at or near the operating temperature during the blowdown. The SIS is actuated to introduce water, at an assumed minimum temperature of 32°F, into the RCS. The SI signal also results in reactor and turbine trips.

3.9.N.1.1.4.2 Large Steam Line Break. The transient is based on the complete rupture of a main steam line. The following conservative assumptions are made:

- The plant is initially at the hot, no-load condition.
- The steam line break results in immediate reactor trip and actuation of the SIS.
- Offsite power is lost at the time of the break. All RCPs are deenergized and coolant flow coasts down to the natural circulation value.
- The SIS operates at design capacity and repressurizes the RCS within a relatively short time.

In the analysis the effects of core decay heat, thick metal stored energy, and steam generator reverse heat transfer are included in order to maximize RCS post-break pressurization. These represent the most conservative cases for reactor vessel design and evaluation. The results are also characterized by high feedwater flow.

The worst case large steam break, with respect to the steam generator tubes and tube sheet, occurs outside containment with the plant initially at full power. The affected steam generator will rapidly blow down to atmospheric pressure. It is assumed that the auxiliary feedwater system will deliver 1410 gal/min at 32°F to the affected steam generator via the auxiliary feedwater nozzle. The primary side repressurizes to the pressurizer safety valve set pressure, resulting in a large differential pressure across the tubes and tube sheet.

3.9.N.1.1.4.3 Feedwater Line Break. This accident involves the double-ended rupture of a main feedwater line, resulting in rapid blowdown of the affected steam generator and termination of feedwater flow to the others. The plant is assumed to be operating at an initial power level of 102% of engineered safeguards design rating when the break occurs. Turbine trip, with immediate reactor trip, occurs on a low-low level signal from the faulted steam generator.

The auxiliary feedwater system is actuated within 1 min and supplies two intact steam generators with flow equivalent to the capacity of one motor-driven auxiliary feedwater pump via the auxiliary feedwater nozzle. Loss of the plant from the grid is assumed to cause a loss of offsite power; all RCPs are deenergized and coolant flow coasts down to the natural circulation value. The SIS is actuated and is assumed to deliver maximum safeguards flow until manually shut off at approximately 600 s.

3.9.N.1.1.4.4 RCP Locked Rotor. This accident is based on the instantaneous seizure of an RCP with the plant operating at full power. The locked rotor can occur in any loop. Reactor trip occurs almost immediately, as the result of low coolant flow in the affected loop.

3.9.N.1.1.4.5 Control Rod Ejection. This accident is based on the single most reactive control rod being instantaneously removed from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the RCS, such that the pressurizer safety valves lift, and also causes a more severe temperature transient in the loop associated with the affected region than in the other loops. For conservatism, the analysis is based on the reactivity insertion and does not include the mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head vacated by the ejected rod.

3.9.N.1.1.4.6 Steam Generator Tube Rupture. This accident is postulated as the double-ended rupture of a single steam generator tube resulting in decreases in pressurizer level and RCS pressure. Eventually the loss of reactor coolant causes a reactor trip (also a turbine trip) on low pressurizer pressure. The ensuing plant cooldown results in SIS actuation due to low pressurizer pressure. The SI signal automatically starts the auxiliary feedwater pumps and isolates the main feedwater lines. The steam line leading from the affected steam generator is isolated manually. When the pressurizer water level is recovered, the operator stops the SI pumps and uses the auxiliary feedwater system to conduct an orderly cooldown to cold shutdown conditions.

Primary side temperatures are assumed to remain constant at their initial values for 10 min following the rupture. After 10 min, the primary side temperatures and the steam temperature are assumed to vary in the same way as for the reactor trip with cooldown and SI transient. Main feedwater flow consistent with the initial power level is assumed for the first 10 min. After 10 min, main feedwater flow is terminated and auxiliary feedwater flow initiated.

3.9.N.1.1.4.7 Simultaneous Steam Line-Feedwater Line Break. This transient is based on the simultaneous, complete severance of both a main steam line and a feedwater line. It is postulated to occur when the pipe whip or missile resulting from the severance of a steam line results in the complete severance of the smaller feedwater line connected to the same steam generator. Since the velocity of the whip or missile can be very high, the two lines are assumed to break simultaneously. To ensure conservatism, this transient is assumed to occur at any power level between hot, zero, and full power.

3.9.N.1.1.4.8 Safe Shutdown Earthquake. The SSE is defined as the maximum vibratory ground motion which can reasonably be predicted from geologic and seismic evidence. The mechanical dynamic or static equivalent loads due to the vibratory motion of the SSE are considered on a component basis.

3.9.N.1.1.5 Test Conditions

The following primary system transients are considered test conditions:

- Primary side hydrostatic test.
- Secondary side hydrostatic test.
- Tube leakage test.

3.9.N.1.1.5.1 Primary Side Hydrostatic Test. The pressure tests covered by this section include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3107 psig, consistent with steam generator secondary side pressure of 0 psig.

The RCS is designed for 10 cycles of these hydrostatic tests, which are performed prior to plant startup. The number of cycles is independent of other operating transients.

Additional hydrostatic tests may be performed to meet the inservice inspection requirements of ASME Section XI, subarticle IWB-5200. A total of four such tests is expected. The increase in the fatigue usage factor caused by these tests is easily covered by the conservative number (200) of primary side leakage tests that are considered for design.

3.9.N.1.1.5.2 Secondary Side Hydrostatic Test. The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F, coincident with the primary side at 0 psig.

For design purposes, it is assumed that the steam generator will experience 10 cycles of this test.

These tests may be performed either prior to plant startup, or subsequently following major repairs, or both. The number of cycles is independent of other operating transients.

3.9.N.1.1.5.3 Tube Leakage Test. During the life of the plant it may be necessary to check the steam generator for tube leakage and tube-to-tube sheet leakage. This is done by visual inspection of the underside (channel head side) of the tube sheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdown. For these tests, the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure; and the primary system remains depressurized. The underside of the tube sheet is examined visually for leaks. If any are observed, the secondary side is depressurized and repairs made by tube plugging. The secondary side is then repressurized (to a higher pressure), and the underside of the tube sheet is again checked for leaks. This process is repeated until all the leaks are repaired. The maximum (final) secondary side test pressure reached is 840 psig. Both the primary and secondary sides of the steam generators are at ambient temperatures during these tests.

Test Pressure (psig)	Number of Occurrences
200	400
400	200
600	120
840	80

The total number of tube leakage test cycles is defined as 800 during the 40-year life of the plant.[a]

3.9.N.1.2 Computer Programs Used in Analyses

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category 1 components and equipment. These are described and verified in references 1 and 3.

- WESTDYN - static, dynamic, and fatigue analysis of redundant piping systems.
- FIXFM-3[b] - time-history response of three-dimensional structures.
- WESDYN-2[b] - piping system stress analysis from time-history displacement data.
- THRUST - hydraulic loads on loop components from blowdown information.
- WECAN - finite element structural analysis.
- MULTIFLEX - thermal-hydraulic structure systems dynamics.

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. Metal fatigue, including the effect of the extended operating term on the number of transient cycles or occurrences, was evaluated as a TLAA for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in subsection 19.4.2.

[b] These capabilities have been incorporated into WESTDYN.

The WESTEMS™ computer program (reference 4) was used for the fatigue analysis that considered the application of the mechanical stress improvement process to the Unit 1 and Unit 2 Loops 1, 2, 3, and 4 reactor vessel outlet nozzles.

3.9.N.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Seismic Category 1 systems or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests as discussed in subsection 3.9.N.2.

3.9.N.1.4 Considerations for the Evaluation of the Faulted Condition

3.9.N.1.4.1 Loading Conditions

The structural stress analyses performed on the RCS consider the loadings specified in table 3.9.B.3-1. These loads may result from any, or all of the following: thermal expansion, pressure, weight, OBE, SSE, system operating transients, LOCA loop hydraulic forces, subcompartment pressurization forces, and reactor vessel loads.

A description of the loads used in the analysis of the RCS follows.

A. Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code.

The term "operating pressure" is used in connection with determination of the system deflections and support forces. The steady-state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the RCL model at changes in direction or flow area.

B. Weight

A weight analysis is performed to meet code requirements by applying a 1.0-g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

C. Seismic

The input for the RCS seismic analysis is in the form of six statistically independent time-history inputs: three translational accelerations and three rotational accelerations. These six accelerations are applied simultaneously to the containment basemat.

For the OBE and SSE seismic analyses, 2- and 4% critical damping, respectively, is used in the RCL/supports system analysis.

D. Loss-of-Coolant Accident

Blowdown loads are developed in the RCL as a result of transient flow and pressure fluctuations following a postulated pipe break at the applicable branch nozzles. Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations. Postulated pipe break locations are given in section 3.6.

Time-history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time dependent hydraulic forcing functions used in the analysis of the RCL for each break case. For a further description of the hydraulic forcing functions, refer to section 3.6.

E. Transients

The ASME Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are discussed in paragraph 3.9.N.1.1.

To provide the necessary high degree of integrity for the RCS, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operating conditions.

3.9.N.1.4.2 RCL Analytical Models and Methods

The analytical methods used in obtaining the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the time-history modal superposition method for seismic dynamic analysis, and time- history integration method for the LOCA dynamic analysis.

The integrated RCL/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the RCL piping and components, the stiffness of supports, and the stiffnesses of auxiliary line piping which affect the system. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

A. Static

The RCL/supports system model, constructed for the WESTDYN computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3.9.N.1-1 shows an isometric line schematic of this mathematical model.

The spatial geometric description of the RCL model is based upon the RCL piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity, the coefficient of thermal expansion, the average temperature change from ambient temperature, and the weight per unit length are specified for each element. The primary equipment supports are represented by stiffness matrices which define restraint characteristics of the supports. Due to the symmetry of the static loadings, the reactor pressure vessel (RPV) centerline is represented by a fixed boundary in the system mathematical model. The vertical thermal growth of the reactor vessel nozzle centerline is considered in the construction of the model.

The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this matter, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for weight, thermal, and general pressure loading conditions are obtained by using the WESTDYN computer program.

B. Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and primary equipment. The containment internals structure and all of the piping loops are included in the coupled building/loop system model. The effect of the equipment motion on the RCL/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

Component lateral supports are inactive during plant heatup, cooldown, and normal plant operating conditions. However, these restraints become active during the rapid motions of the RCL components that occur from the dynamic loadings and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full-power condition.

The total response is obtained using the modal superposition method for time integration of the equations of motion. The results of the analysis are time-history forces and displacements. The time-history displacement response is then used in computing support loads and in performing the RCL piping stress evaluation.

C. Loss-of-Coolant Accident

The mathematical model used in the static analysis is modified for the LOCA analyses by including the mass characteristics of the piping and primary equipment. The natural frequencies and eigenvectors are determined from this.

The time-history hydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural lumped-mass node points.

The dynamic structural solution for the full-power LOCA is obtained by using a modified-predictor-corrector-integration technique and normal mode theory.

When elements of the system can only be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple nonlinear elements can be applied at the same node, if necessary.

The time-history solution is performed in program FIXFM-3. The input to this program consists of the natural frequencies, normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified RCL dynamic model are determined with the WESTDYN program. To properly simulate the release of the strain energy in the pipe, the internal forces in the system at the postulated break location due to the initial steady-state hydraulic forces, thermal forces, and weight forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response of the system from the static equilibrium position). The time-history displacement solution of all dynamic degrees of freedom is obtained using program FIXFM-3 and employing 4% critical damping.

The LOCA displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the RCL. The LOCA analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions and loop mechanical loads. The reactor vessel analysis is described in paragraph 3.9.N.1.4.6.

The asymmetric external pressure loads on the RCP and steam generator resulting from a postulated pipe rupture and pressure buildup in the loop compartments are applied to the same integrated RCL/supports system model used to compute loadings on the components, component supports, and reactor coolant piping, as discussed above. Jet impingement loads on the loop piping, components, and supports resulting from postulated pipe ruptures are also applied to the RCL supports model. The response of the entire system is obtained for the various external loading cases from which the internal member forces and piping stresses are calculated. The equipment support loads and piping stresses resulting from the external loading are added to the support loads and piping stresses calculated using the loop LOCA hydraulic forces and RPV motion.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the RCL piping.

The time-history displacements of the FIXFM-3 (or WESTDYN) program are used as input to WESDYN-2 (or WESTDYN) to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation, the displacements are treated as imposed deflections on the RCL masses.

By application of leak before break technology, the dynamic effects of the pipe rupture in the RCL of Units 1 and 2 are eliminated, as are those of selected Class 1 branch lines in Units 1 and 2. Consequently, the dynamic effects of a break in the RCL piping are not considered in the design verification of the RCL component supports.

D. Fatigue

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the code into three parts: a uniform, a linear, and a nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall, and the nonlinear portion causes a skin stress.

The transients, as defined in paragraph 3.9.N.1.1, are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat transfer program is used to solve the thermal transient problem. The pipe is represented by at least 50 elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. This temperature distribution is used to determine the uniform, linear, and nonlinear portions of the pipe stress for each transient.

A load set is defined as a set of pressure loads, moment loads, and through-wall thermal effects at a given location and time in each transient. The method of load set generation is based on reference 2. The through-wall thermal effects are functions of time and can be subdivided into four parts:

- Average temperature (TA), the average temperature through-wall of the pipe which contributes to general expansion loads.
- Radial linear thermal gradient which contributes to the through-wall bending moment ($\Delta T1$).
- Radial nonlinear thermal gradient ($\Delta T2$) which contributes to a peak stress associated with shearing of the surface.
- Discontinuity temperature (TA - TB) represents the difference in average temperature at the cross sections on each side of a discontinuity.

Each transient is described by at least two load sets representing the maximum and minimum stress state during each transient. The construction of the load sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient.

- ΔT_1.
- ΔT_2.
- $\alpha_A T_A - \alpha_B T_B$.
- Moment loads due to T_A.
- Pressure loads.

This procedure produces at least twice as many load sets as transients for each point.

For all possible load set combinations, the primary- plus-secondary and peak stress intensities, fatigue reduction factors, and cumulative usage factors are calculated. The WESTDYN program is used to perform this analysis in accordance with ASME III, Subsection NB-3650.

Since it is impossible to predict the order of occurrence of the transients over a 40-year life[a], it is assumed that the transients can occur in any sequence. This is a very conservative assumption.

The combination of load sets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

A fatigue analysis was performed that considered the application of the mechanical stress improvement process to the Unit 1 and Unit 2 Loops 1, 2, 3, and 4 reactor vessel outlet nozzles. The WESTEMS™ computer program was used to perform this fatigue analysis in accordance with ASME III, Subsection NB-3650.

3.9.N.1.4.3 Analytical Methods for RCS Class 1 Branch Lines

The analytical methods used to obtain the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectrum method for seismic dynamic analysis, and static or dynamic structural analysis for the effect of the applicable branch nozzle breaks per paragraph 3.6.2.1.1.A.1.

The integrated Class 1 piping/supports system model is the basic system model used to compute loadings on components, component and piping supports, and piping. The system models include the stiffness and mass characteristics of the Class 1 piping components, the RCL, and the stiffness of supports which affect the system response. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

A. Static

The Class 1 piping system models are constructed for the WESTDYN computer program, which numerically describes the physical system. A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads are also computed by multiplying the stiffness matrix by the displacement vector at the support point.

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. Metal fatigue, including the effect of the extended operating term on the number of transient cycles or occurrences, was evaluated as a TLAA for license renewal in accordance with 10 CFR Part 54. The results of this evaluation are provided in subsection 19.4.2.

B. Seismic

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is accomplished by locating the total mass at points in the system which appropriately represent the response of the distributed system. Effects of the primary equipment motion, that is, reactor vessel, steam generator, RCP, and pressurizer on the Class 1 piping system are obtained by modeling the mass and the stiffness characteristics of the primary equipment and loop piping in the overall system model.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Shock suppressors which resist rapid motions are also included in the analysis. The solution for the seismic disturbance employs the response spectra method. This method employs the lumped mass technique, linear elastic properties, and the principle of modal superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response from differential anchor motions. The stresses resulting from the anchor motions are considered to be secondary and, therefore, are included in the fatigue evaluation.

C. Loss-of-Coolant Accident

The mathematical models used in the seismic analyses of the Class 1 lines are also used for three RCL branch nozzle break effect analyses. To obtain the proper dynamic solution for emergency core cooling system (ECCS) lines attached to the unaffected loops, lines 6 in. and larger attached to the affected broken loop, and the surge line, the time-history deflections from the analysis of the RCL are applied at branch nozzle connections. The motion of the RCL is applied statically to non-ECCS lines attached to the unaffected loops and small lines attached to the affected loop which must maintain structural integrity.

D. Fatigue

A thermal transient heat transfer analysis is performed for each different piping component on all the Class 1 branch lines. The normal, upset, and test condition transients identified in paragraph 3.9.N.1.1 are considered in the fatigue evaluation.

The thermal quantities ΔT_1, ΔT_2, and $\alpha_A T_A - \alpha_B T_B$ are calculated on a time-history basis, using a one-dimensional finite difference heat transfer computer program. Stresses due to these quantities are calculated for each time increment using the methods of NB-3650 of ASME III.

For each thermal transient, two load sets are defined, representing the maximum and minimum stress states for that transient.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus ensuring the most conservative combinations of seismic loads are used in the stress evaluation.

The WESTDYN computer program is used to calculate the primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors, and cumulative usage factors for all possible load set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of load sets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.N.1.4.4 Primary Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped- parameter, multimass structural models. The equipment support structure models are dual-purpose since they are required to quantitatively represent the elastic restraints which the supports impose upon the loop, and to evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

A description of the supports is found in subsection 5.4.14. Detailed models are developed using beam elements and plate elements, where applicable. The reactor vessel supports are modeled using the WECAN computer program. Steam generator and RCP supports are normally modeled as linear or nonlinear springs.

For each operating condition, the loads (obtained from the RCL analysis) acting on the support structures are appropriately combined. The adequacy of each member of the steam generator supports, RCP supports, and pressurizer supports is verified by solving the ASME III Subsection NF stress and interaction equations. The adequacy of the RPV support structure is verified using the WECAN computer program and comparing the resultant stresses to the criteria given in ASME III, Subsection NF.

3.9.N.1.4.5 Analysis of Primary Components

Equipment that serves as part of the pressure boundary in the RCL includes the steam generators, the RCPs, the pressurizer, and the reactor vessel. This equipment is American Nuclear Society (ANS) Safety Class 1, and the pressure boundary meets the requirements of the ASME Code. This equipment is evaluated for the loading combinations outlined in table 3.9.B.3-1. The equipment is analyzed for:

- The normal loads of weight, pressure, and thermal.
- Mechanical transients of OBE, SSE, and pipe ruptures.
- Pressure and temperature transients outlined in paragraph 3.9.N.1.1.

The results of the RCL analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analyses, a set of loads are determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the umbrella load are handled by individualized analysis.

Seismic analyses are performed individually for the RCP, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator, RCP, and pressurizer are performed using 2% damping for the OBE and 4% damping for the SSE. The RPV is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III. These valves are identified in table 3.9.N.3-2.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the lines connect to the primary system piping are orificed to a 3/8-in. hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

3.9.N.1.4.6 Dynamic Analysis of RPV for Postulated LOCA

3.9.N.1.4.6.1 Introduction. This section presents the method of computing the reactor pressure vessel response to a postulated LOCA. Since VEGP has leak-before-break (LBB), the LOCA analyses of the reactor vessel system due to main line breaks to include the dynamic effects are not required in accordance with the rules of GDC-4. Then the most limiting breaks to be considered are the branch line breaks which consist of accumulator line in the cold leg, pressurizer surge line in the hot leg, and residual heat removal (RHR) line in the hot leg. Of these branch line breaks, the most limiting breaks which were considered for the dynamic analysis of the VEGP reactor vessel system are the accumulator line break and the pressurizer surge line break. These breaks were used for RPV dynamic analysis though the pressurizer line break has been eliminated on both units and the accumulator and RHR line breaks have been eliminated on Unit 2 as described in section 3.6.

The structural analysis of the reactor vessel system for a postulated LOCA considers simultaneous application of the time-history loads resulting from the RCL mechanical loads and internal hydraulic pressure transients. The vessel is restrained by reactor vessel supports beneath four of the reactor vessel nozzles and the RCLs with the primary supports of the steam generators and the RCPs.

3.9.N.1.4.6.2 Loading Conditions. Following a postulated pipe rupture at the applicable branch nozzle breaks, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of two phenomena:

- RCL mechanical loads.
- Reactor internal hydraulic forces.

The RCL mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. The reactions on the nozzles of the RCL piping are applied to the vessel in the RPV blowdown analysis.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomena has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8709[3] and are discussed in more detail in subsection 3.9.N.2.5.

3.9.N.1.4.6.3 Reactor Vessel and Internals Modeling. The mathematical model of the RPV is a three-dimensional nonlinear finite element model which represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN computer code. The model consists of three concentric structural submodels connected by nonlinear impact elements and stiffness matrices. The first submodel, figure 3.9.N.1-2, represents the reactor vessel shell and associated components. The reactor vessel is restrained by the four reactor vessel supports and by the attached primary coolant piping. Each reactor vessel support is modeled by a linear horizontal stiffness and a vertical impact element. The attached piping is represented by a stiffness matrix.

The second submodel, figure 3.9.N.1-3, represents the reactor core barrel, neutron panels, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first and is connected to it by a stiffness matrix at the internals support ledge. Core-barrel- to-vessel-shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost submodel, figure 3.9.N.1-4, represents the lower support plate, guide tubes, support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and second by stiffness matrices and nonlinear elements.

3.9.N.1.4.6.4 Analytical Methods. The time-history effects of the internals loads and loop mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time-history input to the dynamic RCL blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

3.9.N.1.4.6.5 Analytical Results. The results from the nonlinear time history LOCA analysis show that the reactor pressure vessel system component response (displacements and loads) obtained from the auxiliary line breaks (accumulator line and pressurizer surge line) are enveloped by the response obtained from original analysis of the main line breaks.

3.9.N.1.4.7 Evaluation of Control Rod Drive Mechanisms and Supports

The control rod drive mechanisms (CRDMs) and CRDM support structure are evaluated for the loading combinations outlined in table 3.9.B.3-1. The CRDMs and CRDM support structures are described in section 4.6 and subsection 5.4.14, respectively.

A detailed finite element model of the CRDMs and CRDM supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis, nonlinearities in the structure are represented. These include RPV plate impact, tie rods, and lifting leg clevis/RPV head interface. The time-history motion of the reactor vessel head, obtained from the RPV analysis described in 3.9.N.1.4.6, is input to the dynamic model. Maximum forces and moments in the CRDMs and support structure are then determined. For the seismic analysis, the structural model is linearized, and the floor response spectra corresponding to the CRDM tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the CRDMs for the various loading conditions are compared with maximum allowable moments determined from a detailed finite element stress evaluation of the CRDMs. Adequacy of the CRDM support structure is verified by comparing the calculated stresses to the criteria given in ASME III, Subsection NF.

3.9.N.1.4.8 Stress Criteria for Class 1 Components and Component Supports

The allowable stresses for ASME Class 1 components and supports are given in tables 3.9.B.3-2 through 3.9.B.3-6.

All Class 1 components and supports are designed and analyzed for the design, normal, upset, and emergency conditions and corresponding service level requirements of the ASME Code Section III. The analysis of test methods and associated stress or load allowable limits that are used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code with the following supplementary option.

The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

The reactor vessel support pads are qualified using the test option. The RPV support pads are designed to restrain unidirectional horizontal motion in addition to supporting the vessel. The design of the supports allows radial growth of the vessel but restrains the vessel from horizontal displacements since tangential displacement of the vessel is prevented at each vessel nozzle.

To duplicate the loads that act on the pads during faulted conditions, the tests, which utilized a one-eighth linear scale model, were performed by applying an unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted by a support shoe which was mounted to the test fixture.

The above modeling and application of load thus allows the maximum load capacity of the support pads to be accurately established. The test load, L, was then determined by multiplying the maximum collapse load by 64 (ratio of prototype area to model area) and including temperature effects in accordance with the rules of the ASME Code, Section III.

The loads on the reactor vessel support pads, as calculated in the system analysis for faulted conditions, are limited to the value of 0.80 L_T . The tests performed and the limits established for the test load method ensure that the experimentally obtained value for L_T is accurate and that the support pad design is adequate for its intended function.

Loading combinations for ASME Class 1 components are given in table 3.9.B.3-1.

The methods of load combination for each operating condition are as follows:

A. Design

Loads are combined by algebraic sum.

B. Normal, Upset

These loads are used in the fatigue evaluation in accordance with the methods prescribed in the ASME Code. Loadsets are defined for each transient, including the OBE, and are combined such that the maximum stress ranges are obtained without regard to the order in which the transients occur. (This is discussed in more detail in paragraph 3.9.N.1.4.3). The dynamic loads are combined using the square root of the sum of the squares (SRSS) method.

C. Emergency

Normal operating loads are combined algebraically. The dynamic loads are combined using the SRSS method.

D. Faulted

LOCA and SSE loads are combined using the SRSS method on a load component basis; i.e., the LOCA F_x is combined with the SSE F_x by SRSS, the LOCA F_y is combined with the SSE F_y by SRSS, and likewise for F_z, M_x, M_y, and M_z. The sustained loads, such as weight effects, are combined with the SRSS results by algebraic sum. Other dynamic loads are combined using the SRSS method.

3.9.N.1.5 References

1. The description and verification of the computer codes are in compliance with quality assurance requirements (WCAP-9550, WCAP-9565, and WCAP-9805) and are maintained in the applicable central file.
2. "Sample Analysis of a Class 1 Nuclear Piping System," prepared by ASME Working Group on Piping, ASME Publication, 1972.
3. Takeuchi, K., et al., "Multiflex - A Fortran IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708, February 1976 (Proprietary), and WCAP-8709, February 1976 (Nonproprietary).
4. “Topical Report on ASME Section III Piping and Component Fatigue Analysis Utilizing the WESTEMS™ Computer Code,” WCAP-17577-P-A, Revision 2, October 2013 (Proprietary) and WCAP-17577-NP-A, Revision 2, October 2013 (Nonproprietary).

3.9.B.2 DYNAMIC TESTING AND ANALYSIS

3.9.B.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

A preoperational test program as described in section 14.2 is implemented as required by NB-3622.3, NC-3622, and ND-3611 of Section III of the ASME Boiler and Pressure Vessel Code[1] to verify that the piping and piping restraints will withstand dynamic effects due to transients, such as pump trips and valve trips, and that piping vibrations are within acceptable levels.

The preoperational test program for the Class 1, 2, and 3 and high-energy piping systems is to simulate actual operating modes to demonstrate that the components comprising these systems meet functional design requirements and that piping vibrations are within acceptable levels. Piping systems are checked in three sequential steps or series of tests and inspections.

Construction acceptance, the first step, entails inspection of components for correct installation. During this phase, pipe and equipment supports are checked for correct assembly and setting. The cold locations of reactor coolant system (RCS) components, such as steam generators and reactor coolant pumps, are recorded.

During the second step of testing, plant heatup, the plant is heated to normal operating temperatures. During the heatup, all systems are observed periodically to verify proper expansion; expansion data is recorded at the end of heatup.

During the third step of testing, performance testing, systems are operated and performance of critical pumps, valves, controls, and auxiliary equipment is checked. This phase of testing includes transient tests, such as reactor coolant pump trips, reactor trip, and relief valve testing. During this phase of testing, the piping and piping restraints are observed for vibration and expansion response. Automatic safety devices, control devices, and other major equipment are observed for indications of overstress, excess vibration, overheating, and noise. Each system test includes critical valve operation during transient system modes.

The locations in the piping system selected for observation during the testing, and the respective acceptance criteria, are provided in the detailed preoperational vibration, thermal expansion, and dynamic effects test program plan. These are submitted to the Nuclear Regulatory Commission (NRC) at least 60 days prior to the initiation of the test program.

Provisions are made to verify the operability of essential snubbers by recording hot and cold positions. If vibration during testing exceeds the acceptance criteria, corrective measures are taken and the test rerun to demonstrate adequacy.

Should additional restraints be installed, piping rerouted, or other corrective action taken as a result of the preoperational piping test, the NRC is provided with documentation of such action. The analysis verifying that system response is within acceptable limits will be on file.

Further discussion is warranted in the area of piping vibration because of the nature of the resulting loadings and the methodology used during the testing. The loadings can be placed in two categories, transient-induced vibrations and steady-state vibrations. The first is a dynamic system response to a transient, time-dependent forcing function, such as fast valve closure, while the second is a constant vibration, usually flow induced.

A. Transient Response

1. Dynamic events falling in this category are anticipated operational occurrences. The systems and the transients to be included in the preoperational test program are provided in section 14.2.

2. For those types of transients provided in section 14.2, where a time-dependent dynamic analysis is performed on the system, the stresses thus obtained are combined with system stresses resulting from other operating conditions in accordance with the criteria provided in subsections 3.9.1 and 3.9.3.

3. Details of the program and the pipe monitoring displacement transducers or scratch plates and strain gage or load cells locations, including the criteria for evaluation of data gained, are provided in the test procedures.

B. Steady-State Vibration

1. System vibrations resulting from flow disturbances fall into this category. Positive displacement pumps may cause such flow variation and vibrations.

2. Since the exact nature of the flow disturbance is not known prior to pump operation, no analysis is performed. If the system vibration is evidenced during initial operation, the maximum amplitudes are measured and related to alternating stress intensity levels based on the guidance of ANSI/ASME OM3. The VEGP Preoperational Vibration Monitoring program will include appropriate safety-related instrument lines up to the first anchor. The acceptance criterion is that the maximum alternating stress intensity S_{alt}, calculated from the measured amplitudes, shall be limited as defined below:

a. For ASME Class 1 piping systems:

$$S_{alt} = \frac{C_2 K_2}{Z} M \leq \frac{S_{el}}{\alpha}$$

where

C_2 = secondary stress index as defined in the ASME code.

α = allowable stress reduction factor: 1.3 for materials covered by figure I-9.1; or 1.0 for materials covered by figure I-9.2.1 or I-9.2.2 of the ASME code.

K_2 = local stress index as defined in the ASME code.

M = maximum zero to peak dynamic moment loading due to vibration only, or in combination with other loads as required by the system design specification.

S_{el} = 0.8 S_A where S_A is the alternating stress at 10^6 cycles from Figure I-9.1; or S_A at 10^{11} cycles from Figure I-9.2.2 of the ASME Code. The user shall consider the influence of temperature on the modulus of elasticity.

Z = Section modulus of the pipe.

b. For ASME Class 2 and 3 or ANSI B31.1 piping:

$$S_{alt} = \frac{C_2 K_2}{Z} M \leq \frac{S_{el}}{\alpha}$$

where

$C_2K_2 = 2i$

i = stress intensification factor, as defined in subsection NC and ND of the ASME Code or in ANSI B31.1.

If significant vibration levels are detected during the test program which have not been previously considered in the piping system analysis, consideration should be given to modifying the design specification to reverify applicable code conformance.

3. When required, additional restraints are provided to reduce stresses to below the acceptance criterion levels.

3.9.B.2.2 <u>Seismic Qualification Testing of Safety-Related Mechanical Equipment</u>

The criteria used to decide whether dynamic testing or analysis will be used to qualify Seismic Category 1 mechanical equipment are as follows:

A. Analysis Without Testing

1. Structural analysis without testing will be used if structural integrity alone can ensure the intended design function. Equipment which falls into this category includes:

- Piping.
- Ductwork.
- Tanks and vessels.
- Heat exchangers.
- Filters.
- Passive valves.

The seismic analysis of piping is described in section 3.7.B.

2. Dynamic analysis without testing shall be used to qualify heavy machinery too large to be tested. It must be verified that deformations due to seismic loadings will not cause binding of the moving parts to the extent that the component cannot perform its required safety function. Components which fall into this category include:

- Pumps.
- Turbines.
- Generators.
- Fans.
- Diesel engines.

The seismic qualification of pumps is discussed more fully in paragraph 3.9.B.3.2. The procedure discussed therein is applied, with some variations, to other items in this category.

B. Dynamic Testing

Dynamic testing is used for components with mechanisms that must change position in order to perform their required safety function. Such components include:

- Electric motor valve operators.
- Valve limit switches.

- Similar appurtenances for other active mechanical equipment.

The seismic qualification of Seismic Category 1 electrical equipment is discussed in section 3.10.B.

C. Combinations of Analysis with Testing

A combination of analysis, static testing, and dynamic testing is used for seismic qualification of complex equipment. Such equipment includes:

1. Standby diesel generators.
2. Turbine-driven auxiliary feedwater pumps.
3. Main steam and main feedwater isolation valves.
4. Other active valves:
   a. The manufacturer determines the first natural frequency of an active valve by analysis or test. If analysis is used the analytical model is verified by test.
   b. If the first natural frequency is 33 Hz or higher, a static stress analysis and a static deflection test are performed to verify that deformation due to seismic loadings will not cause binding of internal valve parts which prevents valve operations within specified time limits.
   c. The operability of valve actuator and other appurtenances is qualified by dynamic testing.
   d. If the first natural frequency is less than 33 Hz, a dynamic analysis of the piping system using a finite element model of the flexible valve is required to determine the effect of the valve to the response of the overall system.

The seismic qualification of active valves is discussed more fully in paragraph 3.9.B.3.2.

The acceptance criteria which are used are as follows:

1. Tests, when used, demonstrate that the component is not prevented from performing its required safety function during and after the test.
2. Analysis, when used, verifies that stresses do not exceed the allowables specified for the appropriate loading combinations listed in subsection 3.9.B.3 and that deformations do not exceed those that will permit the component to perform its required safety function.

The results of tests and analyses of safety-related mechanical equipment are available for inspection.

3.9.B.2.3 <u>Dynamic Response Analysis of Reactor Internals Operational Flow Transients and Steady-State Conditions</u>

Refer to paragraph 3.9.N.2.3.

3.9.B.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Refer to paragraph 3.9.N.2.4.

3.9.B.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Condition

Refer to paragraph 3.9.N.2.5.

3.9.B.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Refer to paragraph 3.9.N.2.6.

3.9.B.2.7 Reference

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1974 edition through Summer 1975 addenda. For small bore piping analysis only, Subartical NC-3600 of the Winter 1981 Addenda to the 1980 Edition for class 2 piping, and Subartical ND-3600 of the Summer, 1984 Addenda to the 1983 Edition for class 3 piping.

3.9.N.2 DYNAMIC TESTING AND ANALYSIS

3.9.N.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

A discussion of the preoperational vibration testing program is provided in subsection 3.9.B.2.

3.9.N.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

The operability of Seismic Category 1 mechanical equipment must be demonstrated if the equipment is determined to be active. The operability of active Class 2 and 3 pumps, active Class 1, 2, and 3 valves and their respective drives, operators, and vital auxiliary equipment is shown by satisfying the criteria given in paragraph 3.9.N.3.2.

The testing procedures used in the seismic qualification of instrumentation and electrical equipment are discussed in section 3.10.N.

Inactive Seismic Category 1 equipment is shown to have structural integrity during all plant conditions by analysis satisfying the stress criteria applicable to the particular piece of equipment or by test showing that the equipment retains its structural integrity under the simulated test environment.

A list of Seismic Category 1 equipment is provided in table 3.2.2-1.

3.9.N.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow-induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed-form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be practically performed independently of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitations depends on many factors, such as type and location of component and flow conditions. The effects of these forcing functions have been studied from tests performed on models and prototype plants as well as component tests.(1)(2)(3)

The Indian Point No. 2 plant (Docket No. 50-247) has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant (Docket No. 50-344) instrumentation program and the Sequoyah No. 1 plant (Docket No. 50-327) instrumentation program will provide prototype data applicable to VEGP.(1)(3)

The VEGP reactor internals are similar to Indian Point No. 2; the only significant differences are the modifications resulting from the use of 17 x 17 fuel, replacement of the annular thermal shield with neutron shielding pads, and the change to the upper head injection (UHI) style inverted top hat support structure configuration. These differences are addressed below.

A. 17 x 17 Fuel

The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and control rod driveline. The new 17 x 17 guide tubes are stronger and more rigid; hence, they are less susceptible to flow-induced vibration. The fuel assembly itself is relatively unchanged in mass and spring from the 15 x 15 fuel assembly vibration characteristics.

B. Neutron Shield Panels Lower Internals

The primary cause of core barrel excitation is flow turbulence, generated in the downcomer annulus.(3) The vibration levels due to core barrel excitation for Trojan and VEGP, both having neutron shielding panels, are expected to be similar. Since the VEGP plants have greater velocities than Trojan, vibration levels due to the core barrel excitation are expected to be somewhat greater than that for Trojan (proportional to flow velocity raised to a small power).(2) However, scale model test results(2) and results from Trojan(1) show that core barrel vibration of plants with neutron shield panels is significantly less than that of plants with thermal shields. This information and the fact that low core barrel stresses and large safety margins were measured at Indian Point No. 2 (thermal shield configuration) lead to the conclusion that stresses approximately equal to those of Indian Point No. 2 result on the VEGP internals with the attendant large safety margins.

C. UHI-Style Inverted Top Hat Upper Support Configuration

The components of the upper internals are excited by turbulent forces due to axial and cross-flows in the upper plenum.(3) Sequoyah and VEGP have the same basic upper internals configuration; therefore, the general vibration behavior is not changed.

Results from Sequoyah 1 plant testing(4) show high factors of safety for upper internals components. These results, supported by scale model test results and analytical work are used to demonstrate the adequacy of the VEGP upper internals.

The original test and analysis of the four-loop configuration are augmented by references 1, 2, and 3 to cover the effects of successive hardware modifications.

3.9.N.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the VEGP reactor internals design configuration is well characterized, as is discussed in paragraph 3.9.N.2.3, it is not considered necessary to conduct instrumented tests of the VEGP hardware. The recommendations of Regulatory Guide 1.20 are satisfied by conducting the confirmatory pre- and post-hot functional examination for integrity. Conformance with Regulatory Guide 1.20 is summarized in section 1.9. This examination includes in excess of 30 features illustrated on drawing AX6DD401 with special emphasis on the following areas:

A. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

B. The lateral, vertical, and torsional restraints provided within the vessel.

C. The locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

D. The other locations on the reactor internal components which are similar to those that were examined on the prototype designs.

E. The inside of the vessel is inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign material is in evidence.

A particularly close inspection is made on the following items or areas, using a 5X or 10X magnifying glass or other appropriate inspection.

A. Lower Internals

1. Upper barrel to flange girth weld.
2. Upper barrel to lower barrel girth weld.
3. Upper core plate aligning pin. Examine bearing surface for shadow marks, burnishing, buffing, or scoring. Inspect welds for integrity.
4. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.
5. Baffle assembly locking devices. Check for lockweld integrity.
6. Lower barrel to core support girth weld.
7. Neutron shield panel screw locking devices and dowel pin lockwelds. Examine the interface surfaces for evidence of tightness. Check for lockweld integrity.

8. Radial support key welds.
9. Insert screw locking devices. Examine soundness of lockwelds.
10. Core support columns and instrumentation guide tubes. Check the joints for tightness and soundness of the locking devices.
11. Secondary core support assembly screw locking devices for lockweld integrity.
12. Lower radial support keys and inserts. Examine bearing surfaces for shadow marks, burnishing, buffing, or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. Subsequent to the hot functional testing, the bearing surfaces of the key and keyway show burnishing, buffing, or shadow marks which indicate pressure loading and relative motion between these parts. Minor scoring of engaging surfaces is also possible and acceptable.
13. Gaps at baffle joint. Check gaps between baffle-to-baffle joints.

B. Upper Internals

1. Thermocouple conduits, clamps, and couplings.
2. Guide tube, support column, and thermocouple assembly locking devices.
3. Support column and thermocouple conduit assembly clamp welds.
4. Upper core plate alignment inserts. Examine bearing surface for shadow marks, burnishing, buffing, or scoring. Check the locking devices for integrity of lockwelds.
5. Thermocouple conduit fitting locktab and clamp welds.
6. Guide tube enclosure and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals are subjected to a total operating time at greater than normal full-flow conditions (four-pump operation) of at least 240 h. This provides a cyclic loading of approximately 107 cycles on the main structural elements of the internals. In addition, there is some operating time with only one, two, and three pumps operating.

Pre- and post-hot functional inspection results serve to confirm that the internals are well behaved. When no signs of abnormal wear or harmful vibrations are detected and no apparent structural changes take place, the four-loop core support structures are considered to be structurally adequate and sound for operation.

3.9.N.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The reactor internals analysis under faulted events considers the following conditions:

- LOCA (RCL auxiliary line breaks as considered in paragraph 3.9.N.1.4.6.1)
- SSE (Safe Shutdown Earthquake)

The criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that geometry remains substantially intact. Consequently, the limitations established for the internals are concerned with the deflections and stability of the parts in addition to stress criteria to assure integrity of the components.

3.9.N.2.5.1 Reactor Internals Analysis Methods

The evaluation of the reactor internals is composed of two parts. The first part is the three-dimensional response of the reactor internals resulting from the RCL branch pipe break conditions as mentioned in paragraph 3.9.N.1.4.6. The reactor internals response is taken from the WECAN RPV and internals system response as described in paragraph 3.9.N.1.4.6.3. The second part of this evaluation is the component stress evaluations. Maximum stresses and displacements under LOCA and SSE conditions are obtained for the reactor internal components and are combined by the square root sum of the squares. Then these maximum stresses and displacements are compared to the allowable values for the faulted conditions.

Analysis of the reactor internals for blowdown loads resulting from a loss-of-coolant accident is based on the time-history response of the internals to simultaneously applied blowdown forcing functions. The forcing functions are defined at points in the system where changes in cross-section of direction of flow occur such that differential loads are generated during the blowdown transient. The dynamic mechanical analysis can employ the displacement method, lumped parameters, and stiffness matrix formulations. Because of the complexity of the system and the components, it is necessary to use finite element stress analysis codes to provide more detailed information at various points.

Multiflex, a blowdown digital computer program,[(5)] which was developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a LOCA, is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM,[(6)] which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. Multiflex is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically, using a fixed mesh in both space and time.

Although spatially one-dimensional conservation laws are employed, the code can be applied to describe three-dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc., are considered.

The Multiflex code evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the programs LATFORCE and FORCE 2, which utilize a detailed geometric description in evaluating the loadings on the reactor internals.

Each reactor component for which calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

A. The pressure differential across the element.

B. Flow stagnation on and unrecovered orifice losses across the element.

C. Friction losses along the element.

Input to the code, in addition to the blowdown pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The reactor internals analysis has been performed using the following assumptions:

- The mechanical and hydraulic analyses have considered the effect of hydroelasticity.
- The reactor internals are represented by concentric pipes, 3-D beams, and a multimass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components.
- The model described is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the horizontal and vertical directions.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loading on the components. A 1-ms severance time is taken as the limiting case.

After a postulated break at the cold leg branch line, a rarefaction wave propagates along the reactor inlet pipe entering through the inlet nozzle into the region between the core barrel and the reactor vessel. This region is called the downcomer annulus. The initial wave propagates up, around, and down the downcomer annulus, then up through the region circumferentially enclosed by the core barrel; that is, the fuel region.

The region of the downcomer annulus close to the break depressurizes rapidly but, because of restricted flow areas and finite wave speed (approximately 3000 feet per second), the opposite side of the core barrel remains at a high pressure. This results in a net horizontal force on the core barrel and the reactor pressure vessel. As the depressurization wave propagates around the downcomer annulus and up through the core, the barrel differential pressure reduces and, similarly, the resulting hydraulic forces drop.

In the case of a postulated break in the hot leg, the wave follows a similar depressurization path, passing though the outlet nozzle and directly into the upper internals region, depressurizing the core and entering the downcomer annulus from the bottom exit of the core barrel. Thus, after the hot leg break, the downcomer annulus would be depressurized with very little difference in pressure across the outside diameter of the core barrel.

A hot leg break produces less horizontal force because the depressurization wave travels directly to the inside of the core barrel (so that the downcomer annulus is not directly involved) and internal differential pressures are not as large as for the cold leg break. Since the differential pressure is less for a hot leg break, the horizontal force applied to the core barrel is less for a hot leg break than for a cold leg break. For breaks in both the hot leg and cold leg, the depressurization waves would continue to propagate by reflection and translation through the reactor vessel and loops.

3.9.N.2.5.2 Reactor Internals Modeling and Analysis Method

The finite element modeling of the reactor pressure vessel and its internals system has been discussed in paragraph 3.9.N.1.4.6.3 for the nonlinear time history LOCA analysis. The output from this analysis consists of time history component loads and displacements.

If a simultaneous seismic event with the intensity of the safe shutdown earthquake is postulated with the LOCA transient, the combined effect is determined by considering the maximum stresses for each condition and combining them by the square root sum of the squares.

A summary of the analysis for major components is presented in the following paragraphs.

3.9.N.2.5.2.1 Core Barrel. For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

The barrel is then analyzed for dynamic buckling, using the following conservative assumptions:

- The effect of the fluid environment is neglected.
- The shell is treated as simply supported.

During cold leg blowdown, the barrel is subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of the barrel response to cold leg blowdown is performed as follows:

A. The core barrel is analyzed as a shell with two variable sections to model the support flange and core barrel.

B. The barrel, with the core and neutron shield panels, is analyzed as a beam supported at the top and supported at the bottom by the lower radial support; and the dynamic response is obtained.

3.9.N.2.5.2.2 Guide Tubes. The dynamic loads on rod cluster control assembly (RCCA) guides are more severe for a LOCA caused by hot leg branch pipe rupture than for an accident caused by cold leg branch pipe rupture, since the cold leg break leads to much smaller changes in the transverse coolant flow over the RCCA guides. The guide tubes in closest proximity to the outlet nozzle of the ruptured loop are the most severely loaded during a blowdown. The transverse guide tube forces decrease with increasing distance from the ruptured nozzle location. The stresses due to the SSE (vertical and horizontal components) are combined with the blowdown (LOCA) stresses in order to obtain principal stresses and stress intensities to be compared with the allowables.

3.9.N.2.5.2.3 Upper Support Columns. Upper support columns located close to the broken nozzle during a hot leg break are subjected to transverse loads due to cross-flow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes (i.e., by taking into consideration the increase in flow across the column during the accident). The columns are studied as beams with variable sections, and the resulting stresses are obtained using the reduced section modulus and appropriate stress risers for the various sections.

The stresses due to the SSE (vertical and horizontal components) are combined with the blowdown stresses in order to obtain principal stresses and deflection.

3.9.N.2.5.2.4 Results of Reactor Internals Analysis. The system seismic analysis of the reactor vessel and its internals can either be performed by a response spectrum analysis method or by a time history integration method. Both of these analyses techniques are the NRC accepted methodologies. For certain systems or components, when time dependent seismic response is desired, the nonlinear time history analysis is used. The seismic time history analysis technique is essentially the same as that discussed earlier for the LOCA analysis, except that in seismic analysis, time history accelerations are used as the forcing function. The seismic response is then combined with the LOCA response in a conservative manner in order to obtain the maximum stresses and deflections.

All reactor internal components were found to be within acceptable stress and deflection limits for both the cold leg and hot leg LOCA transients when combined with the SSE condition.

3.9.N.2.5.2.5 Control Rod Insertability. During full power plant operation, all RCCAs and the corresponding drive rod assemblies are held at a fully withdrawn position by their respective control rod drive mechanisms (CRDMs). During certain accident conditions, such as small break LOCA (SBLOCA) and/or an SSE condition, all RCCAs are assumed to drop from their fully inserted position. It has been confirmed that all of the guide tubes are designed to maintain the function of the control rods for SBLOCA (break size of 144 in^2. and smaller). No credit for the function of the control rods is assumed for large breaks (break size areas above 144 in^2.). However, the design of the guide tubes permits control operation in all but four control rod positions, which is sufficient to maintain the core in a subcritical configuration, for break sizes up to a double-ended hot leg break. This double-ended hot leg break imposes the limiting lateral guide tube loading.

3.9.N.2.6 Correlation of Reactor Internals Vibration Tests with the Analytical Results

As stated in paragraph 3.9.N.2.3, it is not considered necessary to conduct instrumented tests of the VEGP reactor vessel internals. References 2 and 3 describe predicted vibration behavior based on studies performed prior to the plant tests. These studies, which utilize analytical models, scale model test results, component tests, and results of previous plant tests, are used to characterize the forcing functions and establish component structural characteristics so that the flow-induced vibratory behavior and response levels for VEGP reactor internals are estimated. These estimates are supported by values deduced from plant test data obtained from the Sequoyah and Trojan internals vibration measurement programs. Adequacy of the VEGP internals is verified by the use of Sequoyah and Trojan results supported by scale model tests and analytical work.

3.9.N.2.7 References

1. Bloyd, C. N., Ciaramitaro, W., and Singleton, N. R., "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8766 (Proprietary) and WCAP-8780, (Nonproprietary), May 1976.

2. Lee, H., "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," WCAP-8803-P-A (Proprietary) and WCAP-8317-A (Nonproprietary), July 1975.

3. Bloyd, C. N., and Singleton, N. R., "UHI Plant Internals Vibration Measurement Program and Pre- and Post-Hot Functional Examinations," WCAP-8516-P (Proprietary) and WCAP-8517 (Nonproprietary), March 1975.

4. Altman, D. A., and Singleton, N. R., "Preliminary Results of Sequoyah Unit 1 Internals Vibration Measurement Program," Westinghouse Letter, PEE-RI-957, September 10, 1979, (Proprietary Class 2).

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3.9.B.3 AMERICAN SOCIETY OF MECHANICAL ENGINEERS CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURE

The following subsection, 3.9.3, applies to all components and component supports including Westinghouse scope of supply except as noted.

For plant conditions and loading combinations, the requirements of Regulatory Guide 1.48 are met as summarized in section 1.9.

The following ASME Editions and Addenda are applicable to piping stress analysis and pipe support design:

- The design of Code Class 1 piping is in accordance with Subsection NB of the 1977 Edition, including Addenda through Summer 1979, with the following exception:

  Code Class 1 as-built reconciliation analysis of pipe supports on the pressurizer safety and relief valve system is evaluated in accordance with NF-3000 from the 1983 Edition.

- Code Class 2 and 3 piping is designed to Subsections NC and ND of the 1974 Edition, including Addenda through Summer of 1975 with the following exceptions:

  1. Code Class 2 and 3 flanged joints are evaluated in accordance with NC/ND-3658 from the 1977 Edition with Addenda through Summer 1979.
  2. Coefficients of thermal expansion and modulus of elasticity values may be taken from Appendix I, Table I-5.0 and I-6.0 of the 1977 Edition with Addenda through Summer 1979.
  3. Stress indices and stress intensification factors may be taken from Subparagraph NC/ND-3673.2 of the 1980 Edition, Winter 1980 Addendum. For circumferential fillet welded or socket welded joints, the stress intensification factors may be taken from figure NC/ND-3673.2(b)-1 of the 1983 Edition, Summer 1983 Addendum.
  4. Code Class 2 small bore piping stress analysis may be evaluated in accordance with NC-3652, -3653, 3654, and -3655 from the 1980 Edition with Addenda through Winter 1981, and Code Class 3 small bore piping

stress analysis may be evaluated in accordance with ND-3652, -3653, -3654, and -3655 from the 1983 Edition with Addenda through Summer 1984.

5. Code Class 2 and 3 piping design for single nonrepeated anchor movement (e.g., predicted building settlement) are evaluated in accordance with NC/ND-3650 from the 1977 Edition with Addenda through Summer 1979.

6. Later editions of the ASME Code may be used in accordance with 10 CFR 50.55a(b)(1) when properly reconciled to the Code of Record.

- Code Class 1, 2, and 3 component supports are designed to Subsection NF of the 1974 Edition, including Addenda through Summer of 1975, with the following exceptions:

1. The stress limitation of 0.3 Sy for subparagraphs NF-3226.5, NF-3321.1(c), NF-3392.1, and Mandatory Appendix XVII, Figure XVII-2211(c) of the 1974 Edition with Addenda through Summer 1975 are excluded.

2. Nuclear service shock arrestors/snubbers are designed to Subsection NF of the 1977 Edition including Addenda through Winter of 1977.

3. The jurisdictional boundary between the pressure-retaining component and the component support is established in accordance with the applicable subparagraphs of Paragraphs NF-1131 and NF-1132 of the Code.

4. Supporting structures are extensions of the building and are not considered within the jurisdiction of ASME Section III, Subsection NF. The component supports are attached to these members. The supporting structures are constructed in accordance with the rules of the AISC codes. The pipe support configurations shown in figures 3.9.B.3-1 through 3.9.B.3-3 are samples and are only intended to show weld locations and NF jurisdictional boundaries. Embedded anchor bolts, washers, nuts, and concrete expansion anchors are defined as part of the building structure.

- In general, the basis used for the design and construction of both the component support and the supplementary steel are as follows.

1. All materials used in the supporting structural steel members, both safety related and nonsafety related, are ASME Code SA-36 or equivalent ASTM type material and are furnished with certificates of compliance and/or certified material test reports as required by the structural steel procurement/construction specification.

2. The structural tubing used for pipe support is A-500 GR. B material.

3. Embedded anchor bolts and concrete expansion anchors are not within the scope of ASME Section III. The stress analysis of the bolt (diameter sizing) is in accordance with Subsection NF using the design loads derived from the component ASME III design specification and the allowable stresses specified in Subsection NF for the selected material.

4. The embedded anchor bolt material is selected from the materials listed in Appendix I of ASME Section III, Division 1 Code or requested in accordance with NA/NCA-1140. Concrete expansion anchors are made

from materials having mechanical properties adequate to support the expected loadings.

5. Supporting structural steel members are welded by ASME code Section IX qualified welders.
6. Welding procedures are qualified in accordance with ASME Section IX and welding materials conform to ASME Section II, Part C requirements.
7. All supporting structural steel members are examined in accordance with the acceptance criteria and methods of Article NF-5000 of the Code.
8. The component support materials are selected from the materials listed in Appendix I of Section III, Division 1 of the ASME Boiler & Pressure Vessel Code.
9. All supports are designed to carry the loads imposed by the piping systems.
10. The amount of clearance shown on the pipe support drawing is as applicable to the erected (cold) system, but calculated based on the pipe at operating temperature and the framing structure at ambient temperature.
11. Friction is considered if known thermal movement is larger than 1/16 of an inch. The load due to friction is applied at the point of contact in the direction of thermal movement.

    Friction reducing devices are used when engineering judgement indicates the friction load may impose excessive stress in the pipe. The equation for determining friction load is:

    $$F = \mu N$$

    where:

    F = friction load acting in the direction of pipe thermal movement.

    N = normal load due to thermal and dead weight.

    μ = coefficient of friction.

    The following coefficient of friction (μ) values are used in calculating the friction load:

Carbon steel against carbon steel	=	0.3
Teflon-based low friction plates	=	0.1

3.9.B.3.1 Loading Combinations, Design Transients, and Stress Limits

American Society of Mechanical Engineers (ASME) Class 1, 2, and 3 components and component supports are designed to an appropriate combination of plant conditions and design loadings. The plant conditions are design, normal, upset, emergency, faulted, and thermal conditions. The design loadings are pressure, temperature, deadweight, seismic, and dynamic loads.

The plant conditions and design loading combinations for ASME Code Class 1 piping and supports are indicated in table 3.9.B.3-1. Stress limits for Class 1 piping and supports are indicated in tables 3.9.B.3-2 and 3.9.B.3-6, respectively.

The plant conditions and design loading combinations for ASME Code Class 2 and 3 components and supports are indicated in table 3.9.B.3-1. Stress limits for Class 2 and 3 piping vessels, valves, pumps, and supports are indicated in tables 3.9.B.3-2, 3.9.B.3-3, 3.9.B.3-4, 3.9.B.3-5, and 3.9.B.3-7, respectively.

The methodology used for combining responses meets the requirements of NUREG-0484, Revision 1.

Piping components in essential ASME Code Class 1, 2, and 3 piping systems designed to level C or D service limits will be shown to retain functionability for emergency and faulted plant conditions by meeting the screening criteria found in reference 1.

3.9.B.3.1.1 Design Load Combinations and the Associated Operating Plant Condition

3.9.B.3.1.2 Design and Normal Conditions

The design and normal conditions are as defined in NB-3112 and NB-3113 of ASME Boiler and Pressure Vessel Code, Section III.

3.9.B.3.1.3 Upset Condition

Loads that are considered in upset plant operating conditions (defined in ASME Section III as those having a high probability of occurrence) include the following:

- Operating pressure.
- Operating temperature.
- Deadweight.
- Open relief valve thrust.
- Transient pressure effects.
  - Fast valve closure.
  - Closed relief valve discharges.
- Operating basis earthquake (OBE).

Although the occurrence of an earthquake cannot be considered highly probable, the number of cycles associated with a seismic event are considered with the low stress allowables of the upset plant condition.

It should be noted that two different types of relief valves are categorized, open and closed discharge, as described in paragraph 3.9.B.3.3.

The open discharge relief valve has a continuous blowdown thrust that can occur for a period of tens of seconds to minutes. The associated piping must be designed for this thrust. Since the maximum stress due to the relief valve thrust occurs over a significant period of time, coincident earthquake and relief stresses are assumed.

However, the case for a closed discharge system is much different. During the long-term blowdown following the establishment of steady-state flow, the reactions on the discharge piping, relief valve, and inlet piping are balanced; and no stresses are introduced as a result of relief valve blowdown. The time duration for the stresses induced during the transient preceding steady-state flow is approximately 500 ms. After this period of time, the effects from valve blowdowns are damped out.

The results of time-history dynamic analysis, using very conservative damping ratios (0.5%), show that 3 to 10 stress cycles occur before steady-state blowdown.

It may be argued that an earthquake can cause a plant trip and consequential relief valve actuation. However, the probability of the maximum stresses from these transients occurring at the same location and at the same time is extremely low.

Further, the number of cycles (3 to 10) during which both are occurring also is extremely low.

Therefore, it is concluded that the combination of loads resulting from the OBE and the transient induced by relief valve actuation in a closed discharge system is not an upset plant condition. The same argument is made for the fast valve closure event. An example of this is the trip of the main steam turbine stop valves.

3.9.B.3.1.4 Emergency Condition

Load combinations falling into this category have a low probability of occurrence. Therefore, a higher design stress is allowed since the number of cycles is low. The coincident effects of OBE and transient pressures (discussed under upset) are evaluated as an emergency plant condition.

3.9.B.3.1.5 Faulted Condition

For the faulted condition, ASME piping, vessels, pumps, and valves are analyzed to the design loading combinations shown in tables 3.9.B.3-1.

3.9.B.3.1.6 Design Stress Limits

3.9.B.3.1.6.1 ASME Code Class 1 Piping. Stress limits on ASME Code Class 1 piping, vessels, valves, and pumps are given in tables 3.9.B.3-2, 3.9.B.3-3, 3.9.B.3-4, and 3.9.B.3-5, respectively.

3.9.B.3.1.6.2 ASME Code Class 2 and 3 Piping. Stress limits on piping are given in the subsequent listing:

A. Normal Condition

Calculated stresses due to sustained loads and thermal expansion shall conform to the requirements of ASME Code, Section III, NC-3600 or ND-3600. For calculated stresses due to occasional loads, the following shall be used.

B. Upset Condition

The sum of stresses produced by loading combinations shown in table 3.9.B.3-1 for the upset condition shall not exceed 1.2 times the allowable stress values given in Tables I-7.1, I-7.2, and I-7.3 of Appendix I of the ASME Code Section III or ND-3600. Under upset conditions equation (9) of NC-3650 or ND-3650 shall be met (permissible pressure ≤1.1 P).

C. Emergency Condition

The sum of the stresses produced by the loading combinations shown in table 3.9.B.3-1 for emergency conditions shall not exceed 1.8 times the allowable stress values given in Tables I-7.1, I-7.2, and I-7.3 of Appendix I of the ASME Code, Section III. Under emergency conditions, Equation (9) of NC-3650 or ND-3650 shall be met using a stress limit of 1.8 S. Equations (8), (10), and (11) shall not be considered.

The permissible pressure shall not exceed 1.5 times the design pressure (P) calculated in accordance with equation (4) of NC-3641.1.

D. Faulted Condition (Code Case 1606)

The sum of the stresses produced by the loading combinations shown in table 3.9.B.3-1 for the faulted condition shall not exceed 2.4 times the allowable stress values given in Tables I-7.1, I-7.2, and I-7.3 of the ASME Code, Section III, Appendix I. Under faulted conditions, Equation (9) of NC-3650 shall be met using a stress limit of 2.4 S. Equations (8), (10), and (11) shall not be considered.

The permissible pressure shall not exceed 2.0 times the design pressure (P) calculated in accordance with Equation (4) of NC-3641.1.

3.9.B.3.1.6.3 ASME Code Class 2 and 3 Components. Stress limits on vessels, valves, and pumps are given in tables 3.9.B.3-3, 3.9.B.3-4, and 3.9.B.3-5, respectively.

3.9.B.3.2 Pump and Valve Operability Assurance

3.9.B.3.2.1 Pumps

The balance of plant (BOP) safety-related active pumps are listed in table 3.9.B.3-8. Safety-related active pumps are subjected to inshop tests which include hydrostatic tests of casing to 150% of the design pressure, and performance tests to determine the following:

- Total developed head.
- Minimum and maximum head.
- Net positive suction head (NPSH) requirements except as noted below.
- Other pump/motor characteristics.

Where applicable, bearing temperature and vibration are monitored during the performance tests. (For the diesel fuel oil transfer pumps, the NPSH data is developed by actual developmental testing of pumps of the same size.) After the pump is installed, it undergoes hydrostatic testing, construction acceptance tests, and preoperational tests. Where applicable, periodic inservice inspection and testing will be performed to verify and assure the functional ability and reliability of the pump for the design life of the plant. Except as noted, those pumps listed in table 3.9.B.3-8 will be included in the Inservice Testing Program (ASME OM Code~~,~~

~~Section XI~~). For those pumps which are not included in the Inservice Testing Program and are listed in Table 3.9.B.3-8, the capability to perform their safety related function will be demonstrated through inclusion in plant maintenance programs, plant procedures and/or Technical Specifications.

In addition to the required testing, the pumps are designed and supplied in accordance with the following specified criteria:

A. In order to ensure that the active pump will not be damaged during the seismic event, the pump manufacturer must demonstrate by test or analysis that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis is performed. The natural frequency of the support is determined and used in conjunction with the project seismic response spectra. The deflection determined from the static shaft analysis is compared with the applicable clearances.

   If the natural frequency is found to be below 33 Hz, a dynamic analysis is performed using a finite element model to determine the amplified input accelerations necessary to perform the shaft analysis. The shaft deflection analyses are performed using the adjusted accelerations and the deflections compared with allowable shaft clearances. Assumptions used for generating the analytical model are verified by test.

B. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to ensure that unacceptable system misalignment cannot occur.

C. To complete the pump qualification, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation within their specified environment, as well as during the maximum seismic event in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975.

   From this, it is concluded that the safety-related pump/motor assemblies will not be damaged, will continue operating under safe shutdown earthquake (SSE) loadings, and will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

3.9.B.3.2.2 Valves

The BOP safety-related active valves are listed in table 3.9.B.3-9. Safety-related active valves are subjected to the following tests:

- Hydrostatic test in accordance with ASME Code, Section III requirements.
- Main seat leakage tests.
- Functional tests to verify that the valve will open and close within the specified time limits when subjected to the design pressure.
- Operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident, environment simulation, etc.) in accordance

with the general format and qualification procedure of IEEE 382-1972, IEEE 323-1974, and IEEE 344-1975.

In addition, static deflection tests on active valve assemblies are performed to ensure that valve internals will not bind because of loads experienced during an SSE. Normal and abnormal environmental conditions to which the valves are subjected are listed in section 3.11.B.

After installation, the valves undergo hydrostatic tests, construction acceptance tests, and preopertional tests. Where applicable, periodic inservice tests are performed to verify and ensure the functional ability of the valve. These tests demonstrate reliability of the valve for the design life of the plant. Except as noted, those valves listed in table 3.9.B.3-9 will be included in the Inservice Testing Program (ASME OM Code~~, Section XI~~). For those valves which are not included in the Inservice Testing Program and are listed in Table 3.9.B.3-9, the capability to perform their safety related function will be demonstrated through inclusion in plant maintenance programs, plant procedures, and/or Technical Specifications.

The valves are designed using either stress analysis (described by ASME Code, Section III) or standard design rules for minimum wall thickness requirements. On rigid (natural frequency above 33 Hz) active valves with extended top works, an analysis is also performed for static equivalent SSE loads applied at the center of gravity of the extended structure. The maximum stress limits allowed in the analyses are those recommended by the ASME for the particular ASME class of valve analyzed.

In addition to these tests and analyses, the operability of the valve during SSE is demonstrated by satisfying the following criteria:

A. Active valves with extended top works are designed to have a first natural frequency greater than 33 Hz. This may be shown by test or by analysis verified by test.

B. The structural integrity of the valve is qualified by requirements that the nozzle loads due to the SSE (emergency or faulted plant condition) are considered as a normal load in the seismic analysis of the active valve, including supports when required to operate after SSE.

C. The motor operators and other electrical appurtenances necessary for operation are qualified for operability during SSE as described above. Seismic qualification of Seismic Category 1 instrumentation and electrical equipment is presented in section 3.10.

D. The complete valve assembly is qualified by test or analysis or both for operability during the SSE. The valve assembly is only qualified by analysis in cases where structural integrity alone is required, as in the case of a nonactive valve.

For the static deflection test, the valve is mounted in a manner that conservatively represents a typical plant installation. The valve includes the actuator and all appurtenances normally attached to the valve when in service.

The extended top works of the valve are subjected to a statically applied equivalent seismic load of 4.5 g horizontally and vertically for line-mounted valves. Vertical load in the static test may be excluded when it is not a contributing factor. Actual g loadings, if known, may also be used for this test with a safety margin.

The load is applied at the center of gravity of the operator, in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static load tests.

The valve is then operated while the equivalent seismic static load is applied, i.e., from the normal operating status to the faulted operating status. The valve must perform its safety-related function within the specified operating time limits.

If the frequency of the valve, by test or analysis, is less than 33 Hz, a dynamic analysis of the piping system using a finite element model of the flexible valve is performed to determine the equivalent acceleration, considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The equivalent acceleration is then used in the static analysis and the valve seismic operability test.

The static deflection test applies only to valves with overhanging structures, e.g., the operator. The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types (e.g., motor-operated valve, solenoid-operated valve) are tested. Valve sizes that cover the range of sizes in service are qualified by the tests, and the results are used to qualify all valves within the intermediate range of sizes. Stress and deformation analyses are used to support the interpolation.

Degraded conditions as discussed in SRP Section 3.10, paragraph II.1.a(2) are minimized during system design to preclude the presence of debris, impurities, and contaminants in the fluid system. For example, containment sump design considers the presence of debris as described in Final Safety Analysis Report (FSAR) subsection 6.2.2. Other degraded conditions, such as motive power fluctuations, air pressure, etc., are addressed by testing or analysis showing that sufficient margin was included in the design of the equipment to perform its function.

Safety-related valves that can be classified as not having an overhanging structure, such as some check valves, are considered below. With the exceptions of counterweighted check valves, which have overhung structures, check valves are characteristically simple in design. The effects from seismic accelerations or the maximum applied nozzle loads on their operation are negligible. The check valve design is compact, and there are no extended structures or masses whose motion can cause distortions that can restrict operation of the valve. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve, since the valve disc is designed to be isolated from the casing wall. The clearance supplied around the disc by the design prevents the disc from becoming bound or restricted because of any casing distortions caused by nozzle loads. Therefore, the structural integrity of these valves is assured by standard design or analysis methods, and the ability of the valve to operate is ensured by the design features. In addition to these design considerations, the valve also undergoes the following tests and analyses:

- Stress analysis as a part of the piping system including the SSE loads.
- Inshop hydrostatic test.
- Inshop seat leakage test.
- Periodic <u>in situ</u> valve exercising and inspection to ensure the functional ability of the valve as noted in table 3.9.B.3-9.

The following criteria are used to qualify BOP active check valves for their service conditions. Load conditions are addressed through the dynamic analysis of the piping in which the valve is installed as described in FSAR subsection 3.9.2. Valve suppliers assure by analysis that the valve is stronger than the pipe (i.e., section modulus is at least 110% of the connected pipe and the allowable stress for the valve body material is equal to or greater than the allowable stress of the connected piping material). Reverse flow conditions (transients) are considered for active valves required to close to perform their safety function. For check valves containing critical environmentally degradable organic components, environmental conditions are also addressed through stringent selection of materials for use under specific environmental conditions. Individual check valve drawings and bills of material were reviewed to identify all age-sensitive parts and their materials. The parts were then evaluated as to their criticality within the check valves. After identifying the critical parts, the check valve location was reviewed for environmental conditions. Qualification testing of check valves is not required for service conditions because of their design configuration.

Turbulence is minimized as much as practicable by the following guidelines. BOP check valves are specified as equal to the line size. This minimizes pressure loss in the system. Furthermore, piping velocity guidelines, used by the project to produce cost-effective system designs, generally envelop the velocities necessary to fully open system check valves.

Check valves are generally located in horizontal pipe runs and the disks are oriented in a vertical position. Also, check valves are located to minimize the effects of flow disruptive devices while considering other important factors such as containment isolation.

Conformance to Regulatory Guide 1.148, concerning active valve assemblies is addressed in table 3.9.B.3-10. In addition, procurement specifications for replacement components (active safety related) will be consistent with the original purchased equipment to the extent of conformance discussed.

3.9.B.3.3 Design and Installation Criteria, Pressure-Relieving Devices

Pressure vessels are protected by pressure-relieving devices to meet applicable code requirements such as ASME Codes, Sections III and VIII, and American National Standards Institute (ANSI) B31.1 with exception to the following relief systems:

The relief systems discharging into the recycle hold-up tank and the thermal relief systems for the regenerative heat exchanger and residual heat removal loop suction valves have manual block valves installed which are physically locked in the open position. These valves are verified to be locked open by plant procedure to ensure overpressure protection is not defeated during normal plant operations. These manual block valves are typically used to facilitate maintenance or hydrostatic testing.

The design of pressure-relieving devices generally can be grouped in two categories, open and closed systems.

3.9.B.3.3.1 Open System

Design requirements for this piping follow the recommendations of Regulatory Guide 1.67 and its referent, Code Case 1569 (N-40).

The relief valve discharge system to atmosphere for the main steam lines is an example of an open system.

Each main steam line is designed to withstand the maximum loads possible from any or all relief valves discharging at full capacity. The design of a safety and relief valve system includes consideration of all components of the system:

- Safety or relief valve.
- Upstream piping or header.
- Downstream or vent piping.

- System support.
- Structures or building to which the supports are attached.

The most severe load combination is considered as follows:

- Internal pressure.
- Deadweight
- Seismic.
- Thermal.
- Reaction forces of blowing valves including entrainment.

The maximum allowable internal pressure in the main steam piping or header at the safety valve inlet nozzle is 110% of the steam generator shell design pressure as specified by the ASME Code.

Relief valve connections will be spaced on the header so that there is no local interaction.

Reaction force and moment effects on the steam header, supports, and connecting nozzles for each valve blowing and for combinations of valves blowing shall be considered. The steady blowdown load is not transmitted to the header but is carried by the structure, using a piston-type design.

The reaction force of the flowing valve shall be obtained from the valve manufacturer; however, the manufacturer's reaction force is verified by the total hydraulic reaction force analysis for a discharging jet of fluid, comprised of a pressure area contribution and fluid momentum contribution, referring to the outlet plane of the flow geometry.

Dynamic amplification of the reaction force is considered using a dynamic load factor of 2.0.

Stress analysis of the safety and relief valve system is conducted including evaluation of the header local stresses due to reaction moment when applicable. The stresses are categorized according to the appropriate code.

Material thicknesses are selected to accommodate expected loads and maintain stresses within allowable limits.

3.9.B.3.3.2 Closed System

A closed discharge system is characterized by piping between the valve and a tank or some other terminal. Under steady-state conditions, there are no net unbalanced forces. The initial transient response and resulting stresses are determined, using either a time-history computer solution or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included. If required, water slug effects are also included.

3.9.B.3.4 <u>Component Supports</u>

3.9.B.3.4.1 Supports Furnished with the Nuclear Steam Supply System (NSSS)

Refer to paragraph 3.9.N.3.4.

3.9.B.3.4.2 Supports Not Furnished with the NSSS

The loadings, as specified in the design specifications, are taken into account in designing component supports for ASME Code constructed items. These loadings include but are not limited to the following:

A. Weight of the component and normal contents under operating and test conditions.

B. Weight of the component support.

C. Superimposed loads and reactions induced by the adjacent system components.

D. Dynamic loads, including loads caused by earthquake vibration.

E. Restrained thermal expansion.

F. Anchor and support movement effects.

The combinations of loadings categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted which are specified for the design of supports for ASME Code constructed items are presented in table 3.9.B.3-1. The stress limits which are provided for each plant operating condition are provided in tables 3.9.B.3-6 and 3.9.B.3-7.

All ASME Section III, Class 1, 2, and 3 supports are designed as welded attachments to embedded or surface-mounted plates. Bolting for plates is designed according to American Institute of Steel Construction allowables. In no case do the tensile stresses in bolts exceed the yield stress of the bolting material at temperature.

3.9.B.3.4.2.1 Snubbers Used as Component Supports. The location and size of the snubbers are determined by stress analysis. The stress analysis uses the computer program mentioned in subsection 3.9.B.1 and the loading combinationgiven in table 3.9.B.3-1. The location and line of action of a snubber are selected based on the necessity of limiting seismic stresses in the piping and nozzle loads on equipment. Snubbers are chosen in lieu of rigid supports where restricting thermal growth would induce excessive thermal stresses in the piping or nozzle loads or equipment. The snubbers are constructed to ASME Boiler and Pressure Vessel Code, Section III, Subsection NF standards.

Pertinent requirements have been included in design specifications to demonstrate that the snubbers will perform their required safety function. These requirements include:

- Seismic requirements.
- Normal environmental parameters.
- Accident/post-accident environmental parameters.
- Full-scale performance test to measure pertinent performance requirements.
- Instructions for periodic maintenance (in technical manuals).

Both hydraulic and mechanical snubbers are approved for use in the non-NSSS piping support design. Details of the contents of mechanical snubber design specification are provided below. The hydraulic snubber design specification has similar requirements.

The design specification for mechanical snubbers requires consideration of the following:

A. The mechanical snubber is considered a linear support. Design is in accordance with Subarticle NF-3200 of Section III.

B. A certified stress report or certified load capacity data sheet is furnished showing the load capabilities of the snubber. Verification of the load carrying capability of the snubber is in accordance with NF-3132 of Section III.

C. The service loading of the snubber is equal to or less than the design strength established under listing B above for the particular loading condition.

D. The frictional resistance due to normal thermal movement does not exceed 1% or 5 lb, whichever is greater, for acceleration limiting snubbers and 2% or 10 lb, whichever is greater, for velocity limiting snubbers.

E. The total movement during cyclic loading including lost motion and structural deflection, does not exceed +0.06 in. at any load up to rated load when subjected to cyclic loading in the frequency range of 3 to 33 Hz.

F. The snubbers are designed for normal operation with a temperature range of 40° to 140°F and are capable of providing normal performance when exposed to an abnormal environmental temperature of 400°F for a period no longer than 15 min.

G. The total travel of mechanical snubbers shall be equal to or greater than the requirements shown on the pipe support drawings.

H. The design, procurement, manufacture, inspection, handling, testing, storage, and shipping of units and their component parts are performed in accordance with ASME Code and the quality assurance program and the vendor's standard quality assurance procedures.

The design specification requires that an installation manual be provided by the manufacturer to ensure correct installation, including dimensional detailed drawings giving materials of construction with installation and adjustment instruction. Visual confirmation and inspection are required in the field. Also, the hot and cold position of the snubbers will be measured during the preoperational testing stage.

There are no formal provisions for accessibility for inspection, testing, and repair or replacement of snubbers. Snubbers are located in order to most efficiently minimize stresses in the components and piping. However, access will be provided for inspection, testing, repair, or replacement by removing obstructions, if necessary.

All non-NSSS snubbers are of the mechanical or hydraulic type. The fabricator of the mechanical non-NSSS snubbers is the Pacific Scientific Company/Anchor Darling Company. The fabricator of the hydraulic non-NSSS snubbers is Lisega. The function of the mechanical and hydraulic snubbers is shock arrest.

Two types of tests are performed on the snubber.

A. Production tests are made on every unit:

   1. Check unit to confirm that acceleration or velocity (as applicable) levels are less than specified maximum.

   2. Check unit to confirm that it operates freely over the total stroke.

   3. Measure and record the force required to initiate motion over the stroke in tension and compression.

   4. Measure and record lost motion of the snubber mechanisms.

B. Qualification tests are performed on randomly selected production models. The tests are used to demonstrate the required load performance (load rating).

These tests include dynamic load cycling, low temperature, high temperature, humidity, salt spray, sand, dust, life test, and faulted load test.

In the piping system seismic stress analysis, the snubbers are modeled as stops. Where necessary, the snubber spring rates are incorporated into the analysis. There is no impact on the performance of the mechanical snubber by entrapped air or temperature on fluid properties.

Hydraulic snubbers approved for use have design features to prevent air entrapment. Changes in hydraulic snubber performance due to temperature effects on fluid viscosity have been evaluated and found to be acceptable.

The recommendations of Regulatory Guide 1.124 applicable to the service limits and loading combinations for Class 1 linear supports are met as discussed in table 3.9.B.3-6.

Supports for active pumps and valves are included in the overall design and qualification of the component.

3.9.B.3.5 <u>Standard Review Plan Evaluation</u>

Pipe support deformation is controlled by the following methods to ensure that pipe deflections are negligible and have no effect upon the operability requirements of active pumps and valves:

A. Actual pipe support stiffness values are included in the piping analysis computer model of Class 1 piping and interfacing non-Class 1 branch lines, up to and including the first anchor point. Some Class 2 and 3 lines are also analyzed in this manner. With this method, calculated piping nozzle loads reflect any effects due to support deformation.

B. For the balance of the ASME Class 2 and 3 lines, pipe support stiffness or deflection is controlled by either of the two following methods to ensure that component nozzle loads are not affected by pipe support deformation:

   1. Pipe support miscellaneous steel deflections are limited for deadweight and seismic loading to 0.04 in. in each restrained direction for large bore piping (nominal diameter > 2.0 in.) and to 0.0625 in. for small bore piping (nominal diameter ≤ 2.0 in.). These deflections are defined with respect to the structure (e.g., embed, primary structural members) to which the miscellaneous steel is attached. These deflection limits, which ensure adequate stiffness for seismic analysis, are small enough to also ensure that nozzle loads are unaffected by pipe support deformation.

   2. Some small bore (≤ 2.0 in. diameter) piping supports are designed as essentially rigid structures with fundamental frequencies equal to or greater than 25 Hz.

These deformation limits are derived from the guidelines presented in the Bechtel Pipe Support Design Manual (Volume I-1, Section 3.12.2).

As shown in table 3.9.B.3-1, the design basis pipe break is considered a faulted condition. The faulted condition stresses are adequately conservative to ensure safety-related systems perform their function when subjected to faulted condition loads.

3.9.B.3.6 Reference

1. "Functional Capability Criteria for Essential Mark II Piping," GE Topical Report, NEDO-21985, September 1978 (reference NRC Memorandum to R. L. Tedesco, Assistant Director for Licensing, from J. P. Knight, Assistant Director for Components and Structures Engineering, dated July 1980).

3.9.N.3 AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

The ASME Class components are constructed in accordance with the ASME Code, Section III.

A detailed discussion of ASME Code Class 1 components is provided in subsection 3.9.N.1. For core support structures, design loading conditions are given in paragraph 3.9.5.2. Loading conditions are discussed in paragraph 3.9.5.1.

3.9.N.3.1 Loading Combinations, Design Transients, and Stress Limits (for ASME Code Class 2 and 3 Components)

Design pressure, temperature, and other loading conditions that provide the bases for the design of fluid system Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9.N.3.1.1 Design Loading Combinations

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in table 3.9.B.3-1. The design loading combinations are categorized with respect to normal, upset, emergency, and faulted conditions. The responses for all loading combinations defined in table 3.9.B.3-1 for Class 2 and 3 components are combined by the absolute sum method. Stress limits for each of the loading combinations are component oriented and are presented in table 3.9.B.3-3 for vessels, table 3.9.B.3-5 for pumps, table 3.9.B.3-4 for valves, table 3.9.B.3-7 for component supports, and table 3.9.B.3-2 for piping. Active pumps and valves are discussed in paragraph 3.9.B.3.2 and listed in tables 3.9.B.3-8, 3.9.B.3-9, 3.9.N.3-1, and 3.9.N.3-2.[a] Design of component supports is discussed in paragraph 3.9.N.3.4.

3.9.N.3.1.2 Design Stress Limits

The design stress limits established for the components are sufficiently low to ensure that violation of the pressure-retaining boundary does not occur. These limits, for each of the loading combinations, are component oriented and are presented in tables 3.9.B.3-2 through 3.9.B.3-6.

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[a] Active components are those whose operability is relied upon to perform a safety function (as well as a reactor shutdown function) during transients or events considered in the respective operating condition categories.

Inactive components are those whose operability is not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.

3.9.N.3.2 Pump and Valve Operability Assurance

Refer to paragraph 3.10.N.2.2 for discussion of the pump and valve operability program.

3.9.N.3.3 Design and Installation Details in Mounting of Pressure Relief Devices

Safety valves and relief valves are analyzed in accordance with the ASME Section III Code.

The method of analysis for safety valves and relief valves suitably accounts for the time-history of loads acting immediately following a valve opening, i.e., the first few milliseconds. The fluid-induced forcing functions are calculated for each safety valve and relief valve using one-dimensional equations for the conservation of mass, momentum, and energy.

The calculated forcing functions are applied at locations along the associated piping where a change in fluid flow direction occurs. Application of these forcing functions to the associated piping model constitutes the dynamic time-history analysis, referred to as a hydraulic transient analysis, which calculates the dynamic response of the piping system to the forcing functions. Therefore, a dynamic amplification factor is inherently accounted for in the analyses.

Snubbers or strut-type restraints are used, as required, to maintain the stresses resulting from the loads produced by the sudden opening of a relief or safety valve when combined with stress due to other upset loads within allowable limits of the ASME Section III Code for upset conditions. Also, the analyses show that the loads applied to the nozzles of the safety and relief valves do not exceed the maximum loads specified by the manufacturer.

3.9.N.3.3.1 Pressurizer Safety and Relief System – General Description

A. General Description

Special considerations for pressurizer safety and relief valve systems are discussed here.

The pressurizer safety and relief valve discharge piping systems provide overpressure protection for the reactor coolant system. The three springloaded safety valves, located on top of the pressurizer, are designed to prevent system pressure from exceeding design pressure by more than 10%. The two power-operated relief valves, also located on top of the pressurizer, are designed to prevent system pressure from exceeding the normal operating pressure by more than 100 psi. A water seal is maintained upstream of each valve to minimize leakage. Condensate accumulation on the inlet side of each valve prevents any leakage of hydrogen gas or steam through the valves.

The pressurizer safety valves, manufactured by Crosby, are self-actuated, spring-loaded valves with back pressure compensation. The power-operated relief valves, manufactured by Garrett, are solenoid-operated valves, capable of automatic operation via high- pressure signal or remote manual operation. The safety valves and relief valves are located in the pressurizer cubicle and are supported by the attached piping which, in turn, is supported by a system of beams, struts, and snubbers.

If the pressure exceeds the setpoint and the valves open, the water slug from the loop seal discharges. The water slug, driven by high system pressure, generates transient thrust forces. The valve discharge conditions conservatively considered in the analysis of the pressurizer safety and relief valve piping systems are as follows:

1. The three safety valves are assumed to open simultaneously while the relief valves remain closed.
2. The two relief valves open simultaneously while the safety valves are closed.

In addition to these two cases which consider water seal discharge (water slug) followed by steam, solid water from the pressurizer (cold overpressure) is also analyzed.

B. Plant Hydraulic Model

When the pressurizer pressure reaches the set pressure (2475 psia for a safety valve and 2350 psia for a relief valve) and the valve opens, the high pressure steam in the pressurizer forces the water in the water seal loop through the valve and down the piping system to the pressurizer relief tank. Additionally, the power-operated relief valves are subjected to water discharge transients when used for cold overpressure mitigation. For the pressurizer safety and relief piping system, analytical hydraulic models are developed to represent the conditions described above. A Westinghouse proprietary computer program ITCHVALVE is used to perform the transient hydraulic analysis for the system. This program uses the Method of Characteristics approach to generate fluid parameters as a function of time. One-dimensional fluid calculations applying both the implicit and explicit characteristic methods are performed. Using this approach, the piping network is input as a series of single pipes. The network is generally joined together at one or more places by two- or three-way junctions.

Each of the single pipes has associated with its friction factors, angles or elevation and flow areas.

Conservation equations are converted into characteristic equations. The ITCHVALVE computer program incorporates special provisions to allow analysis of valve opening and closing situations.

Fluid acceleration inside the pipe generates reaction forces on all segments of the line. Reaction forces resulting from fluid pressure and momentum variations are calculated. These forces can be expressed in terms of the fluid properties available from the transient hydraulic analysis. The unbalanced forces are calculated using the momentum balance method.

C. Valve Thrust Analysis

The mathematical model used in the seismic analysis is modified for the valve thrust analysis to represent the safety and relief valve discharge. The time-history hydraulic forces from the aforementioned hydraulic analysis are then applied to the piping system lump mass points. The dynamic solution for the valve thrust is obtained by using a modified-predictor-corrector- integration technique and normal mode theory.

The time-history solution is found using program FIXFM3. The input to this program consists of natural frequencies, normal modes, and applied forces. The natural frequencies and normal modes for the pressurizer safety and relief line dynamic model are determined with the WESTDYN program.

Subsequently the time-history displacements of the FIXFM3 program are used as input to the WESTDYN2 program to determine the time-history internal forces and deflections at each end of the piping elements. For this calculation, the displacements are treated as imposed deflections on the pressurizer safety and relief line masses. The solution is stored on tape for later use in the piping stress evaluation and piping support load evaluation.

The time-history internal forces and displacements of the WESTDYN2 program are used as input to the POSDYN2 program to determine the maximum forces, moments, and displacements that exist at each end of the piping elements and the maximum loads for piping supports. The results from program POSDYN2 are saved for future use in piping stress analysis and support load evaluation.

The major structural analyses programs utilized in this static and dynamic analyses are described in WCAP-8252. This was reviewed and approved by the Nuclear Regulatory Commission (NRC letter, April 7, 1981, R. L. Tedesco to T. M. Anderson).

D. Comparison to EPRI Test Results

Piping load data have been generated from the tests conducted by EPRI at the Combustion Engineering test facility. Pertinent tests simulating dynamic opening of the safety valves for representative commercial upstream environments were carried out. The resulting downstream piping loadings and responses were measured. Upstream environments for particular valve opening cases of importance, which envelop the commercial scenarios, are as follows:

1. Cold water discharge followed by steam - steam between the pressure source and the loop seal - cold loop seal between the steam and the valve.
2. Hot water discharge followed by steam - steam between the pressure source and the loop seal - hot loop seal between the steam and the valve.
3. Steam discharge - steam between the pressure source and the valve.

A discussion of the methodology for generating the thermal hydraulic forcing functions and a comparison of analytically determined hydraulic force results to test data was presented in the following article:

> L. C. Smith and K. S. Howe, "Comparison of EPRI Safety Valve Test Data with Analytically Determined Hydraulic Results," The Inter- national Conference on Structural Mechanics in Reactor Technology, Chicago, Illinois, August 22-28, 1983, Volume F, 2/6, pp. 89-96.

A discussion of the methodology utilized in performing a safety valve discharge structural analysis and a comparison of analytical results to structural test results were presented in the following article.

> L. C. Smith and T. M. Adams, "Comparison of Analytically Determined Structural Solutions with EPRI Safety Valve Test Results," 4th National Congress on Pressure Vessel and Piping Technology, Portland, Oregon, June 19-24, 1983, PVP-Volume 74, pp. 193-199.

E. System Evaluation

In order to evaluate the pressurizer safety and relief valve piping, appropriate load combinations and acceptance criteria were developed. The load combinations and acceptance criteria are identical to those recommended by the piping subcommittee of the PWR PSARV EPRI test program and are outlined in tables 3.9.N.3-3 and 3.9.N.3-4 with a definition of load abbreviations provided in table 3.9.N.3-5.

The structural evaluation of the Class 1 piping is conducted consistent with the rules outlined in NB-3650 of the ASME Boiler and Pressure Vessel Code Section III. The piping configuration and downcomer pipe support layout is illustrated in figures 3.9.N.3-1 through 3.9.N.3-3. The piping between the valves and the pressurizer relief tank is analyzed to satisfy the requirements of the appropriate equations of the ANSI B31-1 Code. The load combinations defined in tables 3.9.N.3-3 and 3.9.N.3-4 are utilized. All static and dynamic cases are considered in the support design and evaluation. All pressurizer nozzles, valve flanges, and weld attachments are evaluated per ASME Code rules.

A comparison of calculated valve end loads to the design umbrella values tabulated in the piping design specification is conducted to verify acceptability. Maximum principal stress maximizes bending stress and maximum torsional stress are compared to umbrella operability values if valve operability is required to be demonstrated. For loading cases where only structural integrity has to be demonstrated, a comparison of maximum principal stresses is conducted.

The accelerations at the center of gravity of the valve are limited to the values provided in the design specification for dynamic loadings.

In summary, the operability and structural integrity of the as-built system is assured for all applicable loadings and load combinations including all pertinent safety and relief valve discharge cases.

3.9.N.3.4 <u>Component Supports</u>

The criteria for Westinghouse-supplied supports for ASME Code Class 1 mechanical equipment are presented in subsection 3.9.N.1 and table 3.9.B.3-6.

The criteria for Westinghouse-supplied supports for ASME Code Class 2 and 3 mechanical equipment are discussed in the following paragraphs. (See also the summary in table 3.9.B.3-7.)

3.9.N.3.4.1 Supports for Vessels Procured after July 1, 1974

Class 2 and 3 vessel supports are designed and analyzed to the rules and requirements of ASME III, Subsection NF.

3.9.N.3.4.2 Supports for Vessels Procured Prior to July 1, 1974

A. Linear

1. Normal - The allowable stresses of American Institute of Steel Construction (AISC)-69 Part 1 are employed for normal condition allowables.
2. Upset - Stress limits for upset conditions are 33% higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of AISC-69 Part 1, which permits one-third increase in allowable stresses for wind or seismic loads.
3. Emergency - Not applicable.
4. Faulted - Stress limits for faulted conditions are the same as for the upset condition.

B. Plate and Shell

1. Normal - Normal condition limits are those specified in ASME Section VIII, Division 1 of AISC-69 Part 1.
2. Upset - Stress limits for upset conditions are 33% higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of AISC Part 1, which permits one-third increase in allowable stresses for wind or seismic loads.
3. Emergency - Not applicable.
4. Faulted - Stress limits for faulted conditions are the same as for the upset condition.

3.9.N.3.4.3 Plate and Shell Supports for Pumps

The stress limits used for ASME Code Class 2 and 3 plate and shell component supports are identical to those used for the supported component. These allowable stresses are such that the design requirements for the components and system structural integrity are maintained.

3.9.N.3.4.4 Snubbers Used as Component Supports

The nuclear steam supply system vendor has included pertinent requirements in design specifications to demonstrate that the snubbers will perform their required safety function. These requirements include:

- Seismic requirements.
- Normal environmental parameters.
- Accident/post-accident environmental parameters.
- Full-scale performance test to measure pertinent performance requirements.
- Instructions for periodic maintenance (in technical manuals).

3.9.N.3.5 Standard Review Plan Evaluation

The VEGP FSAR does not fully specify deformation limits of supports where the component supports may affect the operability of the component. The deformation of component supports for active components is not permitted by Westinghouse.

The design basis pipe break is defined by VEGP as a faulted rather than emergency condition. Westinghouse defines design basis pipe break as a faulted (level D) condition consistent with the classification system for plant conditions in American Nuclear Society Standard 18.2. The faulted condition stress limits are sufficiently conservative to ensure the structural integrity and operability of the components and piping to perform their intended safety functions when subjected to faulted condition loads.

3.9.4 CONTROL ROD DRIVE SYSTEM (CRDS)

3.9.4.1 Descriptive Information of CRDS

3.9.4.1.1 Control Rod Drive Mechanism (CRDM)

CRDMs are located on the dome of the reactor vessel. They are coupled to rod control clusters which have absorber material over the entire length of the control rods. The CRDM is shown in figures 3.9.4-1 and 3.9.4-2.

The primary function of the CRDM is to insert or withdraw rod cluster control assemblies (RCCAs) within the core to control average core temperature and during startup and shutdown to control reactivity.

The CRDM is a magnetically operated jack. A magnetic jack is an arrangement of three electromagnets which are energized in a controlled sequence by a power cycler to insert or withdraw RCCAs in the reactor core in discrete steps. Rapid insertion of the RCCAs occurs when electrical power is interrupted.

The CRDM consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

A. The pressure vessel includes a latch housing and a rod travel housing which are connected by a threaded, seal-welded, maintenance joint which facilitates replacement of the latch assembly. The closure at the top of the rod travel housing is a threaded plug with a canopy seal weld for pressure integrity. This closure contains a threaded plug used for venting.

   The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

B. The coil stack assembly includes the coil housings, electrical conduit and connector, and three operating coils: the stationary gripper coil, the movable gripper coil, and the lift coil.

The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

C. The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches: the movable gripper latches and the stationary gripper latches.

The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8-in. steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8-in. step.

D. The drive rod assembly includes a coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

The drive rod has 5/8-in. grooves which receive the latches during holding or moving of the drive rod. The coupling is attached to the drive rod and provides the means for coupling to the RCCA.

The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the RCCA and permit remote disconnection of the drive rod.

The CRDM is designed to release the drive rod and RCCA during any part of the power cycler sequencing if electrical power to the coils is interrupted. When released from the CRDM, the drive rod and RCCA fall by gravity into a shutdown position. The CRDM is threaded and seal welded on an adapter on top of the reactor vessel and is coupled to the RCCA directly below.

The mechanism is capable of raising or lowering a 360-lb load (which includes the drive rod weight) at a rate of 45 in./min. Withdrawal of the drive rod and RCCA is accomplished by magnetic forces, while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 psia. The three operating coils are designed to operate at 392°F with forced-air cooling required to maintain the coil internal temperature at or below 392°F.

The CRDM, shown schematically in figure 3.9.4-2, withdraws and inserts an RCCA as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon-controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation the stationary gripper coil of the drive mechanism holds the RCCA in a static position until a stepping sequence is initiated, at which time the movable gripper coil and lift coil are energized sequentially.

3.9.4.1.2 RCCA Withdrawal

The RCCA is withdrawn by repetition of the following sequence of events (figure 3.9.4-2). The sequence, starting with the stationary gripper energized in the hold position, is as follows:

A. Movable Gripper Coil B - ON

The latch-locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 1/16-in. axial clearance exists between the latch teeth and the drive rod.

B. Stationary Gripper Coil A - OFF

The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 1/16 in. until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

C. Lift Coil C - ON

The 5/8-in. gap between the movable gripper pole and the lift pole closes, and the drive rod assembly raises one step length (5/8 in.).

D. Stationary Gripper Coil A - ON

The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 in. The 1/16-in. vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

E. Movable Gripper Coil B - OFF

The latch-locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

F. Lift Coil C - OFF

The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5/8 in. to a position adjacent to a drive rod assembly groove.

G. Repeat Step

The sequence described above (items A through F) is termed one step or one cycle. The RCCA moves 5/8 in. for each step or cycle. The sequence is repeated at a rate of up to 72 steps/min, and the drive rod assembly (which has a 5/8-in. groove pitch) is raised 72 grooves/min. The RCCA is thus withdrawn at a rate of up to 45 in./min.

3.9.4.1.3 RCCA Insertion

The sequence for RCCA insertion is similar to that for control rod withdrawal, except that the timing of lift coil C ON and OFF is changed to permit the lowering of the control assembly. The sequence, starting with the stationary gripper energized in the hold position, is as follows:

A. Lift Coil C - ON

The 5/8-in. gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

B. Movable Gripper Coil B - ON

The latch-locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A 1/16-in. axial clearance exists between the latch teeth and the drive rod assembly.

C. Stationary Gripper Coil A - OFF

The force of gravity, acting upon the drive rod assembly and attached RCCA, causes the stationary gripper latches and plunger to move downward 1/16 in. until the load of the drive rod assembly and attached RCCA is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

D. Lift Coil C - OFF

The force of gravity and spring force separate the movable gripper pole from the lift pole, and the drive rod assembly and attached RCCA drop down 5/8 in.

E. Stationary Gripper A - ON

The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 in. The 1/16-in. vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

F. Movable Gripper Coil B - OFF

The latch-locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

G. Repeat Step

The sequence is repeated, as for RCCA withdrawal, up to 72 times/min which gives an insertion rate of 45 in./min.

3.9.4.1.4 Holding and Tripping of the Control Rods

During most of the plant operating time, the CRDMs hold the RCCAs withdrawn from the core in a static position. In the holding mode, only one coil, stationary gripper coil A, is energized on each mechanism. The drive rod assembly and attached RCCAs hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weights of the drive rod assembly and the RCCA (plus the stationary gripper return spring) are sufficient to move the latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses, and the stationary gripper plunger half is forced down by the weight of the stationary gripper return spring and the weight acting upon the latches. After the RCCA is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9.4.2 Applicable CRDS Design Specifications

For those components in the CRDS comprising portions of the reactor coolant pressure boundary (RCPB), conformance with the General Design Criteria and 10 CFR 50, section 50.55a is discussed in sections 3.1 and 5.2. Conformance with Regulatory Guides pertaining to materials suitability is discussed in section 4.5 and subsection 5.2.3.

3.9.4.2.1 Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

3.9.4.2.2 Design Stresses

The CRDS is designed to withstand stresses originating from various operating conditions as summarized in table 3.9.B.3-1.

3.9.4.2.2.1 Allowable Stresses. For normal operating conditions Section III of the American Society of Mechanical Engineers (ASME) Code is used. All pressure boundary components are analyzed as Class 1 components.

3.9.4.2.2.2 Dynamic Analysis. The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the CRDS.

3.9.4.2.3 Control Rod Drive Mechanisms

The CRDM pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transient are included in the analysis.

A dynamic seismic analysis is required on the CRDMs when a seismic disturbance has been postulated to confirm the ability of the pressure housing to meet ASME Code, Section III, allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

3.9.4.2.4 CRDM Operational Requirements

The basic operational requirements for the CRDMs are:

A. 5/8-in. step.

B. 147-in. travel.

C. 360-lb maximum load.

D. Step in or out at 45 in./min (72 steps/min).

E. Electrical power interruption initiating release of drive rod assembly.

F. Trip delay time of less than or equal to 150 ms. Free fall of drive rod assembly shall begin less than 150 ms after power interruption, no matter what holding or stepping action is being executed, with any load and coolant temperature of 100°F to 550°F.

G. 40-year design life with normal refurbishment.[a]

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9.4.3.1 Pressure Vessel

The pressure-retaining components are analyzed for loads corresponding to normal, upset, emergency, and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

- Control rod trip (equivalent static load).
- Differential pressure.
- Spring preloads.
- Coolant flow forces (static).
- Temperature gradients.
- Differences in thermal expansion:
  - Due to temperature differences.
  - Due to expansion of different materials.
- Interference between components.
- Vibration (mechanically or hydraulically induced).
- All operational transients listed in table 3.9.N.1-1.
- Pump overspeed.
- Seismic loads (operating basis earthquake (OBE) and safe shutdown earthquake (SSE)).
- Blowdown forces (due to cold and hot leg break).

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

The main objectives of the analysis are to satisfy allowable stress limits, to ensure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to ensure that peak stresses will not reach unacceptable values but also to limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied, using experimental test data and experience from operating reactors.

3.9.4.3.2 Drive Rod Assembly

All postulated failures of the drive rod assemblies, either by fracture or uncoupling, lead to a reduction in reactivity. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with and is guided by the RCCA. This always results in reactivity decrease.

3.9.4.3.3 Latch Assembly and Coil Stack Assembly

3.9.4.3.3.1 <u>Results of Dimensional and Tolerance Analysis.</u>

With respect to the CRDM system as a whole, critical clearances are present in the following areas:

A. Latch assembly - thermal clearances.

B. Latch arm - drive rod clearances.

C. Coil stack assembly - thermal clearances.

D. Coil fit in coil housing.

The following discussion defines clearances that are designed to provide reliable operation in the CRDM in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

A. Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of type 410 stainless steel fit over parts made of type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearances of these parts at 68°F is 0.011 in. At the maximum design temperature of 650°F, minimum clearance is 0.0045 in.; at the maximum expected operating temperatures of 550°F, minimum clearance is 0.0057 in.

B. Latch Arm - Drive Rod Clearances

The CRDM incorporates a load transfer action. The movable or stationary gripper latches are not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9.4-3 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9.4-4 shows clearance variations over the design temperature range.

C. Coil Stack Assembly - Thermal Clearances

The assembly clearances of the coil stack assembly over the latch housing were selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F the inside diameter of the coil stack is 7.308 to 7.298 in. The outside diameter of the latch housing is 7.260 to 7.270 in.

Thermal expansion of the mechanism due to operating temperature of the CRDM results in minimum inside diameter of the coil stack being 7.310 in. at 222°F and the maximum latch housing diameter being 7.302 in. at 532°F.

Under the extreme tolerance conditions listed above, it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

Four coil stack assemblies were removed from four hot CRDMs mounted on 11.035-in. centers on a 550°F test loop, allowed to cool, and then replaced without incident as a test to prove the preceding.

D. Coil Fit in Core Housing

CRDM and coil housing clearances are selected so that coil heatup results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot CRDM.

3.9.4.4 CRDM Performance Assurance Program

The ability of the pressure housing components to perform throughout the design lifetime as defined in the equipment specification is confirmed by the stress analysis report required by the ASME Code, Section III.

Internal components subjected to wear have withstood a minimum of 3,000,000 steps without refurbishment as confirmed by life tests.[(1)] Latch assembly inspection is recommended after 2.5×10^6 steps have been accumulated on a single CRDM.

To confirm the mechanical adequacy of the fuel assembly, the CRDM, and the RCCA, functional test programs have been conducted on a full scale 12-ft control rod. The 12-ft prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 h. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test, the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow-excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the CRDMs meets the design requirement of 2.7 s from start of RCCA motion to dashpot entry. This trip time requirement will be confirmed for each CRDM prior to initial reactor operation and at periodic intervals after initial reactor operation, as required by the Technical Specifications.

There are no significant differences between the prototype CRDMs and the production units. Design materials, tolerances, and fabrication techniques are the same (section 4.5).

These tests have been reported in reference 1.

It is expected that all CRDMs will meet specified operating requirements for the duration of plant life with normal refurbishment. However, a technical specification pertaining to an inoperable

RCCA has been set. Latch assembly inspection is recommended after 2.5 x 10^6 steps have been accumulated on a single CRDM.

If an RCCA cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one RCCA can be tolerated. More than one inoperable RCCA could be tolerated but would impose additional demands on the plant operator. Therefore, the number of inoperable RCCAs has been limited to one, as discussed in the Technical Specifications.

In order to demonstrate proper operation of the CRDM and to ensure acceptable core power distributions, RCCA partial movement checks are performed (Technical Specifications). In addition, periodic drop tests of the RCCA are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical RCCA ejection. During these tests, the acceptable drop time of each assembly is not greater than 2.7 s, at full flow and operating temperature, from the beginning of motion to dashpot entry.

Actual experience in operating Westinghouse plants indicates excellent performance of CRDMs.

All units are production tested prior to shipment to confirm ability of the CRDM to meet design specification-operation requirements.

Each production CRDM undergoes a production test as listed below:

Test	Acceptance Criteria
Cold (ambient) hydrostatic	ASME Code, Section III
Confirm step length and load transfer (stationary gripper to movable gripper or movable gripper to stationary gripper)	Step length: 0.625+0.015-in. axial movement
	Load transfer: 0.047-in. nominal axial movement
Cold (ambient) performance test at design load - five full travel excursions	Operating speed: 45 in./min
	Trip delay: Free fall to drive rod to begin within 150 ms

3.9.4.5 Reference

1. Cooper, F. W., Jr., "17 x 17 Drive Line Components Tests - Phase 1B 11, 111 D-Loop Drop and Deflection," WCAP-8446 (proprietary) and WCAP-8449 (nonproprietary), December 1974.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS[a]

3.9.5.1 Design Arrangements

The VEGP reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support assembly (including the entire core barrel and neutron shield pad assembly), the upper core support assembly, and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms (CRDMs), direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

3.9.5.1.1 Lower Core Support Assembly

The major containment and support member of the reactor internals is the lower core support assembly, shown in figure 3.9.5-1. This assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support, which is welded to the core barrel. All the major material for this structure is type 304 stainless steel. The lower core support assembly is supported at its upper flange from a ledge in the reactor vessel flange, and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support assembly, and principally the core barrel, serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are located. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of type 304 stainless steel and are approximately 48 in. wide by 148 in. long by 2.8 in. thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material

[a] Fatigue of the reactor pressure vessel internals is evaluated as a TLAA for license renewal in accordance with 10 CFR 54.21 (see paragraph 19.4.2.6).

surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guide by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in reference 1.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support and thence through the core barrel shell to the core barrel flange supported by the vessel flange. Transverse loads from earthquake acceleration, coolant cross-flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins, which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by key and keyway joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. Another Inconel insert block is bolted to each of these blocks and has a keyway geometry. Opposite each of these is a key which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. Correct positioning of the internals is ensured by the installation equipment (lifting rig) guide tubes and bushings. With this design, the internals are provided with a support at the furthest extremity and may be viewed as a beam supported at the top and bottom.

Radial and axial expansion of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within American Society of Mechanical Engineers (ASME) Code, Section III, limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy-absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy-absorbing devices of the internals to the vessel.

The energy absorber base plate is contoured on its bottom surface to the reactor vessel bottom geometry. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy-absorbing devices.

3.9.5.1.2 Upper Core Support Assembly

The VEGP upper core support assembly, shown in figures 3.9.5-2 and 3.9.5-3, consists of the upper support, the upper core plate, the support columns, and the guide tube assemblies. The support columns establish the spacing between the upper support and the upper core plate. They are fastened at top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouples. The guide tube assemblies sheath and guide the control rod drive shafts and control rods. They are fastened to the upper support and are restrained by pins in the upper core plate for proper orientation and support. The pins are held in place by a nut locking device assembly. Operation with a missing nut and broken pin has been evaluated and it was concluded that the guide tube assembly would continue to perform its required function.

The upper core support assembly is positioned in its proper orientation with respect to the lower core support assembly by flat-sided pins in the core barrel flange. At an elevation in the core barrel where the upper core plate is positioned, four equally spaced flat-sided pins are located.

Four mating sets of inserts are located in the upper core plate at the same positions. As the upper support assembly is lowered into the lower support assembly, the inserts engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support assembly, the upper core support assembly, the fuel assemblies, and control rods is thereby ensured by this system of locating pins and guidance arrangement. The upper and lower core support assemblies are restrained from any axial movements by a large circumferential spring, which rests between the upper barrel flange and the upper core support assembly and is compressed by installation of the reactor vessel head.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns to the upper support and then into the reactor vessel head. Transverse loads from coolant cross-flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support and upper core plate. The upper support plate is particularly stiff to minimize deflection.

3.9.5.1.3 Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom. (Figure 7.7.1-9 shows the basic flux-mapping system.)

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip connected to inline columns that are in turn fastened to the upper support. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom instrumentation columns which carry the retractable, cold-worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to the thimble seal table. The minimum bend radii are about 144 in., and the trailing ends of the thimbles (at the seal table) are extracted approximately 15 ft during refueling of the reactor. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal table. During normal operation, the retractable thimbles are stationary. They are moved only during refueling or for maintenance, at which time a space of approximately 15 ft above the seal table is cleared for the retraction operation.

The incore instrumentation support structure is designed for support of instrumentation during reactor operation and is rugged enough to resist damage under the conditions imposed during the refueling sequence.

3.9.5.2 <u>Design Loading Conditions</u>

3.9.5.2.1 Normal and Upset Conditions

The normal and upset loading conditions that provide the basis for the design of the reactor internals are:

- Fuel and reactor internals weight.
- Fuel and core component spring forces, including spring preloading forces.
- Differential pressure and coolant flow forces.
- Temperature gradients.
- Vibratory loads including operating basis earthquake (OBE) seismic loads.
- Normal and upset operational thermal transients listed in table 3.9.N.1-1.
- Control rod trip (equivalent static load).
- Loads due to loop(s) out of service.
- Loss of load/pump overspeed.

3.9.5.2.2 Emergency Conditions

The emergency loading conditions that provide the basis for the design of the reactor internals are small loss-of-coolant accident (LOCA), small steam break, and complete loss of flow.

3.9.5.2.3 Faulted Conditions

The faulted loading conditions that provide the basis for the design of the reactor internals are large LOCA and safe shutdown earthquake (SSE).

3.9.5.2.4 Design Loading Categories

The combination of design loadings fit into either the normal, upset, emergency, or faulted conditions as defined in the ASME Code, Section III, as indicated by Figures NG-3221.1 and NG-3224.1 and by Appendix F, Rules for Evaluating Faulted Conditions.

The VEGP units are considered noncode plants with respect to their core support structures and as such are not certified to the requirements of subsection NG of the ASME code. However, in design and manufacture it is the Westinghouse policy to meet the intent of article NG-3000. Concerning other internal structures, article NG-3000 does not specifically apply, but as with core support structures Westinghouse does meet the intent of the code.

3.9.5.3 Design Bases

The scope of the stress analysis problem is large, requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

3.9.5.3.1 Allowable Deflections

Loads and deflections imposed on components as a result of shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses resulting from these dynamic loads and reflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions, as summarized in table 3.9.N.1-1.

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For LOCA plus the SSE condition, the deflection criteria of critical internal structures are the limiting values given in table 3.9.5-1. The corresponding no-loss-of-function limits are included in table 3.9.5-1 for comparison purposes with the allowed criteria.

The criteria for the core drop accident are based upon analyses which determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately 1/2 in. An additional displacement of approximately 3/4 in. would occur from the strain of the energy-absorbing devices of the secondary core support; thus, the total drop distance is about 1 1/4 in., which is insufficient to permit the tips of the rod cluster control assembly (RCCA) to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident involving the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This structure limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15%, after which a positive stop is provided to ensure support.

3.9.5.3.2 Mechanical Design Bases

The design bases for the mechanical design of the VEGP reactor vessel internals components are as follows:

A. The reactor internals in conjunction with the fuel assemblies direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head is provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

B. In addition to neutron shielding provided by the reactor coolant, a separate neutron pad assembly is provided to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.

C. Provisions are made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

D. The core internals are designed to withstand mechanical loads arising from the OBE, SSE, and pipe ruptures and to meet the requirements of item E below.

E. The reactor has mechanical provisions which are sufficient to adequately support the core and internals and to ensure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

F. Following the design basis accident, the plant is capable of being shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in table 3.9.5-1. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in table 3.9.5-1.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in subsection 3.9.N.2.

3.9.5.4 Reference

1. Kraus, S., "Neutron Shielding Pads," WCAP-7870, June 1972.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

Inservice testing of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves will be performed in accordance with ~~Section XI of the ASME Boiler and Pressure Vessel Code~~the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) and applicable addenda, as required by 10 CFR 50.55a(f) (specific edition and addenda of the code delineated in each program), except where specific written relief has been granted by the Nuclear Regulatory Commission (NRC) by 10 CFR 50.55a(f)(6)(i). Additionally, Class 1 component examinations are addressed in subsection 5.2.4, while Class 2 and 3 component examinations are addressed in section 6.6.

The initial inservice test (IST) programs for each unit were submitted to the NRC prior to their respective commercial operation dates. These IST programs were approved and relief granted pursuant to 10 CFR 50.55a(f)(6)(i) where full compliance with ~~the~~ the code then in effect, ASME Code, Section XI, ~~is~~ was not practical. The IST programs include baseline preservice testing and periodic inservice testing to ensure that all applicable pumps and valves are in a state of operational readiness to perform their safety function throughout the life of the plant.

3.9.6.1 Inservice Testing of Pumps

The IST programs include lists of all safety-related Class 1, 2, and 3 pumps that are provided with an emergency power source and are necessary to safely shut down the plant or mitigate the consequences of an accident. The pump testing portion of the IST program conforms to the requirements of ~~Subsection IWP of~~ the ASME OM Code, ~~Section XI,~~ to the extent practical, and complies with all applicable portions of 10 CFR 50.55a(f). In addition, the hydraulic and mechanical test parameters to be measured or observed are included in the IST programs.

3.9.6.2 Inservice Testing of Valves

The IST programs include lists of all safety-related (i.e., those valves necessary to safely shut down the plant or mitigate the consequences of an accident) Class 1, 2, and 3 valves subject to operational readiness and indicate the test parameters to be measured or observed. The valve testing portion of the IST program conforms to the requirements of ~~Subsection IWV of~~ the ASME OM Code, ~~Section XI,~~ to the extent practical, and complies with all applicable portions of 10 CFR 50.55a(f).

3.9.6.3 Relief Request

Relief from the testing requirements of ~~Section XI~~the ASME OM Code will be requested when full compliance with requirements of the code is not practical. In such cases, specific information will be provided which identifies the applicable code requirements, justification for the relief request, and the testing method to be used as an alternative.

TABLE 3.9.N.1-1 (SHEET 1 OF 3)

SUMMARY OF RCS DESIGN TRANSIENTS

Operating Conditions	Occurrences
RCP startup and shutdown	4000
Heatup and cooldown at 100°F/h (pressurizer cooldown at 200°F/h)	200 (each)
Unit loading and unloading between 0 and 15 percent of full power	500 (each)
Unit loading and unloading at 5 percent of full power/min	13,200 unloading 11,200 loading
Reduced temperature return to power	2000
Step-load increase and decrease of 10 percent of full power	2000 (each)
Large step-load decrease with steam dump	200
Steady-state fluctuations	
Initial fluctuations	1.5×10^5
Random fluctuations	3.0×10^6
Boron concentration equalization	26,400
Feedwater cycling	2000
Loop out of service	
Normal loop shutdown	80
Normal loop startup	70
Refueling	80
Turbine roll test	20
Primary side leakage test	200

TABLE 3.9.N.1-1 (SHEET 2 OF 3)

Upset Conditions	Occurrences
Secondary side leakage test	80
Loss of load without immediate reactor trip	80
Loss of power	40
Partial loss of flow	80
Reactor trip from full power	
With no inadvertent cooldown	230
With cooldown and no SI	160
With cooldown and SI	10
Inadvertent RCS depressurization	20
Inadvertent startup of an inactive loop	10
Control rod drop	80
Inadvertent SI actuation	60
Excessive feedwater flow	30
OBE (5 earthquakes of 10 cycles each)	50 cycles
Excessive bypass feedwater flow	30
RCS cold overpressurization	10
Emergency Conditions	
Small LOCA	5
Small steam line break	5
Complete loss of flow	5
Faulted Conditions	
Reactor coolant pipe break (large LOCA)	1

TABLE 3.9.N.1-1 (SHEET 3 OF 3)

Faulted Conditions (continued)	Occurrences
Large steam line break	1
Feedwater line break	1
RCP locked rotor	1
Control rod ejection	1
Steam generator tube rupture	1
Simultaneous steam line - feedwater line break	1
SSE	1
Test Conditions	
Primary side hydrostatic test	10
Secondary side hydrostatic test	10
Tube leakage test	800

TABLE 3.9.N.1-2 (SHEET 1 OF 3)

MONITORED COMPONENTS CYCLIC OR TRANSIENT LIMITS

	TRANSIENT	DESCRIPTION	CYCLIC OR TRANSIENT DESIGN LIMIT
1	RCS heatup	Heatup cycle: T_{avg} from ≤ 200°F to ≥ 500°F	200 heatup cycles at ≤ 100°F/h
2	RCS cooldown	Cooldown cycle: T_{avg} from ≥ 550°F to ≤ 200°F	200 cooldown cycles at ≤ 100°F/h
3	Pressurizer heatup	Pressurizer heatup cycle temperatures from ≤ 200°F to ≥ 650°F	200 pressurizer heatup cycles at ≤ 200°F/h
4	Pressurizer cooldown	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F	200 pressurizer cooldown cycles at ≤ 200°F/h
5	RCS leak tests	Pressurized to ≥ 2485 psig.	200 leak tests
6	RCP-1 startup/shutdown	RCP-1 starts during routine operations plant heatups/cooldowns and due to loop out of service, LOSP, etc.	1000 cycles
7	RCP-2 startup/shutdown	RCP-2 starts during routine operations, plant heatups/cooldowns and due to loop out of service, LOSP, etc.	1000 cycles
8	RCP-3 startup/shutdown	RCP-3 starts during routine operations, plant heatups/cooldowns and due to loop out of service, LOSP, etc.	1000 cycles
9	RCP-4 startup/shutdown	RCP-4 starts during routine operations, plant heatups/cooldowns and due to loop out of service, LOSP, etc.	1000 cycles
10	Turbine trip without reactor trip	Loss of load ≥ 15% of rated thermal power to 0% of rated thermal power	80 loss of load cycles
11	Loss of offsite power	Loss-of-offsite ac electrical ESF system	40 cycles

TABLE 3.9.N.1-2 (SHEET 2 OF 3)

	COMPONENT	DESCRIPTION	CYCLIC OR TRANSIENT DESIGN LIMIT
12	Loss of charging flow in Loop 1	Charging flow terminated to Loop 1	CUF of Normal Charging Nozzle ≤ 1.0
13	Loss of charging flow in Loop 4	Charging flow terminated to Loop 4	CUF of Alternate Charging Nozzle ≤ 1.0
14	Loss of RCS flow in a loop at power	Loss of only one reactor coolant pump	80 cycles
15	Auxiliary spray with ΔT > 320°F	Spray water/ pressurizer temperature differential > 320°F and < 625°F	10 auxiliary spray actuation cycles
16	Plant loading between 0% and 15%	Represented by continuous and uniform ramp power changes	500 cycles
17	Plant unloading loading between 0% and 15%	Represented by continuous and uniform ramp power changes	500 cycles
18	S/G-1 feedwater cycling	Intermittent slug feeding of 32°F auxiliary feedwater at hot standby or no load conditions	CUF of S/G-1 feedwater nozzle and S/G-1 auxiliary feedwater nozzle ≤ 1.0
19	S/G-2 feedwater cycling	Intermittent slug feeding of 32°F auxiliary feedwater at hot standby or no load conditions	CUF of S/G-2 feedwater nozzle and S/G-2 auxiliary feedwater nozzle ≤ 1.0
20	S/G-3 feedwater cycling	Intermittent slug feeding of 32°F auxiliary feedwater at hot standby or no load conditions	CUF of S/G-3 feedwater nozzle and S/G-3 auxiliary feedwater nozzle ≤ 1.0
21	S/G- 4 feedwater cycling	Intermittent slug feeding of 32°F auxiliary feedwater at hot standby or no load conditions	CUF of S/G-4 feedwater nozzle and S/G-4 auxiliary feedwater nozzle ≤ 1.0
22	Refueling	RCS cooled to 140°F, head removed, refueling canal filled	80

TABLE 3.9.N.1-2 (SHEET 3 OF 3)

	COMPONENT	DESCRIPTION	CYCLIC OR TRANSIENT DESIGN LIMIT
23	Secondary side hydrostatic tests S/G-1	Secondary pressurized to ≥ 1481 psig, min water temperature of 120°F	10
24	Secondary side hydrostatic tests S/G-2	Secondary pressurized to ≥ 1481 psig, min water temperature of 120°F	10
25	Secondary side hydrostatic tests S/G-3	Secondary pressurized to ≥ 1481 psig, min water temperature of 120°F	10
26	Secondary side hydrostatic tests S/G-4	Secondary pressurized to ≥ 1481 psig, min water temperature of 120°F	10
27	Reactor trip (with no cooldown)	No immediate cooldown following trip	230
28	Reactor trip (with cooldown and no SI)	Plant cooled down following trip due to continued FWTR flow	160
29	Reactor trip (with cooldown and SI)	Plant cooled down following trip due to continued FWTR flow with safety injection initiation	10
30	Inadvertent safety injection	Spurious safety injection with actuation of high-head centrifugal charging pumps	60
31	Loss of letdown and RTS	Letdown terminated with continued charging	CUF of Normal and Alternate Charging Nozzles ≤ 1.0

TABLE 3.9.N.1-3

MONITORED COMPONENTS FATIGUE LOCATIONS

	Component
1	Alternate charging nozzle
2	Normal charging nozzle
3	Hot leg surge nozzle
4	Pressurizer spray line
5	Pressurizer spray line nozzle
6	Pressurizer surge nozzle
7	Pressurizer surge line
8	S/G 1 auxiliary feedwater nozzle
9	S/G 2 auxiliary feedwater nozzle
10	S/G 3 auxiliary feedwater nozzle
11	S/G 4 auxiliary feedwater nozzle
12	High pressure SI nozzle
13	RCP inlet nozzle
14	RCP outlet nozzle
15	RHR suction nozzle
16	Reactor vessel flange studs
17	Reactor vessel inlet nozzle
18	Reactor vessel outlet nozzle
19	Excess letdown nozzle
20	S/G 1 feedwater nozzle
21	S/G 2 feedwater nozzle
22	S/G 3 feedwater nozzle
23	S/G 4 feedwater nozzle

TABLE 3.9.B.3-1 (SHEET 1 OF 5)

LOAD COMBINATIONS FOR ASME SECTION III, DIVISION 1,
CODE CLASS 1, 2, AND 3 COMPONENT AND COMPONENT SUPPORTS

Plant Condition	Service Level	Components[a]	Component Supports[a][e][q]
Class 1 Components and Component Supports			
Design		PD	DW
Normal	A	PO + DW + TH[d][e]	DW + TH
Upset	B	PO + DW + OBE[e][f]	DW + TH + OBE + Eq
		PO + DW + (OBE + > RVC)[e][f][j]	DW + TH + [(OBE + Eq) + >RVC][j]
		PO + DW + DU[e]	DW + TH + DU
Emergency	C	PO + DW + DE	DW + TH + DE
		PO + DW + SVC[j]	DW + TH + SVC[j]
Faulted	D	PO + DW + SSE	DW + TH + SSE + Eq SSE
		PO + DW + (SSE + >LOCA)	DW + TH + [(SSE + Eq SSE) + >LOCA]
		PO + DW + (SSE + >RVC)[j]	DW + TH + [(SSE + Eq SSE) + >RVC][j]
		PO + DW + (SSE + >SVC)[j]	DW + TH + [(SSE + Eq SSE) + >SVC][j]
		PO + DW + HEB	DW + TH + HEB
		PO + DW + DF	DW + TH + DF

TABLE 3.9.B.3-1 (SHEET 2 OF 5)

Plant Condition	Service Level	Components[a][b]	Component Supports[p][q]
Class 2 and 3 Components and Component Supports			
Design		PD[c]	PD + DW[c][h]
Normal		PO + DW	DW + TH + BS
Upset		PO + DW + OBE[g]	DW + TH + OBE + Eq + BS
		PO + DW + RVC	DW + TH + RVC + Eq + BS
		P0 + DW + (OBE + >RVO)[g]	DW + TH + [(OBE + Eq) + >RVO] + BS
		PO + DW + FV	DW + TH + FV + BS
		PO + DW + DU	DW + TH + DU + BS
Emergency		PO + DW + DE	DW + TH + DE + BS
		PO + DW + (OBE + >RVC)[k][g]	DW + TH + [(OBE + Eq) + >RVC] + BS[k]
		PO + DW + SVC[k]	DW + TH + SVC + BS[k]
Faulted		PO + DW + SSE	DW + TH + SSE + Eq SSE + BS
		PO + DW + (SSE + >LOCA)[n]	DW + TH + [(SSE + Eq SSE) + >LOCA] + BS[n]
		PO + DW + (SSE + >RVO)	DW + TH + [(SSE + Eq SSE) + >RVO] + BS
		PO + DW + FV1	DW + TH + [(SSE + Eq SSE) + >FV1] + BS
		PO + DW + (SSE + >RVC)[k]	DW + TH + [(SSE + Eq SSE) + >RVC] + BS[k]
		PO + DW + (SSE + >SVC)[k]	DW + TH + [(SSE + Eq SSE) + >SVC] + BS[k]
		PO + DW + HEB	DW + TH + HEB + BS
		PO + DW + DF	DW + TH + DF + BS
Thermal		Se + Eq	

a. A "+" represents absolute summation of loading effects other than thermal and weight, which are combined algebraically. Thermal loads are considered only if the sign is the same as the direction for which the loads are calculated. Entire downward weight loads are considered for calculation of downward loads. Seventy-five percent of downward weight loads are considered for calculations of upward loads. A "+ >" represents the square root sum of the squares of loading effects.

b. Also applicable to B31.1 piping; where Seismic Category 1 design requirements do not apply, delete OBE/SSE terminology but include seismic effects per Uniform Building Code requirements.

TABLE 3.9.B.3-1 (SHEET 3 OF 5)

c. Loading effects resulting from internal pressure include those caused by unbalanced expansion joints where applicable; this note applies to all pressure terms (PD or PO).

d. TH (for Class 1 components only) includes localized loadings due to thermal transient events.

e. TH should be added for fatigue evaluation. (Note d applies.)

f. OBE includes effects due to inertially induced motions and seismic anchor motions (for Class 1 fatigue evaluation only); for Class 1 primary stress evaluation, only effects due to inertially induced motions are included in OBE.

g. Loading effects of seismic anchor motions are either included with OBE or added to TH (thermal plant condition).

h. For certain components (e.g., valves designed by standard design rules) utilizing pressure/temperature ratings for design conditions, loading effects due to deadweight are not applicable for the design condition loading combination.

i. For Westinghouse equipment, loads caused by the thermal expansion of attached piping are included in all load combinations.

j. Applicable to pressurizer safety and relief valve piping, Class 1 portion only.

k. Applicable to pressurizer safety and relief valve piping, downstream portion only.

l. Also applicable to Class 2, 3 and ANSI B31.1 piping up to the first anchor, which are included in Class 1.

m. Thermal plant condition. This condition is associated with the thermal expansion stresses and stresses associated with seismic anchor point motion (piping only).

n. For 26-in. mainsteam, 16-in. main feedwater, 6-in. auxiliary feedwater and Class 2 and 3 lines included in Class 1 analysis only.

o. Building settlement piping stress is evaluated independent of plant condition.

p. Piping system loading due to thermal expansion, dynamic events not concurrent with seismic events, and seismic excitation of support mass need not be included in load combinations used to satisfy deflection limits for pipe support miscellaneous steel.

q. The analysis of a piping and support system includes several thermal “cases” which represent different operating conditions of the plant and piping systems.

For example, the analysis of the residual heat removal (RHR) system includes four thermal cases which represent the plant at the following conditions: (1) during RHR operation, (2) during plant at 100% power, (3) during post LOCA cooling, and (4) during an RCS transient event. In addition, the analysis of the safety injection (SI) system includes three thermal cases which represent the plant at the following conditions: (1)during plant at 100% power, (2) during post LOCA cooling, and (3) during an RCS transient event.

Following is the example for faulted conditions for Loop 1 & 4 for RHR piping system due to MSIP loads. (See Westinghouse Calculation Note CN-PAFM-13-10, “Vogtle Unit 1 Auxiliary Piping Evaluation with MSIP Applied to Reactor Vessel Outlet Nozzles,” Rev. 1 (SNC Calculation 1X6MSIP1310, Version 1.0), for details of all load combinations applicable to this note):

Detailed Load Combinations:

Faulted Min = DW + MSIP + TH 100% Power - SRSS [(SSE + SAM) + (LOCA + Jet)]
Faulted Max = DW + MSIP + TH 100% Power + SRSS [(SSE + SAM) + (LOCA + Jet)]

Faulted Min (RHR) = DW + MSIP + Min (TH RHR Normal Operation, or TH RHR Post LOCA) - [(SSE + SAM)]
Faulted Max (RHR) = DW + MSIP + Max (TH RHR Normal Operation, or TH RHR Post LOCA) + [(SSE + SAM)]

TABLE 3.9.B.3-1 (SHEET 4 OF 5)

Abbreviations for Loading Terms

PD	Loadings associated with the design pressure.
PO	Loadings associated with operating pressures including, where applicable, any transient pressures associated with the loading condition event under consideration.
DW	Loadings associated with deadweight and liveweight.
OBE	Inertial loadings associated with the OBE.
Eq	Anchor point displacement loading associated with OBE earthquake.
AN	Single nonrepeated anchor movements (building settlement).
SSE	Inertial loadings associated with the SSE.
Eq SSE	Anchor point displacement loading associated with SSE.
RVC	Transient loadings associated with relief valve blowdown in a closed system.
SVC	Transient loading associated with safety valve in a closed system.
RVO	Sustained loadings associated with relief valve in an open system.
FV	Transient loadings associated with fast valve closure.
TH	Loadings associated with thermal expansion. (Appropriate to each plant condition).[q]
Se	Thermal expansion stress.
LOCA	Loadings associated with pressurizer surge line nozzle, residual heat removal line nozzle, and accumulator line nozzle breaks; such loadings result from the absolute summation of the effects from RCL hydraulic loads, jets, and RCL motions which occur during the break. See section 5.4.14.1.
HEB	Loadings associated with high-energy line pipe breaks other than LOCA (includes loadings from jet impingement, pipe motion, and pipe impact).
DU	Loadings associated with other transient dynamic event classified as an upset condition.
DE	Loadings associated with other transient dynamic event classified as an emergency condition.
DF	Loadings associated with other transient dynamic event classified as a faulted condition.
FV1	Load resulting from main feedwater check valve closure due to pipe break upstream of check valve.

TABLE 3.9.B.3-1 (SHEET 5 OF 5)

BS	-	Loadings associated with building settlement.
MSIP	-	Loadings associated with axial elongation of the RCS hot leg pipe due to the application of the Mechanical Stress Improvement Process.
RHR	-	Residual heat removal (system).

TABLE 3.9.B.3-2

STRESS CRITERIA FOR ASME III CLASS 1, 2, AND 3 PIPING

Stress Limits[a]

Loading Condition	Class 1	Class 2 and 3[b]
Design	NB-3652 (Design)	NC/ND-3611.2(b) (Design)
Normal	NB-3653 (Level A)	NC/ND-3611.2(c) (1) (Level B)
Upset	NB-3654 (Level B)	NC/ND-3611.2(c) (2) (Level B)
Emergency	NB-3655 (Level C)	NC/ND-3611.2(c) (3) (Level C)
Faulted	NB-3656 (Level D)	NC/ND-3611.2(c) (4) (Level D)

a. Limits identified refer to subsections of the ASME Code, Section III.

b. Stress limits of Code Case 1606-1 have been used, the Code Case limits have subsequently been included in ASME Section III (Winter 76 Addenda).

TABLE 3.9.B.3-3

STRESS CRITERIA FOR ASME III
CLASS 1, 2 AND 3 VESSELS

Stress Limits[a]

Loading Condition	Class 1[b]	Class 2[c][d]	Class 2 and 3[d][e]	Class 2 and 3[f]
Design	NC-3221	NB-3217	NC/ND-3321	The vessel shall conform to the requirements of ASME Section VIII, Division 1.
Normal	NB-3222	NC-3217	NC/ND-3321	The vessel shall conform to the requirements of ASME Section VIII, Division 1.
Upset	NB-3223	NC-3217	NC/ND-3321	$\sigma_m \leq 1.1$ s; $(\sigma_m$ or $\sigma_\ell) + \sigma_b \leq 1.65$ s
Emergency	NB-3224	NC-3217	NC/ND-3321	$\sigma_m \leq 1.5$ s; $(\sigma_m$ or $\sigma_\ell) + \sigma_b \leq 1.8$ s
Faulted	NB-3225	NC-3217	NC/ND-3321	$\sigma_m \leq 2.0$ s; $(\sigma_m$ or $\sigma_\ell) + \sigma_b \leq 2.4$ s

a. Limits identified refer to subsections of the ASME Code, Section III.

b. Limits identified also apply to ASME Code Class 2 vessels designed to Division 2 of Section VIII of the ASME Code.

c. Apply for vessels designed in accordance with ASME III, NC-3200.

d. For those vessels procured prior to the incorporation of stress limits into the ASME Code, Section III, the stress limits of Code Case 1607 were used. The code case limits have subsequently been included in ASME III.

e. Apply for vessels designed in accordance with ASME III, NC-3300 or ND-3300.

f. Apply for vessels designed in accordance with the ASME Code, Division 1 of Section III.

TABLE 3.9.B.3-4

STRESS CRITERIA FOR ASME III CLASS 1, 2, AND 3 VALVES

Loading Condition	Stress Limits[a] Class 1	Class 2 and 3[c]
Design	NB-3520 (Design)	NC/ND-3510 (Design)
Normal	NB-3525 (Level A)	NC/ND-3510 (Level A)
Upset	NB-3525 (Level B)	NC/ND-3520 (Level B)
Emergency	NB-3526 (Level C)	NC/ND-3520 (Level C)
Faulted	(b)	NC/ND-3520 (Level D)

a. Limits identified refer to subsections of the ASME Code, Section III.

b. Class 1 valve service level D criteria:

Active	Inactive
Calculate P_m from paragraph NB3545.1 with internal pressure P_s = 1.25 PS $P_m \leq 1.5\ S_m$	Calculate P_m from paragraph NB3545.1 with internal pressure P_s = 1.50 PS $P_m \leq 2.4\ S_m$ or 0.7 S_u
Calculate from S_n from paragraph NB3545.2 with C_p = 1.5; P_s = 1.25PS; Qt2 = 0 P_{ed} = 1.3X value of P from equations of 3545.2(b) (1)	Calculate S_n from paragraph NB3545.2 with C_p = 1.5; P_s = 1.50PS; Qt2 = 0; P_{ed} = 1.3X value of P_{ed} from equations of NB3545.2(b) (1)
$S_n \leq 3\ S_m$	$S_n \leq 3\ S_m$

c. For those valves procured prior to the incorporation of stress limits into the ASME Code, Section III, the stress limits of Code Case N69 (1635-1) were used. The code case limits have subsequently been included in ASME III.

TABLE 3.9.B.3-5

STRESS CRITERIA FOR ASME SECTION III, DIVISION 1,
CODE CLASS 1, 2, AND 3 PUMPS

Loading Condition	Class 1[b]	Stress Limits[a][c] Class 2 and 3 (Active)	Class 2 and 3 (Inactive)
Design	NB-3221 (Design)	ASME Section III NC/ND-3400	ASME Section III NC/ND-3400
Normal	NB-3222 (Level A)	ASME Section III NC/ND-3400	ASME Section III NC/ND-3400
Upset	NB-3223 (Level B)	$\sigma_m \leq 1.0$ S (σ_m or σ_L) + $\sigma_b \leq 1.5$ S $P_{max} \leq 1.1$ PD	$\sigma_m \leq 1.1$ S (σ_m or σ_L) + $\sigma_b \leq 1.65$ S $P_{max} \leq 1.1$ PD
Emergency	NB-3224 (Level C)	$\sigma_m \leq 1.1$ S (σ_m or σ_L) + $\sigma_b \leq 1.65$ S[d] $P_{max} \leq 1.2$ PD	$\sigma_m \leq 1.5$ S (σ_m or σ_L) + $\sigma_b \leq 1.8$ S $P_{max} \leq 1.2$ PD
Faulted	NB-3225 (Level D)	$\sigma_m \leq 1.2$ S (σ_m or σ_L) + $\sigma_b \leq 1.8$ S[d] $P_{max} \leq 1.5$ PD	$\sigma_m \leq 22.0$ S (σ_m or σ_L) + $\sigma_b \leq 2.4$ S $P_{max} \leq 1.5$ PD

S = Allowable stress
σ_m = General primary membrane stress
σ_L = Local membrane stress
σ_b = Primary bending stress
PD = Design pressure
P_{max} = Maximum pressure resulting from upset, emergency, or faulted conditions

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a. Limits identified refer to subsections of the ASME Code, Section III.
b. There are no active Class 1 pumps.
c. For those pumps procured prior to the incorporation of stress limits into the ASME Code, Section III, the stress limits of Code Case N-70 (1636-1) were used. The Code Case limits have subsequently been included in ASME III.
d. Calculated stresses will be verified to be below yield stress. Otherwise, an analysis must be performed showing that the faulted loading condition will cause only minor rubbing or interference of rotating and stationary parts for the duration of the earthquake only and that the pump will return to design point operation immediately following the earthquake.

TABLE 3.9.B.3-6

STRESS LIMITS FOR CLASS 1 COMPONENT SUPPORTS

Support Type	Loading Conditions Design	Normal	Upset	Emergency	Faulted
Plate and shell design by analysis	NF-3221	NF-3222	NF-3223	NF-3224	NF-3225
Linear[a]	NF-3231.1[a]	NF-3231.1[a]	NF-3231.1[a]	NF-3231.1[b]	NF-3231.1[c]
Component standard supports design by analysis	NF-3240	NF-3240	NF-3240	NF-3240	NF-3240
Component supports design by load rating	NF-3260	NF-3260	NF-3260	NF-3260	NF-3260

a. Paragraph numbers refer to ASME Code, Section III, Subsection NF.

b. In instances where the determination of allowable stress values utilizes S (ultimate tensile stress) at temperatures not included in ASME III, S shall be calculated using one of the methods provided in Regulatory Guide 1.124, Revision 1.

c. For equipment procured prior to July 1, 1974, the procedures of AISC-69 or ASME Section VIII are used.

TABLE 3.9.B.3-7 (SHEET 1 OF 3)

STRESS LIMITS FOR CLASS 2 AND 3 COMPONENT SUPPORTS

	Loading Conditions				
Support Type	Design	Normal	Upset	Emergency[f]	Faulted[f]
Plate and shell design by analysis	NF-3321.1	NF-3321.2[a]	NF-3321.2[b]	NF-3321.2[c]	NF-3321.2[d]
Linear	NF-3231.1[a]	NF-3231.1[a]	NF-3231.1[a]	NF-3231.1[b]	NF-3231.1[c]
Component standard supports design by analysis	NF-3321.1 or NF-3231.1[a]	NF-3321.2[a] or NF-3231.1[a]	NF-3321.2[b] or NF-3231.1[a]	NF-3321.2[c] or NF-3231.1[b]	NF-3321.1[d] or NF-3231.1[c]
Component supports design by load rating	NF-3260	NF-3260	NF-3260	NF-3260	NF-3260
Pipe Support Member	Allowable stresses on support members are based on the AISC (AISI for Tube Track) specification. The yield stress at a selected environmental temperature is based on ASME Section III, Appendix I, Table 13.0. However, the minimum permitted design temperature shall be the applicable environmental temperature for the location of the support member. The design temperature for the support member in close proximity to piping shall consider the piping line temperature when it exceeds the environmental temperature.				
Welds	Weld stresses for pipe supports are limited to the allowable tensile stress for plates and elements of rolled sections in the through thickness direction which is 0.4Sy. Sy is the material yield strength at a design temperature of 400°F. In some cases, however, due to the manner of loading, the throat of the weld may be the more critically stressed area. Therefore, weld allowables associated with the applicable code (i.e., ASME III, Subsection NF, Table 3292.1-1[e]; ANSI B31.1; AISC) are used.				

a. Paragraph numbers refer to ASME Code, Section III.

b. In instances where the determination of allowable stress values utilizes S_u (ultimate tensile stress) at temperatures not included in ASME III, S_u shall be calculated using one of the methods provided in Regulatory Guide 1.124, Revision 1.

c. For equipment procured prior to July 1, 1974, the procedures of AISC-69 or ASME Section VIII are used as defined in paragraph 3.9.N.3.4.

d. Supports for active pumps utilize the same limits as those used for the pumps.

e. Use of Table 3292.1-1 is approved by NRC letter from Youngblood to Richard Conway, dated September 29, 1986.

f. The allowable stress values for safety-related supports may be increased in accordance with Subsection NF for items within the ASME III jurisdictional boundary when evaluating emergency and faulted conditions. For items outside the ASME III jurisdictional boundary allowable stresses may be increased by 33.3 percent above the normal allowable. NRC shall be notified when cases of non-ASME supports exceeding the normal stress allowables by more than 33.3 percent occur.

TABLE 3.9.B.3-7 (SHEET 2 OF 3)

(Allowable Axial Stress for Different kℓ/r Ratio)

$$Fa = \frac{\left[1 - \frac{k\ell/(r)^2}{2C_C}\right] Fy}{\frac{5}{3} + \frac{3(k\ell/r)}{8C_C} - \frac{(k\ell/r)^3}{8C_C^{\ 3}}}$$

AISC Equation (1.5 - 1)

when $k\ell/r \le C_C$

where $C_C = \sqrt{\frac{2\pi^2 E_S}{Fy}} = 131.544$

$$Fa = \frac{12\pi^2 E_S}{23(k\ell/r)^2}$$

E a (1 2)

when $k\ell/r > C_C$

ES = 27000 ksi

Fy = 30.8 ksi

kℓ/r	Fa	kℓ/r	Fa	kℓ/r	Fa	kℓ/r	Fa	kℓ/r	Fa
1	18.448	31	17.078	61	15.037	91	12.432	121	9.283
2	18.415	32	17.020	62	14.959	92	12.336	122	9.168
3	18.381	33	16.961	63	14.880	93	12.239	123	9.052
4	18.346	34	16.902	64	14.800	94	12.142	124	8.936
5	18.310	35	16.841	65	14.720	95	12.044	125	8.819
6	18.273	36	16.780	66	14.640	96	11.946	126	8.701
7	18.236	37	16.719	67	14.559	97	11.846	127	8.582
8	18.197	38	16.656	68	14.477	98	11.747	128	8.463
9	18.158	39	16.593	69	14.395	99	11.647	129	8.343
10	18.117	40	16.530	70	14.312	100	11.546	130	8.223
11	18.076	41	16.465	71	14.228	101	11.444	131	8.101
12	18.034	42	16.400	72	14.144	102	11.342	132	7.797
13	17.991	43	16.334	73	14.059	103	11.239	133	7.860
14	17.947	44	16.268	74	13.974	104	11.136	134	7.743
15	17.903	45	16.200	75	13.888	105	11.032	135	7.629
16	17.857	46	16.133	76	13.801	106	10.928	136	7.517
17	17.811	47	16.064	77	13.714	107	10.823	137	7.408
18	17.763	48	15.995	78	13.626	108	10.717	138	7.301
19	17.716	49	15.925	79	13.538	109	10.610	139	7.196
20	17.667	50	15.854	80	13.449	110	10.503	140	7.094
21	17.617	51	15.783	81	13.360	111	10.396	141	6.993
22	17.567	52	15.711	82	13.269	112	10.287	142	6.895
23	17.515	53	15.639	83	13.179	113	10.178	143	6.799
24	17.463	54	15.566	84	13.087	114	10.069	144	6.705
25	17.411	55	15.492	85	12.996	115	9.958	145	6.613
26	17.357	56	15.418	86	12.903	116	9.847	146	6.522
27	17.303	56	15.343	87	12.810	117	9.736	147	6.434
28	17.248	58	15.267	88	12.716	118	9.623	148	6.347
29	17.192	59	15.191	89	12.622	119	9.511	149	6.262
30	17.135	60	15.114	90	12.527	120	9.397	150	6.179

TABLE 3.9.B.3-7 (SHEET 3 OF 3)

Fa	=	allowable axial stress.
k	=	effective length factor.
ℓ	=	length of member.
r	=	minimum radius of gyration.
C_C	=	column slenderness ratio separating elastic and inelastic buckling.
ES	=	modules of elasticity.
Fy	=	yield stress

TABLE 3.9.B.3-8 (SHEET 1 OF 2)

BOP ACTIVE PUMPS

Pump	Equipment Tag Number	Drawing Number
Nuclear service cooling water (NSCW) pump	1-1202-P4-001	1X4DB133-1
NSCW pump	1-1202-P4-002	1X4DB133-2
NSCW pump	1-1202-P4-003	1X4DB133-1
NSCW pump	1-1202-P4-004	1X4DB133-2
NSCW pump	1-1202-P4-005	1X4DB133-1
NSCW pump	1-1202-P4-006	1X4DB133-2
NSCW transfer pump	1-1202-P4-007	1X4DB133-1
NSCW transfer pump	1-1202-P4-008	1X4DB133-2
Component cooling water (CCW) pump	1-1203-P4-001	1X4DB136
CCW pump	1-1203-P4-002	1X4DB136
CCW pump	1-1203-P4-003	1X4DB136
CCW pump	1-1203-P4-004	1X4DB136
CCW pump	1-1203-P4-005	1X4DB136
CCW pump	1-1203-P4-006	1X4DB136
Auxiliary feedwater turbine-driven pump	1-1302-P4-001	1X4DB161-2
Auxiliary feedwater motor-driven pump	1-1302-P4-002	1X4DB161-2
Auxiliary feedwater motor-driven pump	1-1302-P4-003	1X4DB161-2
Diesel fuel oil storage tank pump[a]	1-2403-P4-001	1X4DB170-1
Diesel fuel oil storage tank pump[a]	1-2403-P4-002	1X4DB170-1

TABLE 3.9.B.3-8 (SHEET 2 OF 2)

Pump	Equipment Tag Number	Drawing Number
Diesel fuel oil storage tank pump[a]	1-2403-P4-003	1X4DB170-2
Diesel fuel oil storage tank pump[a]	1-2403-P4-004	1X4DB170-2
Control building ESF chilled water pump	1-1592-P7-001	1X4DB221
Control building chilled water pump	1-1592-P7-002	1X4DB221

a. For these pumps, the capability to perform their safety related function will be demonstrated through inclusion in plant maintenance programs, plant procedures, and/or Technical Specifications in lieu of the Inservice Testing Program.

TABLE 3.9.B.3-9 (SHEET 1 OF 7)

BOP ACTIVE VALVES

Valve No.	System	Active Function[a]	Drawing Number
HV-10950	SIS	(1)	1X4DB120
HV-10951	SIS	(1)	1X4DB120
HV-10952	SIS	(1)	1X4DB120
HV-10953	SIS	(1)	1X4DB120
HV-10957	SIS	(2)	1X4DB121
HV-10958	SIS	(2)	1X4DB121
1204U4262	SIS	(2)	1X4DB121
1204U4263	SIS	(2)	1X4DB121
HV-1668A	NSCW	(2)(3)	1X4DB133-1
HV-1668B	NSCW	(2)(3)	1X4DB133-1
CV-9446	NSCW	(2)(3)	1X4DB133-1
1202U4025	NSCW	(2)(3)	1X4DB133-1
1202U4031	NSCW	(2)(3)	1X4DB133-1
1202U4035	NSCW	(2)(3)	1X4DB133-1
HV-1669B	NSCW	(2)(3)	1X4DB133-2
HV-1669A	NSCW	(2)(3)	1X4DB133-2
CV-9447	NSCW	(2)(3)	1X4DB133-2
1202U4027	NSCW	(2)(3)	1X4DB133-2
1202U4033	NSCW	(2)(3)	1X4DB133-2
1202U4037	NSCW	(2)(3)	1X4DB133-2
1202U4463	NSCW	(2)(3)	1X4DB134
1202U4464	NSCW	(2)(3)	1X4DB133-2
1202U4465	NSCW	(2)(3)	1X4DB133-2
1202U4466	NSCW	(2)(3)(4)	1X4DB135-1
1202U4467	NSCW	(2)(3)(4)	1X4DB135-1
1202U4468	NSCW	(4)	1X4DB135-1
1202U4469	NSCW	(2)(3)	1X4DB135-1
1202U4470	NSCW	(2)(3)	1X4DB135-1
1202U4471	NSCW	(2)(3)(4)	1X4DB135-2
1202U4472	NSCW	(2)(3)(4)	1X4DB135-2
1202U4473	NSCW	(4)	1X4DB135-2
1202U4474	NSCW	(2)(3)	1X4DB135-2
HV-1806[b][c]	NSCW	(2)(3)	1X4DB135-1
HV-1808[b][c]	NSCW	(2)(3)	1X4DB135-1
HV-1822[b][c]	NSCW	(2)(3)	1X4DB135-1
HV-1830[b][c]	NSCW	(2)(3)	1X4DB135-1
HV-2134	NSCW	(2)(3)	1X4DB135-1

TABLE 3.9.B.3-9 (SHEET 2 OF 7)

Valve No.	System	Active Function	Drawing Number
HV-2138	NSCW	(2)(3)	1X4DB135-1
HV-1807[(b) (c)]	NSCW	(2)(3)	1X4DB135-2
HV-1809[(b) (c)]	NSCW	(2)(3)	1X4DB135-2
HV-1823[(b) (c)]	NSCW	(2)(3)	1X4DB135-2
HV-1831[(b) (c)]	NSCW	(2)(3)	1X4DB135-2
HV-2135	NSCW	(2)(3)	1X4DB135-2
HV-2139	NSCW	(2)(3)	1X4DB135-2
HV-1975	ACCW	(4)	1X4DB138-1
HV-1979	ACCW	(4)	1X4DB138-1
HV-1974	ACCW	(4)	1X4DB138-2
HV-1978	ACCW	(4)	1X4DB138-2
HV-19051	ACCW	(4)	1X4DB138-2
HV-19053	ACCW	(4)	1X4DB138-2
HV-19055	ACCW	(4)	1X4DB138-2
HV-19057	ACCW	(4)	1X4DB138-2
1217U4113	ACCW	(1)	1X4DB138-2
1217U4084	ACCW	(4)	1X4DB138-2
1217U4085	ACCW	(4)	1X4DB138-2
1217U4086	ACCW	(4)	1X4DB138-2
1217U4087	ACCW	(4)	1X4DB138-2
HV-3502	Normal sampling	(1)	1X4DB140
HV-3507	Normal sampling	(1)	1X4DB140
HV-3508	Normal sampling	(1)	1X4DB140
HV-3513	Normal sampling	(1)	1X4DB140
HV-3514	Normal sampling	(1)	1X4DB140
HV-3548	Normal sampling	(1)	1X4DB140
HV-780	Drains	(1)	1X4DB143
HV-781	Drains	(1)	1X4DB143
HV-8211	PASS	(1)	1X4DB110
HV-8212	PASS	(1)	1X4DB110
HV-9453	Main steam	(4)	1X4DB159-1
HV-9454	Main steam	(4)	1X4DB159-1
HV-15212C	Main steam	(4)	1X4DB159-1
HV-15212D	Main steam	(4)	1X4DB159-1
HV-3006A/B	Main steam	(4)	1X4DB159-2
HV-3016A/B	Main steam	(4)	1X4DB159-2
HV-3026A/B	Main steam	(4)	1X4DB159-2
HV-3036A/B	Main steam	(4)	1X4DB159-2

TABLE 3.9.B.3-9 (SHEET 3 OF 7)

Valve No.	System	Active Function	Drawing Number
HV-3009	Main steam	(2)	1X4DB159-2
HV-3019	Main steam	(2)	1X4DB159-2
PV-3000	Main steam	(3)	1X4DB159-2
PV-3010	Main steam	(3)	1X4DB159-2
PV-3020	Main steam	(3)	1X4DB159-2
PV-3030	Main steam	(3)	1X4DB159-2
1301U4008	Main steam	(2)	1X4DB159-2
1301U4404	Main steam	(2)	1X4DB159-2
PSV-3001	Main steam	(4)	1X4DB159-2
PSV-3002	Main steam	(4)	1X4DB159-2
PSV-3003	Main steam	(4)	1X4DB159-2
PSV-3004	Main steam	(4)	1X4DB159-2
PSV-3005	Main steam	(4)	1X4DB159-2
PSV-3011	Main steam	(4)	1X4DB159-2
PSV-3012	Main steam	(4)	1X4DB159-2
PSV-3013	Main steam	(4)	1X4DB159-2
PSV-3014	Main steam	(4)	1X4DB159-2
PSV-3015	Main steam	(4)	1X4DB159-2
PSV-3021	Main steam	(4)	1X4DB159-2
PSV-3022	Main steam	(4)	1X4DB159-2
PSV-3023	Main steam	(4)	1X4DB159-2
PSV-3024	Main steam	(4)	1X4DB159-2
PSV-3025	Main steam	(4)	1X4DB159-2
PSV-3031	Main steam	(4)	1X4DB159-2
PSV-3032	Main steam	(4)	1X4DB159-2
PSV-3033	Main steam	(4)	1X4DB159-2
PSV-3034	Main steam	(4)	1X4DB159-2
PSV-3035	Main steam	(4)	1X4DB159-2
HV-9451	Main steam	(4)	1X4DB159-3
HV-9452	Main steam	(4)	1X4DB159-3
HV-15212A	Main steam	(4)	1X4DB159-3
HV-15212B	Main steam	(4)	1X4DB159-3
HV-5113	AFW	(2)(3)	1X4DB161-2
HV-5118	AFW	(2)(3)	1X4DB161-2
HV-5119	AFW	(2)(3)	1X4DB161-2
HV-5120	AFW	(2)(3)	1X4DB161-2
HV-5122	AFW	(2)(3)	1X4DB161-2
HV-5125	AFW	(2)(3)	1X4DB161-2
HV-5127	AFW	(2)(3)	1X4DB161-2
HV-5132	AFW	(2)(3)	1X4DB161-2
HV-5134	AFW	(2)(3)	1X4DB161-2
HV-5137	AFW	(2)(3)	1X4DB161-2

TABLE 3.9.B.3-9 (SHEET 4 OF 7)

Valve No.	System	Active Function	Drawing Number
HV-5139	AFW	(2)(3)	1X4DB161-2
FV-5154	AFW	(2)	1X4DB161-2
FV-5155	AFW	(2)	1X4DB161-2
1302U4013	AFW	(2)(3)	1X4DB161-2
1302U4014	AFW	(2)(3)	1X4DB161-2
1302U4001	AFW	(2)(3)	1X4DB161-2
1302U4002	AFW	(2)(3)	1X4DB161-2
1302U4017	AFW	(2)(3)	1X4DB161-2
1302U4020	AFW	(2)(3)	1X4DB161-2
1302U4023	AFW	(2)(3)	1X4DB161-2
1302U4026	AFW	(2)(3)	1X4DB161-2
1302U4033	AFW	(2)(3)	1X4DB161-2
1302U4037	AFW	(2)(3)	1X4DB161-2
1302U4040	AFW	(2)(3)	1X4DB161-2
1302U4043	AFW	(2)(3)	1X4DB161-2
1302U4046	AFW	(2)(3)	1X4DB161-2
1302U4051	AFW	(2)(3)	1X4DB161-2
1302U4052	AFW	(2)(3)	1X4DB161-2
1302U4058	AFW	(2)(3)	1X4DB161-2
1302U4061	AFW	(2)(3)	1X4DB161-2
HV-5106	AFW	(2)(3)	1X4DB161-3
PV-15129[(b) (d)]	AFW	(2)(3)	1X4DB161-3
SV-15133[(b) (d)]	AFW	(2)(3)	1X4DB161-3
HV-5227	Feedwater	(4)	1X4DB168-3
HV-5228	Feedwater	(4)	1X4DB168-3
HV-5229	Feedwater	(4)	1X4DB168-3
HV-5230	Feedwater	(4)	1X4DB168-3
1305U4071[(b)]	Feedwater	(4)	1X4DB168-3
1305U4073[(b)]	Feedwater	(4)	1X4DB168-3
1305U4075[(b)]	Feedwater	(4)	1X4DB168-3
1305U4077[(b)]	Feedwater	(4)	1X4DB168-3
FV-0510	Feedwater	(4)	1X4DB168-3
FV-0520	Feedwater	(4)	1X4DB168-3
FV-0530	Feedwater	(4)	1X4DB168-3
FV-0540	Feedwater	(4)	1X4DB168-3
LV-5242	Feedwater	(4)	1X4DB168-3
LV-5243	Feedwater	(4)	1X4DB168-3
LV-5244	Feedwater	(4)	1X4DB168-3
LV-5245	Feedwater	(4)	1X4DB168-3
1302U4113	AFW	(2)(3)	1X4DB168-3
1302U4114	AFW	(2)(3)	1X4DB168-3
1302U4115	AFW	(2)(3)	1X4DB168-3
1302U4116	AFW	(2)(3)	1X4DB168-3

TABLE 3.9.B.3-9 (SHEET 5 OF 7)

Valve No.	System	Active Function	Drawing Number
1302U4117	AFW	(4)	1X4DB168-3
1302U4118	AFW	(4)	1X4DB168-3
1302U4119	AFW	(4)	1X4DB168-3
1302U4120	AFW	(4)	1X4DB168-3
1302U4125	AFW	(2)(3)	1X4DB168-3
1302U4126	AFW	(2)(3)	1X4DB168-3
1302U4127	AFW	(2)(3)	1X4DB168-3
1302U4128	AFW	(2)(3)	1X4DB168-3
HV-15196	AFW	(4)	1X4DB168-3
HV-15197	AFW	(4)	1X4DB168-3
HV-15198	AFW	(4)	1X4DB168-3
HV-15199	AFW	(4)	1X4DB168-3
HV-27901	Fire protection	(1)	1X4DB174-4
2301U4036	Fire protection	(1)	1X4DB174-4
HV-9378	Instrument air	(1)	1X4DB186-4
2420U4049	Instrument air	(1)	1X4DB186-4
TV-12124[(b) (e)]	Essential chilled water	(4)	1X4DB233
TV-12125[(b) (e)]	Essential chilled water	(4)	1X4DB234
1203U4030	CCW	(2)(3)	1X4DB136
1203U4032	CCW	(2)(3)	1X4DB136
1203U4034	CCW	(2)(3)	1X4DB136
1203U4055	CCW	(2)(3)	1X4DB136
1203U4057	CCW	(2)(3)	1X4DB136
1203U4059	CCW	(2)(3)	1X4DB136
1592U4188	Essential chilled water	(4)	1X4DB221
1592U4192	Essential chilled water	(4)	1X4DB221
HV-15214[(b)]	CVCS	(4)	1X4DB114

TABLE 3.9.B.3-9 (SHEET 6 OF 7)

Valve No.	System	Active Function	Drawing Number
HV-15216 A/B/C/D	Main steam	(4)	1X4DB159-1,3
HV-13005 A/B	Main steam	(4)	1X4DB159-2
HV-13006 A/B	Main steam	(4)	1X4DB159-2
HV-13007 A/B	Main steam	(4)	1X4DB159-2
HV-13008 A/B	Main steam	(4)	1X4DB159-2
HV-2041	ACCW	(2)(3)	1X4DB138-2
HV-11600	NSCW	(2)(3)	1X4DB133-1
HV-11605	NSCW	(2)(3)	1X4DB133-1
HV-11606	NSCW	(2)(3)	1X4DB133-1
HV-11607	NSCW	(2)(3)	1X4DB133-2
HV-11612	NSCW	(2)(3)	1X4DB133-2
HV-11613	NSCW	(2)(3)	1X4DB133-2
TV-11675[b]	NSCW	(2)(3)	1X4DB135-2
TV-11740[b]	NSCW	(2)(3)	1X4DB134
1202U4A07	NSCW	(4)	1X4DB133-1
1202U4A08	NSCW	(4)	1X4DB133-1
1202U4A09	NSCW	(4)	1X4DB133-1
1202U4A13	NSCW	(4)	1X4DB133-2
1202U4A14	NSCW	(4)	1X4DB133-2
1202U4A15	NSCW	(4)	1X4DB133-2
HV-2626A	CTB Normal Purge	(5)	1X4DB213-1
HV-2627A	CTB Normal Purge	(5)	1X4DB213-1
HV-2628A	CTB Normal Purge	(5)	1X4DB213-1
HV-2629A	CTB Normal Purge	(5)	1X4DB213-1
HV-2626B	CTB Minipurge	(1)	1X4DB213-1
HV-2627B	CTB Minipurge	(1)	1X4DB213-1
HV-2628B	CTB Minipurge	(1)	1X4DB213-1
HV-2629B	CTB Minipurge	(1)	1X4DB213-1
1513U4001	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
1513U4002	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2790A	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2790B	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2791A	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2791B	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2792A	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2792B	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2793A	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-2793B	CTB Hydrogen Recombiner	(1)(4)	1X4DB213-2
HV-12975	Radiation Monitor	(1)(4)	1X4DB213-2
HV-12976	Radiation Monitor	(1)(4)	1X4DB213-2
HV-12977	Radiation Monitor	(1)(4)	1X4DB213-2
HV-12978	Radiation Monitor	(1)(4)	1X4DB213-2
1302U4085	AFW	(2)(3)	1X4DB161-1
1302U4086	AFW	(2)(3)	1X4DB161-1
1302U4087	AFW	(2)(3)	1X4DB161-1

TABLE 3.9.B.3-9 (SHEET 7 OF 7)

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a. Active Function
   (1) Containment isolation.
   (2) Emergency cooling operation.
   (3) Safety-grade cold shutdown.
   (4) Miscellaneous safety-related operations.
   (5) Containment isolation, modes 5 and 6 only.

b. For these valves, the capability to perform their safety-related function will be demonstrated through inclusion in plant maintenance programs, plant procedures, and/or Technical Specifications in lieu of the Inservice Testing (IST) Program.

c. Receives SI signal but does not stroke for an active safety function.

d. These valves are not in the scope of ASME Section III but are included for table completeness.

e. These valves provide an ASME Section III pressure boundary. Valve position is not a time critical function. Operators will have ample time to control cooling.

NOTE: The valve table as shown reflects the IST Program as modified by the second 10-year interval update. The updated IST Program will be fully implemented by May 31, 1998.

TABLE 3.9.B.3-10 (SHEET 1 OF 4)

EVALUATION OF NRC REGULATORY GUIDE 1.148 (MARCH 1981)

Regulatory Guide 1.148 Position	Position on BOP and NSSS Valves
The requirements delineated in ANSI N278.1-1975 (1) are generally acceptable to the NRC staff for functional specifications of active valve assemblies whose operability must be ensured and (2) provide an adequate basis for complying with those requirements of Criterion 1 of Appendix A and Section III of Appendix B to 10 CFR 50 relating to the correct translation into specifications, as supplemented or modified by the following:	In general, the requirements delineated by ANSI N278.1-1975 are addressed as part of the design specification for power-actuated valves designed in accordance with ASME section for Class 1, 2, and 3. Clarifications are given below:
1. Applicability and Relationship With Other Standards	
a. The scope of ANSI N278.1-1975, as stated in Section 1 of the standard, should be supplemented to include active manually operated valve assemblies in systems import to safety.	1.a. No manually operated valves are classified as active. Active valves consist of selected relief valves, check valves, and power-actuated valves. Refer to tables 3.9.N.3-2 and 3.9.B.3-9.
b. The phrase design specification relationship, used in Section 2 of the standard to indicate an inter-relationship with other codes and standards, should be broadly interpreted to include a document that contains sufficient detail to serve as a "complete basis for construction" and may, as appropriate, be called an equipment specification, procurement specification, or some similar term that includes the design specification required by the ASME Code. However, when valve operability is a requirement, the functional specification required by N278.1-1975 should be provided either as part of, or concurrent with, the controlling document that serves as the "complete basis for construction." For ASME Code Class 1, 2, and 3 valves, the "valve design specification" identified in Section 2 of N278.1-1975 should be that required by NCA-3250, Provision of Design Specifications, of Subsection NCA, General Requirements, of Section III of the Code.	1.b. The BOP and NSSS active valve specifications are performance specifications, which are considered a "complete basis for construction." ASME Code Class 1, 2, and 3 valve specifications include design information as required by NCA-3250 of the ASME Code. However, these specifications address installation by reference to drawings and do not include maintenance requirements. Maintenance requirements are included in technical manuals.
c. The valve functional specification should meet the following requirements in the ASME Code:	
(1) It should be uniquely identified and referenced in the design specification in accordance with NCA-3252(a) (6) of Section III of the Code.	1.c. For active valves, one design specification is prepared which meets the applicable ASME Code requirements and contains the functional requirements necessary to ensure that the valve will perform its intended safety function.

TABLE 3.9.B.3-10 (2 OF 4)

Regulatory Guide 1.148 Position	Position on BOP and NSSS Valves
(2) It should reference the applicable design specification to permit identification of both documents.	
(3) It should meet those portions of the filing requirements of NCA-3256 that call for a copy to be filed at the location of the installation and made available to the enforcement authorities having jurisdiction over the plant installation.	
d. The functional specification prepared in accordance with N278.1-1975 for valve assemblies classified as Quality Group D in Regulatory Guide 1.26 should be cross-referenced with the document that serves as a "complete basis for construction." (See Regulatory Position 1.b of this guide.)	1.d. See item 1a.
e. When the valve design specification for ASME Code Class 1, 2, and 3 valve assemblies requires use of ASME Code Cases such as N-62-2, the functional specification should consider this aspect, and usage should be consistent with Regulatory Guide 1.84, Design and Fabrication Code Case Acceptability, ASME Section III Division 1, and Regulatory 1.85, Materials Code Case Acceptability, ASME Section III Division 1.	1.e. Conform.
2. Specific Considerations	
a. Section 3.1, Valve Application Characteristics, should be supplemented by the following:	
(1) "Manually operated valves" (see Regulatory Position 1.a) as a separate item to the listing.	2.a.(1) See position 1.a.
(2) The functional specification should identify the relationship or correspondence between the "application characteristics" of the subject standard and the valve categories A, B, C, and D in IWV-2200 of Subsection IWV, Inservice Testing of Valves in Nuclear Power Plants, of Section XI of the Code.	2.a.(2) Valve specifications do not identify the relationship between the "application characteristics" of the Code. Valve categories are identified for valves subject to Section XI of the Code in the VEGP Preservice Test Program (ISI-P-004).
b. Section 3.2, Structural Requirements, should be supplemented by the following:	2.b.(1) and (2) In general, valve specifications do not completely address the interdependence and number of cycles of the parameters cited or the time relationship between seismic loadings and other loadings.

TABLE 3.9.B.3-10 (3 OF 4)

Regulatory Guide 1.148 Position	Position on BOP and NSSS Valves
(1) The interdependence and number of cycles, if applicable, of time, temperatures, pressures, and dynamic loading resulting from plant transients.	
(2) The time relationship between applied seismic loadings and other concurrent loadings.	
(3) The frequency response spectra for the OBE and the SSE as well as other potential forcing functions such as those from attached piping, pumps, restraints, or other equipment as applied to valves.	2.b.(3) These requirements are provided for active valves in the form of accelerations and nozzle loads which must be met in piping analyses.
(4) The maximum static and dynamic differential pressure (considering all plant operating conditions) that exists across the closure device, including potential water hammer, for which valve assembly operation is to be ensured.	2.b.(4) The maximum differential pressure is provided in specifications for power-actuated valves. Check valve specifications identify the operating and design conditions, but not the maximum differential pressure. The design condition for check valves implies a maximum differential pressure that corresponds to the ANSI pressure-temperature rating in the specifications. Water hammer and other plant conditions are analyzed for selective active valves, if required; but this information is not part of the specification.
c. Section 3.3, Operational Requirements, should be supplemented by the following:	
(1) The first paragraph should be supplemented to require that the desired position of the valve assembly in the event of loss of actuator power be specified, e.g., fail open, fail closed, fail as is.	2.c.(1) Conform.
(2) In item e of Section 3.3.1, Operating Conditions, the phrase "normal and abnormal plant operation" should be interpreted to include the events covered by plant "operational modes (condition)," and by the transient and accident classification of Chapter 15, Accident Analysis, of Regulatory Guide 1.70. The functional specification should state whether the specific valve assembly safety function applies to events defined in the plant operational modes (condition) or in the transient and accident classification. The specification should also indicate whether the actual valves assembly operation (open, close, or regulate fluid flow) occurs during or after the specific event.	2.c.(2) Valve specifications generally do not state whether the valve safety function applies to events defined in the plant operational modes or in the transient and accident classification. The appropriate service conditions are provided in the specification along with the actual valve assembly operation (open, close, or regulate fluid flow).

TABLE 3.9.B.3-10 (4 OF 4)

Regulatory Guide 1.148 Position	Position on BOP and NSSS Valves
(3) Instead of item e of 3.3.2, the following should be used: "e. motor power and duty requirements, including stall current."	2.c.(3) Conform.
(4) Instead of Section 3.3.2.1, Electrical Power for Valve Actuators and Control Elements, the following should be used: "Electrical power shall be identified as ac (single phase or three phase) or dc. Applicable voltage and frequency operating ranges shall be specified."	2.c.(4) Conform.
(5) In Section 3.3.3, Environmental Conditions, the terms "normal" and "abnormal" environmental conditions should be interpreted to mean the environmental conditions that will exist as a result of specified plant conditions. The term "environmental conditions" should be interpreted to include both the environment external to the valve assembly and that of the controlled fluid inside the valve assembly.	2.c.(5) Conform.
d. Section 3.4., Seat Leakage Limits, should be supplemented, as applicable, by the following:	
(1) The leakage limits identified in paragraph b of this section should include identification of fluid, temperature, and differential pressure for which the limit applies.	2.d.(1) Conform.
(5) If valve assembly function requires a limit on overall leakage (e.g., leakage in addition to that of the main seat, such as stem packing and flange), such overall leakage limit should be specified in this section.	2.d.(2) Conform.

TABLE 3.9.N.3-1

ACTIVE PUMPS

Pump	Item Number	System	ANS Safety Class	Normal Mode	Post-Accident Mode	Basis[a]
Centrifugal charging pumps 1 and 2	APCH	CVCS	2	On/Off	On	(1) (2)
Boric acid transfer pumps 1 and 2	APBA	CVCS	3	On/Off	On/Off	(2)
Residual heat removal pumps 1 and 2	APRH	RHRS	2	Off	On	(1) (2)
Safety injection pumps 1 and 2	APSI	SIS	2	Off	On	(1)
Containment spray pumps 1 and 2	APCS	CSS	2	Off	On	(3)
Spent fuel pool[b] pumps 1 and 2	APSF	SFPCS	3	On	On	Spent fuel decay heat removal

a. Basis are defined as follows:

1. Emergency core cooling system safeguards operation.
2. Safety grade cold shutdown operation.
3. Containment spray system safeguards operation.

b. For these pumps, the capability to perform their safety related function will be demonstrated through inclusion in plant maintenance programs, in lieu of the Inservice Testing Program.

TABLE 3.9.N.3-2 (SHEET 1 OF 8)

ACTIVE VALVES

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis [a]
8010A/B/C	RCS	Self-actuated	6	Safety/1	Closed	(1)
PV-455A	RCS	Solenoid	3	Globe/1	Closed	(1) (4)
PV-456A	RCS	Solenoid	3	Globe/1	Closed	(1) (4)
8000A/B	RCS	Motor	3	Gate/1	Open	(1) (4)
8028	RCS	Air	3	Diaphragm/2	Closed	(2)
8033	RCS	Air	1	Diaphragm/2	Closed	(2)
1201U6112	RCS	ΔP	3	Check/2	NA	(2)
8047	RCS	Air	1	Diaphragm/2	NA	(2)
8095A/B	RCS	Solenoid	1	Globe/1	Closed	(1) (4)
8096A/B	RCS	Solenoid	1	Globe/1	Closed	(1) (4)
HV-442A/B	RCS	Solenoid	1	Globe/2	Closed	(4)
8100	CVCS	Motor	2	Globe/2	Open	(2)
8103A/B/C/D	CVCS	Motor	1/2	Globe	Open	(2) (4)
8104	CVCS	Motor	2	Globe/2	Closed	(4)
8105	CVCS	Motor	3	Gate/2	Open	(2) (3) (4)
8106	CVCS	Motor	3	Gate/2	Open	(3) (4)
8110	CVCS	Motor	2	Globe/2	Open	(3) (4)
8111A/B	CVCS	Motor	2	Globe/2	Open	(3) (4)
8112	CVCS	Motor	2	Globe/2	Open	(2)
1208U4021	CVCS	ΔP	3/4	Check/2	NA	(2)
8116	CVCS	Motor	1	Globe/2	Closed	(4)

TABLE 3.9.N.3-2 (SHEET 2 OF 8)

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis
8152	CVCS	Air	3	Globe/2	Open	(2)
8160	CVCS	Air	3	Globe/2	Open	(2)
1208U4437[b]	CVCS	ΔP	1/2	Check/1	NA	(1) (3) (4)
1208U4438[b]	CVCS	ΔP	1/2	Check/1	NA	(1) (3) (4)
1208U4439[b]	CVCS	ΔP	1/2	Check/1	NA	(1) (3) (4)
1208U4440[b]	CVCS	ΔP	1/2	Check/1	NA	(1) (3) (4)
1208U4006[b]	CVCS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1208U4359[b]	CVCS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1208U4360[b]	CVCS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1208U4361[b]	CVCS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1208U4004	CVCS	ΔP	1 1/2	Check/2	NA	(2) (3) (4)
1208U4353	CVCS	ΔP	1 1/2	Check/2	NA	(2) (3) (4)
1208U4354	CVCS	ΔP	1 1/2	Check/2	NA	(2) (3) (4)
1208U4355	CVCS	ΔP	1 1/2	Check/2	NA	(2) (3) (4)
1208U4033[b]	CVCS	ΔP	2	Check/1	NA	(1)
1208U6035[b]	CVCS	ΔP	3	Check/1	NA	(1) (4)

TABLE 3.9.N.3-2 (SHEET 3 OF 8)

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis
1208U6036[(b)]	CVCS	ΔP	3	Check/1	NA	(1) (4)
1208U6037[(b)]	CVCS	ΔP	3	Check/1	NA	(1) (4)
1208U6038[(b)]	CVCS	ΔP	3	Check/1	NA	(1) (4)
1208U6032	CVCS	ΔP	3	Check/2	NA	(2) (4)
8438[(b)]	CVCS	Motor	4	Gate/2	Open	(3) (4)
1208U6124	CVCS	ΔP	4	Check/2	NA	(4)
1208U4185	CVCS	ΔP	2	Check/2	NA	(4)
1208U4299	CVCS	ΔP	2	Check/3	NA	(4)
1208U4140	CVCS	ΔP	2	Check/2	NA	(4)
1208U4147	CVCS	ΔP	2	Check/2	NA	(4)
1208U6142	CVCS	ΔP	4	Check/2	NA	(3) (4)
1208U6149	CVCS	ΔP	4	Check/2	NA	(3) (4)
8471A/B	CVCS	Motor	6	Gate/2	Open	(4)
8485A/B[(b)]	CVCS	Motor	4	Gate/2	Open	(3) (4)
1208U4284	CVCS	ΔP	2	Check/3	NA	(4)
1208U6129	CVCS	ΔP	3	Check/2	NA	(3) (4)
8508A/B	CVCS	Motor	2	Globe/2	Closed/Open	(3)
8509A/B	CVCS	Motor	2	Globe/2	Closed/Open	(3)
1208U6189	CVCS	ΔP	8	Check/2	NA	(3) (4)

TABLE 3.9.N.3-2 (SHEET 4 OF 8)

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis
HV-190A/B	CVCS	Solenoid	1	Globe/2	Closed	(4)
LV-112B/C	CVCS	Motor	4	Gate/2	Open	(3) (4)
LV-112D/E	CVCS	Motor	8	Gate/2	Closed	(3) (4)
LV-0459[(b),(c)]	CVCS	Air	3	Globe/1	Open	(1)
LV-0460[(b),(c)]	CVCS	Air	3	Globe/1	Open	(1)
8701A	RHRS	Motor	12	Gate/1	Closed	(1) (2) (4)
8701B	RHRS	Motor	12	Gate/1	Closed	(1) (4)
8702A	RHRS	Motor	12	Gate/1	Closed	(1) (2) (4)
8702B	RHRS	Motor	12	Gate/1	Closed	(1) (4)
1201U4251	RHRS	ΔP	3/4	Check/2	NA	(4)[(d)]
1201U4252	RHRS	ΔP	3/4	Check/2	NA	(4)
8716A/B	RHRS	Motor	8	Gate/2	Open	(3) (4)
1205U6009	RHRS	ΔP	8	Check/2	NA	(3) (4)
1205U6010	RHRS	ΔP	8	Check/2	NA	(3) (4)
FV-610	RHRS	Motor	3	Gate/2	Open/Closed	(3) (4)
FV-611	RHRS	Motor	3	Gate/2	Open/Closed	(3) (4)
8801A/B	SIS	Motor	4	Gate/2	Closed	(2) (3) (4)
8802A/B	SIS	Motor	4	Gate/2	Closed	(2) (3)
8804A/B	SIS	Motor	8	Gate/2	Closed	(3)
8806	SIS	Motor	8	Gate/2	Open	(3)
8807A/B	SIS	Motor	6	Gate/2	Closed	(3)

TABLE 3.9.N.3-2 (SHEET 5 OF 8)

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis
8809A/B	SIS	Motor	8	Gate/2	Open	(2) (3) (4)
8811A/B	SIS	Motor	14	Gate/2	Closed	(3)
8812A/B	SIS	Motor	12	Gate/2	Open	(3) (4)
8813	SIS	Motor	2	Globe/2	Open	(3)
8814	SIS	Motor	1 1/2	Globe/2	Open	(3)
1204U6013	SIS	ΔP	3	Check/1	NA	(1) (2) (3) (4)
1204U6147	SIS	ΔP	6	Check/1	NA	(1) (2) (3) (4)
1204U6148	SIS	ΔP	6	Check/1	NA	(1) (2) (3) (4)
1204U6149	SIS	ΔP	6	Check/1	NA	(1) (2) (3) (4)
1204U6150	SIS	ΔP	6	Check/1	NA	(1) (2) (3) (4)
1204U4143	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1204U4144	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1204U4145	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1204U4146	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1205U4122	SIS	ΔP	14	Check/2	NA	(3)
1205U4123	SIS	ΔP	14	Check/2	NA	(3)
8821A/B	SIS	Motor	4	Gate/2	Open	(3)
8823	SIS	Air	3/4	Globe/2	Closed	(2)
8824	SIS	Air	3/4	Globe/2	Closed	(2)
8825	SIS	Air	3/4	Globe/2	Closed	(2)
8835	SIS	Motor	4	Gate/2	Open	(2) (3)
8840	SIS	Motor	12	Gate/2	Closed	(2) (3)

TABLE 3.9.N.3-2 (SHEET 6 OF 8)

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis
1204U6128	SIS	ΔP	8	Check/1	NA	(1) (2) (3)
1204U6129	SIS	ΔP	8	Check/1	NA	(1) (2) (3)
8843	SIS	Air	3/4	Globe/2	Closed	(2)
8871	SIS	Air	3/4	Globe/2	Closed	(2)
8875A/B/C/D/ E/F/G/H	SIS	Solenoid	1	Globe/2	Closed	(4)
8880	SIS	Air	1	Globe/2	Closed	(2)
8881	SIS	Air	3/4	Globe/2	Closed	(2)
8888	SIS	Air	3/4	Globe/2	Closed	(2)
8890A/B	SIS	Air	3/4	Globe/2	Closed	(2)
1204U4026	SIS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1204U4027	SIS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1204U4028	SIS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1204U4029	SIS	ΔP	1 1/2	Check/1	NA	(1) (3) (4)
1204U4120	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1204U4121	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1204U4122	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1204U4123	SIS	ΔP	2	Check/1	NA	(1) (2) (3)
1204U4093	SIS	ΔP	1 1/2	Check/2	NA	(3)
1204U4094	SIS	ΔP	1 1/2	Check/2	NA	(3)
1204U6098	SIS	ΔP	4	Check/2	NA	(3)
1204U6099	SIS	ΔP	4	Check/2	NA	(3)

TABLE 3.9.N.3-2 (SHEET 7 OF 8)

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis
8923A/B[(b)]	SIS	Motor	6	Gate/2	Open	(3)
8920	SIS	Motor	1 1/2	Globe/2	Open	(3)
8924	SIS	Motor	6	Gate/2	Open	(3)
1204U6090	SIS	ΔP	8	Check/2	NA	(3)
1204U6083	SIS	ΔP	10	Check/1	NA	(1) (3)
1204U6084	SIS	ΔP	10	Check/1	NA	(1) (3)
1204U6085	SIS	ΔP	10	Check/1	NA	(1) (3)
1204U6086	SIS	ΔP	10	Check/1	NA	(1) (3)
1204U6124	SIS	ΔP	6	Check/1	NA	(1) (3)
1204U6125	SIS	ΔP	6	Check/1	NA	(1) (3)
1204U6126	SIS	ΔP	6	Check/1	NA	(1) (3)
1204U6127	SIS	ΔP	6	Check/1	NA	(1) (3)
1204U6079	SIS	ΔP	10	Check/1	NA	(1) (3)
1204U6080	SIS	ΔP	10	Check/1	NA	(1) (3)
1204U6081	SIS	ΔP	10	Check/1	NA	(1) (3)
1204U6082	SIS	ΔP	10	Check/1	NA	(1) (3)
1205U6001	SIS	ΔP	12	Check/2	NA	(3) (4)
1205U6002	SIS	ΔP	12	Check/2	NA	(3) (4)
8964	SIS	Air	3/4	Globe/2	Closed	(2)
2402U4017	SIS	ΔP	1	Check/2	NA	(2)
1208U6436	SIS	ΔP	8	Check/2	NA	(3)
1204U6163	SIS	ΔP	8	Check/2	NA	(3)

TABLE 3.9.N.3-2 (SHEET 8 OF 8)

Valve Number	System	Actuated By	Size (in.)	Type/ANS Safety Class	Normal Position	Basis
HV-943A/B	SIS	Solenoid	1	Globe/2	Closed	(4)
7126	WPS	Air	3/4	Diaphragm/2	Open	(2)
7136	WPS	Air	3	Diaphragm/2	Open	(2)
7150	WPS	Air	3/4	Diaphragm/2	Open	(2)
7699	WPS	Air	3	Diaphragm/2	Open	(2)
9001A/B	CSS	Motor	8	Gate/2	Closed	(2) (5)
9002A/B	CSS	Motor	10	Gate/2	Closed	(2) (5)
9003A/B	CSS	Motor	10	Gate/2	Closed	(5)
1206U6015	CSS	ΔP	8	Check/2	NA	(2) (5)
1206U6016	CSS	ΔP	8	Check/2	NA	(2) (5)
9017A/B	CSS	Motor	10	Gate/2	Open	(5)
1206U6001	CSS	ΔP	10	Check/2	NA	(5)
1206U6008	CSS	ΔP	10	Check/2	NA	(5)
7603A/B/C/D	SGBD	Air	3	Globe/2	Open	(2) (3)

a. Basis are defined as follows:
   1. Reactor coolant system pressure boundary isolation.
   2. Containment isolation.
   3. ECCS safeguards operations.
   4. Safety grade cold shutdown operation.
   5. CSS safeguards operations.

b. For these valves, the capability to perform their safety related function will be demonstrated through inclusion in plant maintenance programs, plant procedures, and/or Technical Specifications.

c. Receives a nonsafety-related control grade signal, failure to close has no significant impact on plant safety.

d. Unit 1 only.

NOTE: The valve table as shown reflects the inservice test program as modified by the second 10-year interval update. The updated IST program will be fully implemented by May 31, 1998.

TABLE 3.9.N.3-3

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR
PRESSURIZER SAFETY AND RELIEF VALVE PIPING - UPSTREAM OF VALVES

Combination	Plant/System Operating Condition	Load Combination	Piping Allowable Stress Intensity
1	Normal	N	1.5 S_m
2	Upset	N + OBE + SOT_U	1.8 S_m/1.5 S_y
3	Emergency	N + SOT_E	2.25 S_m/1.8 S_y
4	Faulted	N + SSE + SOT_F	3.0 S_m

1. Table 3.9.N.3-5 contains SOT definitions and other load abbreviations.

2. SRSS is to be used for combining dynamic load responses.

TABLE 3.9.N.3-4

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING SEISMICALLY DESIGNED DOWNSTREAM PORTION

Combination	Plant/System Operating Condition	Load Combination	Piping Allowable Stress Intensity
1	Normal	N	1.0 S_h
2	Upset	N + SOT_U	1.2 S_h
3	Upset	N + OBE + SOT_U	1.8 S_h
4	Emergency	N + SOT_E	1.8 S_h
5	Faulted	N + SSE + SOT_F	2.4 S_h

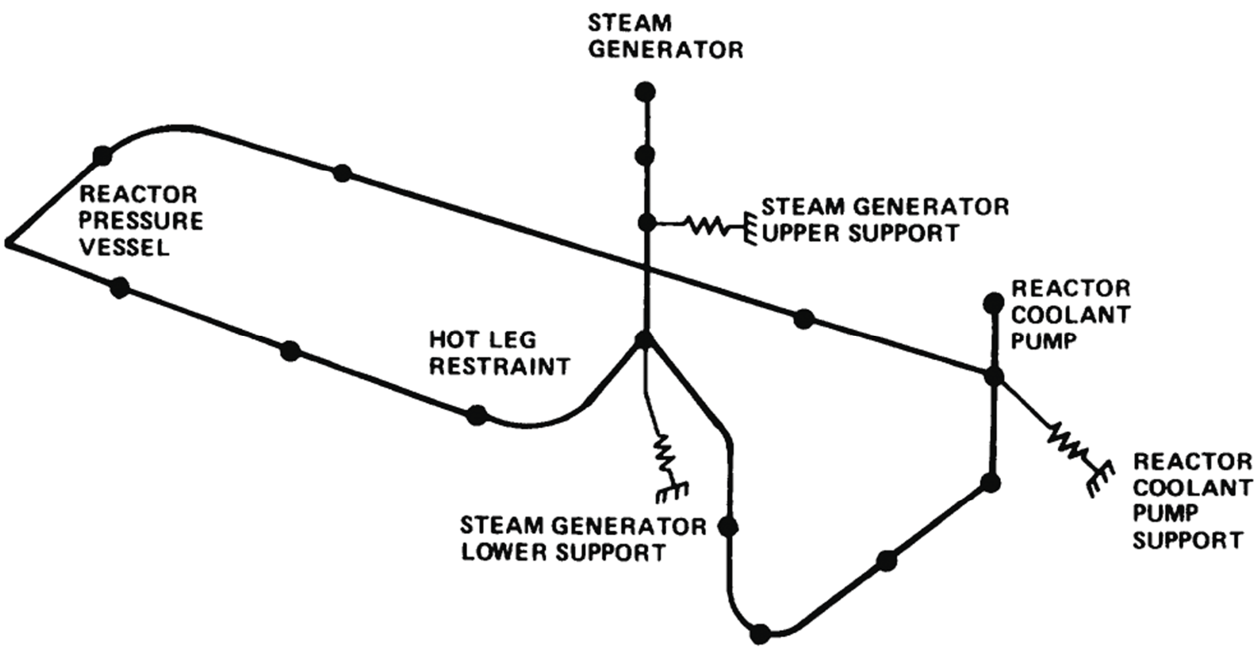
1. Table 3.9.N.3-5 contains SOT definitions and other load abbreviations.

2. SRSS is to be used for combining dynamic load responses.

TABLE 3.9.N.3-5

DEFINITIONS OF LOAD ABBREVIATIONS

N	=	Sustained loads during normal plant operation
SOT	=	System operating transient
SOT_U	=	Relief valve discharge transient
SOT_E	=	Safety valve discharge transient
SOT_F	=	Max (SOT_U, SOT_E), or transition flow
OBE	=	Operating basis earthquake
SSE	=	Safe shutdown earthquake
S_h	=	Basic material allowable stress at maximum (hot) temperature
S_m	=	Allowable design stress intensity
S_y	=	Yield strength value



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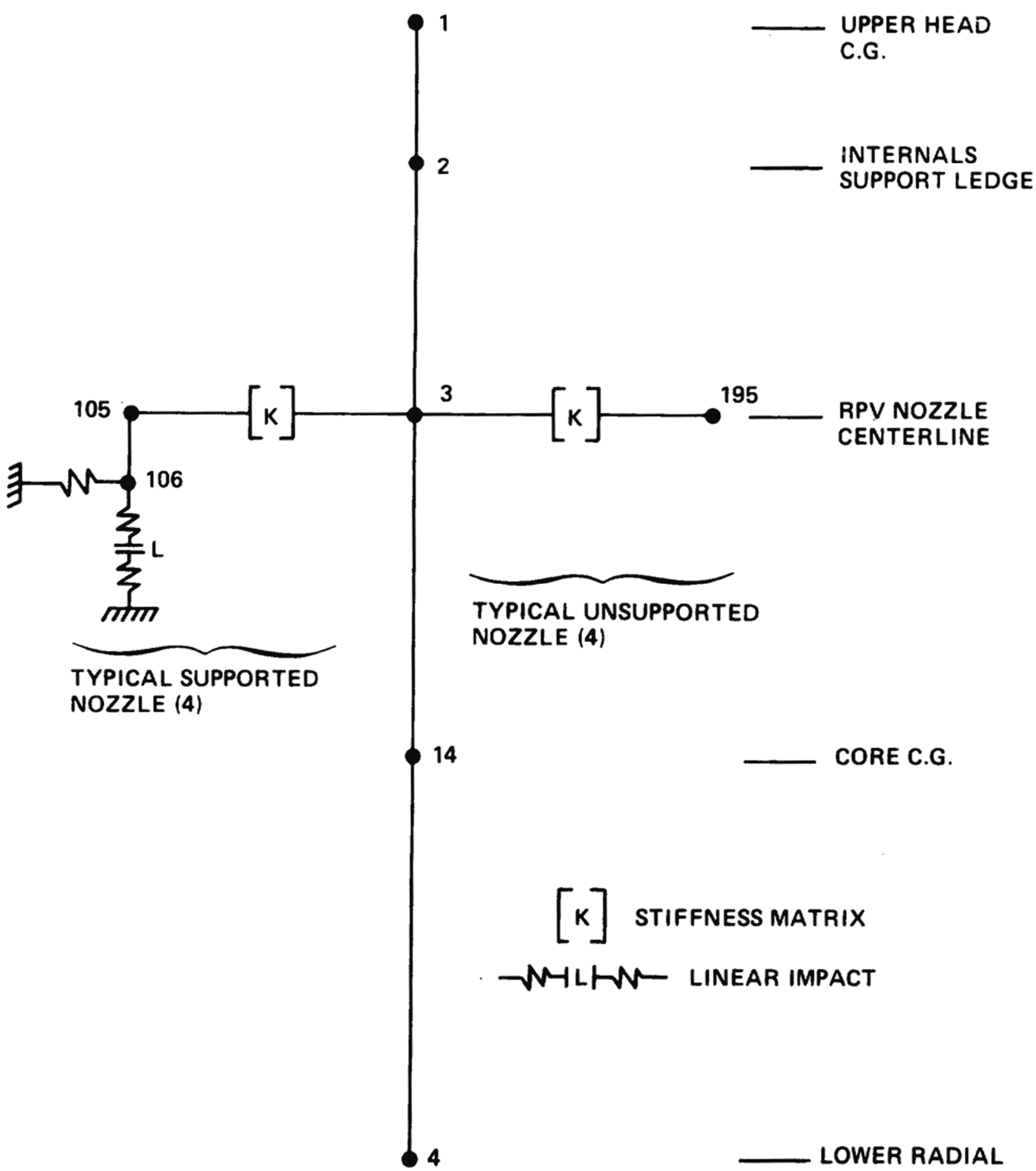
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	REACTOR COOLANT LOOP SUPPORTS SYSTEM DYNAMIC STRUCTURAL MODEL FIGURE 3.9.N.1–1



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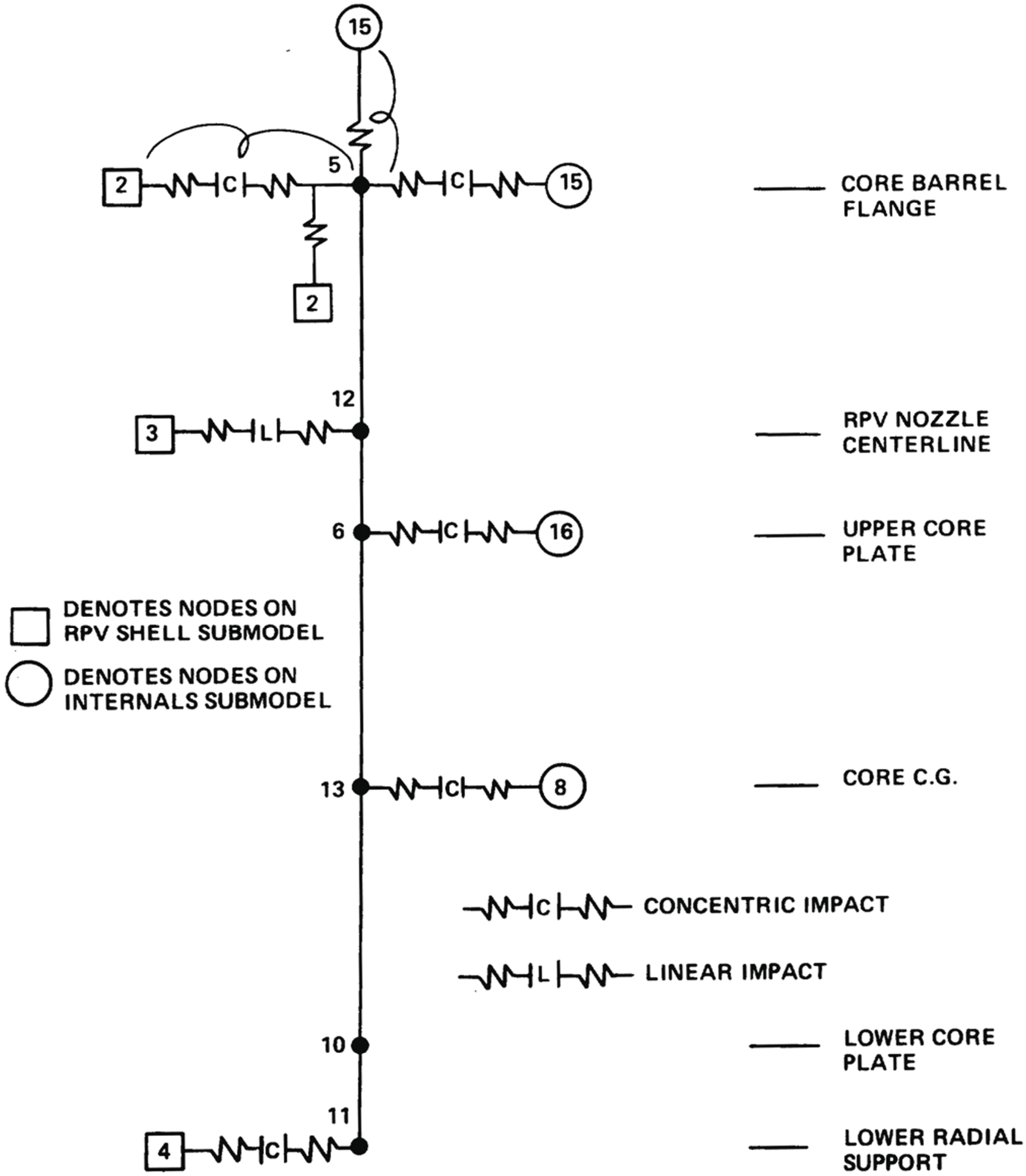
RPV SHELL SUBMODEL

FIGURE 3.9.N.1–2



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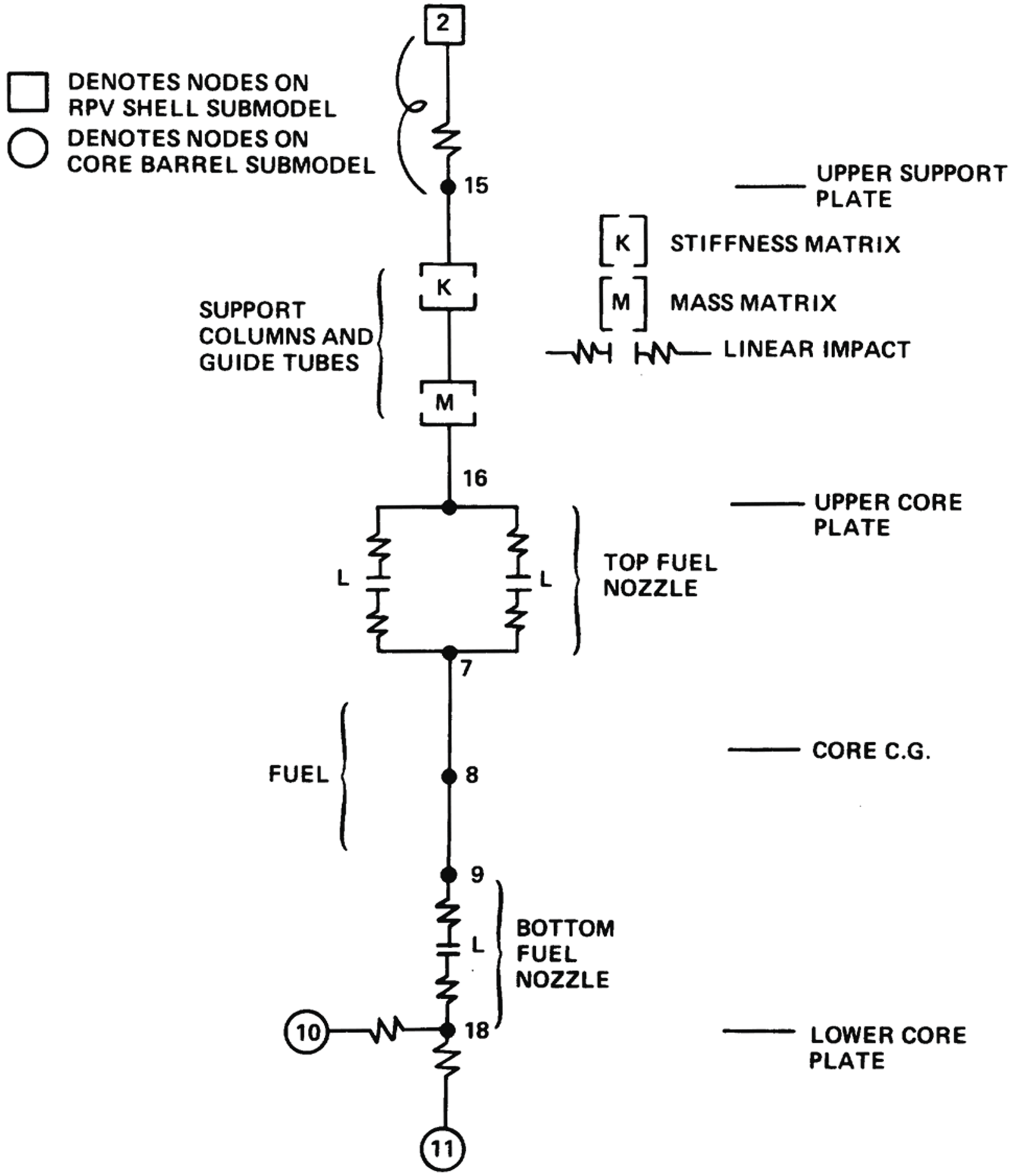
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CORE BARREL SUBMODEL FIGURE 3.9.N.1–3



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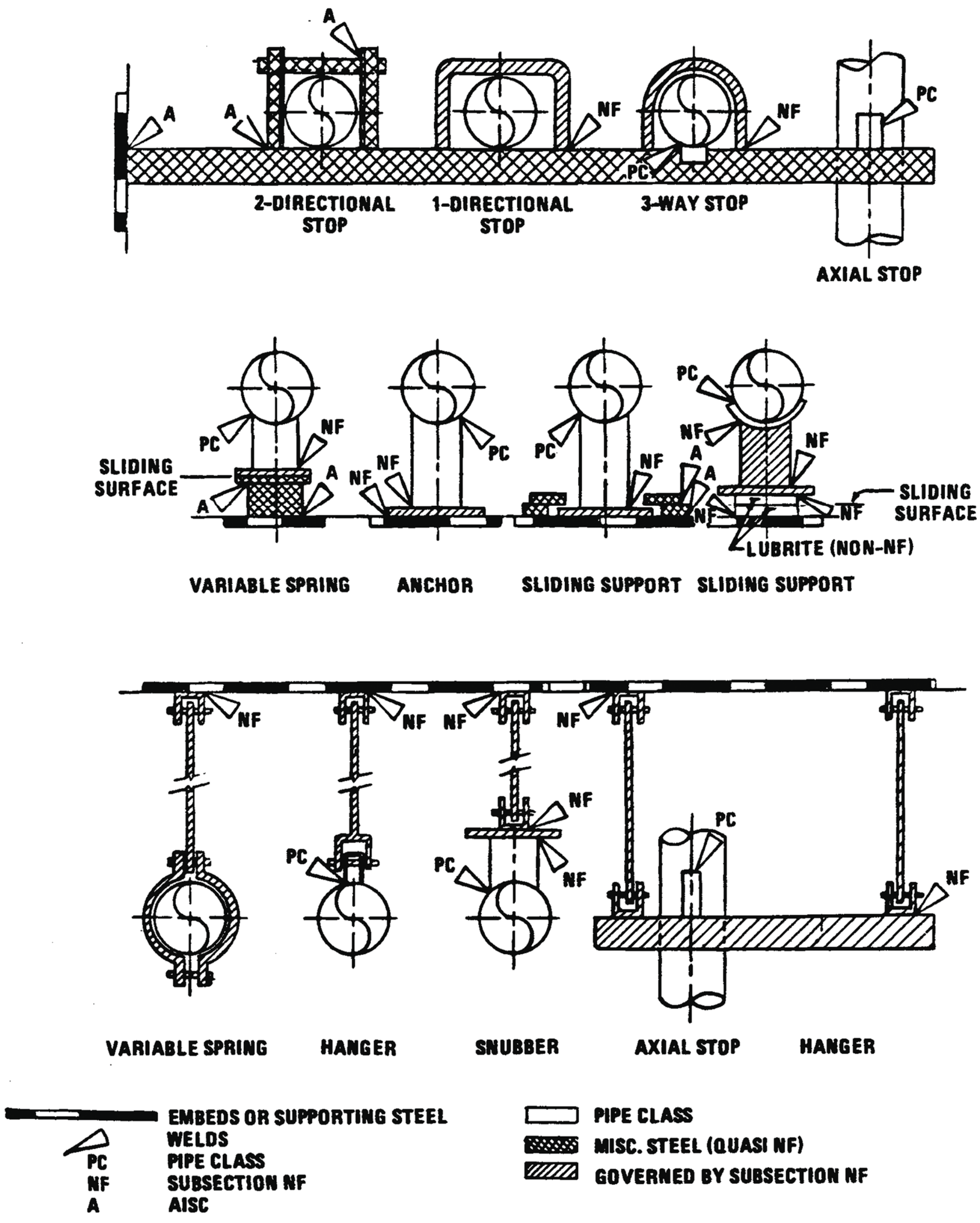
UNIT 1 AND UNIT 2

INTERNALS SUBMODEL

FIGURE 3.9.N.1–4



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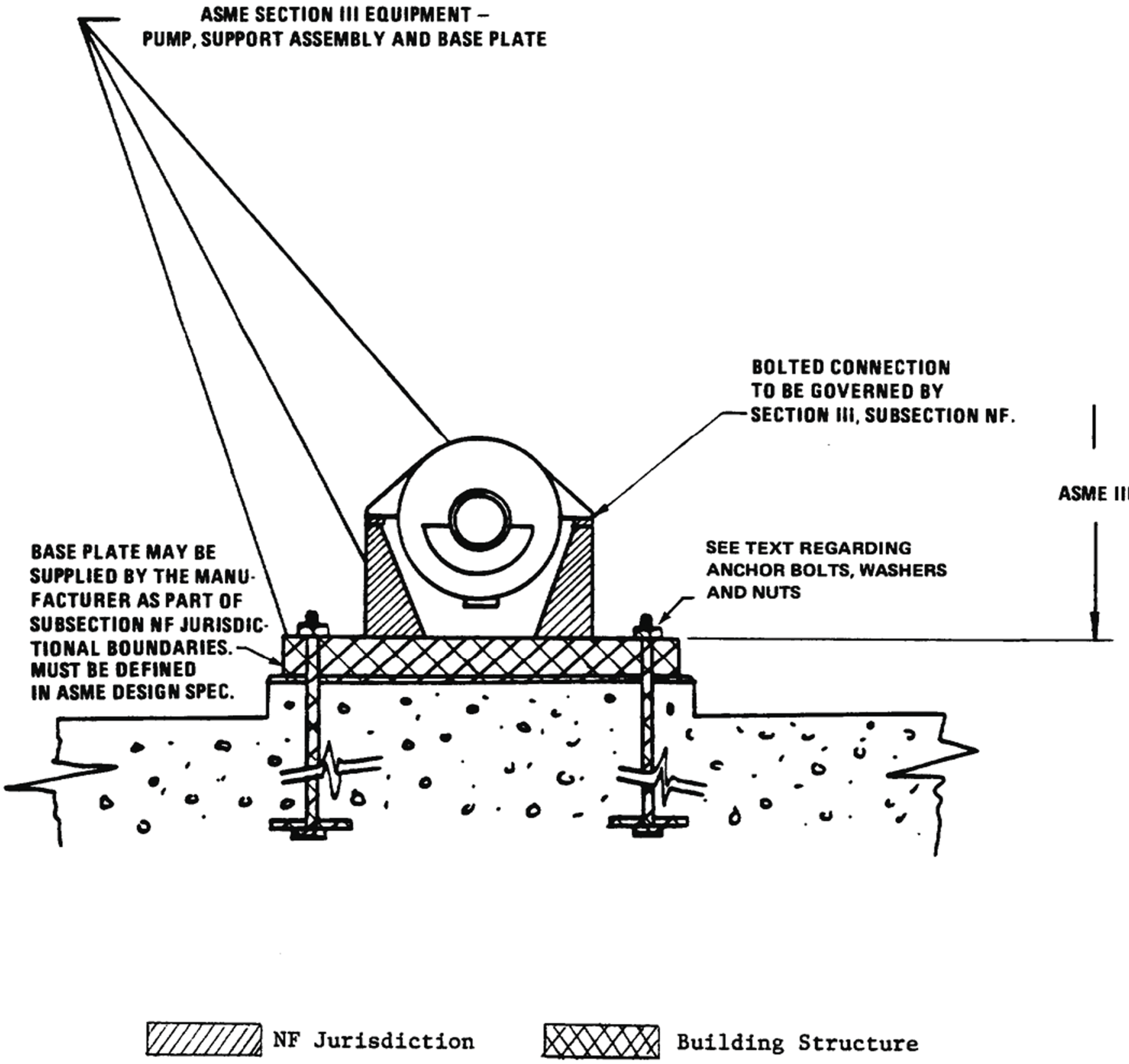
PIPE SUPPORT REFERENCE SKETCHES

FIGURE 3.9.B.3–1



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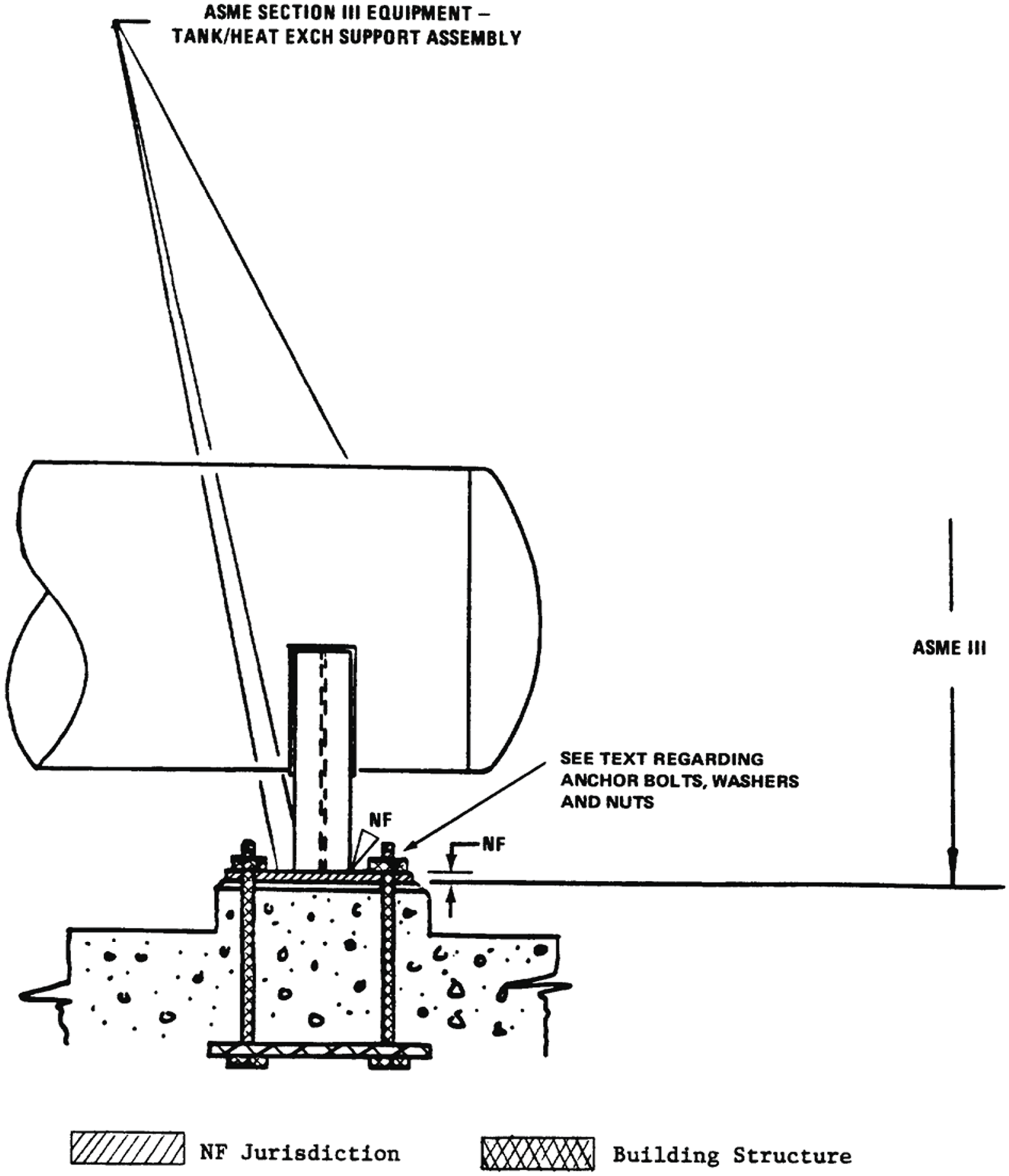
SOUTHERN COMPANY *Energy to Serve Your World®* — VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PUMP COMPONENTS SHOWING ASME SECTION III JURISDICTIONAL BOUNDARY (TYPICAL SKETCH)
	FIGURE 3.9.B.3–2



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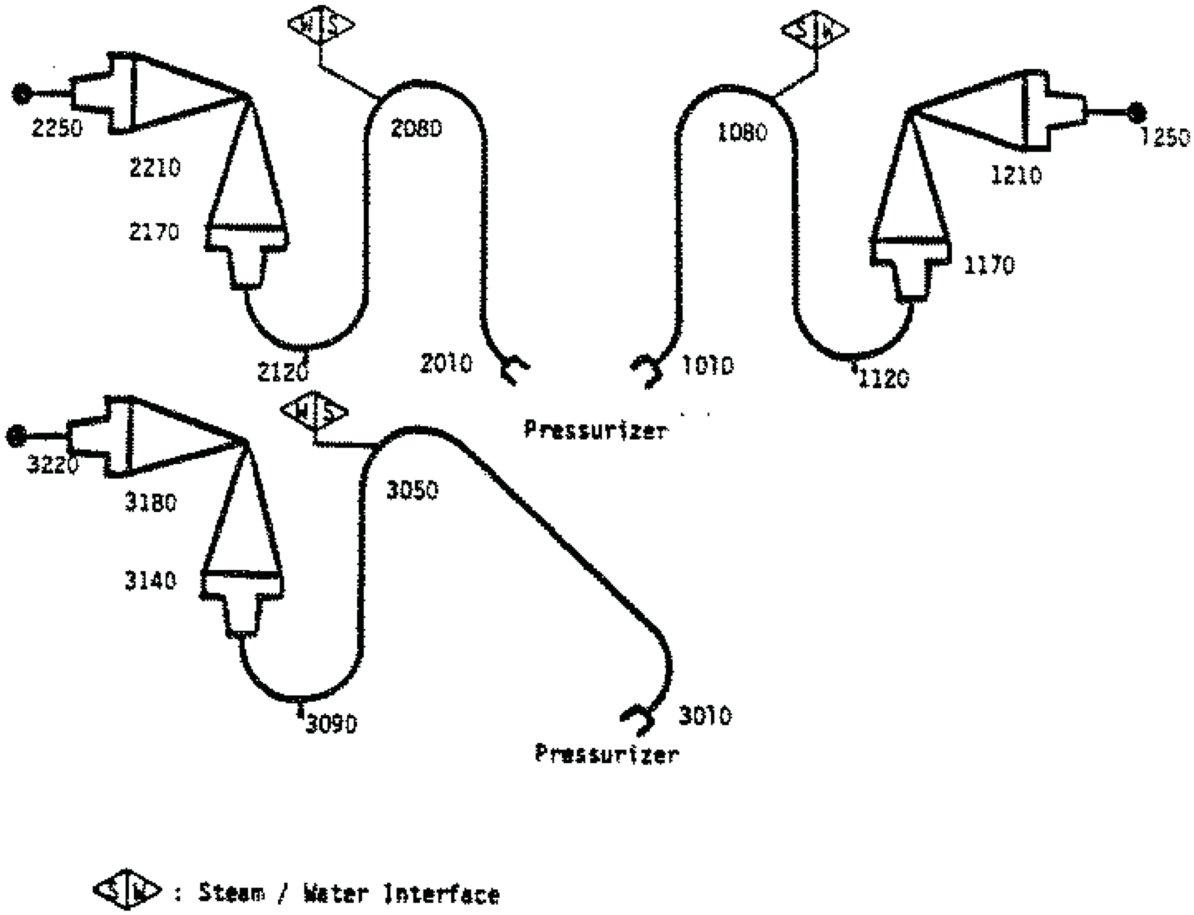
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	COMPONENTS SUCH AS HEAT EXCHANGERS & TANKS SHOWING ASME SECTION III NF JURISDICTIONAL BOUNDARY (TYPICAL SKETCH) FIGURE 3.9.B.3–3



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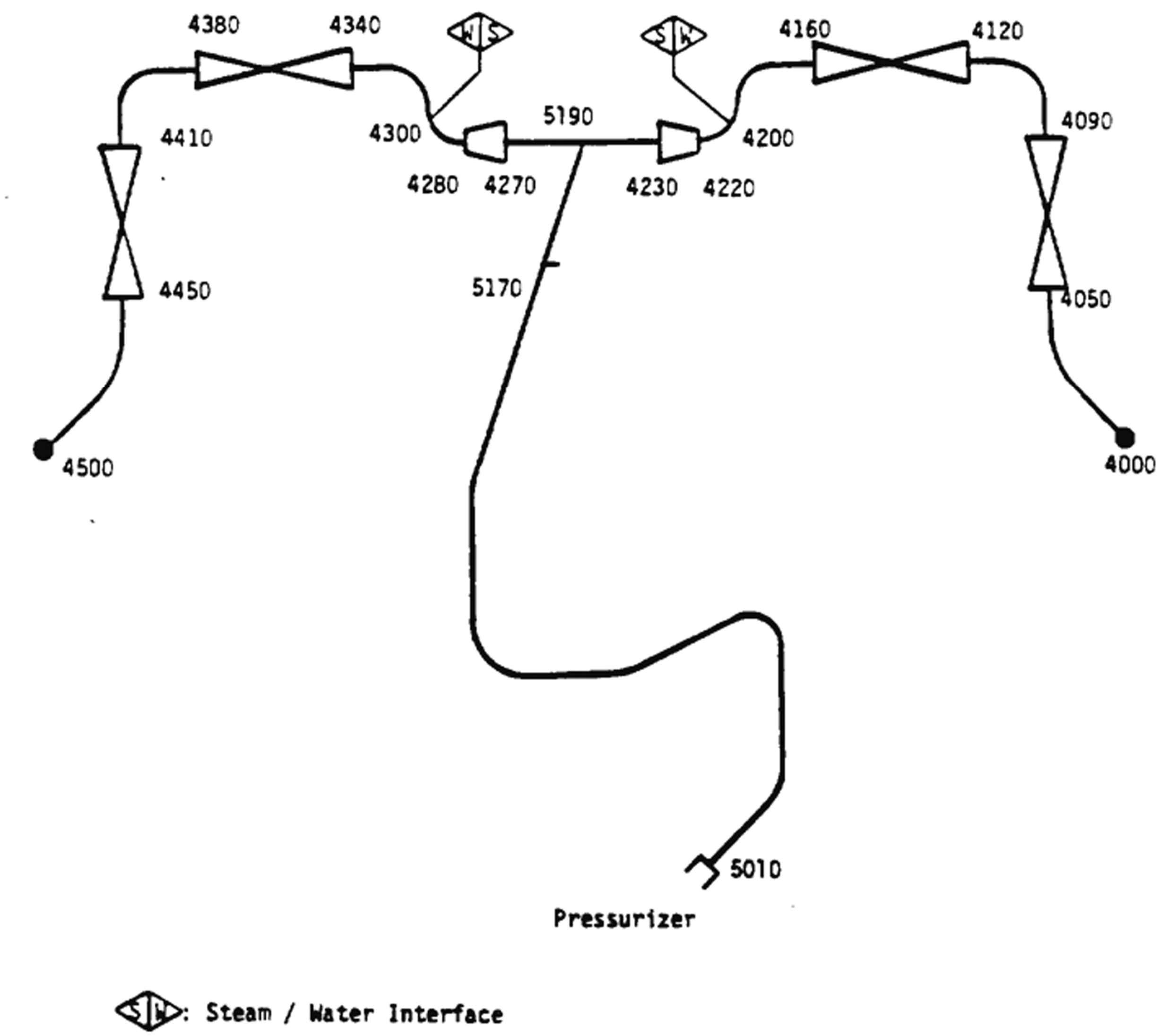
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CLASS 1 PIPING LAYOUT– SAFETY LINES FIGURE 3.9.N.3–1



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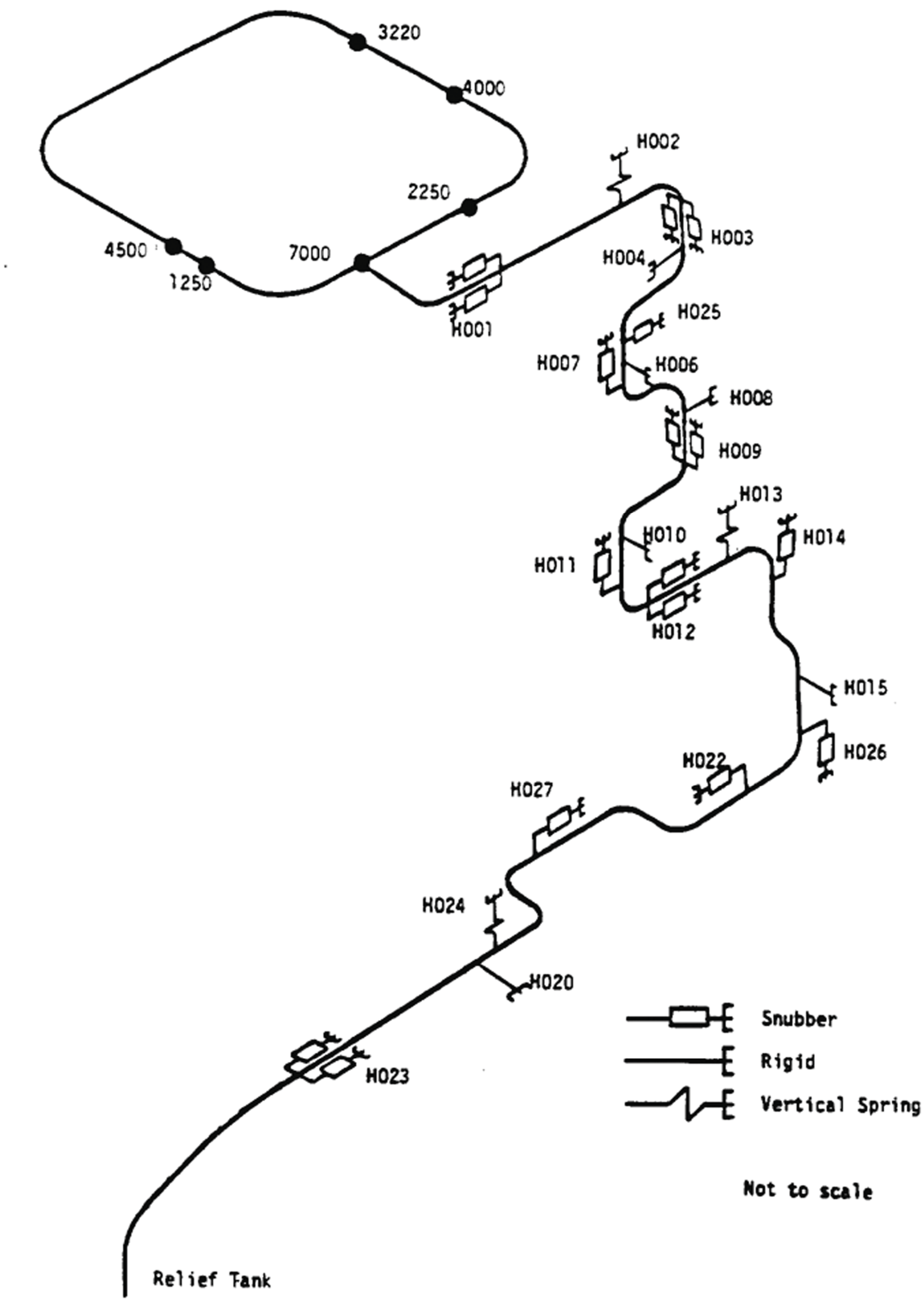
CLASS 1 PIPING LAYOUT–
RELIEF LINES

FIGURE 3.9.N.3–2



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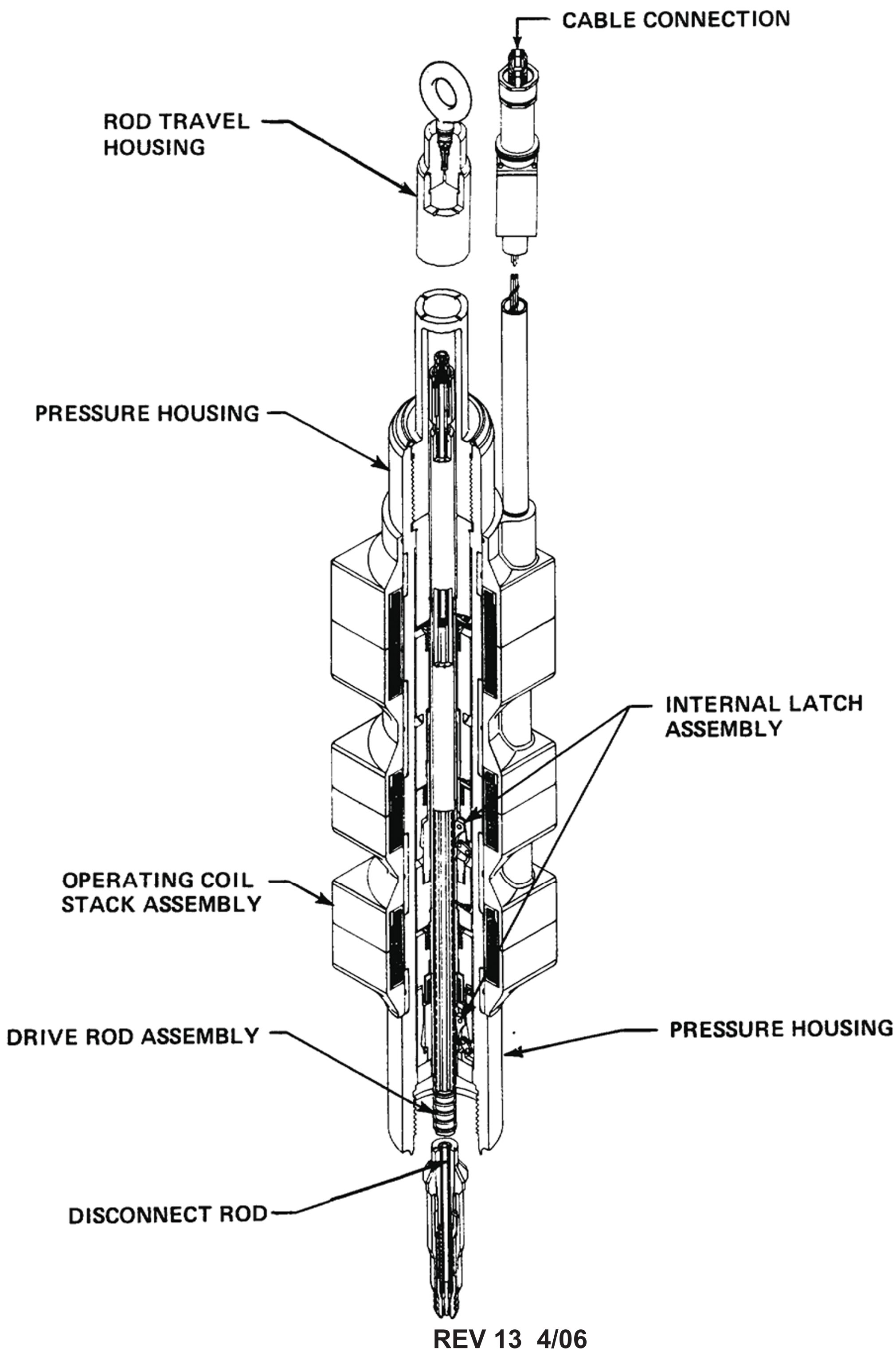
DOWNSTREAM PIPING LAYOUT–
SAFETY AND RELIEF LINES

FIGURE 3.9.N.3–3



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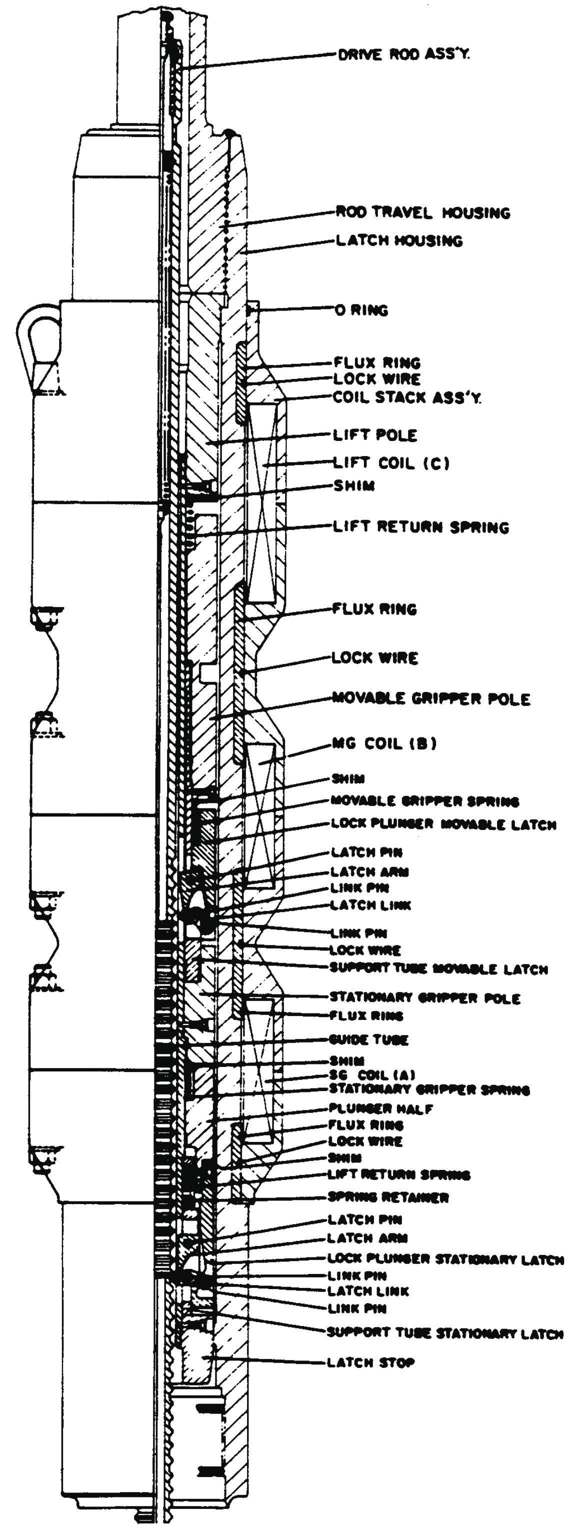
UNIT 1 AND UNIT 2

FULL–LENGTH CONTROL ROD DRIVE MECHANISM

FIGURE 3.9.4-1



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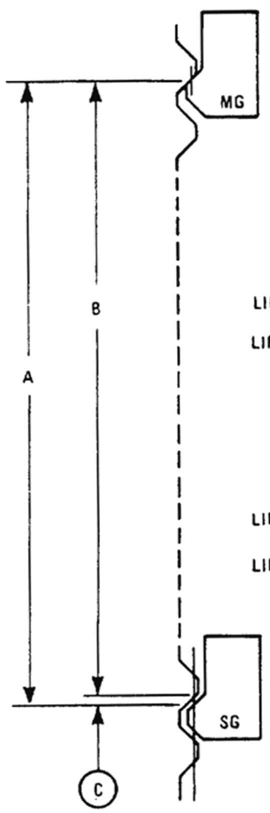
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	FULL–LENGTH CONTROL ROD DRIVE MECHANISM SCHEMATIC FIGURE 3.9.4–2

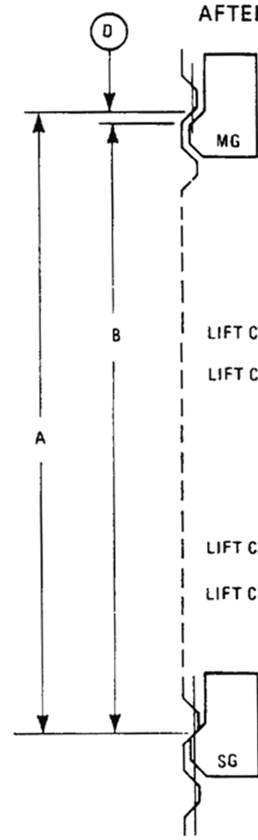
BEFORE LOAD TRANSFER



	AT 70F		
	A	B	C
LIFT COIL OFF	13.133	13.125	0.008
LIFT COIL ON	13.758	13.750	0.008

	AT 650F		
	A	B	C
LIFT COIL OFF	13.203	13.174	0.029
LIFT COIL ON	13.848	13.802	0.046

AFTER LOAD TRANSFER



	AT 70F		
	A	B	D
LIFT COIL OFF	13.125	13.070	0.055
LIFT COIL ON	13.750	13.695	0.055

	AT 650F		
	A	B	D
LIFT COIL OFF	13.174	13.123	0.051
LIFT COIL ON	13.802	13.768	0.034

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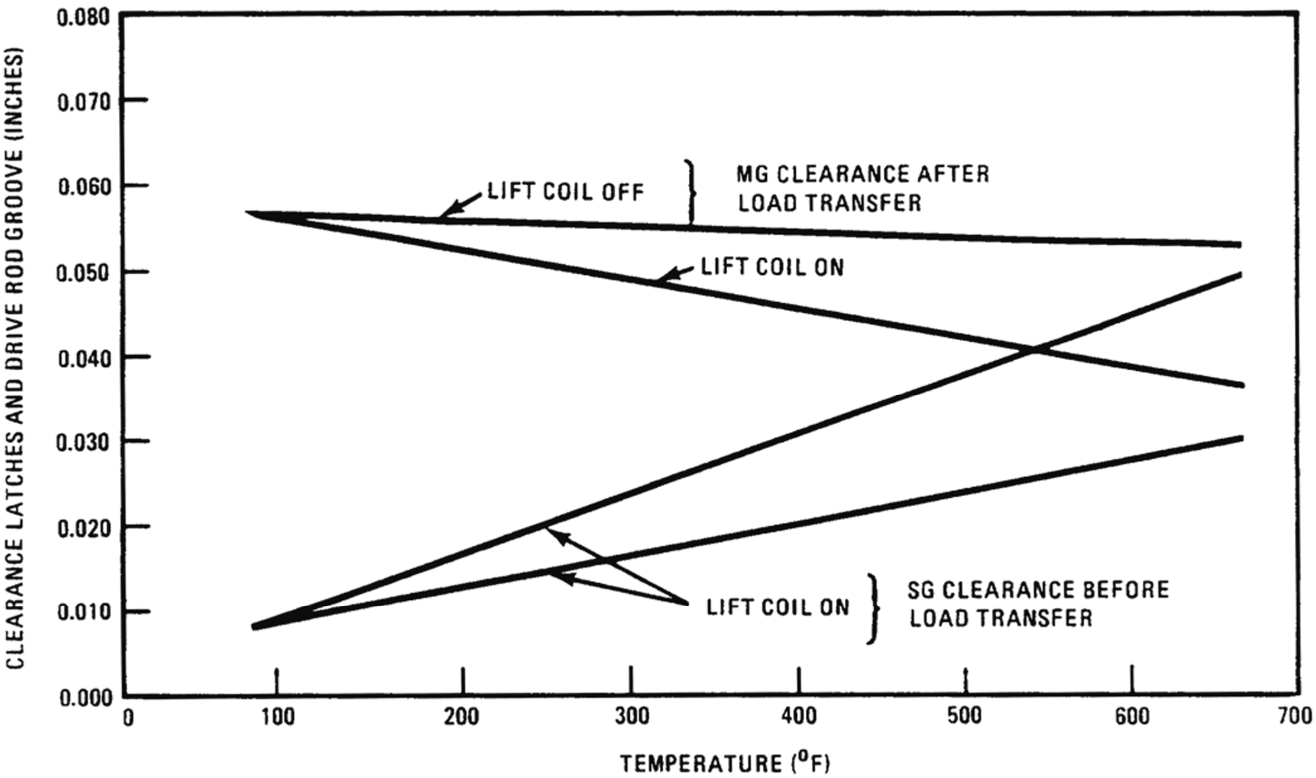
UNIT 1 AND UNIT 2

NOMINAL LATCH CLEARANCE AT MINIMUM AND MAXIMUM TEMPERATURES

FIGURE 3.9.4–3



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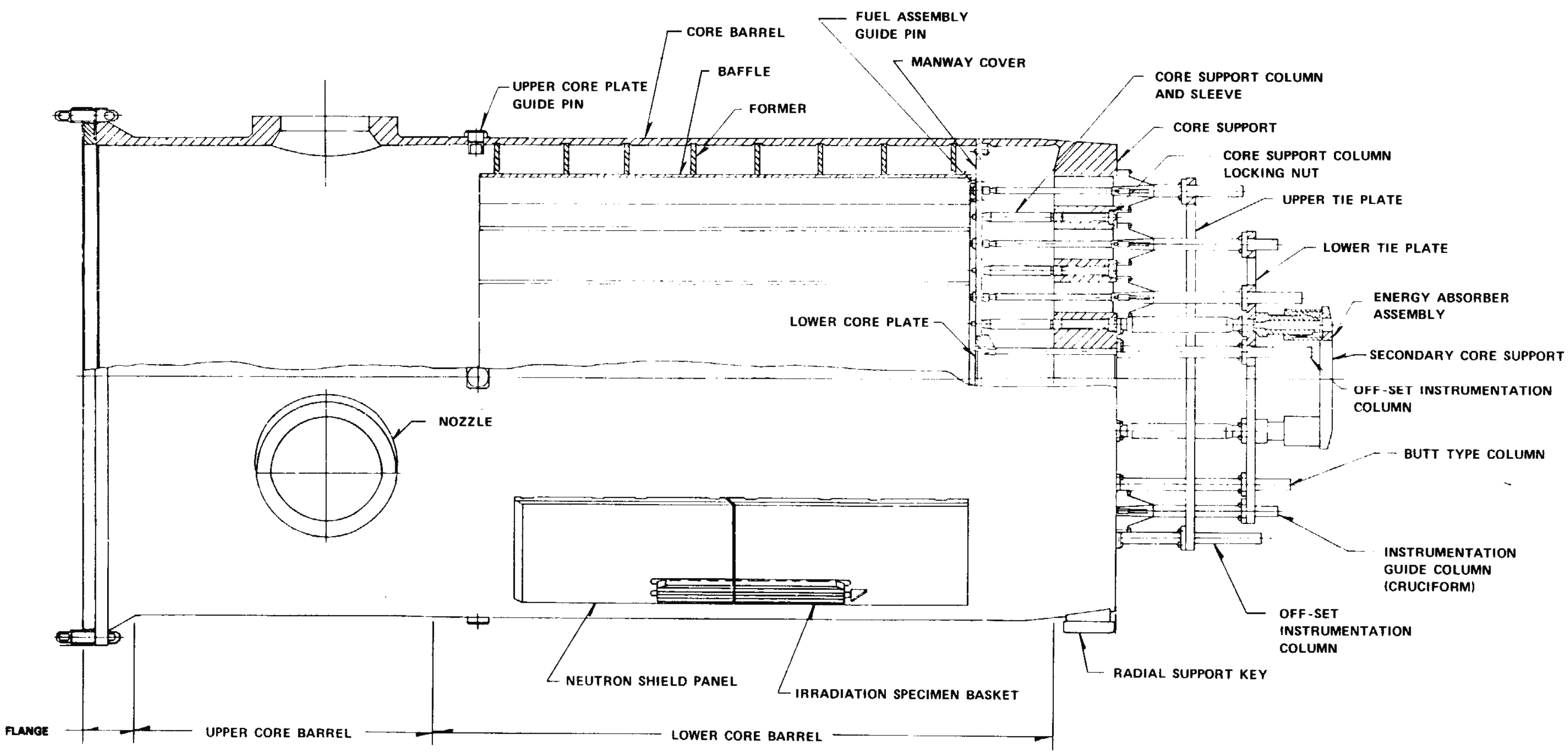
UNIT 1 AND UNIT 2

CONTROL ROD DRIVE MECHANISM LATCH CLEARANCE THERMAL EFFECT

FIGURE 3.9.4–4



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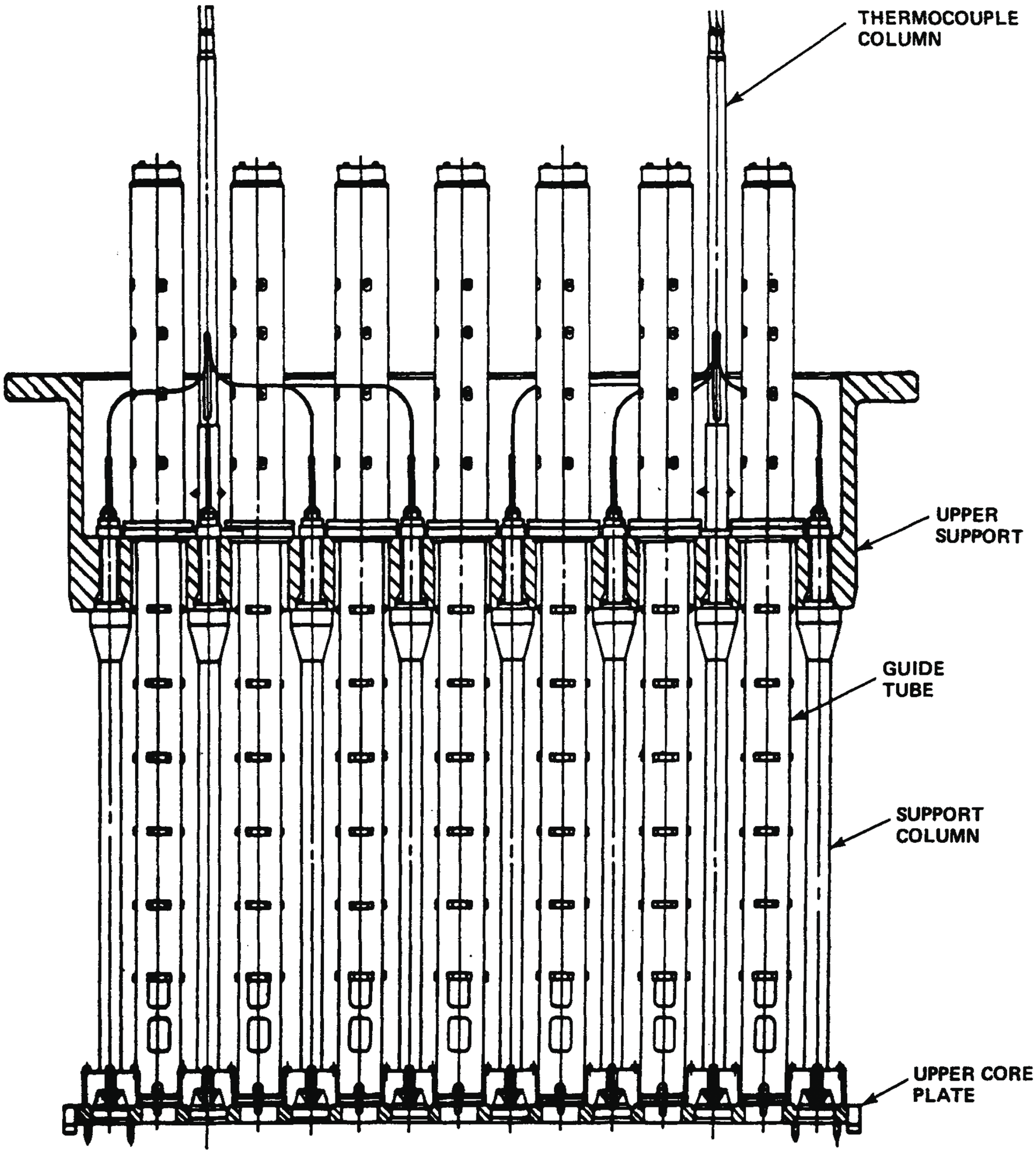
UNIT 1 AND UNIT 2

LOWER INTERNALS ASSEMBLY

FIGURE 3.9.5-1



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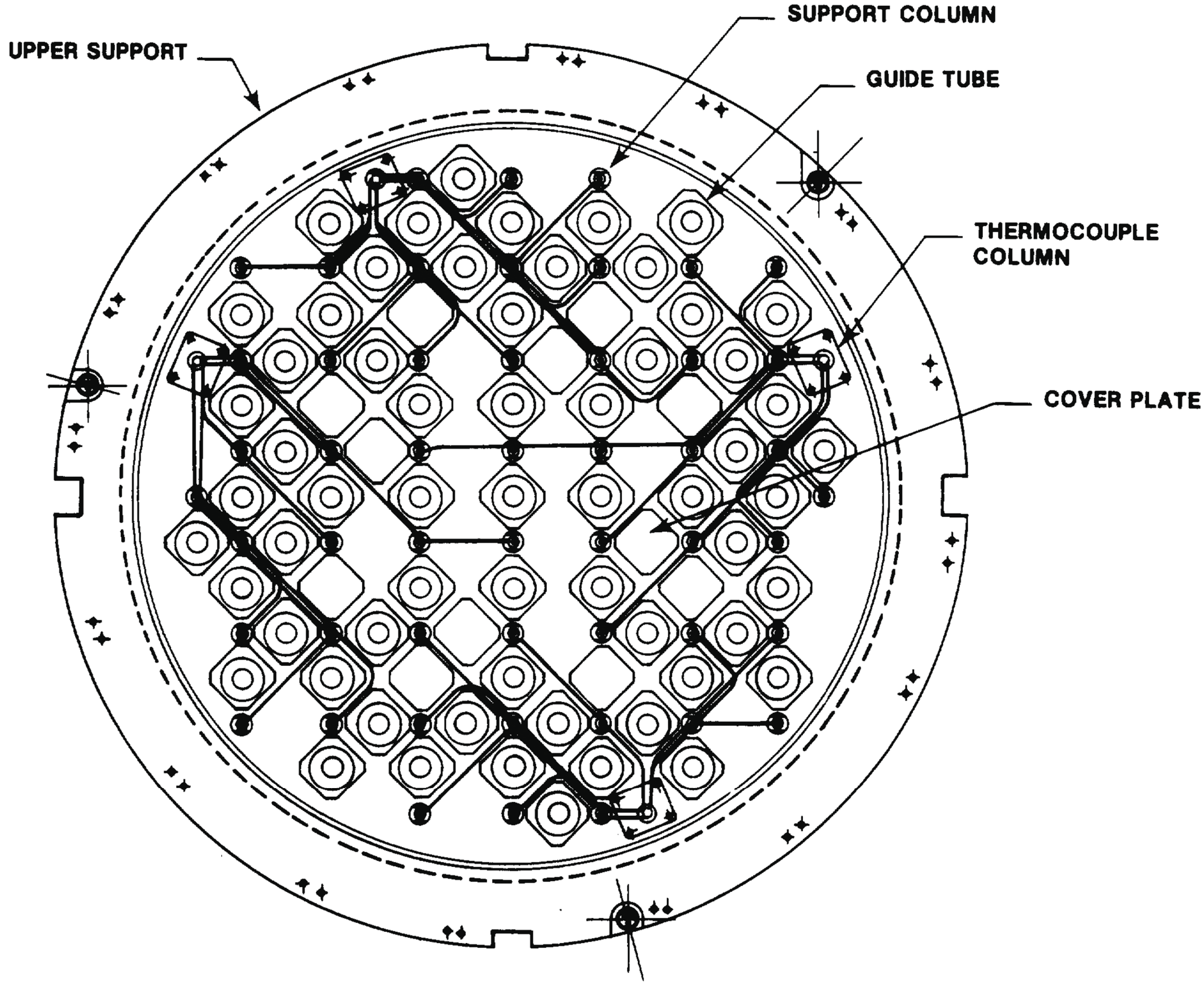
UNIT 1 AND UNIT 2

UPPER CORE SUPPORT ASSEMBLY

FIGURE 3.9.5–2



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ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PLAN VIEW OF UPPER CORE SUPPORT ASSEMBLY

FIGURE 3.9.5–3

3.10.B SEISMIC QUALIFICATION OF SEISMIC CATEGORY 1 MECHANICAL AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that safety-related mechanical and electrical equipment, instrumentation, and, where applicable, their supports classified as Seismic Category 1 are capable of performing their designated safety-related functions, considering the full range of normal and accident loadings in the event of an earthquake. The information presented includes:

- Identification of the Seismic Category 1 equipment.
- The criteria and methods of seismic qualification.
- Definition of the applicable seismic environment.
- The designated safety-related functional requirements.
- The definition of other normal and accident loadings.
- Documentation of the qualification process which demonstrates the required seismic capability.

Seismic Category 1 structures, components, and systems are identified in table 3.2.2-1. A master list and summary of the seismic qualification of the balance of plant (BOP) safety-related, Seismic Category 1 mechanical and electrical equipment will be maintained as part of the equipment qualification file. Refer to section 3.10.N for information concerning the seismic qualification of Seismic Category 1 mechanical and electrical equipment within the Westinghouse nuclear steam supply system (NSSS).

3.10.B.1 SEISMIC QUALIFICATION CRITERIA

The extent to which the VEGP design criteria meet the general requirements for the seismic qualification of safety-related mechanical and electrical equipment, as discussed in General Design Criteria (GDC) 1, 2, 4, 14, and 30 of Appendix A to 10 CFR 50, is discussed in section 3.1.

The general methods of implementing the requirements of Appendix A to 10 CFR 100 as it relates to qualifying equipment to withstand the effects of natural phenomena, such as earthquakes, and the requirements of Appendix B to 10 CFR 50 are discussed in section 2.5 and chapter 17, respectively.

The seismic qualification and documentation procedures used for safety-related equipment and their supports are in conformance with Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975, Recommended Practices for Seismic Qualification for Class 1E Equipment for Nuclear Power Generating Stations, and Nuclear Regulatory Commission (NRC) Regulatory Guide 1.100. The VEGP position regarding conformance to NRC Regulatory Guide 1.100 is given in section1.9.

Seismic Category 1 safety-related mechanical and electrical equipment is qualified to withstand the effects of seismic loads resulting from the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE), considering the full range of normal and accident loadings. The parameters used to develop seismic loadings and the seismic criteria for qualification of Seismic Category 1 structures, systems, and components are described in section 3.7.

The acceptance criteria for qualification of Seismic Category 1 safety-related mechanical and electrical equipment are specified to the supplier in the design purchase specification.

Testing is the preferred method to qualify equipment. Both dynamic as well as static test approaches are used to assure structural integrity and operability of mechanical and electrical equipment in the event of an SSE preceded by a number of occurrences of the OBE. Test fixtures are designed to simulate the actual service mounting. Test samples are selected according to type, load level, and size, as well as other pertinent factors on a prototype basis.

Analysis using mathematical modeling techniques correlated to tests performed on similar equipment or structures and/or verified analytical approaches are also used to qualify equipment. Combined analysis and testing may also be used to qualify equipment.

The analytical approach to seismic qualification without testing is used:

- If only maintaining structural integrity is required for the safety function.
- If the equipment is too large or heavy to obtain a representative test input at existing test facilities. (The essential control devices and electrical parts of large equipment are tested separately if required.)
- If the interfaces (e.g., interconnecting cables to the cabinet or other complex inputs) cannot be conservatively considered during testing.
- If the response of the equipment is essentially linear or has a simple nonlinear behavior which can be predicted by conservative analytical methods.

A combination of testing and analysis is used when complete testing is not practical and is incorporated into a test and analysis operability program.

Equipment that has been previously qualified by means of test and analysis equivalent to those described herein are acceptable provided that proper documentation is submitted.

Analysis and/or tests are performed for all Seismic Category 1 safety-related mechanical and electrical equipment to assure their structural and functional capability as required for the seismic criteria described in section 3.7.

The guidance provided in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, is followed in the design of safety-related Seismic Category 1 mechanical equipment to assure the structural integrity of pressure boundary components.

3.10.N SEISMIC AND DYNAMIC QUALIFICATION OF SEISMIC CATEGORY 1 MECHANICAL AND ELECTRICAL EQUIPMENT

The classification of Seismic Category 1 includes three basic types of equipment:

A. Seismic Category 1 instrumentation and electrical equipment.

B. Seismic Category 1 mechanical equipment that must perform a mechanical motion during the course of accomplishing a system safety function. These devices are defined as active and include the control rod drive mechanisms (CRDMs), snubbers, and certain pumps and valves within the nuclear steam supply system (NSSS) scope of supply.

C. Other Seismic Category 1 mechanical equipment whose only safety function is to maintain structural integrity.

This section presents information to demonstrate that mechanical equipment, electrical equipment, instrumentation, and, where applicable, their supports classified as Seismic Category 1 are capable of performing their designated safety-related functions under the full range of normal and accident (including seismic) loadings. This equipment includes devices associated with systems that are essential to the safe shutdown of the reactor containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment or mitigating the consequences of accidents. The information presented includes:

- Identification of the Seismic Category 1 instrumentation, electrical equipment, and appropriate mechanical equipment that are within the scope of the Westinghouse NSSS.
- The qualification criteria employed for each type of equipment.
- The designated safety-related functional requirements.
- Definition of the applicable seismic environment.
- Definition of other normal and accident loadings.
- Documentation of the qualification process employed to demonstrate the required structural integrity and operability of mechanical and electrical equipment and instrumentation in the event of a safe shutdown earthquake (SSE) after a number of postulated occurrences of the operating basis earthquake (OBE) in combination with other relevant dynamic and static loads.

3.10.N.1 SEISMIC AND DYNAMIC QUALIFICATION CRITERIA

3.10.N.1.1 Qualification Standards

The methods of meeting the general requirements for the seismic and dynamic qualification of Seismic Category 1 mechanical and electrical equipment and instrumentation as described by General Design Criteria (GDC) 1, 2, 4, 14, 23, and 30 are described in section 3.1. The general methods of implementing the requirements of Appendix B to 10 CFR 50 are described in chapter 17. The general methods of implementing the requirements of Appendix A to 10 CFR 100 as they relate to qualifying equipment to withstand the effects of natural phenomena such as earthquakes are discussed in section 2.5.

The Nuclear Regulatory Commission (NRC) recommendations concerning the methods to be employed for seismic qualification of instrumentation and electrical equipment are contained in Regulatory Guide 1.100, Seismic Qualification of Electric Equipment for Nuclear Power Plants, which endorses the Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.

Westinghouse Electric Corporation meets this standard, as modified by Regulatory Guide 1.100, by either type testing, analysis, or an appropriate combination of these methods. Westinghouse meets this commitment by employing the methodology described in WCAPs 8587 and 9714.(1)(2)

The guidance provided in the ASME Boiler and Pressure Vessel Code Section III is followed in the design of Seismic Category 1 mechanical equipment to assure the structural integrity of pressure boundary components. In addition, Westinghouse has implemented an operability program for active pumps and valves as originally required by Regulatory Guide 1.48, Design

Limits and Loading Combinations for Seismic Category 1 Fluid System Components. Regulatory Guide 1.148 is addressed in section 1.9.

3.10.N.1.2 Performance Requirements for Seismic Qualification

WCAP-8587 Supplement 1[3] contains an equipment qualification data package (EQDP) for every item of instrumentation and electrical equipment classified as Seismic Category 1 within the Westinghouse NSSS scope of supply. Table 3.2.2-1 identifies the Seismic Category 1 electrical equipment and instrumentation supplied by Westinghouse and the equipment qualification central file identifies the applicable EQDP in reference 3 or the appropriate qualification test report. Each EQDP contains a section entitled "Performance Specification." This specification establishes the safety-related functional requirements of the equipment to be demonstrated during and after a seismic event. The test response spectrum employed by Westinghouse for generic seismic qualification is also identified in the specification, as applicable. The spectra employed have been selected to envelop the plant specific required response spectra defined in section 3.7.

For active Seismic Category 1 mechanical components, the performance requirements are defined in the appropriate design and equipment specifications. System functional requirements are described in other sections of the FSAR that define other performance requirements for these components. For active pumps and valves, additional requirements are discussed in paragraph 3.10.N.2.2, and the equipment qualification reports are referenced in the equipment qualification central file. For other Seismic Category 1 mechanical components, the only performance requirement is to maintain structural integrity under all appropriate loading conditions.

A master list and summary of seismic qualification of safety-related Category I electrical and mechanical equipment will be maintained as part of the equipment qualification file.

3.10.N.1.3 Acceptance Criteria

Seismic and dynamic loading qualification must demonstrate that Category 1 instrumentation and electrical equipment as well as active pumps and valves are capable of performing their designated safety-related functions under all plant loading conditions including the SSE. The qualification must also demonstrate the structural integrity of other Seismic Category 1 pumps, valves, mechanical supports, and structures at the OBE level. Some permanent deformation of supports and structures is acceptable at the SSE level, provided that the ability to perform the designated safety-related functions is not impaired.

3.10.N.1.4 References

1. Butterworth, G., and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587 Revision 6, April 1983.
2. Kelly, R. E., and McInerney, J. J., "Methodology for the Seismic Qualification of Westinghouse WRD Supplied Equipment," WCAP-9714, May 1980.
3. "Equipment Qualification Data Package," Supplement 1 to WCAP-8587, April 1983.

3.10.B.2 METHODS AND PROCEDURES FOR QUALIFYING MECHANICAL AND ELECTRICAL EQUIPMENT

3.10.B.2.1 Means of Qualification

Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975, Recommended Practices for Seismic Qualification for Class 1E Equipment for Nuclear Power Generating Stations, and Nuclear Regulatory Commission (NRC) Regulatory Guide 1.100 are used for seismic qualifications. See section 3.10.N for information regarding the Westinghouse nuclear steam supply system (NSSS) equipment.

The horizontal and vertical operating basis earthquake (OBE) and safe shutdown earthquake (SSE) required response spectra (RRS) curves, as discussed in section 3.7, form the basis for the seismic qualification of equipment, systems, and components. The RRS curves are identified with the building elevation. These curves are integrated into the purchase specification, along with the equipment location or locations, and the acceptance criteria for the safety-related functions for each item of equipment.

Seismic qualification of equipment is demonstrated by one of the following:

A. Qualifying a complete assembly identical to that being installed.

B. Qualifying a complete assembly similar to that being installed with adequate justification for differences.

C. Qualifying the individual devices and/or components separately and demonstrating the adequacy of the complete equipment assembly.

All interfaces and the effects of the amplification within the equipment due to the interfaces and supporting structure are considered in the seismic qualification.

Seismic qualification plans/procedures are prepared by the equipment suppliers and submitted for review before testing and/or analysis are performed. The seismic qualification reports, when prepared by the supplier and submitted for review, demonstrate (in accordance with subsection 3.10.B.1) that the equipment performs its required safety-related function before, during, and after (as required) multiple occurrences of the OBE followed by one SSE. The OBEs shall contain a minimum of 50 c of maximum stress.

For components that have been previously tested to generic criteria, multifrequency inputs developed by the test laboratory are reviewed to assure conservatism in amplitude and frequency content and to assure that the test response spectra (TRS) envelops the RRS over the critical frequency range. Test reports are reviewed to confirm the operability of essential equipment.

For active mechanical equipment (i.e., pumps and valves) a combination of test and analysis is used to demonstrate operability and structural integrity of components. Other Seismic Category 1 safety-related mechanical equipment is qualified by analysis to demonstrate structural integrity. The criteria used to decide which method is appropriate are discussed in paragraph 3.9.B.2.2.

Qualification of BOP safety-related active pumps is based on an evaluation of the entire pump/motor (including coupling system) assembly considering its interface with the mounting structure. Simultaneous dynamic interactions between pump, motor, and pedestal/mounting structure are addressed through seismic analysis. Frequency search tests of the pump/motor assembly are used to verify the mathematical modeling technique used for equipment qualification.

Load combinations, combining of dynamic responses for mechanical equipment, and the pump and valve operability assurance program are discussed in subsection 3.9.B.3.

The pump and valve operability assurance program is discussed in paragraph 3.9.B.3.2.

3.10.B.2.2 Method of Qualification

The methods for seismic qualification are listed below:

- Analysis.
- Test.
- Combination of analysis and test.

3.10.B.2.2.1 Analysis

Mathematical analyses without testing are acceptable if the structural integrity alone ensures the intended design function of the equipment. (See subsection 3.10.B.1.) The procedures used are in accordance with Section 5 of IEEE 344-1975.

When an equivalent static coefficient analysis is performed, justification for its use is provided by the supplier. See paragraph 3.7.B.3.5 for additional information on use of equivalent static load method of analysis. All equipment qualification performed by mathematical analysis is based on mathematical methods correlated with tests of similar equipment or structures or verified analytical techniques.

Analytical results are evaluated for mechanical strength, fatigue, alignment, and noninterruption of function as related to the functional requirements of the equipment during an SSE event. Maximum stresses under all loading are computed and compared with the allowables. Interference effects as well as interaction effects are considered in the analysis when significant.

3.10.B.2.2.2 Testing

Seismic qualification by testing is performed using either multifrequency or single frequency inputs. These test inputs and methods are in accordance with IEEE 344-75, Section 6.

The multifrequency test method is used for floor- and wall-mounted equipment. In addition, in special cases it is used for equipment mounted on structural steel, piping, ducts, or other types of supports or equipment where an analysis or test has been performed to determine the RRS at the equipment mounting location. The test or analysis considers the dynamic amplification characteristics of the support system.

For equipment qualified by multifrequency testing, the measured TRS envelops the RRS in the critical frequency range as shown in the appropriate BOP equipment qualification data package (EQDP).

Single-frequency tests are used for line-mounted equipment, which includes equipment mounted in piping systems and in ducts. The equipment is tested to a required input motion

(RIM). The RIM is the peak acceleration of the input motion (sine wave or sine beats) as a fraction of input frequency.

The piping and duct systems are designed and supported to limit the peak acceleration experienced by the equipment to a value less than the specified RIM acceleration.

Single-frequency tests may also be used for other types of equipment as permitted by IEEE 344-75 and Regulatory Guide 1.100.

3.10.B.2.2.3 Combined Test and Analysis

When the equipment could not be qualified practically by analysis or testing because or its size or complexity, combined analysis and testing were utilized. This method of qualification is applied to equipment such as cabinets that may contain several different configurations of internally mounted devices.

The combined analysis and test method is in accordance with Section 7 of IEEE 344-75, and the equipment qualification method of paragraphs 3.10.B.2.2.1 and 3.10.B.2.2.2 apply.

Equipment that has been previously qualified by means of tests and analyses equivalent to those described herein is acceptable if proper documentation is provided.

3.10.B.2.2.4 Test Sequence Verification

As defined in Part B of Regulatory Guide 1.100, IEEE 344-1975 is an ancillary standard of IEEE 323-1974. In accordance with this standard, seismic testing as part of the overall qualification is performed in its proper sequence as indicated in Section 6 of IEEE 323-1974.

3.10.N.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT, INSTRUMENTATION, AND MECHANICAL COMPONENTS

Seismic qualification of Seismic Category 1 instrumentation and electrical equipment is demonstrated by either type testing or a combination of test and analysis methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including practicability, complexity of equipment, economics, and availability of previous seismic qualification to earlier standards. The qualification method employed for a particular item of instrumentation or electrical equipment is identified in the individual equipment qualification data package (EQDP) of reference 1.

For active pumps and valves Westinghouse utilizes a combination of test and analysis to demonstrate the structural integrity and operability of such components. Other Seismic Category 1 mechanical equipment is qualified by analysis to demonstrate structural integrity.

Westinghouse methods of load combination and methods of combining dynamic responses for mechanical equipment are discussed in subsection 3.9.N.3. For instrumentation and electrical equipment, the only dynamic loads considered in testing are seismic loads and vibratory loads where applicable. Other dynamic loads which instrumentation and electrical equipment may be subjected to are enveloped by this testing or are addressed by analysis.

Provided below is a description of the methods used to qualify Seismic Category 1 equipment.

3.10.N.2.1 Seismic Qualification of Instrumentation and Electrical Equipment

3.10.N.2.1.1 Type Testing

Prior to the implementation of Institute of Electrical and Electronics Engineers (IEEE) 344-1975, Westinghouse utilized single-axis sine-beat inputs as specified by IEEE 344-1971 to qualify equipment. Westinghouse demonstrated the conservatism of this test method with respect to the modified methods of testing for complex equipment recommended by IEEE 344-1975 through a supplemental test program. As a result, no additional qualification testing was required on a limited number of equipment items defined in Table 7.1 of reference 2.

For other Seismic Category 1 instrumentation and electrical equipment, seismic qualification by test is performed in accordance with IEEE 344-1975. Where testing is utilized, multifrequency, multiaxis inputs are developed by the general procedures outlined in reference 3. The test results contained in the individual EQDPs of reference 1 demonstrate that the measured test response spectrum envelops the generic required response spectrum defined in Section 1 of the EQDP. Qualification for VEGP use is established by verification that the generic required response spectrum specified by Westinghouse envelops the applicable VEGP response spectrum.

Alternative test methods, such as single-frequency, single-axis inputs for line-mounted equipment, are used in selected cases as permitted by IEEE 344-1975 and Regulatory Guide 1.100. These methods are further described in reference 3.

3.10.N.2.1.2 Test and Analysis

Westinghouse also utilizes a combination of test and analysis to qualify Seismic Category 1 instrumentation and electrical equipment. The test methods utilized are similar to those described above for type testing. Available test results are employed in combination with the analysis methods described in IEEE 344-1975 to demonstrate seismic qualification. The analytical methods used include both static and dynamic techniques which are described in detail in reference 3 and the qualification reports identified in the equipment qualification central file.

3.10.N.2.2 Seismic and Operability Qualification of Active Mechanical Equipment

Active mechanical equipment is qualified for both structural integrity and operability under all its intended service conditions by a combination of test and analysis. These methods address such loading conditions as thermal transients, flow loads where significant, and degraded flow conditions if applicable. The test and analysis methods utilized in qualification of these components provide adequate assurance of operability under all required plant conditions.

Qualification methods utilized for active pumps and valves are described below. The qualification methods utilized for control rod drive mechanisms are described in section 3.9.N, and the qualification methods utilized for snubbers are described in section 3.9.N and section 5.2.

3.10.N.2.2.1 Nuclear Steam Supply System Pumps

All active pumps listed in table 3.9.N.3-1 are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include:

A. Hydrostatic tests of pressure-retaining parts to 150 percent of the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature.

B. Seal leakage tests.

C. Performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump parameters.

Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer based on the bearing material, clearances, oil type, and rotational speed. These limits are approved by Westinghouse. After the pump is installed in the plant, it undergoes the cold hydro tests, hot functional tests, and, where applicable, periodic inservice inspection and operation. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant. Except as noted, those pumps listed in table 3.9.N.3-1 will be included in the Inservice Testing Program (ASME Code Section XI). For those pumps which are not included in the Inservice Testing Program and are listed in table 3.9.N.3-1, the capability to perform their safety related function will be demonstrated through inclusion in plant maintenance programs and/or plant procedures.

In addition to these tests, the safety-related active pumps are qualified for operability by assuring that the pump will start, continue operating, and not be damaged during the faulted condition.

The pump manufacturer is required to show by analysis correlated by tests, prototype tests, or existing documented data that the pump will perform its safety function when subjected to loads imposed by the maximum seismic accelerations and the maximum faulted nozzle loads. It is required that test or dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered essentially rigid. This frequency is sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative safe shutdown earthquake (SSE) accelerations of 2.1 g in two orthogonal horizontal directions and 2.1 g vertical acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration. If rubbing or impact is predicted, it is required that it be shown by prototype tests or existing documented data that the pump will not be damaged or cease to perform its design function. The effect of impact on the operation of the pump is evaluated by comparison of the impacting surfaces of the pump to similar surfaces of pumps which have been or will be tested.

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited as specified in tables 3.9.B.3-2 through 3.9.B.3-7. In addition, the pump casing stresses caused by the maximum faulted nozzle loads are limited to the stresses outlined in table 3.9.B.3-5. The changes in operating rotor clearances caused by casing distortions due to these nozzle loads are considered. The maximum seismic nozzle loads combined with the loads imposed by the seismic accelerations are also considered in an analysis of the pump supports. Furthermore, the calculated misalignment is shown to be less than that misalignment which could cause pump malfunction. The stresses in the supports are below those in tables 3.9.B.3-6 and 3.9.B.3-7; thus the support distortion is elastic and of short duration (equal to the duration of the seismic event).

Performing these analyses with the conservative loads stated and with the restrictive stress limits of table 3.9.B.3-5 as allowables assures that critical parts of the pump will not be damaged during the short duration of the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event. To complete the seismic qualification procedures, the pump motor is qualified for operation during the maximum seismic event.

In many instances, pumps and motors are seismically qualified independently of each other. Where applicable, loads that may be transmitted between pump and motor as a result of seismic interaction are included in the qualification. Most pump assemblies employ a coupling between the driver and the pump. Westinghouse specifies flexible, limited end float couplings for use in nuclear applications. This design compensates for end or axial movement of the coupled shafts preventing either shaft from exerting excessive thrust on the other. Under severe loading conditions such as a strong seismic event, a small amount of thrust is transmitted through the coupled shafts. Westinghouse pump designs include thrust collars to absorb forces in the shaft axial direction. This prohibits the pump from tansmitting thrust loads to the motor. Most motor designs use ball or journal bearing, either of which are very effective in limiting shaft axial movement. Should loads be transmitted from the motor to the pump, again the thrust collar would provide the ability to absorb them.

Westinghouse has performed dynamic seismic testing on a multistage centrifugal pump/gear/motor assembly. The couplings performed superbly throughout the test. Disassembly and visual examination with tolerance measuring showed no excessive wear or damage resulting from the seismic events.

The program above gives the required assurance that the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings and, therefore, will perform their intended functions. The proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is assured, since only normal operating loads and steady-state nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the postfaulted-condition operating loads will be identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and postfaulted conditions are limited by the magnitudes of the normal condition nozzle loads. The postfaulted-condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.10.N.2.2.2 Nuclear Steam Supply System Valves

Safety-related active valves, listed in table 3.9.N.3-2 must perform their mechanical motion in times of an accident. Assurance is supplied that these valves will operate during a seismic event. Tests and analyses are conducted to qualify active valves.

The safety-related valves are subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to

American Society of Mechanical Engineers (ASME) Section III requirements, back-seat and main seat leakage tests, disc hydrostatic tests, and operational tests to verify that the valve will open and close. For the qualification of motor operators for environmental conditions refer to section 3.11.N. After installation, the valves undergo hydro tests, construction acceptance tests, and preoperational tests. Where applicable, periodic inservice inspections, and periodic inservice operations are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. Except as noted, those valves listed in Table 3.9.N.3-2 will be included in the Inservice Testing Program (ASME Code, Section XI). For those valves which are not included in the Inservice Testing Program and are listed in table 3.9.N.3-2, the capability to perform their safety related function will be demonstrated through inclusion in plant maintenance programs, plant procedures, and/or Technical Specifications. The valves are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. On active valves, an analysis of the extended structure is performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure. The maximum stress limits used for active Class 1, 2, and 3 valves are shown in subsection 3.9.B.3.

In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated plant faulted-condition event by demonstrating operational capabilities within the specified limits. A representative valve of a specific design type is identified for this testing by the specification (e.g., globe valve, motor-operated valve, etc.) for that particular type of valve. A stratification of design is further made based upon the valve size, pressure rating, type of operator, and previous operability testing to evaluate the need for additional testing of a particular design type. The testing procedures are described below.

The valve is mounted in a manner which conservatively represents typical valve installations. The valve includes the operator pilot, solenoid valves, and limit switches when such devices are normally attached to the valve in service. The faulted-condition nozzle loads are considered in the test in either of two ways: loads equivalent to the faulted-condition nozzle loads are simultaneously applied to the valve through its mounting during the below described test, or by analysis, the nozzle loads are shown not to affect the operability of the valve. Interface requirements are specified to limit nozzle loads such that deflection or deformation of the valve materials will not affect the operability of the valve. The operability of a rigid valve (natural frequency equal to or greater than 33 Hz) is demonstrated by satisfying the following criteria:

A. The actuator and yoke of the valve system are statically deflected by applying a load or loads equivalent to the resultant faulted seismic loads at the extended structure center of gravity in the direction of the lowest determined natural frequency. The design pressure of the valve will be simultaneously applied to the valve during the static deflection tests.

B. The valve is cycled while in the deflected position. The time required to open or close the valve in the deflected position will be compared to similar data taken in the undeflected condition to evaluate the significance of any change.

C. Motor operators, external limit switches, and pilot solenoid valves necessary for operation are qualified by IEEE 344-1975 as described in paragraph 3.10.N.2.1.1.

The accelerations which are used for the static valve qualification shall be equivalent, as justified by analysis, to 2.1 g in two orthogonal horizontal directions and 2.1 g vertical. The piping design must maintain the operator accelerations to these levels.

If the natural frequency of the valve is less than 33 Hz, a dynamic analysis of the valve will be performed to determine the equivalent acceleration which will be applied during the static test. The analysis will provide the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant piping response spectra. The adjusted accelerations will be determined using the same conservatisms contained in the 2.1-g-horizontal and 2.1-g-vertical accelerations used for rigid valves. The adjusted acceleration will then be used in the static deflection test and the valve operability will be assured by the methods outlined above, using the modified acceleration input.

The above testing program applies to valves with extended structures. The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types are tested. Valve sizes, which cover the range of sizes in service, are qualified by the tests, and the results are used to qualify all valves within the intermediate range of sizes.

Valves that are safety-related but can be classified as not having an extended structure, such as check valves and safety valves, are considered separately.

Check valves are characteristically simple in design, and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact, and there are no extended structures or masses whose motion could cause distortion which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve, since the valve disc is typically designed to be isolated from the body wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any body distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard methods, the ability of the valve to operate is assured by the design features. The valve will also undergo in-shop hydrostatic test, in-shop seat leakage test, and periodic in situ valve exercising and inspection to assure the functional ability of the valve.

For NSSS check valves, generic testing has been performed to determine the performance characteristics for various sizes of check valves. The performance characteristics include flow required to open, pressure drop, etc. These tests demonstrate the valves will be fully open during the design conditions, therefore precluding cycling of the valve which results in wear.

In addition, the ability of the valve to open is assured by its inherent design characteristics. The swing check design and the clearance between the disc hanger assembly and body preclude the possibility of binding.

The methodology used for system layout is per Westinghouse document 1.12, "Systems Standard Design Criteria NSSS Layout Guidelines." In addition, valve sizing is determined by line size and flowrates at which the valve is required to operate.

The pressurizer safety valves are qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves): stress and deformation analyses of critical items which may affect operability for faulted condition loads, in-shop hydrostatic and seat leakage tests, and periodic in situ valve inspection. In addition to these tests, a static load equivalent to that applied by the faulted condition is applied at the top of the bonnet, and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capabilities during a seismic event.

Using these methods, all the safety-related valves in the system are qualified for operability during a faulted event. These methods outlined above conservatively simulate the seismic event and assure that the active valves will perform their safety-related function when necessary.

Degraded conditions as discussed in the Standard Review Plan Section 3.10, paragraph II.1.a(2) are minimized during system design to preclude the presence of debris, impurities, and containments in the fluid system. For example, containment sump design considers the presence of debris as described in Final Safety Analysis Report subsection 6.2.2. Other degraded conditions, such as motive power fluctuations, air pressure, etc., are addressed by testing or analysis showing that sufficient margin was included in the design of the equipment to perform its function.

For the types of valves supplied for nuclear steam supply system active service, supplemental tests and analytical data confirm that flow-induced loadings and thermal loads, when considered in conjunction with seismic and operating loads, have no significant effect on valve operability. The same is true for valve end loads.

Conformance to Regulatory Guide 1.148, concerning active valve assemblies, is addressed in table 3.9.B.3-10. In addition, procurement specifications for replacement components (active safety related) will be consistent with the original purchased equipment to the extent of conformance discussed.

3.10.N.2.2.3 Pump Motor and Valve Operator Qualification

Active pump motors, vital pump appurtenances, active valve motor operators, limit switches, and solenoid valves are seismically qualified in accordance with IEEE 344-1975 as discussed in the appropriate EQDPs of reference 1.

3.10.N.2.3 <u>Seismic Qualification of Other Seismic Category 1 Mechanical Equipment</u>

For Seismic Category 1 mechanical equipment not defined as active, Westinghouse utilizes analysis to demonstrate structural integrity. The analysis methods used by Westinghouse are described in sections 3.7.N and 3.9.N and reference 3.

3.10.N.2.4 <u>References</u>

1. "Equipment Qualification Data Package," Supplement 1 to <u>WCAP-8587</u>, April 1983.
2. Butterworth, G., and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment," <u>WCAP-8587</u> Revision 6, April 1983.
3. Kelly, R. E., and McInerney, J. J., "Methodology for the Seismic Qualification of Westinghouse WRD Supplied Equipment," <u>WCAP-9714</u>, May 1980.

3.10.B.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF MECHANICAL AND ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Analyses or tests are performed for all safety-related, Seismic Category 1 electrical and mechanical equipment supports to ensure their structural capability to withstand seismic excitation.

Information concerning the structural integrity of pressure- retaining components, their supports, and core supports is presented in subsection 3.9.B.3. The following bases are used in the design and analysis of cable tray supports and instrument tubing supports.

A. The methods used in the seismic analysis of cable tray supports are described in subsection 3.7.B.3. The amplification of seismic loads due to the flexibility of the supporting system, if any, is accounted for in the design of the cable trays.

B. The Seismic Category 1 instrument tubing systems are supported so that the allowable stresses permitted by Section III of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are not exceeded when the tubing is subjected to the loads specified in section 3.9 and Regulatory Guide 1.48 for Class 2 and 3 piping.

For field-mounted instruments the supports are tested or analyzed to meet the following:

- The field mounting supports for Seismic Category 1 instruments excluding line-mounted instruments have a fundamental frequency of 33 Hz or greater, with the weight of the instrument included. If, however, the mounting should be flexible (i.e., frequency <33 Hz), the dynamics of the support are considered in the qualification of the supported instrument.
- The stress level in the mounting support or piping does not exceed the material allowable stress when subjected to the maximum acceleration level of the mounting location. The weight of the instrument is included.

In some cases, panels and racks supporting Seismic Category 1 devices are tested and/or analyzed with equipment installed. If the equipment is in an inoperative mode during the support test, the response at the equipment mounting location is monitored. In such a case, devices are qualified separately, and the actual input to the equipment is more conservative in amplitude and frequency content than the response monitored at the equipment location. The required response spectra (RRS) for devices (i.e., in-cabinet response spectra) are generated and as shown in the individual EQDPs applicable to the device and the test response spectra to which the device is qualified envelops the RRS measured at the device mounting location.

3.10.N.3 METHOD AND PROCEDURES FOR QUALIFYING SUPPORTS OF ELECTRICAL EQUIPMENT, INSTRUMENTATION, AND MECHANICAL COMPONENTS

The equipment qualification data packages (EQDPs) contained in reference 1 identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure subsequent in-plant installation does not prejudice the qualification established by Westinghouse.

Westinghouse provides interface requirements to the installer for the mounting of mechanical equipment. Where Westinghouse supplies component supports, the support configuration is addressed in the seismic and dynamic qualification. The criteria used to ensure the structural integrity of component supports are defined in section 3.9.N.

3.10.N.3.1 Test Configurations

Westinghouse seismic qualification testing configurations are designed to represent typical plant installation for the tested component. Interface requirements are defined based on the test configuration and other design requirements. Installation is then completed in accordance with the component interface and installation requirements. Any dynamic coupling effects that result from mounting the component in accordance with these interface criteria are adequately simulated during the test program.

3.10.N.3.2 Reference

1. "Equipment Qualification Data Package," Supplement 1 to WCAP-8587, April 1983.

3.10.B.4 OPERATING LICENSE REVIEW

3.10.B.4.1 Qualification and Documentation Procedures

Qualification and documentation procedures for safety-related Seismic Category 1 mechanical and electrical equipment are in accordance with the recommendations contained in the Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975 and Regulatory Guide 1.100.

EQDPs for balance of plant equipment are prepared for all safety-related Seismic Category 1 electrical and mechanical equipment. Balance of plant EQDPs include qualification plan/procedures and reports documenting that criteria established in subsections 3.10.B.1, 3.10.B.2, and 3.10.B.3 have been satisfied.

3.10.B.4.2 Standard Review Plan Evaluation

The following summary describes the Standard Review Plan differences in regard to seismic and dynamic qualification of mechanical and electrical equipment:

A. The Standard Review Plan requires that equipment be tested in the operational condition and that loadings simulating normal plant conditions should be superimposed on seismic and dynamic loads. This includes flow-induced loads and degraded flow conditions. VEGP tests for nuclear steam supply system (NSSS) equipment are made in the operational conditions where practical, simulated as appropriate, or addressed by analysis. Flow loads are not superimposed on seismic loads for valve operability tests.

   Full operational conditions are simulated because of the impracticality of using actual conditions during testing. Flow loads for valve static operability tests are calculated, and, if significant, the effects are simulated in the test.

B. Evaluation is performed to ensure that test configurations conservatively simulate actual field mountings.

   Seismic qualification testing configurations are designed to represent typical plant installation. Interface requirements are simulated during the test in accordance with field installation. For generic testing, evaluation is performed to ensure test configurations conservatively simulate actual field mounting.

C. The VEGP does not apply end loadings to active valves during static deflection tests. End loads are evaluated in the active valve analysis.

D. The VEGP does not analyze valve discs for Δp or impact energy resultant from a loss-of-coolant accident (LOCA), except for certain cases where a significant impact from the LOCA is expected.

Westinghouse performs design verification on vendor-supplied valves to ensure that the valves are designed properly and meet the stress acceptance criteria in the equipment specification and in the American Society of Mechanical Engineers Code standards. For balance of plant vendors, loadings simulating Δp conditions are calculated and included in the static operability test, if significant.

E. The VEGP does not utilize Regulatory Guide 1.92 guidance for combination of multimodal or multidirectional responses.

For balance of plant equipment, VEGP uses a Bechtel topical report (BC-TOP-4A) as guidance for combination of multimodal response. This report meets the intent of Regulatory Guide 1.92. For NSSS equipment, Westinghouse uses methods previously justified and accepted by the Nuclear Regulatory Commission.

F. The program for environmental qualification of active mechanical equipment is based on a combination of design, testing, and analysis of critical components as discussed in the Final Safety Analysis Report paragraph 3.11.B.2.

G. The VEGP documentation file conforms to the Standard Review Plan guidelines with the exception of certain NSSS documentation which remains in an auditable file at Westinghouse.

Information that Westinghouse considers proprietary will be summarized in EQDPs located in a central file. Detailed documentation will be available for auditing at Westinghouse.

3.10.N.4 OPERATING LICENSE REVIEW

The results of tests and analyses that ensure that the criteria established in subsection 3.10.N.1 have been satisfied, employing the qualification methods described in subsections 3.10.N.2 and 3.10.N.3, are included in the individual equipment qualification data packages and test reports referenced in the equipment qualification central file.

3.10.N.4.1 Documentation

Seismic qualification of equipment is documented in test reports, analysis reports, calculation notes, etc., contained in the Westinghouse files. Westinghouse satisfies existing regulatory requirements for documentation as further described in chapter 17.

3.10.N.4.2 Standard Review Plan Evaluation

The following summary describes the Standard Review Plan differences in regard to seismic and dynamic qualification of mechanical and electrical equipment:

A. The Standard Review Plan requires that equipment be tested in the operational condition and that loadings simulating normal plant conditions should be superimposed on seismic and dynamic loads. This includes flow-induced loads and degraded flow conditions. VEGP tests for nuclear steam supply system (NSSS) equipment are made in the operational conditions where practical, simulated as appropriate, or addressed by analysis. Flow loads are not superimposed on seismic loads for valve operability tests.

Full operational conditions are simulated because of the impracticality of using actual conditions during testing. Flow loads for valve static operability tests are calculated, and, if significant, the effects are simulated in the test.

B. Evaluation is performed to ensure that test configurations conservatively simulate actual field mountings.

Seismic qualification testing configurations are designed to represent typical plant installation. Interface requirements are simulated during the test in accordance with field installation. For generic testing, evaluation is performed to ensure test configurations conservatively simulate actual field mounting.

C. The VEGP does not apply end loadings to active valves during static deflection tests. End loads are evaluated in the active valve analysis.

End loads are evaluated in the valve analysis.

D. The VEGP does not analyze valve discs for Δp or impact energy resultant from a loss-of-coolant accident (LOCA), except for certain cases where a significant impact from the LOCA is expected.

Westinghouse performs design verification on vendor-supplied valves to ensure that the valves are designed properly and meet the stress acceptance criteria in the equipment specification and in the American Society of Mechanical Engineers Code standards. For balance of plant vendors, loadings simulating Δp conditions are calculated and included in the static operability test, if significant.

E. The VEGP does not utilize Regulatory Guide 1.92 guidance for combination of multimodal or multidirectional responses.

For balance of plant equipment, VEGP uses a Bechtel topical report (BC-TOP-4A) as guidance for combination of multimodal response. This report meets the intent of Regulatory Guide 1.92. For NSSS equipment, Westinghouse uses methods previously justified and accepted by the Nuclear Regulatory Commission.

F. The program for environmental qualification of active mechanical equipment is based on a combination of design, testing, trending, and analysis of critical components as discussed in Final Safety Analysis Report paragraph 3.11.B.2. This program will be supported by periodic plant testing and maintenance/surveillance programs. This program meets the intent of the Standard Review Plan.

G. The VEGP documentation file conforms to the Standard Review Plan guidelines with the exception of certain NSSS documentation which remains in an auditable file at Westinghouse.

Information that Westinghouse considers proprietary will be summarized in equipment qualification documentation packages located in a central file. Detailed documentation will be available for auditing at Westinghouse.

3.11.B ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section provides environmental qualification[a] information which verifies the capability of safety-related mechanical and electrical equipment to perform its design functions under all normal, abnormal, and design basis accident (DBA) conditions. This section provides the information for the balance of plant (BOP) equipment. Section 3.11.N provides the information for the nuclear steam supply system (NSSS). Mechanical and electrical equipment covered by this section includes equipment located in a harsh environment and associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal.

With the exception of diaphragm valves, which are addressed in table 1.9-1, the replacement/refurbishment of safety-related mechanical and electrical equipment in a mild environment and safety-related mechanical equipment in a harsh environment is based on a combination of design life, trending, and periodic maintenance and surveillance.

The information in this section includes a definition of applicable environmental conditions, requirement for documentation of the qualification tests and analysis performed on equipment, and demonstration of adequacy of the equipment qualification program.

The seismic qualification of mechanical and electrical equipment is presented in section 3.10.B.

3.11.B.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Table 3.11.B.1-1 and figure 3.11.B.1-1 identify applicable normal, abnormal, and DBA environmental conditions conforming to 10 CFR 50, Appendix A, General Design Criterion 4. These environmental conditions are associated with various plant areas by building, by an environmental designator. The environmental designator is composed of the following elements:

- IA and IB - Containment (upper and lower).
- II - Main steam isolation valve (MSIV)/main feedwater isolation valve (MFIV) area.
- III - Outside areas.
- IV - Diesel generator structures.
- V - Equipment building.
- VI - Auxiliary feedwater pump area.
- VII - Fuel handling building.
- VIII - Auxiliary building.
- IX - Control building.

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[a] The renewed operating licenses authorize an additional 20-year period of extended operation for both VEGP units resulting in a plant operating life of 60 years. The EQ program is credited to continue to manage aging effects associated with the EQ equipment for the period of extended operation (see subsections 19.3.1 and 19.4.3). Applicable EQ evaluations based on a 40-year design life were evaluated as time-limited aging analyses (TLAAs) for license renewal and will be revised as necessary to reflect the 60-year plant operating life before the units enter the period of extended operation.

X - Turbine building.

XI - Not used.

XII - Nuclear service cooling water structure.

XIII - Radwaste structures (abandoned in-place).

For mild environments, the area conditions do not change as the result of an accident; as a result, there are no degrading environmental effects that could lead to common mode failure of safety-related equipment.

The environmental conditions identified in table 3.11.B.1-1 are defined as follows:

A. Normal operating environmental conditions are defined as those conditions that exist during routine plant operations for which the equipment is expected to perform its safety functions, as required, on a continuous basis.

B. Abnormal/test environmental conditions are those plant conditions for which the equipment is designed to operate for a period of time without accelerating normal periodic tests, inspections, and maintenance schedules for that equipment. These conditions are reviewed on a case-by-case basis for severity and duration when required. The maximum and minimum conditions identified as the abnormal condition are based on the design temperature limits for the affected areas.

C. DBA and post-DBA conditions are those plant conditions resulting from various postulated equipment and piping failures during which safety-related equipment must operate without impairment of the safety-related function. The DBA and post-DBA conditions as shown on figure 3.11.B.1-1 are enveloping values. Qualification may be based on component specific calculations when necessary.

Compatibility of equipment with the specified environmental conditions is assured by the following:

A. Systems and components required to mitigate the consequences of a DBA or to perform safe shutdown operation are qualified to remain functional after exposure to the environmental conditions in table 3.11.B.1-1.

B. Safety-related systems and components have a qualified life goal of 41 years[a] (including 1 year of post-DBA operating time). Demonstration of qualified life by test and/or analysis with adequate justification is provided by equipment suppliers, including the effects of aging when applicable. For critical components susceptible to aging, a qualified life was established which includes the effects of the total integrated radiation dose experienced at their respective locations within the plant. When a 41-year[a] qualified life was not possible, a shorter qualified life was established, and a replacement program was implemented.

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[a] The renewed operating licenses authorize an additional 20-year period of extended operation for both VEGP units resulting in a plant operating life of 60 years. The EQ program is credited to continue to manage aging effects associated with the EQ equipment for the period of extended operation (see subsections 19.3.1 and 19.4.3). Applicable EQ evaluations based on a 40-year design life were evaluated as TLAAs for license renewal and will be revised as necessary to reflect the 60-year plant operating life before the units enter the period of extended operation.

C. Equipment qualification has taken into account the most severe environmental conditions resulting from the design basis high-energy line break. Included in these conditions are the short-term peak transient temperature following a main steam line break (MSLB), radiation exposure, temperature, and chemical spray due to a loss-of-coolant accident (LOCA) within the reactor containment. For equipment located inside containment, the surface temperature does not exceed the saturation temperature at containment peak steam partial pressure.

For equipment located outside containment, the effects of post-LOCA recirculating fluids outside containment are included in the total integrated doses to the equipment in the affected areas. Postulated high energy line failures (as defined in paragraph 3.6.2.1.2) are assumed in all areas where high energy lines (greater than 1 in.) are routed. The main steam and main feedwater isolation valve areas have been evaluated using VEGP specific blowdown data which includes superheat.[(1)] The main steam piping in the MSIV compartments is designed to the break exclusion (superpipe) criteria of Branch Technical Position MEB 3-1 item B.1.b for the portions of piping passing through the primary containment and extending to the first five-way restraint past the MSIVs as discussed in paragraph 3.6.2.1.1.D.

The following cases were reanalyzed using a VEGP specific model and input.

- Two power levels were assumed: 70% and 102% of 3579 MWt.
- Ten break sizes were assumed: from a larger break of 1.0 ft^2 to a smaller break of 0.1 ft^2 at increments of 0.1 ft^2.

These break sizes include the following:

— 1.0 ft^2 is the largest postulated break in superpipe.

— 0.5 ft^2 is the largest branchline break.

- Two auxiliary feedwater (AFW) conditions were assumed: three AFW pumps (nominal and maximum flow) and two AFW pumps (nominal flow)

Cases were run with and without the turbine-driven pump available to model superpipe breaks where a single failure is not considered and branchline breaks where a single failure is considered. Additional cases were run reflecting minimum and maximum initial steam generator inventory.

The analyses of the environmental response of each MSIV compartment to MSLBs with superheated steam blowdown is consistent with the requirements of NUREG-0588. These analyses were completed using the EPRI computer code "GOTHIC." The GOTHIC code calculates heat transfer to the surrounding structure surfaces.

A facility response evaluation was performed to determine if the equipment was essential for an MSLB in the area, and the environmental qualification test reports for the essential equipment were reviewed to ensure that the equipment was qualified for the MSLB event. For four components (MSIVs, MSIV bypass valves, steam generator atmospheric relief valves, and auxiliary feedwater discharge valves), the maximum MSLB environmental temperatures achieved during the qualification tests did not envelope the maximum MSLB environmental temperature profiles considering superheat developed for the control building and auxiliary building MSIV compartments. A thermal lag analysis was performed on these components to demonstrate that the actual safety-related component temperature achieved under the VEGP superheated MSLB conditions is less than the component temperature reached in the qualification testing program.

The essential equipment for an MSLB in the auxiliary and control building MSIV compartments has successfully completed environmental qualification test programs which, in conjunction with thermal lag analysis, demonstrate that the equipment is qualified for the maximum MSLB environmental temperature postulated in these compartments. It is concluded that no required safety components are precluded from performing their safety function in the event of an MSLB in either of the MSIV compartments. Therefore, no safety implications exist to prevent safe shutdown of the VEGP.

Where post-DBA functional requirements after performing the safety-related function are different from the DBA requirements, the following are considered:

1. Such components inside the containment consider the effects of temperature, pressure, relative humidity, radiation, and chemical parameters during the post-DBA period.
2. Such components outside the containment are designed for the required temperature pressure and other environmental conditions.

Specific supporting qualification information for each mechanical and electrical equipment component is contained in the applicable BOP equipment qualification data package (EQDP).

D. Rooms with safety-related equipment necessary for safe shutdown were evaluated to assure that the temperatures would be within acceptable values in order to determine the necessity for temperature surveillance. Rooms that met any of the following criteria do not require surveillance:

- Rooms with safety-related HVAC systems.
- Rooms with a high temperature alarm that annunciates in the main control room when the normal design temperature is exceeded. High temperature alarms require the same level of action for exceeding the design temperature limits as those described in section 16.3, Area Temperature Monitoring.
- Rooms classified as mild environment where the calculated room temperature with a loss of normal HVAC for 7 days is less than 150°F.
- Rooms with natural ventilation and designed for ambient conditions (e.g., NSCW pumphouse).

Rooms that contain safety-related equipment necessary for safe shutdown that do not meet any of the above criteria were also evaluated based on historical records of temperatures. These records indicated that the temperatures remain within their environmental qualification limits with the normally provided HVAC in operation, but that upon failure of the HVAC it is possible for the room temperature to exceed the values assumed for equipment environmental qualification. HVAC systems were operated in a degraded condition during high ambient conditions during the summer to determine what minimum ventilation is required to maintain temperatures below the limits listed in FSAR table 16.3-6. Therefore, surveillance of these rooms is conducted in accordance with section 16.3, Requirement 7 - Area Temperature Monitoring, in the event of a loss of the normal HVAC.

The qualified life of environmentally qualified equipment is based on the environmental conditions indicated in table 3.11.B.1-1 and/or the results of temperature monitoring. Area temperature monitoring in conjunction with the evaluation described above provides reasonable assurance that the normal temperature used for environmental qualification of safety-related equipment necessary for safe shutdown is not exceeded without an evaluation of equipment operability.

Nonactive mechanical equipment whose only safety function is structural integrity is designed in accordance with the American Society of Mechanical Engineers Code. The accident and post-accident environmental effects are considered in the design of such structural components as pump casings and valve bodies. The environmental qualification program, therefore, is restricted to evaluating the design of critical nonmetallic subcomponents of active devices in a harsh environment where failure could result in loss of the active component.

Active mechanical equipment (pumps and valves) is qualified for operability as discussed in subsection 3.9.B.3 and section 3.10.B. This operability program, combined with the qualification of the electrical appurtenances (motors, valve operators, solenoids, limit switches etc.), demonstrates qualification under all required environmental conditions. Active mechanical equipment is defined as that equipment that must perform a mechanical motion as part of its safety function.

3.11.B.1.1 Standard Review Plan Evaluation

The VEGP addresses the guidelines of NUREG-0588 by providing environmental qualification of mechanical equipment using a combination of test and analysis rather than by testing alone.

The environmental qualifications of mechanical equipment is addressed through the stringent selection of materials for use under adverse environmental conditions. This selection is supported by partial type testing and materials analysis and evaluation to confirm the adequacy of the materials used. If sufficient documentation is not found on materials, then equipment is qualified by testing using the guidance of NUREG-0588.

3.11.B.1.2 Reference

1. Georgia Power Company letter from D. O. Foster to NRC, dated June 25, 1986.

3.11.N ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that the mechanical and electrical portions of the engineered safety features (ESF) and the reactor protection systems are capable of performing their designated safety-related functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. The information presented includes identification of the safety-related equipment that is within the scope of the Westinghouse nuclear steam supply system (NSSS) and, for each item of equipment, the designated safety-related functional requirements, definition of the applicable environmental parameters, and documentation of the qualification process employed to demonstrate the required environmental capability. The seismic qualification of NSSS-supplied safety-related mechanical and electrical equipment is presented in section 3.10.N.

3.11.N.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

3.11.N.1.1 Equipment Identification

A complete list of safety-related electrical and active mechanical equipment within the Westinghouse NSSS scope of supply that is essential to emergency reactor shutdown, containment isolation, reactor core cooling, or containment and reactor heat removal or that is otherwise essential in preventing significant release of radioactive material to the environment is provided in table 3.2.2-1. A master list of safety-related electrical and mechanical equipment along with a summary of electrical equipment qualification results will be maintained as part of the equipment qualification central file.

3.11.N.1.2 Definition of Environmental Conditions

The plant-specific normal, abnormal, accident and post-accident conditions are defined in subsection 3.11.B.1. The parameters considered for the Westinghouse qualification program are included in the equipment qualification data packages (EQDPs) found in WCAP-8587, Supplement 1.

3.11.N.1.3 Equipment Operability Times

For Westinghouse NSSS supplied Class 1E electrical and active mechanical equipment, post-accident operability times are generally defined as follows:

Equipment	Required Post-Accident Operability
Equipment necessary to perform trip functions	5 min
Equipment that is located outside containment, is accessible, and can be repaired, replaced, or recalibrated	2 weeks
Equipment located inside containment that is inaccessible and is required for post-accident monitoring	4 months (This number is based on an acceptable amount of time to allow the instrument to be repaired, replaced, or recalibrated, or for an equivalent indication to be obtained.)
Equipment that is located inside containment, is inaccessible, or cannot be repaired, replaced, or recalibrated	1 year
Equipment located in a mild environment following an accident	Continuous

Specific information for each device qualified as part of Westinghouse's Institute of Electrical and Electronic Engineers (IEEE) 323-1974 qualification program is contained in the appropriate EQDP.

In general, the active mechanical component is qualified for operability as discussed in section 3.10.N utilizing test, analysis, or a combination of tests and analyses. This operability program, combined with the qualification of the electrical appurtenances (motors and valve operators, etc.) discussed in the appropriate EQDPs, demonstrates qualification for the required post-accident times.

3.11.N.1.4 Standard Review Plan Evaluation

The VEGP addresses the guidelines of NUREG-0588 by providing environmental qualification of mechanical equipment using a combination of test and analysis rather than by testing alone.

The environmental qualification of mechanical equipment is addressed through the stringent selection of materials for use under adverse environmental conditions. The selection is supported by partial type testing and materials analysis and evaluation to confirm the adequacy of the materials used. If sufficient documentation is not found on materials, then equipment is qualified by testing using the guidance of NUREG-0588.

3.11.B.2 QUALIFICATION TESTS AND ANALYSES

Qualification of safety-related equipment located in a harsh environment is based on type testing of actual or similar equipment or by analysis. Testing is used to demonstrate the ability of the equipment to perform its required safety-related function for a definable period. The type test includes, as a minimum, thermal and mechanical aging, radiation, and exposure to extremes of environmental, seismic, and vibration effects. Type testing is done with representative samples of the production line equipment and simulated service conditions generally in accordance with the sequence indicated in the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974 to the specified service conditions, including margin, and takes into account normal and abnormal plant operation and DBA and post-DBA operations.

A master list of safety-related electrical and mechanical equipment along with a summary of electrical equipment qualification results will be maintained as part of the equipment qualification file for harsh environment only.

IEEE Standard 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, and NRC Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants, are used to establish acceptance criteria for environmental qualification for all safety-related equipment located in a harsh environment. Other regulatory guides providing guidance for meeting the requirements of 10 CFR 50 Appendix A, General Design Criteria 1, 4, 23, and 50 and Appendix B, Criterion III to 10 CFR 50 include Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment; Regulatory Guide 1.40, Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants; Regulatory Guide 1.63, Electrical Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants; Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants; and Regulatory Guide 1.100, Seismic Qualification of Electric Equipment for Nuclear Power Plants; and Regulatory Guide 1.131, Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants.

Additional information regarding conformance with each of these Regulatory Guides is given in section 1.9.

When sufficiently reliable data and proven analytical methods are available, environmental qualification may be based on analysis.

Qualification by analysis includes justification of the methods, theories, and assumptions used (i.e., mathematical or logical proof based on actual test data) that the equipment meets or exceeds its specified performance when subjected to normal, abnormal, and design basis event (DBE) environmental conditions.

For mechanical equipment in the harsh environmental, post-DBA environmental effects on structural components such as valve and pump casings and bodies is considered negligible. Critical subcomponents are evaluated and qualified as required. Organic materials, such as gaskets, seals, and O-rings, are evaluated for their function and evaluated separately. Combined test and analysis therefore is the qualification approach used for this equipment if located in harsh environments.

With the exception of diaphragm valves, which are addressed in table 1.9-1, replacement/refurbishment intervals for safety-related equipment located in a mild environment and safety-related mechanical equipment in a harsh environment are based on a combination of design life, trending, and periodic maintenance and surveillance.

Prior to fuel loading, VEGP will implement administrative controls of component qualification involving an equipment qualification file, handling of documentation, internal acceptance review procedures, and a maintenance surveillance program.

Planned maintenance and surveillance is a VEGP program that schedules equipment maintenance, calibration, and surveillance activities. Its purpose is to maintain equipment in a condition safe for operations, minimize unplanned outages due to breakdown, and provide a mechanism by which greater than anticipated degradation of safety-related equipment can be detected and remedied. The program is being developed using personnel experience in the area of maintenance and surveillance of electrical, mechanical, and instrumentation and controls equipment.

Under the program, a planned maintenance and surveillance checklist is prepared for each piece of safety-related equipment that identifies the maintenance and surveillance tasks to be performed. If the task requires removing components for internal inspection, an equipment-specific procedure is referenced describing how the removal and inspection is to be performed. If the task only requires visual inspection of the outer areas of the equipment, it is described on the checklist. The content of the program is derived from the following sources:

- Manufacturer/vendor recommendations.
- Lubrication requirements.
- Calibration requirements.
- Field verification of equipment descriptions.
- Industry experience.
- Qualification test/analysis results.

The information is then used in preparing equipment qualification data packages (EQDPs) which include maintenance and surveillance requirements specified by vendors in their test reports. Safety-related equipment located in a mild environment does not require environmental qualification documentation for inclusion in the EQDPs.

All EQDPs were originally transmitted to the equipment qualification task force (EQTF), which reviewed the vendor's information and the architect-engineer's evaluation for completeness and validity. Upon approval, the EQTF transmitted a copy of the EQDP to the equipment qualification (EQ) group. The EQ group identified all equipment tag numbers included in the EQDP and transmitted this information along with the maintenance and surveillance requirements from the EQDP to the maintenance performance teams. Maintenance performance teams then prepared replacement schedules, planned maintenance and surveillance checklists, and procedures (if necessary).

When required, corrective maintenance will be performed to assure that equipment will operate satisfactorily. Such corrective maintenance will become part of an equipment history file. Proper documentation of corrective maintenance actions will highlight recurring situations in similar equipment and will provide data to identify component past-performance trends.

Furthermore, equipment or component failures detected in other nuclear power plants will be available to VEGP through industry event reports, Nuclear Regulatory Commission inspection and enforcement bulletins, information notices, letters, and directives, and manufacturer's information notices. VEGP will review these reports to determine their applicability and will modify its maintenance and surveillance program accordingly.

In addition to that required by the planned maintenance and surveillance checklists, surveillance and operability testings are performed by the VEGP inservice testing and the Technical Specifications surveillance programs to address mechanical aging. The inservice testing is conducted in accordance with American Society of Mechanical Engineers (ASME) Section XI. All safety-related pumps with a Class 1E power source and selected safety-related active valves required by ASME Section XI are tested on a regular basis. The pump testing includes measurement of differential pressure at a predetermined flow rate, vibration amplitude, bearing temperatures and inlet pressure. The valve testing includes determination of leak rates for certain isolation valves, stroke times for certain power operated valves, fail safe verification, position indication verification, safety valve set point verification, and opening force or torque for check valves with external lever arms. The Technical Specification surveillance program covers all equipment required by the VEGP Technical Specifications. The VEGP Technical Specifications specify requirements for the test frequency, acceptability of testing, and measured parameters.

As described in FSAR paragraph 3.11.B.1.1 and in conformance with NUREG-0588 guidelines, VEGP has prepared and issued specifications identifying qualification methods and requirements for safety-related mechanical and electrical equipment. This specification requires suppliers to meet IEEE 323-1974 section 6 paragraph 6.3.2 sequence for safety-related equipment located in harsh environments. Subsections of paragraph 6.3.2 regarding "Aging" or "Qualified-Life" do not apply to safety-related equipment located in mild environments. The specification also required suppliers to simulate mechanical aging by operating equipment to simulate the expected mechanical wear.

For qualification tests of electric valve operators, the specification requires vendors to follow the direction given in Section 4 of IEEE Standard 382-1972 as committed to in VEGP FSAR. The vendors are also required to provide complete justification that the sequence used is the most severe for the item being tested.

For NSSS equipment, the impact of mechanical aging effects on equipment qualification has been addressed in the design process and/or as part of specific qualification test program. Significant mechanical aging effects can result from system piping vibration and cycling of equipments. Equipment where such effects may be significant are active pumps and line-mounted equipment (active valves, resistance temperature detectors, etc.).

Relative to system/piping vibration, FSAR paragraph 3.9.B.2 describes the preoperational piping vibration program which is being implemented on VEGP. This program assures that piping system vibration limits are within ASME code limits. This testing program includes transient testing such as the opening and closing of valves and the stopping and starting of pumps. This program assures that piping vibration effects are not significant.

Relative to active valves, the testing program described above assures that piping vibration effects on the mechanical portions of the valve are insignificant. For valve appurtenances, vibration aging is included in the qualification test program.

Requirements for the cycling of valves are included in valve specifications. This requirement has been factored into valve designs by selecting proper materials, assuring proper clearances, and internal part interfaces. Also, several valve vendors have performed valve cycle tests to demonstrate that the Class 1 valves have performed and design transients for Class 2 and 3 valves are evaluated. Finally, the valves supplied to VEGP by Westinghouse are similar to these and other Westinghouse plants which have been operating for extended periods of time.

3.11.N.2 QUALIFICATION TESTS AND ANALYSIS

3.11.N.2.1 Environmental Qualification Criteria

The methods of meeting the general requirements for environmental design and qualification of safety-related equipment as described by General Design Criteria (GDC) 1, 2, 4, 23, and 50 are described in section 3.1. Additional specific information concerning the implementation of GDC 23 and 50 is provided in paragraph 7.2.2.2 and section 6.2, respectively. The general methods of implementing the requirements of Appendix B to 10 CFR 50 are described in chapter 17. Regulatory Guides 1.40, 1.73, and 1.89 concerning environmental qualification are addressed in section 1.9. Administrative control of component qualification is addressed in paragraph 3.11.B.2.

3.11.N.2.2 Environmental Design of Mechanical Equipment

Westinghouse-supplied safety-related mechanical components have been designed to perform their required safety functions under the appropriate environmental effects of normal, abnormal, accident, and post-accident conditions as required by GDC 4. For mild environments, the area conditions do not change as the result of an accident; as a result, there are no degrading environmental effects that could lead to common mode failure of safety-related equipment. However, mechanical equipment located in harsh environmental zones must be designed to perform under the appropriate environmental conditions.

For safety-related mechanical equipment, there are two basic categories of components as follows:

A. Active equipment - equipment that must perform a mechanical motion as part of its safety function.

B. Nonactive equipment - equipment whose only safety function is structural integrity. Nonactive components are designed for structural integrity in accordance with Section III of the American Society of Mechanical Engineers (ASME) Code as discussed in section 3.9.N.

Based on the above information, the VEGP program for environmental qualification of mechanical equipment discussed in this section pertains strictly to active components that must perform their safety functions in a harsh environment. The post-accident environmental effects are considered to be negligible on structural components such as pump casings and valve bodies. Therefore, the program is restricted to evaluating the design of critical subcomponents of active devices whose failure could result in loss of the active components.

The design of these components is based in part on the stringent selection of materials utilized in safety-grade mechanical components. Organic materials, such as gaskets, seals, and O-rings, are selected for use based on their capability to perform in a nuclear environment. In some cases, partial type tests have been performed to confirm the adequacy of selected materials or subcomponents for use under adverse environmental conditions. Mechanical aging of safety-related pumps and valves is addressed in paragraph 3.11.B.2.

The program for environmental qualification of active mechanical components is based on a combination of design, test, trending, and analysis of critical subcomponents, which is supported by periodic plant test and maintenance/surveillance programs. This program addresses the requirements of GDC 4 and provides adequate assurance that active components will perform their required functions under all normal, abnormal, accident, and post-accident environmental conditions.

3.11.N.2.3 Environmental Design of Electrical Equipment

The Westinghouse approach for environmental qualification of nuclear steam supply system (NSSS)-supplied Class 1E equipment is outlined in reference 2. This methodology was developed based on the guidelines provided in the Institute of Electrical and Electronics Engineers (IEEE) Standards 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, and 344-1975, IEEE Recommended Practices for Seismic Qualifications of Class 1E Equipment for Nuclear Power Generating Stations.

Westinghouse meets the IEEE Standard 323-1974 (augmented by Regulatory Guide 1.89), including IEEE Standard 323a-1975, the Nuclear Power Engineering Committee Position Statement of July 24, 1975, by either type test, operating experience, analysis, or an appropriate combination of these methods. Westinghouse meets this commitment by employing the methodology described in WCAP-8587. The Westinghouse program outlined in WCAP-8587 has been approved as an acceptable program for demonstrating qualification of Class 1E equipment to the requirements of Regulatory Guides 1.89 and 1.100 and IEEE Standards 323-1974 and 344-1975(3).

3.11.N.2.4 References

1. Buttersworth, G., and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587, Revision 3, May 5, 1980.
2. Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment, WCAP-8587.
3. NRC letter from C. O. Thomas to E. P. Rahe, Jr., Westinghouse Electric Corporation, dated November 10, 1983.

3.11.B.3 QUALIFICATION TEST RESULTS

The results of the qualification program for the balance of plant (BOP) equipment are provided in the appropriate BOP equipment qualification data packages (EQDP).

Each BOP EQDP contains qualification procedures and reports demonstrating qualification and provides adequate assurance that safety-related equipment can perform its required safety functions under all required plant conditions.

In addition, table 3.11.B.3-1 provides information relating VEGP conformance to the requirements in NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.

3.11.N.3 QUALIFICATION PROGRAM RESULTS

Qualification program test results for the nuclear steam supply system supplied Class 1E equipment are provided in the equipment qualification central file.

The program for the environmental qualification of safety-related mechanical equipment is outlined in paragraph 3.11.N.2.2. This program was developed to meet the intent of General Design Criterion 4 and provides adequate assurance that safety-related mechanical equipment can perform its required safety functions under all required plant conditions. Qualification results are provided in the equipment qualification central file.

A comparison of NUREG-0588 against WCAP-8587, Supplement 1, is contained in table 3.11.N.3-1.

3.11.4 LOSS OF VENTILATION

The maximum temperatures considered in the sizing of air-conditioning systems serving safety-related systems are determined by analysis of the following factors:

A. Outdoor design temperatures for the VEGP site (both wet bulb and dry bulb readings).

B. Conservative piping thermal loads for the room, using maximum operating temperatures for the pipe contents and maximum footage of active pipe for each mode of operation.

C. Conservative electrical heat loads, assuming full lighting for the room and using the maximum control and equipment resistance losses for each mode of operation.

D. Conservative heat transfer from miscellaneous equipment surfaces, if applicable, e.g., outer surface of the diesel generator.

E. Conservative heat transfer from the surfaces of open pools and tanks, if applicable, using the maximum operating temperature of the contents.

F. Conservative heat transfer from the surfaces of the room including walls, floor, and ceiling or roof.

Safety-related air-conditioning, air handling, or ventilation systems described in section 9.4 are powered from Class 1E electrical power supplies and are provided for the following areas:

- Control room envelope.
- Control building auxiliary relay rooms.
- Control building safety features electrical equipment rooms (including Class 1E battery rooms).

- Class 1E motor control center and switchgear rooms.
- Piping penetration areas.
- Class 1E electrical tunnels.
- Residual heat removal pump, safety injection pump, and containment spray pump rooms.
- Control building control room ESF chiller rooms.
- Centrifugal charging pump rooms.
- Remote shutdown panel rooms.
- Diesel generator buildings.
- Component cooling water pump rooms.
- Auxiliary feedwater pump rooms.
- Spent fuel pool heat exchanger and pump rooms.
- Fuel handling building.
- Containment.

Safety-related air-conditioning systems are designed such that the single failure of an active component, after a design basis accident (DBA), cannot impair the ability of the systems located within the area served by the redundant train of air-conditioning equipment to fulfill their safety functions. Should one train of the safety-related air-conditioning system become inoperative during normal operation, sufficient equipment capacity is still available to mitigate the consequences of a DBA.

Four redundant 100-percent-capacity essential air filtration units are provided in the control building for the common control room complex. It is not considered a credible event to lose all control room air-conditioning simultaneously.

The worst-case environments are presented in table 3.11.B.1-1.

3.11.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

The plant-specific estimates of the radiation dose incurred by equipment during normal operation and the estimated doses and chemical conditions following a loss-of-coolant accident are defined in table 3.11.B.1-1.

3.11.5.1 Chemical Environment

Engineered safety feature (ESF) systems and components are qualified to perform their safety-related functions in the temperature and pressure conditions described in table 3.11.B.1-1. In addition, safety-related components inside the containment are designed to perform their safety-related functions in long-term contact with a combined boric acid- trisodium phosphate solution recirculated through the emergency core cooling system and the containment spray system.

3.11.5.2 Radiation Environment

Safety-related equipment is qualified to perform safety-related functions in the radiation environments present during normal and design basis accident conditions. The normal operational exposure is based upon design source terms presented in chapter 11 and subsection 12.2.1, and the equipment and shielding configurations presented in section 12.3. Post-accident ESF system and component radiation exposures are dependent on the location of the equipment in the plant and include the effect of recirculatory fluid for equipment outside the containment. Source terms and other accident parameters are presented in subsection 12.2.1 and chapter 15 and are consistent with the recommendations of Regulatory Guides 1.89, 1.4, and 1.7. The maximum combined integrated radiation dose for inside containment is based on the integrated effects (beta and gamma) of the normally expected radiation environment over the equipment's installed life plus that associated with the most severe design basis event during or following which the equipment is required to remain functional. Source term and methodology are consistent with that discussed in NUREG-0588. (See table 3.11.B.2-1.) Normal and accident radiation exposures based on the above assumptions are presented in table 3.11.B.1-1.

TABLE 3.11.B.1-1 (SHEET 1 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test[b]		DBA/Post-DBA[c]			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[h]	Pressure[h]	Int Dose (rads)[d]	Normal (%)	DBA (%)
SPRAYED VAPOR REGION[e]											
IB-R-301H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-302H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-101H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-102H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-103H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-104H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-105H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-106H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-109H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-111H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A01H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
UNSPRAYED VAPOR REGION[e]											
IB-R-107H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-108H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-110H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-115H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A02H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A03H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A04H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A05H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A06H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A07H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-A08H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 6	Sheet 11	1.8×10^8	50	100
IB-R-A09H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100

CONTAINMENT BUILDING
(Sheet 1 of 4)

TABLE 3.11.B.1-1 (SHEET 2 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test[b]		DBA/Post-DBA[c]			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[h]	Pressure[h]	Int Dose (rads)[d]	Normal (%)	DBA (%)
UNSPRAYED VAPOR REGION[e]											
IB-R-B03H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-B10H	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 9	Sheet 11	1.8×10^8	50	100
IB-R-B12H[f]	1/2	amb	14.7 psia	-	-	11.7 psia	-	-	-	-	-
IB-R-B13H[f]	1/2	amb	14.7 psia	-	-	11.7 psia	-	-	-	-	-
IB-R-B14H[f]	1/2	amb	14.7 psia	-	-	11.7 psia	-		-	-	-
IB-R-B15H[f]	1/2	amb	14.7 psia	-	-	11.7 psia	-		-	-	-
SUMP REGION[e]											
IA-R-B01H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B02H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B04H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B05H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B06H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B07H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B08H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B09H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B11H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-B16H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-C03H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100
IA-R-C10H[g]	1/2	120	17.7-13.2 psia	2×10^6	120/60	60 psig	Sheet 10	Sheet 11	1.8×10^8	50	100

CONTAINMENT BUILDING
(Sheet 2 of 4)

TABLE 3.11.B.1-1 (SHEET 3 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- Total integrated dose (TID) > 1 x 10^4 rad.

\- = No high-energy line or safety-related equipment located in the room.

b. The containment test pressure is 60 psig.

c. Includes normal doses.

d. Integrated dose = 3 x 10^7 rads (γ), 1.5 x 10^8 rads (β). (Calculated for center of containment building but envelope all areas.)

e. Spray Exposure

The containment building consists of three general regions, defined as follows:

- Sprayed Vapor Region - That area above the operating deck that would be exposed to the direct effects of the spray system.

- Unsprayed Vapor Region - That area of containment that is below the operating deck and above the containment flood level that may be exposed to the indirect effects of the spray system.

CONTAINMENT BUILDING
(Sheet 3 of 4)

TABLE 3.11.B.1-1 (SHEET 4 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES (continued)

- Sump Region - That area of containment that would be flooded post-accident.

  Sprayed and unsprayed vapor regions pH at injection phase (0-100 min) = 4.5.

  Sprayed and unsprayed vapor regions pH at recirculation phase (100 min - 24 h) = 7.0 - 10.5.

  Sump region pH at 0 min - 24 h = 7.0 - 10.5.

  Chemicals - Spray solution is boric acid (2400 - 2600 ppm boron) plus sufficient trisodium phosphate (TSP) to achieve the desired pH.

f. Tendon access shafts which are outside the containment pressure boundary.

g. If equipment is located above flood level, sheet 9 shall be used for temperature. (Flood level is el 181 ft 2 in.)

h. Sheet numbers refer to figure 3.11.B.1-1.

CONTAINMENT BUILDING
(Sheet 4 of 4)

TABLE 3.11.B.1-1 (SHEET 5 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[c]	DBA/Post-DBA Pressure[c]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
CONTROL BUILDING											
II-R-A08H	2	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-A09H	2	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-A56H	1/com	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-A57H	1/com	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-121H	1/com	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-122H	1/com	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-123H	1/com	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-121H	2	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-122H	2	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-123H	2	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
AUXILIARY BUILDING											
II-R-A06H	1/com	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-A11H	1/com	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-A12H	1/com	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-A20H	1/com	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-A99H	2	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-A104H	2	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-A105H	2	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-A107H	2	115	atm	1×10^3	126/17[b]	11.7 psia	Sheet 7	Sheet 8	1×10^3	90	100
II-R-108H	1/com	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100
II-R-159H	2	115	atm	5×10^5	126/17[b]	11.7 psia	Sheet 7	Sheet 8	5×10^6	90	100

MSIV AREA
(Sheet 1 of 2)

TABLE 3.11.B.1-1 (SHEET 6 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

  \- = No high-energy line or safety-related equipment located in the room.

b. During plant operation, the minimum room temperature is 47°F.

c. Sheet numbers refer to figure 3.11.B.1-1.

MSIV AREA
(Sheet 2 of 2)

TABLE 3.11.B.1-1 (SHEET 7 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
EQUIPMENT BUILDING											
V-R-114	1/com	120	atm	1×10^3	135/40	11.7 psia	-	-	1×10^3	90	-
V-R-116	1/com	120	atm	1×10^3	135/40	11.7 psia	-	-	1×10^3	90	-
V-R-117H[b]	1/com	120	atm	5×10^5	125/40	11.7 psia	-	-	1×10^7	90	-
V-R-118	1/com	120	atm	1×10^3	120/40	11.7 psia	-	-	5×10^3	90	-
V-R-120	1/com	120	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
V-R-124	1/com	120	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
V-R-125H[b]	1/com	120	atm	1×10^3	122/40	11.7 psia	-	-	1×10^7	90	-
V-R-114	2	120	atm	1×10^3	135/40	11.7 psia	-	-	1×10^3	90	-
V-R-116	2	120	atm	1×10^3	135/40	11.7 psia	-	-	1×10^3	90	-
V-R-117H[b]	2	120	atm	5×10^5	125/40	11.7 psia	-	-	1×10^7	90	-
V-R-118	2	120	atm	1×10^3	120/40	11.7 psia	-	-	5×10^3	90	-
V-R-120	2	120	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
V-R-124	2	120	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
V-R-125H[b]	2	120	atm	1×10^3	122/40	11.7 psia	-	-	1×10^7	90	-

EQUIPMENT BUILDING
(Sheet 1 of 2)

TABLE 3.11.B.1-1 (SHEET 8 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

- = No high-energy line or safety-related equipment located in the room.

b. Room is included in the area temperature monitoring program (FSAR 16.3).

EQUIPMENT BUILDING
(Sheet 2 of 2)

TABLE 3.11.B.1-1 (SHEET 9 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL D											
VIII-R-D01H[b]	2	100	atm[c]	5 x 10^5	120/40	atm	120	-	5 x 10^5	60	-
VIII-R-D02H[b]	2	100	atm[c]	5 x 10^5	120/40[d]	atm	130	-	5 x 10^6	60	-
VIII-R-D03H[b]	2	120	atm[c]	5 x 10^5	130/40[e]	atm	-	-	5 x 10^7	60	-
VIII-R-D04H[b]	2	100	atm[c]	5 x 10^5	120/40[d]	atm	120	-	2 x 10^6	60	-
VIII-R-D05H[b]	2	100	atm[c]	5 x 10^5	120/40[d]	atm	120	-	2 x 10^6	60	-
VIII-R-D06	2	100	atm[c]	1 x 10^3	106/40	atm	-	-	5 x 10^3	60	-
VIII-R-D07	2	100	atm[c]	1 x 10^3	120/40	atm	-	-	1 x 10^3	60	-
VIII-R-D08	2	100	atm[c]	1 x 10^3	120/65	atm	-	-	1 x 10^3	60	-
VIII-R-D09H	2	100	atm[c]	5 x 10^5	120/65	atm	-	-	1 x 10^8	60	-
VIII-R-D10H	2	100	atm[c]	1 x 10^3	120/40	atm	Sheet 5	Sheet 6/A	1 x 10^3	60	100
VIII-R-D11	2	100	atm[c]	1 x 10^3	105/40	atm	Sheet 5	Sheet 6/C	1 x 10^3	60	100
VIII-R-D12H	2	100	atm[c]	5 x 10^5	120/40	atm	Sheet 5	Sheet 6/C	1 x 10^8	60	100
VIII-R-D13H	2	100	atm[c]	5 x 10^5	120/40	atm	-	-	1 x 10^8	60	-
VIII-R-D14H	2	100	atm[c]	5 x 10^5	120/40	atm	Sheet 5	Sheet 6/C	1 x 10^8	60	100
VIII-R-D15H	1/com	100	atm[c]	1 x 10^3	120/40	atm	Sheet 3	Sheet 6/C	1 x 10^3	60	100
VIII-R-D16H	2	100	atm[c]	5 x 10^5	120/40	atm	Sheet 5	Sheet 6/C	5 x 10^5	60	100
VIII-R-D17H	2	100	atm[c]	5 x 10^5	120/40	atm	Sheet 5	Sheet 6/C	1 x 10^8	60	100
VIII-R-D18H	2	100	atm[c]	5 x 10^5	120/40	atm	Sheet 5	Sheet 6/C	1 x 10^8	60	100
VIII-R-D19	2	100	atm[c]	1 x 10^3	120/40	atm	Sheet 3	Sheet 6/C	1 x 10^3	60	100

TABLE 3.11.B.1-1 (SHEET 10 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL D (Continued)											
VIII-R-D20H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	5×10^5	60	100
VIII-R-D21H[b]	2	100	atm[c]	1×10^7	120/40[d]	atm	-	-	5×10^7	60	-
VIII-R-D22H[b]	2	100	atm[c]	1×10^7	120/40[d]	atm	-	-	5×10^7	60	-
VIII-R-D23	void										
VIII-R-D24H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D25H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D26H	1/com	100	atm[c]	5×10^5	112/40	atm	-	-	1×10^8	60	-
VIII-R-D27H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-D28H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D29H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D30H	2	100	atm[c]	5×10^4	120/40	atm	Sheet 3	Sheet 6/C	5×10^5	60	100
VIII-R-D31	void										
VIII-R-D32	2	100	atm[c]	1×10^3	110/40	atm	Sheet 5	Sheet 6/C	1×10^3	60	100
VIII-R-D33H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D34H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100
VIII-R-D35H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100
VIII-R-D36H	1/com	100	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-D37H	2	100	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-D38H	1/com	100	atm[c]	5×10^4	120/40	atm	Sheet 5	Sheet 6/C	5×10^4	60	100
VIII-R-D39H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100

TABLE 3.11.B.1-1 (SHEET 11 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL D (Continued)											
VIII-R-D40H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	Sheet 5	Sheet 6/C	1 x 10^{8}	60	100
VIII-R-D41	2	100	atm[c]	1 x 10^{3}	110/40	atm	113	-	5 x 10^{3}	60	100
VIII-R-D42H	1/com	100	atm[c]	5 x 10^{4}	120/40	atm	Sheet 5	Sheet 6/C	5 x 10^{4}	60	100
VIII-R-D43H	1/com	100	atm[c]	1 x 10^{3}	120/40	atm	Sheet 1	Sheet 6/C	1 x 10^{3}	60	100
VIII-R-D44H	1/com	100[f]	atm[c]	1 x 10^{3}	120/40	atm	Sheet 3	Sheet 6/C	1 x 10^{3}	60	100
VIII-R-D45H	1/com	100	atm[c]	5 x 10^{5}	102/40	atm	Sheet 2	Sheet 6/C	1 x 10^{8}	60	100
VIII-R-D46H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	-	-	1 x 10^{8}	60	-
VIII-R-D47H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	Sheet 2	Sheet 6/C	1 x 10^{8}	60	100
VIII-R-D48H[b]	1/com	100	atm[c]	1 x 10^{7}	120/40[d]	atm	-	-	5 x 10^{7}	60	-
VIII-R-D49H[b]	1/com	100	atm[c]	1 x 10^{7}	120/40[d]	atm	-	-	5 x 10^{7}	60	-
VIII-R-D50H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	-	-	1 x 10^{8}	60	-
VIII-R-D51H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	-	-	1 x 10^{8}	60	-
VIII-R-D52H	1/com	100	atm[c]	1 x 10^{3}	116/40	atm	Sheet 2	Sheet 6/C	1 x 10^{3}	60	100
VIII-R-D53H[h]	1/com	100	atm[c]	1 x 10^{3}	110/40	atm	Sheet 5	Sheet 6/E	5 x 10^{3}	60	100
VIII-R-D54H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	Sheet 5	Sheet 6/C	1 x 10^{8}	60	100
VIII-R-D55H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	Sheet 5	Sheet 6/C	5 x 10^{5}	60	100
VIII-R-D56H	1/com	100	atm[c]	1 x 10^{3}	110/40	atm	Sheet 5	Sheet 6/C	1 x 10^{3}	60	100
VIII-R-D57H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	-	Sheet 6/C	1 x 10^{8}	60	-
VIII-R-D58H	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	Sheet 5	-	1 x 10^{8}	60	100

TABLE 3.11.B.1-1 (SHEET 12 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL D (Continued)											
VIII-R-D59H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100
VIII-R-D60H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	5×10^5	60	100
VIII-R-D61H	1/com	100[f]	atm[c]	1×10^3	120/40	atm	Sheet 3	Sheet 6/C	1×10^3	60	100
VIII-R-D62H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100
VIII-R-D63H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D64H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100
VIII-R-D65H	1/com	100[f]	atm[c]	5×10^4	120/40	atm	Sheet 3	Sheet 6/D	5×10^5	60	100
VIII-R-D66H	1/com	100[f]	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	1×10^3	60	100
VIII-R-D67H[h]	1/com	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/C	1×10^3	60	100
VIII-R-D68	1/com	100	atm[c]	1×10^3	106/40	atm	-	-	1×10^3	60	-
VIII-R-D69H	1/com	100	atm[c]	5×10^5	120/65	atm	-	-	1×10^8	60	-
VIII-R-D70	1/com	100	atm[c]	1×10^3	120/65	atm	-	-	1×10^3	60	-
VIII-R-D71	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-D72H[h]	1/com	100	atm[c]	1×10^5	125/65	atm	-	-	1×10^5	60	-
VIII-R-D73H[h]	1/com	100	atm[c]	5×10^5	121/40	atm	-	-	5×10^5	60	-
VIII-R-D74H[b]	1/com	100	atm[c]	5×10^5	120/40	atm	120	-	5×10^5	60	-
VIII-R-D75H	1/com	100	atm[c]	1×10^3	106/40	atm	-	-	5×10^3	60	-
VIII-R-D76H[b]	1/com	100	atm[c]	5×10^5	120/40[d]	atm	120	-	2×10^6	60	-
VIII-R-D77H[b]	1/com	100	atm[c]	5×10^5	120/40[d]	atm	120	-	2×10^6	60	-

AUXILIARY BUILDING
(Sheet 4 of 41)

TABLE 3.11.B.1-1 (SHEET 13 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)	DBA/Post-DBA Pressure	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL D (Continued)											
VIII-R-D78H[b]	1/com	120	atm[c]	5×10^5	130/40[e]	atm	-	-	5×10^7	60	-
VIII-R-D79H[b]	1/com	100	atm[c]	5×10^5	120/40[d]	atm	130	-	5×10^6	60	-
VIII-R-D80H	1/com	100	atm[c]	5×10^5	112/40	atm	-	-	5×10^5	60	-
VIII-R-D81H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D82H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D83H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D84H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D85H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D86H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D87H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D88H	1/com	100[i]	atm[c]	5×10^5	110/40	atm	-	-	1×10^8	60	-
VIII-R-D89H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D90H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D91H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D92H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D93H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D94H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D95H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-D96H	1/com	100[i]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-

TABLE 3.11.B.1-1 (SHEET 14 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL D (Continued)											
VIII-R-D97H	1/com	100	atm[c]	1 x 10^3	120/40	atm	Sheet 2	Sheet 6/E	1 x 10^3	60	100
VIII-R-D98H[b]	2	100	atm[c]	5 x 10^5	120/40[e]	atm	120	-	5 x 10^7	60	-
VIII-R-D99H[b]	2	100	atm[c]	5 x 10^5	112/40[e]	atm	Sheet 12/A Sheet 13	Sheet 6/C	5 x 10^7	60	100
VIII-R-D100H[b]	1/com	100	atm[c]	5 x 10^5	120/40[e]	atm	120	-	5 x 10^7	60	-
VIII-R-D101H[b]	1/com	100	atm[c]	5 x 10^5	120/40[e]	atm	Sheet 12/A Sheet 13	Sheet 6/C	5 x 10^7	60	100
VIII-R-D102H	1/com	100	atm[c]	5 x 10^5	110/40	atm	Sheet 5	Sheet 6/C	1 x 10^8	60	100
VIII-R-D103	void										
VIII-R-D104[b]	2	100	atm[c]	1 x 10^3	120/40[d]	atm	-	-	5 x 10^3	60	-
VIII-R-D105[b]	1/com	100	atm[c]	1 x 10^3	120/40[d]	atm	-	-	5 x 10^3	60	-
VIII-R-D106H[h]	2	100	atm[c]	1 x 10^5	125/65	atm	-	-	1 x 10^5	60	-
VIII-R-D107H[h]	2	100	atm[c]	5 x 10^5	105/40	atm	-	-	5 x 10^5	60	-
VIII-R-D108	2	100	atm[c]	1 x 10^3	106/40	atm	-	-	1 x 10^3	60	-
VIII-R-D109	2	100	atm[c]	1 x 10^3	120/40	atm	-	-	1 x 10^3	60	-
VIII-R-D110	1/com	100	atm[c]	1 x 10^3	120/40	atm	-	-	1 x 10^3	60	-
VIII-R-D111H	2	100	atm[c]	5 x 10^5	110/40	atm	Sheet 5	Sheet 6/C	1 x 10^8	60	100
VIII-R-D112H	2	100	atm[c]	5 x 10^5	106/40	atm	-	-	5 x 10^6	60	-
VIII-R-D113H[h]	2	100	atm[c]	5 x 10^5	103/40	atm	-	-	1 x 10^6	60	-
VIII-R-D114H	1/com	100	atm[c]	5 x 10^5	120/40	atm	Sheet 5	Sheet 6/C	5 x 10^5	60	100
VIII-R-D115H	2	100	atm[c]	5 x 10^5	120/40	atm	Sheet 5	Sheet 6/C	1 x 10^8	60	100

TABLE 3.11.B.1-1 (SHEET 15 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL D (Continued)											
VIII-R-D116H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100
VIII-R-D117H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/C	1×10^8	60	100
VIII-R-D118	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-D119H[h]	1/com	100	atm[c]	1×10^5	121/65	atm	Sheet 5	Sheet 6/C	1×10^5	60	100
VIII-R-D120H	2	100	atm[c]	5×10^5	120/40	atm	-	-	5×10^5	60	-
VIII-R-D121H[h]	1/com	100	atm[c]	5×10^5	103/40	atm	-	-	1×10^6	60	-
VIII-R-D122H	1/com	100[f]	atm[c]	5×10^5	106/40	atm	-	-	5×10^6	60	-
VIII-R-D123H[h]	2	100	atm[c]	1×10^5	121/65	atm	-	-	1×10^5	60	-
VIII-R-D124	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-D125H	1/com	100[f]	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/C	1×10^3	60	100
VIII-R-D126H	1/com	100[f]	atm[c]	5×10^4	120/40	atm	Sheet 3	Sheet 6/D	5×10^6	60	100
VIII-R-D127H	2	100	atm[c]	5×10^4	120/40	atm	Sheet 3	Sheet 6/C	5×10^6	60	100
VIII-R-D128H[b]	1/com	100	atm[c]	5×10^5	111/40	atm	106	-	5×10^6	60	-
VIII-R-D129H[b]	2	100	atm[c]	5×10^5	111/40	atm	106	-	5×10^6	60	-
VIII-R-D130H[b]	1/com	100	atm[c]	5×10^5	110/40	atm	-	-	1×10^6	60	-
VIII-R-D131H[b]	2	100	atm[c]	5×10^5	110/40	atm	-	-	1×10^6	60	-
VIII-R-D132H	1/com	100[f]	atm[c]	5×10^5	104/40	atm	-	-	1×10^6	60	-
VIII-R-D133H	2	100	atm[c]	5×10^5	104/40	atm	-	-	1×10^6	60	-
VIII-UCD001H	2	120	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100

TABLE 3.11.B.1-1 (SHEET 16 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL D (Continued)											
VIII-UCD002H	2	120	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCD003H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCD004H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCD005H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCD006H	1	120	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCD006H	2	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCD007H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCD008H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCD009H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCD010H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 12/B	Sheet 6/C	1×10^8	60	100
VIII-UCD011H	2	120	atm[c]	5×10^5	120/40	atm	Sheet 12/B	Sheet 6/C	1×10^8	60	100
AUXILIARY BUILDING - LEVEL C											
VIII-R-C01H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	Sheet 2	Sheet 6/A	5×10^5	60	100
VIII-R-C02H[b]	2	115	atm[c]	1×10^7	120/40[d]	atm	Sheet 2	Sheet 6/A	1×10^7	60	100
VIII-R-C03H[b]	2	114	atm[c]	1×10^3	120/40[d]	atm	Sheet 2	Sheet 6/A	1×10^4	60	100
VIII-R-C04H[b]	2	100	atm[c]	5×10^5	105/40	atm	102, Sheet 2	Sheet 6/A	5×10^5	60	100
VIII-R-C05H[b]	2	106	atm[c]	5×10^5	120/40[e]	atm	Sheet 2 Sheet 6	Sheet 6/A	5×10^7	60	100
VIII-R-C06H	2	128	atm[c]	1×10^3	120/40	atm	154	-	5×10^4	60	-
VIII-R-C07H[b]	2	100	atm[c]	1×10^3	120/40[d]	atm	-	-	5×10^4	60	-

TABLE 3.11.B.1-1 (SHEET 17 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C08H	2	100	atm[c]	1 x 10^3	120/40	atm	Sheet 2	Sheet 6/A	1 x 10^3	60	100
VIII-R-C09H[b]	2	115	atm[c]	5 x 10^5	120/40[d]	atm	-	-	1 x 10^7	60	-
VIII-R-C10H[h]	2	100	atm[c]	1 x 10^3	109/40	atm	-	-	1 x 10^7	60	-
VIII-R-C11H[b]	2	104	atm[c]	5 x 10^5	120/65[d]	atm	120	-	1 x 10^6	60	-
VIII-R-C12H[b]	2	116	atm[c]	5 x 10^5	120/40[d]	atm	117	-	5 x 10^6	60	-
VIII-R-C13	2	100	atm[c]	1 x 10^3	120/40	atm	-	-	5 x 10^3	60	-
VIII-R-C14	2	100	atm[c]	1 x 10^3	120/40	atm	-	-	5 x 10^3	60	-
VIII-R-C15	2	100	atm[c]	1 x 10^3	120/40	atm	-	-	5 x 10^3	60	-
VIII-R-C16H[b]	2	100	atm[c]	5 x 10^5	120/65[d]	atm	-	-	1 x 10^7	60	-
VIII-R-C17H[b]	2	100	atm[c]	5 x 10^5	120/65[d]	atm	-	-	1 x 10^7	60	-
VIII-R-C18H[b]	2	115	atm[c]	5 x 10^5	120/65[d]	atm	Sheet 17	-	1 x 10^6	60	-
VIII-R-C19	2	100	atm[c]	1 x 10^3	106/40	atm	-	-	1 x 10^4	60	-
VIII-R-C20H	2	100	atm[c]	1 x 10^3	120/40	atm	Sheet 5	Sheet 6/A	1 x 10^3	60	100
VIII-R-C21H	2	100	atm[c]	1 x 10^3	120/40	atm	Sheet 4	Sheet 6/D	1 x 10^3	60	100
VIII-R-C22H	2	100	atm[c]	5 x 10^4	120/40	atm	Sheet 3	Sheet 6/D	5 x 10^4	60	100
VIII-R-C23H[b]	2	102	atm[c]	5 x 10^5	120/40[e]	atm	Sheet 12/A Sheet 13	Sheet 6/C	1 x 10^7	60	100
VIII-R-C24H	2	100	atm[c]	1 x 10^3	122/40	atm	Sheet 4	Sheet 6/D	1 x 10^4	60	100
VIII-R-C25H[b]	2	100	atm[c]	1 x 10^7	120/40[d]	atm	112	-	5 x 10^7	60	-
VIII-R-C26H[b]	2	100	atm[c]	1 x 10^7	120/40[d]	atm	112	-	5 x 10^7	60	-

TABLE 3.11.B.1-1 (SHEET 18 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C27H	2	100	atm[c]	5×10^4	104/40	atm	-	-	5×10^7	60	-
VIII-R-C28H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C29H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C30H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	5×10^6	60	100
VIII-R-C31H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 4	Sheet 6/D	1×10^3	60	100
VIII-R-C32H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-C33H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-C34H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-C35H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-C36H	2	100	atm[c]	5×10^5	120/40	atm	-	-	5×10^6	60	-
VIII-R-C37H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-R-C38H[b]	2	100	atm[c]	5×10^4	104/40	atm	-	-	5×10^6	60	-
VIII-R-C39	2	104	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-C40H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 1	Sheet 6/A	1×10^8	60	100
VIII-R-C41H	2	100	atm[c]	5×10^5	118/40	atm	Sheet 1	Sheet 6/A	1×10^8	60	100
VIII-R-C42H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 2	Sheet 6/A	5×10^5	60	100
VIII-R-C43H	2	100	atm[c]	5×10^5	121/40	atm	Sheet 2	Sheet 6/D	1×10^8	60	100
VIII-R-C44H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 4	Sheet 6/D	5×10^3	60	100
VIII-R-C45H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/A	1×10^8	60	100
VIII-R-C46H	2	100	atm[c]	5×10^5	110/40	atm	Sheet 3	Sheet 6/A	1×10^8	60	100

TABLE 3.11.B.1-1 (SHEET 19 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator(a)	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)(i)	DBA/Post-DBA Pressure(i)	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C47H	2	100	atm(c)	5×10^5	120/40	atm	Sheet 3	Sheet 6/A	1×10^8	60	100
VIII-R-C48H	1/com	100	atm(c)	5×10^5	120/40	atm	Sheet 3	Sheet 6/A	1×10^8	60	100
VIII-R-C49H	2	100	atm(c)	1×10^3	113/40	atm	Sheet 3	Sheet 6/A	1×10^3	60	100
VIII-R-C50H	2	100	atm(c)	5×10^5	107/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C51H	2	100	atm(c)	5×10^5	120/40	atm	Sheet 5	Sheet 6/D	5×10^5	60	100
VIII-R-C52H	2	100	atm(c)	5×10^5	100/40	atm	-	-	1×10^8	60	-
VIII-R-C53H	1/com	100(f)	atm(c)	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-R-C54H	2	100	atm(c)	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C55H	2	100	atm(c)	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C56H	2	100	atm(c)	5×10^5	105/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C57H	2	100	atm(c)	5×10^4	120/40	atm	Sheet 3	Sheet 6/D	5×10^4	60	100
VIII-R-C58H	1/com	100	atm(c)	1×10^3	120/40	atm	Sheet 4	Sheet 6/D	1×10^3	60	100
VIII-R-C59H	1/com	100(f)	atm(c)	1×10^3	120/40	atm	Sheet 3	Sheet 6/C	1×10^3	60	100
VIII-R-C60H	1/com	100(f)	atm(c)	5×10^5	120/40	atm	Sheet 4	Sheet 6/D	5×10^6	60	100
VIII-R-C61H	1/com	100	atm(c)	5×10^5	120/65	atm	Sheet 1	Sheet 6/D	1×10^8	60	100
VIII-R-C62H	1/com	100	atm(c)	5×10^4	120/40	atm	Sheet 1	Sheet 6/D	1×10^5	60	100
VIII-R-C63H	1/com	100	atm(c)	5×10^5	120/40	atm	Sheet 1	Sheet 6/D	1×10^8	60	100
VIII-R-C64H	1/com	100	atm(c)	5×10^5	129/65	atm	Sheet 2	Sheet 6/D	1×10^8	60	100

TABLE 3.11.B.1-1 (SHEET 20 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C65H	1/com	100	atm[c]	5×10^5	118/65	atm	Sheet 2	Sheet 6/D	1×10^8	60	100
VIII-R-C66H	1/com	100	atm[c]	5×10^5	124/40	atm	Sheet 2	Sheet 6/D	1×10^8	60	100
VIII-R-C67H	1/com	100	atm[c]	5×10^5	121/40	atm	Sheet 2	Sheet 6/D	1×10^8	60	100
VIII-R-C68H	1/com	100	atm[c]	5×10^5	110/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C69H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C70H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C71H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C72H	1/com	100	atm[c]	5×10^5	105/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C73H	1/com	100	atm[c]	5×10^4	120/40	atm	Sheet 3	Sheet 6/D	5×10^4	60	100
VIII-R-C74H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C75H	1/com	100	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C76H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-R-C77H[b]	1/com	100	atm[c]	5×10^5	100/40	atm	-	-	1×10^8	60	-
VIII-R-C78H	1/com	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/D	5×10^5	60	100
VIII-R-C79H	1/com	100	atm[c]	5×10^5	107/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C80H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-C81H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-C82H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-C83H[h]	1/com	100	atm[c]	1×10^3	113/40	atm	Sheet 4	Sheet 6/D	1×10^3	60	100

AUXILIARY BUILDING
(Sheet 12 of 41)

TABLE 3.11.B.1-1 (SHEET 21 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator(a)	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)(i)	Pressure(i)	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C84H(b)	1/com	100	atm(c)	1×10^7	104/40	atm	Sheet 16	-	5×10^7	60	-
VIII-R-C85H	1/com	100	atm(c)	1×10^3	120/40	atm	Sheet 4	Sheet 6/D	5×10^3	60	100
VIII-R-C86H	1/com	100(f)	atm(c)	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-R-C87H	1/com	100	atm(c)	5×10^5	120/40	atm	Sheet 4	Sheet 6/D	5×10^5	60	100
VIII-R-C88	1/com	104	atm(c)	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-C89H(b)	1/com	100	atm(c)	5×10^4	104/40	atm	-	-	5×10^6	60	-
VIII-R-C90H(b)	1/com	100	atm(c)	1×10^7	120/40(d)	atm	112	-	5×10^7	60	-
VIII-R-C91H(b)	1/com	100	atm(c)	1×10^7	120/40(d)	atm	112	-	5×10^7	60	-
VIII-R-C92H(b)	1/com	100	atm(c)	5×10^4	104/40	atm	-	-	5×10^7	60	-
VIII-R-C93H	1/com	100	atm(c)	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C94H	1/com	100	atm(c)	5×10^5	120/40	atm	Sheet 3	Sheet 6/D	1×10^8	60	100
VIII-R-C95H(b)	1/com	100	atm(c)	5×10^5	120/40(e)	atm	Sheet 12/A Sheet 13	Sheet 6/C	1×10^7	60	100
VIII-R-C96H(b)	1/com	100	atm(c)	1×10^7	100/40	atm	Sheet 16	-	5×10^7	60	-
VIII-R-C97H	1/com	100	atm(c)	1×10^3	122/40	atm	Sheet 4	Sheet 6/D	1×10^4	60	100
VIII-R-C98H	1/com	100	atm(c)	1×10^3	120/40	atm	Sheet 4	Sheet 6/D	1×10^3	60	100
VIII-R-C99H	1/com	100(e)	atm(c)	5×10^4	120/40	atm	Sheet 3	Sheet 6/D	5×10^4	60	100
VIII-R-C100H	1/com	100	atm(c)	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	1×10^3	60	100
VIII-R-C101H	1/com	100	atm(c)	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	1×10^3	60	100
VIII-R-C102H	1/com	100(e)	atm(c)	1×10^3	120/40	atm	Sheet 4	Sheet 6/D	1×10^3	60	100

TABLE 3.11.B.1-1 (SHEET 22 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[a]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C103H[b]	1/com	106	atm[c]	5 x 10^{5}	120/40[e]	atm	Sheet 2 Sheet 16	Sheet 6/A	5 x 10^{7}	60	100
VIII-R-C104H[b]	1/com	128	atm[c]	1 x 10^{3}	128/40	atm	154	-	5 x 10^{4}	60	-
VIII-R-C105H[b]	1/com	100	atm[c]	5 x 10^{5}	120/40	atm	102/Sheet 2	Sheet 6/A	5 x 10^{5}	60	100
VIII-R-C106H[b]	1/com	100	atm[c]	1 x 10^{3}	105/40	atm	Sheet 2	Sheet 6/A	1 x 10^{4}	60	100
VIII-R-C107H[b]	1/com	100	atm[c]	5 x 10^{5}	120/40[d]	atm	Sheet 2	Sheet 6/A	5 x 10^{5}	60	100
VIII-R-C108H[b]	1/com	115	atm[c]	1 x 10^{7}	120/40[d]	atm	Sheet 2	Sheet 6/A	1 x 10^{7}	60	100
VIII-R-C109H[b]	1/com	100	atm[c]	1 x 10^{3}	120/40[d]	atm	-	-	5 x 10^{4}	60	-
VIII-R-C110H[b]	1/com	100[f]	atm[c]	1 x 10^{3}	120/40[d]	atm	Sheet 2	Sheet 6/A	1 x 10^{3}	60	100
VIII-R-C111H[b]	1/com	116	atm[c]	5 x 10^{5}	120/40	atm	117	-	5 x 10^{6}	60	-
VIII-R-C112H[b]	1/com	115	atm[c]	5 x 10^{5}	120/40[d]	atm	-	-	1 x 10^{7}	60	-
VIII-R-C113H[h]	1/com	100	atm[c]	1 x 10^{3}	109/40	atm	-	-	1 x 10^{7}	60	-
VIII-R-C114H[b]	1/com	104	atm[c]	5 x 10^{5}	120/65	atm	120	-	1 x 10^{6}	60	-
VIII-R-C115H[b]	1/com	100	atm[c]	5 x 10^{5}	120/65[d]	atm	-	-	1 x 10^{7}	60	-
VIII-R-C116	1/com	100	atm[c]	1 x 10^{3}	120/40[d]	atm	-	-	5 x 10^{3}	60	-
VIII-R-C117	1/com	100[i]	atm[c]	1 x 10^{3}	120/40	atm	-	-	5 x 10^{3}	60	-
VIII-R-C118H[b]	1/com	100	atm[c]	5 x 10^{5}	120/65[d]	atm	-	-	1 x 10^{7}	60	-
VIII-R-C119H[b]	1/com	115	atm[c]	5 x 10^{5}	120/65[d]	atm	Sheet 17	-	1 x 10^{6}	60	-
VIII-R-C120	1/com	100	atm[c]	1 x 10^{3}	106/40	atm	-	-	1 x 10^{4}	60	-
VIII-R-C121	1/com	100	atm[c]	1 x 10^{3}	120/40	atm	-	-	5 x 10^{3}	60	-

TABLE 3.11.B.1-1 (SHEET 23 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C122H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	5×10^6	60	-
VIII-R-C123H	2	100	atm[c]	1×10^7	104/40	atm	Sheet 16	-	5×10^7	60	-
VIII-R-C124H[b]	2	100	atm[c]	5×10^5	120/40[e]	atm	120/Sheet 2	Sheet 6/A	1×10^7	60	100
VIII-R-C125H	1/com	115	atm[c]	1×10^7	120/40	atm	Sheet 2	Sheet 6/A	1×10^7	60	100
VIII-R-C126H[b]	2	100	atm[c]	5×10^5	120/40[e]	atm	120/Sheet 2	Sheet 6/A	5×10^7	60	100
VIII-R-C127H[b]	2	100	atm[c]	1×10^7	104/40	atm	Sheet 16	-	5×10^7	60	-
VIII-R-C128H[b]	2	100	atm[c]	5×10^5	100/40	atm	120/Sheet 2	Sheet 6/A	5×10^7	60	100
VIII-R-C129H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 2	Sheet 6/A	5×10^7	60	100
VIII-R-C130H	1/com	100[f]	atm[c]	1×10^3	120/40	atm	Sheet 2	Sheet 6/A	5×10^7	60	100
VIII-R-C131H[b]	1/com	100	atm[c]	5×10^5	100/40	atm	120/Sheet 2	Sheet 6/A	5×10^7	60	100
VIII-R-C132	void										
VIII-R-C133H[b]	1/com	100	atm[c]	5×10^5	120/40[e]	atm	120/Sheet 2	Sheet 6/A	5×10^7	60	100
VIII-R-C134H[b]	1/com	100	atm[c]	5×10^5	120/40[e]	atm	120/Sheet 2	Sheet 6/A	1×10^7	60	100
VIII-R-C135H	2	115	atm[c]	1×10^7	120/40	atm	Sheet 2	Sheet 6/A	1×10^7	60	100
VIII-R-C136	void										
VIII-R-C137	void										
VIII-R-C138H	2	100	atm[c]	1×10^9	120/40	atm		-	1×10^9	60	-
VIII-R-C139H	2	100	atm[c]	1×10^9	120/40	atm		-	1×10^9	60	-
VIII-R-C140H	2	100	atm[c]	1×10^9	120/40	atm		-	1×10^9	60	-

TABLE 3.11.B.1-1 (SHEET 24 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-R-C141H	1/com	100[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C142H	1/com	100[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C143H	1/com	100[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C144H	1/com	100[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C145H	1/com	100[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C146H	1/com	100[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C147H	2	100[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-C148	spare										
VIII-R-C149H	2	100	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-UCC01H	1	130	atm[c]	5×10^5	130/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCC01H	2	130	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCC02H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCC03H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCC04H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCC05H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCC06H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCC07H	1/com	130[f] 130	atm[c]	5×10^5	130/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCC08H	2	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCC09H	1/com	130	atm[c]	5×10^5	130/40	atm	Sheet 4	Sheet 6/A	1×10^8	60	100
VIII-UCC10H	2	130[i]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCC11H	1/com		atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/D	1×10^8	60	100

TABLE 3.11.B.1-1 (SHEET 25 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[a]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL C (Continued)											
VIII-UCC12H	1/com	140[f]	atm[c]	5 x 10^5	140/40	atm	Sheet 5	Sheet 6/D	1 x 10^8	60	100
VIII-UCC13H	1/com	140[f]	atm[c]	1 x 10^9	140/40	atm	-	-	1 x 10^9	60	-
VIII-UCC14H	1/com	130[f]	atm[c]	5 x 10^5	130/40	atm	Sheet 12/B	Sheet 6/A	1 x 10^8	60	100
VIII-UCC15H	2	130	atm[c]	5 x 10^5	130/40	atm	Sheet 12/B	Sheet 6/A	1 x 10^8	60	100
VIII-UCC16H	2	140 140	atm[c]	5 x 10^5	140/40	atm	Sheet 5	Sheet 6/A	1 x 10^8	60	100
VIII-UCC17H	2		atm[c]	1 x 10^9	140/40	atm	-	-	1 x 10^9	60	-
AUXILIARY BUILDING - LEVEL B											
VIII-R-B01	1/com	100[i]	atm[c]	1 x 10^3	120/40	atm	-	-	1 x 10^3	60	-
VIII-R-B02H	1/com	100	atm[c]	1 x 10^4	120/40	atm	-	-	5 x 10^5	60	-
VIII-R-B03H[b]	1/com	100	atm[c]	1 x 10^7	104/40	atm	Sheet 16	-	5 x 10^7	60	-
VIII-R-B04H[b]	1/com	100	atm[c]	5 x 10^4	120/40[d]	atm	110	-	1 x 10^5	60	-
VIII-R-B05H[b]	1/com	100	atm[c]	5 x 10^5	120/40[d]	atm	-	-	1 x 10^7	60	-
VIII-R-B06H[b]	1/com	100	atm[c]	5 x 10^5	120/40[d]	atm	-	-	1 x 10^7	60	-
VIII-R-B07H[b]	1/com	100	atm[c]	5 x 10^5	120/40[d]	atm	Sheet 2 Sheet 15	Sheet 6/C	5 x 10^5	60	100
VIII-R-B08H[b]	1/com	100	atm[c]	5 x 10^5	120/40[d] 104/40	atm	Sheet 2 Sheet 15	Sheet 6/C	5 x 10^5	60	100
VIII-R-B09H[b]	1/com	100	atm[c]	5 x 10^4	128/40	atm	115	-	1 x 10^7	60	-
VIII-R-B10H[b]	1/com	128	atm[c]	5 x 10^4	120/40[d]	atm	154	-	1 x 10^8	60	-
VIII-R-B11H[b]	1/com	100	atm[c]	5 x 10^5		atm	115	-	1 x 10^7	60	-

TABLE 3.11.B.1-1 (SHEET 26 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[(i)]	DBA/Post-DBA Pressure[(i)]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B12H[(b)]	1/com	100	atm[(c)]	5×10^4	104/40	atm	115	-	1×10^7	60	-
VIII-R-B13H[(b)]	1/com	100	atm[(c)]	5×10^4	120/40[(d)]	atm	115	-	5×10^6	60	-
VIII-R-B14H[(b)]	1/com	100	atm[(c)]	5×10^5	104/40	atm	-	-	1×10^8	60	-
VIII-R-B15H[(b)]	1/com	100	atm[(c)]	5×10^5	120/40[(d)]	atm	-	-	4×10^6	60	-
VIII-R-B16H[(b)]	1/com	100	atm[(c)]	1×10^3	120/40[(d)]	atm	-	-	1×10^5	60	-
VIII-R-B17H[(b)]	1/com	100	atm[(c)]	1×10^3	114/40	atm	-	-	5×10^6	60	-
VIII-R-B18H[(b)]	1/com	100	atm[(c)]	5×10^5	100/40	atm	-	-	1×10^8	60	-
VIII-R-B19H[(b)]	1/com	100	atm[(c)]	5×10^5	120/40[(d)]	atm	-	-	4×10^6	60	-
VIII-R-B20H[(b)]	1/com	112	atm[(c)]	1×10^7	120/40[(d)]	atm	-	-	5×10^7	60	-
VIII-R-B21H	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	5×10^6	60	-
VIII-R-B22H	1/com	100	atm[(c)]	1×10^3	114/40	atm	-	-	5×10^6	60	-
VIII-R-B23H[(h)]	1/com	100	atm[(c)]	5×10^5	117/40	atm	-	-	1×10^7	60	-
VIII-R-B24H	1/com	100	atm[(c)]	1×10^3	115/40	atm	-	-	5×10^5	60	-
VIII-R-B25H[(b)]	1/com	100	atm[(c)]	5×10^5	120/40[(d)]	atm	-	-	1×10^8	60	-
VIII-R-B26H[(b)]	1/com	100	atm[(c)]	5×10^5	120/40	atm	Sheet 5	Sheet 6/D	5×10^6	60	100
VIII-R-B27H	1/com	100[(f)]	atm[(c)]	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	1×10^3	60	100
VIII-R-B28	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-B29H	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^6	60	-
VIII-R-B30H	1/com	100[(f)]	atm[(c)]	5×10^4	120/40	atm	Sheet 5	Sheet 6/D	5×10^6	60	100

TABLE 3.11.B.1-1 (SHEET 27 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B311H	1/com	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	1×10^3	60	100
VIII-R-B32H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B33H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B34H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B35H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B36H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B37H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B38H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B39H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B40H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B41H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B42H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B43H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B44H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B45H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B46H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B47H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B48H	1/com	100	atm[c]	5×10^4	120/40	atm	-	-	5×10^4	60	-
VIII-R-B49H	1/com	100	atm[c]	5×10^4	120/40	atm	-	-	5×10^4	60	-
VIII-R-B50H[b]	1/com	120	atm[c]	5×10^5	130/40	atm	Sheet 12/A Sheet 13	Sheet 6/C	5×10^7	60	100

TABLE 3.11.B.1-1 (SHEET 28 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B51H[b]	1/com	100	atm[c]	1×10^7	120/40[e]	atm	Sheet 16	-	1×10^8	60	-
VIII-R-B52	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-B53H[b]	1/com	100	atm[c]	1×10^7	120/40[e]	atm	Sheet 16	-	1×10^8	60	-
VIII-R-B54	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-B55	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-B56	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-B57	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-B58H	1/com	100	atm[c]	5×10^5	109/40	atm	113	-	1×10^8	60	100
VIII-R-B59H	1/com	100	atm[c]	5×10^5	110/40	atm	113	-	1×10^8	60	100
VIII-R-B60H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B61H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B62	1/com	104	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-B63H	1/com	100	atm[c]	5×10^4	120/40	atm	-	-	5×10^4	60	-
VIII-R-B64H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B65H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B66H	1/com	100	atm[c]	5×10^4	120/40	atm	-	-	1×10^8	60	-
VIII-R-B67H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B68H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B69H	1/com	100[f]	atm[c]	1×10^3	120/40	atm	Sheet 3	Sheet 6/C	1×10^3	60	100

TABLE 3.11.B.1-1 (SHEET 29 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[(i)]	Pressure[(i)]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B70	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-B71	1/com	100[(f)]	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^4	60	-
VIII-R-B72H	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	5×10^5	60	-
VIII-R-B73H[(b)]	2	100	atm[(c)]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	5×10^6	60	100
VIII-R-B74H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B75H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B76H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B77	2	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^4	60	-
VIII-R-B78H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B79H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B80H	2	100	atm[(c)]	5×10^4	120/40	atm	-	-	5×10^4	60	-
VIII-R-B81H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B82H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B83H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B84H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B85H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B86H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B87H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B88H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B89H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-

TABLE 3.11.B.1-1 (SHEET 30 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B90H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B91H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B92	2	100	atm[c]	1×10^3	103/40	atm	-	-	1×10^3	60	-
VIII-R-B93H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-B94H	2	100	atm[c]	5×10^4	120/40	atm	Sheet 5	Sheet 6/D	5×10^6	60	100
VIII-R-B95H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-B96H	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^6	60	-
VIII-R-B97H	2	100	atm[c]	5×10^4	120/40	atm	-	-	1×10^8	60	-
VIII-R-B98H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B99H	2	100	atm[c]	5×10^5	120/40	atm		-	1×10^8	60	-
VIII-R-B100H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B101H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-B102H	2	100	atm[c]	5×10^4	120/40	atm	-	-	5×10^4	60	-
VIII-R-B103H	2	100	atm[c]	5×10^4	120/40	atm	-	-	5×10^4	60	-
VIII-R-B104H	2	100	atm[c]	5×10^4	120/40	atm	-	-	1×10^5	60	-
VIII-R-B105	2	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-B106H[b]	2	100	atm[c]	1×10^7	120/40[e]	atm	Sheet 16	-	1×10^8	60	-
VIII-R-B107	2	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-B108H[b]	2	100	atm[c]	1×10^7	120/40[e]	atm	Sheet 16	-	1×10^8	60	-
VIII-R-B109	2	104	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 31 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B110H	2	120	atm[c]	5×10^5	130/40[e]	atm	Sheet 12/A	Sheet 6/C	5×10^7	60	100
VIII-R-B111H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 13	-	5×10^6	60	-
VIII-R-B112H	2	100	atm[c]	1×10^3	115/40	atm	-	-	5×10^5	60	-
VIII-R-B113H[h]	2	100	atm[c]	5×10^5	117/40	atm	-	-	1×10^7	60	-
VIII-R-B114H	2	100	atm[c]	1×10^3	114/40	atm	-	-	5×10^6	60	-
VIII-R-B115H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	-	-	1×10^8	60	-
VIII-R-B116H[b]	2	112	atm[c]	1×10^7	120/40[d]	atm	-	-	5×10^7	60	-
VIII-R-B117H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	-	-	4×10^6	60	-
VIII-R-B118H[b]	2	100	atm[c]	5×10^5	104/40	atm	-	-	1×10^8	60	-
VIII-R-B119H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	-	-	4×10^6	60	-
VIII-R-B120H[b]	2	100	atm[c]	5×10^5	104/40	atm	-	-	1×10^8	60	-
VIII-R-B121H[b]	2	100	atm[c]	1×10^3	114/40	atm	-	-	5×10^6	60	-
VIII-R-B122H[b]	2	100	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^5	60	-
VIII-R-B123H[b]	2	100	atm[c]	5×10^4	120/40[d]	atm	115	-	5×10^6	60	-
VIII-R-B124H[b]	2	100	atm[c]	1×10^7	104/40	atm	Sheet 16	-	5×10^7	60	-
VIII-R-B125H[b]	2	100	atm[c]	5×10^4	120/40[d]	atm	110	-	1×10^5	60	-
VIII-R-B126H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	-	-	1×10^7	60	-
VIII-R-B127H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	115	-	1×10^7	60	-
VIII-R-B128H[b]	2	128	atm[c]	5×10^4	128/40	atm	154	-	1×10^8	60	-
VIII-R-B129H[b]	2	100	atm[c]	5×10^4	104/40	atm	115	-	1×10^7	60	-

TABLE 3.11.B.1-1 (SHEET 32 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B130H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	Sheet 2 Sheet 15	Sheet 6/C	5×10^5	60	100
VIII-R-B131H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	Sheet 2 Sheet 15	Sheet 6/C	5×10^5	60	100
VIII-R-B132H[b]	2	100	atm[c]	5×10^4	104/40	atm	115	-	1×10^7	60	-
VIII-R-B133H	2	100	atm[c]	5×10^5	120/40[d]	atm	-	-	1×10^7	60	-
VIII-R-B134	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-B135H	2	100	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VIII-R-B136H	2	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^5	60	-
VIII-R-B137H[b]	1/com	100	atm[c]	5×10^5	120/65	atm	Sheet 3 Sheet 14	Sheet 6/C	1×10^8	60	100
VIII-R-B138H[b]	2	100	atm[c]	5×10^5	120/65	atm	Sheet 3 Sheet 14	Sheet 6/C	1×10^8	60	100
VIII-R-B139H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	-	-	1×10^7	60	-
VIII-R-B140H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B141H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B142H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B143H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B144H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	-	-	1×10^7	60	-
VIII-R-B145H	1/com	130[f]	atm[c]	1×10^9	130/40	atm	-	-	1×10^9	60	-
VIII-R-B146H	1/com	120[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B147H	1/com	120[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B148H	1/com	120[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-

TABLE 3.11.B.1-1 (SHEET 33 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL B											
VIII-R-B149H	1/com	120[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B150H	1/com	130[f]	atm[c]	1×10^9	130/40	atm	-	-	1×10^9	60	-
VIII-R-B151H[b]	1/com	115	atm[c]	1×10^9	115/40	atm	-	-	1×10^9	60	-
VIII-R-B152H[b]	1/com	120	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B153H[b]	1/com	120[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B154H	1/com	120[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B155H[b]	1/com	137	atm[c]	1×10^9	137/40	atm	-	-	1×10^9	60	-
VIII-R-B156H	1/com	140[f]	atm[c]	1×10^9	140/40	atm	-	-	1×10^9	60	-
VIII-R-B157H	1/com	155[f]	atm[c]	1×10^9	155/40	atm	-	-	1×10^9	60	-
VIII-R-B158H[b]	1/com	115	atm[c]	5×10^5	120/40[d]	atm	Sheet 16	-	1×10^7	60	-
VIII-R-B159H[b]	1/com	113	atm[c]	5×10^5	120/40[d]	atm	Sheet 16	-	1×10^7	60	-
VIII-R-B160H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B161H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B162H[b]	1/com	135	atm[c]	5×10^5	135/40	atm	-	-	1×10^7	60	-
VIII-R-B163H	1/com	140[f]	atm[c]	5×10^5	140/40	atm	-	-	1×10^7	60	-
VIII-R-B164H	1/com	155[f]	atm[c]	5×10^5	155/40	atm	-	-	1×10^7	60	-
VIII-R-B165H	2	130	atm[c]	5×10^5	130/40	atm	-	-	1×10^7	60	-
VIII-R-B166	spare										
VIII-R-B167	spare										

TABLE 3.11.B.1-1 (SHEET 34 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL B											
VIII-R-B168H	2	120	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B169H	2	120	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B170H	2	130	atm[c]	5×10^5	130/40	atm	-	-	1×10^7	60	-
VIII-R-B171H	2	130	atm[c]	1×10^9	130/40	atm	-	-	1×10^9	60	-
VIII-R-B172	spare										
VIII-R-B173	spare										
VIII-R-B174H	2	120	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B175H	2	120	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B176H	2	130	atm[c]	1×10^9	130/40	atm	-	-	1×10^9	60	-
VIII-R-B177H	2	140	atm[c]	1×10^9	140/40	atm	-	-	1×10^9	60	-
VIII-R-B178H[b]	2	137	atm[c]	1×10^9	137/40	atm	-	-	1×10^9	60	-
VIII-R-B179H	2	120	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B180H	2	120	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B181H[b]	2	120	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-B182H[b]	2	115	atm[c]	1×10^9	115/40	atm	-	-	1×10^9	60	-
VIII-R-B183H	2	140	atm[c]	5×10^5	140/40	atm	-	-	1×10^7	60	-
VIII-R-B184H[b]	2	135	atm[c]	5×10^5	135/40	atm	-	-	1×10^7	60	-
VIII-R-B185H	2	120	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B186H	2	120	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-B187H[b]	2	113	atm[c]	5×10^5	120/40[d]	atm	Sheet 16	-	1×10^7	60	-

TABLE 3.11.B.1-1 (SHEET 35 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL B (Continued)											
VIII-R-B188H[b]	2	115	atm[c]	5×10^5	120/40[d]	atm	Sheet 16	-	1×10^7	60	-
VIII-UCB01H	1/2	130	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCB02H	2	130	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCB03H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCB04H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	-	-	1×10^8	60	-
VIII-UCB05H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCB06H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCB07H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	-	-	1×10^8	60	-
VIII-UCB08H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCB09H	2	130[f]	atm[c]	5×10^5	130/40	atm	-	-	1×10^8	60	-
VIII-UCB10H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCB11H[b]	1/com	130[f]	atm[c]	5×10^5	130/40	atm	-	-	1×10^8	60	-
VIII-UCB12H	2	130	atm[c]	5×10^5	130/40	atm	-	-	1×10^8	60	-
VIII-UCB13H[b]	1/com	130	atm[c]	5×10^5	130/40	atm	Sheet 2	Sheet 6/C	1×10^8	60	100
VIII-UCB14H[b]	2	130	atm[c]	5×10^5	130/40	atm	Sheet 2	Sheet 6/C	1×10^8	60	100
AUXILIARY BUILDING - LEVEL A											
VIII-R-A01	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A02	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A03[b]	1/com	100	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 36 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL A (Continued)											
VIII-R-A04	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A05[b]	1/com	100	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-
VIII-R-A07H[b]	1/com	116	atm[c]	1×10^7	120/40[d]	atm	Sheet 12/A	24 psia[g]	5×10^7	60	100
VIII-R-A08H[b]	1/com	119	atm[c]	5×10^5	120/40[d]	atm	Sheet 12/A	20 psia	1×10^8	60	100
VIII-R-A09H[b]	1/com	103	atm[c]	5×10^5	120/40[d]	atm	120 Sheet 12/A	Sheet 6C[g]	1×10^8	60	100
VIII-R-A10H[b]	1/com	120	atm[c]	1×10^4	120/40[d]	atm	126 Sheet 12/A	Sheet 6/C	1×10^4	60	100
VIII-R-A13H[b]	1/com	100	atm[c]	5×10^5	120/40[d]	atm	Sheet 3 Sheet 15	Sheet 6/C	5×10^7	60	100
VIII-R-A14H	1/com	100	atm[c]	5×10^4	112/40	atm	-	-	5×10^5	60	-
VIII-R-A15[b]	1/com	128	atm[c]	1×10^3	128/40	atm	154	-	1×10^4	60	-
VIII-R-A16H[b]	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-A17	1/com	100	atm[c]	1×10^3	112/40	atm	-	-	1×10^4	60	-
VIII-R-A18H[b]	2	100	atm[c]	5×10^5	120/40[d]	atm	Sheet 3 Sheet 15	Sheet 6/C	5×10^7	60	100
VIII-R-A19H[b]	1/com	115	atm[c]	5×10^4	115/40	atm	140	-	1×10^7	60	-
VIII-R-A21H	1/com	100[f]	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	1×10^3	60	-
VIII-R-A22	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A23H	1/com	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	1×10^3	60	-
VIII-R-A24H	1/com	100[f]	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/D	5×10^3	60	100
VIII-R-A25H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/D	5×10^6	60	100
VIII-R-A26H[b]	1/com	115	atm[c]	5×10^5	120/40[d]	atm	128	-	1×10^8	60	-

TABLE 3.11.B.1-1 (SHEET 37 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)	DBA/Post-DBA Pressure	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL A (Continued)											
VIII-R-A27H	1/com	100	atm[c]	5×10^5	109/40	atm	-	-	1×10^8	60	-
VIII-R-A28H	1/com	100	atm[c]	5×10^5	109/40	atm	113	-	1×10^8	60	100
VIII-R-A29H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A30	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-A31H	1/com	100	atm[c]	5×10^5	109/40	atm	-	-	5×10^5	60	-
VIII-R-A32H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A33H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A34H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A35H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A36H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A37H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A38H	1/com	100	atm[c]	5×10^5	112/40	atm	-	-	5×10^5	60	-
VIII-R-A39H	2	100	atm[c]	5×10^4	120/40	atm	-	-	5×10^5	60	-
VIII-R-A40	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-A40A	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-A40B	2	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-A41H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A42H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A43H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A44H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A45H	1/com	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-

TABLE 3.11.B.1-1 (SHEET 38 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[(a)]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[(i)]	Pressure[(i)]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL A (Continued)											
VIII-R-A46H	1/com	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A47H[(h)]	1/com	100	atm[(c)]	5×10^5	120/40	atm	113	-	1×10^8	60	100
VIII-R-A48H	1/com	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A49H[(b)]	1/com	128	atm[(c)]	5×10^5	120/40[(d)]	atm	Sheet 12/A Sheet 13	Sheet 6/C	1×10^8	60	100
VIII-R-A50	1/com	100	atm[(c)]	1×10^3	107/40	atm	-	-	5×10^3	60	-
VIII-R-A51	1/com	104	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A52	1/com	100	atm[(c)]	1×10^3	112/40	atm	-	-	1×10^3	60	-
VIII-R-A53H[(b)]	1/com	100	atm[(c)]	1×10^4	120/40[(d)]	atm	-	-	5×10^5	60	-
VIII-R-A54	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A55H	1/com	100[(f)]	atm[(c)]	1×10^3	120/40	atm	Sheet 3	Sheet 6/C	1×10^5	60	100
VIII-R-A56	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A57	1/com	100[(f)]	atm[(c)]	6×10^3	120/40	atm	-	-	1×10^4	60	-
VIII-R-A58H	1/com	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A59[(b)]	2	128	atm[(c)]	1×10^3	120/40	atm	154	-	1×10^4	60	-
VIII-R-A60	void										
VIII-R-A61	2	100	atm[(c)]	1×10^3	112/40	atm	-	-	1×10^4	60	-
VIII-R-A62H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	5×10^5	60	-
VIII-R-A63H[(b)]	2	115	atm[(c)]	5×10^5	115/40	atm	140	-	1×10^7	60	-
VIII-R-A64H	2	100	atm[(c)]	5×10^5	120/40	atm	-	-	1×10^8	60	-

TABLE 3.11.B.1-1 (SHEET 39 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL A (Continued)											
VIII-R-A65H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A66H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A67H	2	100	atm[c]	5×10^5	109/40	atm	-	-	5×10^5	60	-
VIII-R-A68	2	100	atm[c]	6×10^3	120/40	atm	-	-	1×10^4	60	-
VIII-R-A69	2	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-A70H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A71H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A72H	2	100	atm[c]	5×10^5	109/40	atm	113	-	1×10^8	60	100
VIII-R-A73H	2	100	atm[c]	5×10^5	109/40	atm	-	-	1×10^8	60	-
VIII-R-A74H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-A75	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A76H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-A77H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	5×10^3	60	100
VIII-R-A78H[b]	2	115	atm[c]	5×10^5	120/40[d]	atm	128	-	1×10^8	60	-
VIII-R-A79H	2	100	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	5×10^6	60	100
VIII-R-A80H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A81H[b]	2	128	atm[c]	5×10^5	140/40[e]	atm	Sheet 12/A Sheet 13	Sheet 6/C	1×10^8	60	100
VIII-R-A82H[h]	2	100	atm[c]	5×10^5	107/40	atm	-	-	1×10^8	60	-
VIII-R-A83H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-

TABLE 3.11.B.1-1 (SHEET 40 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL A											
VIII-R-A84H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A85H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A86H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A87H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A88H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^8	60	-
VIII-R-A89	2	100	atm[c]	1×10^3	120/40	atm	-	-	5×10^3	60	-
VIII-R-A90	2	100	atm[c]	1×10^3	112/40	atm	-	-	1×10^3	60	-
VIII-R-A91H[b]	2	100	atm[c]	1×10^4	120/40[d]	atm	-	-	5×10^5	60	-
VIII-R-A92	2	104	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A93	2	100	atm[c]	1×10^3	107/40	atm	-	-	5×10^3	60	-
VIII-R-A94	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A95	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A96[b]	2	100	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-
VIII-R-A97	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-A98[b]	2	100	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-
VIII-R-A100H[b]	2	116	atm[c]	1×10^7	120/40[d]	atm	Sheet 12/A	24 psia[f]	5×10^7	60	100
VIII-R-A101H[b]	2	119	atm[c]	5×10^5	120/40[d]	atm	Sheet 12/A	20 psia[f]	1×10^8	60	100
VIII-R-A102H[b]	2	100	atm[c]	1×10^4	120/40	atm	126 Sheet 12/A	Sheet 6/C	1×10^4	60	100
VIII-R-A103H[b]	2	103	atm[c]	5×10^5	120/40[d]	atm	120 Sheet 12/A	Sheet 6/C	1×10^8	60	100

TABLE 3.11.B.1-1 (SHEET 41 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)	DBA/Post-DBA Pressure	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL A (Continued)											
VIII-R-A106H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-A108	void										
VIII-R-A109H	1/com	100[f]	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-A110H	2	100	atm[c]	5×10^5	120/40	atm	-	-	1×10^7	60	-
VIII-R-A111H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A112H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A113H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A114H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A115H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^3	60	-
VIII-R-A116H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A117H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A118H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A119H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A120H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A121H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A122H	1/com	115[f]	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A123H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A124H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A125H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A126H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-

TABLE 3.11.B.1-1 (SHEET 42 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[a]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL A (Continued)											
VIII-R-A127H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A128H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A129H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A130H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A131H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A132H	2	115	atm[c]	1×10^9	120/40	atm	-	-	1×10^9	60	-
VIII-R-A133H	2	130	atm[c]	1×10^9	130/40	atm	-	-	1×10^9	60	-
VIII-R-A134H	2	130	atm[c]	1×10^9	130/40	atm	-	-	1×10^9	60	-
VIII-UCA01H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA02H	2	130	atm[c]	5×10^5	130/40	atm	-	-	1×10^8	60	-
VIII-UCA03H	2	130	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA04H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA05H	1/2	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA06H	1/2	130	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA07H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA08H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA09H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA10H	1/com	130[f]	atm[c]	5×10^5	130/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-UCA11H	1/com	120[f]	atm[c]	5×10^5	120/40	atm	Sheet 12/A	Sheet 6/A	1×10^8	60	100
VIII-UCA12H	1/com	120[f]	atm[c]	1×10^7	120/40	atm	Sheet 12/A	20 psia[f]	5×10^7	60	100

TABLE 3.11.B.1-1 (SHEET 43 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[a]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL A (Continued)											
VIII-UCA13H	2	120	atm[c]	5×10^5	120/40	atm	Sheet 12/A	Sheet 6/B	1×10^8	60	100
VIII-UCA14H	2	120	atm[c]	1×10^7	120/40	atm	Sheet 12/A	20 psia[f]	5×10^7	60	100
AUXILIARY BUILDING - LEVEL 1											
VIII-R-101	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-102	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-103	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-104	1/com	100	atm[c]	1×10^3	108/40	atm	-	-	1×10^3	60	-
VIII-R-105	1/com	100	atm[c]	1×10^3	107/40	atm	-	-	1×10^3	60	-
VIII-R-106H[b]	1/com	115	atm[c]	1×10^7	120/40[d]	atm	-	-	5×10^7	60	-
VIII-R-107H[b]	1/com	131	atm[c]	5×10^5	131/40	atm	-	-	1×10^8	60	-
VIII-R-109	void										
VIII-R-110	1/com	100	atm[c]	1×10^4	104/40	atm	-	-	1×10^4	60	-
VIII-R-111[b]	1/com	128	atm[c]	1×10^3	128/40	atm	154	-	1×10^3	60	-
VIII-R-112	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-113	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-114	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-115	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-116[b]	1/com	85	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 44 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[(i)]	Pressure[(i)]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL 1 (Continued)											
VIII-R-117	1/com	100[(f)]	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-118[(b)]	1/com	85	atm[(c)]	1×10^3	120/40[(d)]	atm	-	-	1×10^3	60	-
VIII-R-120	1/com	100	atm[(c)]	1×10^3	112/40	atm	-	-	1×10^3	60	-
VIII-R-121	1/com	104	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-122	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-123	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-124	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-125	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-126	1/com	100[(f)]	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-127	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-128	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-129	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-130	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-131	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-132	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-133	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-134H[(b)]	1/com	117	atm[(c)]	5×10^5	120/40	atm	119	-	1×10^8	60	-
VIII-R-135H[(b)]	1/com	106	atm[(c)]	1×10^7	120/40[(d)]	atm	-	-	5×10^7	60	-
VIII-R-136H[(b)]	1/com	100	atm[(c)]	1×10^3	120/40	atm	Sheet 12/B Sheet 13	Sheet 6/C	1×10^3	60	100

TABLE 3.11.B.1-1 (SHEET 45 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[i]	Pressure[i]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL 1 (Continued)											
VIII-R-137	2	100	atm[c]	1×10^3	114/40	atm	-	-	1×10^3	60	-
VIII-R-138	2	104	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-139	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-140	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-141	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-142	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-143H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-144	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-145H	2	100	atm[c]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-146	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-147[b]	2	85	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-
VIII-R-148	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-149[b]	2	85	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-
VIII-R-150	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-151	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-152	2	100	atm[c]	1×10^3	108/40	atm	-	-	1×10^3	60	-
VIII-R-153	2	100	atm[c]	1×10^3	107/40	atm	-	-	1×10^3	60	-
VIII-R-154H[b]	2	115	atm[c]	1×10^7	120/40[d]	atm	-	-	5×10^7	60	-
VIII-R-155H[b]	2	131	atm[c]	5×10^5	131/40	atm	-	-	1×10^8	60	-

TABLE 3.11.B.1-1 (SHEET 46 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[i]	DBA/Post-DBA Pressure[i]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
AUXILIARY BUILDING - LEVEL 1 (Continued)											
VIII-R-156[b]	2	128	atm[c]	1×10^3	128/40	atm	154	-	1×10^3	60	-
VIII-R-157	2	100	atm[c]	1×10^4	104/40	atm	-	-	1×10^4	60	-
VIII-R-158	void										
VIII-R-160H[b]	2	106	atm[c]	1×10^7	120/40[d]	atm	-	-	5×10^7	60	-
VIII-R-161H[b]	2	117	atm[c]	1×10^5	117/40	atm	119	-	1×10^8	60	-
VIII-R-162H[b]	2	100	atm[c]	1×10^3	120/40	atm	Sheet 12/B Sheet 13	Sheet 6/C	1×10^3	60	100
VIII-R-UC101H[b]	1/com	104[f]	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VIII-R-UC102H[b]	2	104	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
AUXILIARY BUILDING - LEVEL 2											
VIII-R-201	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-202	1/com	100	atm[c]	1×10^3	107/40	atm	-	-	1×10^3	60	-
VIII-R-203	1/com	100	atm[c]	1×10^3	107/40	atm	-	-	1×10^3	60	-
VIII-R-204	1/com	100	atm[c]	1×10^3	116/40	atm	-	-	1×10^3	60	-
VIII-R-205[b]	1/com	128	atm[c]	1×10^3	128/40	atm	154	-	1×10^3	60	-
VIII-R-206	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-207[b]	1/com	100	atm[c]	1×10^3	120/40[d]	atm	-	-	1×10^3	60	-
VIII-R-208H[b]	1/com	100	atm[c]	1×10^3	120/40	atm	Sheet 12/B Sheet 13	Sheet 6/C	1×10^3	60	100
VIII-R-209H[b]	1/com	100	atm[c]	1×10^3	105/40[g]	atm	-	-	1×10^6	60	-
VIII-R-210H[b]	1/com	100	atm[c]	1×10^3	104/40	atm	-	-	1×10^6	60	-
VIII-R-211	1/com	104	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 47 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[(i)]	Pressure[(i)]	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL 2 (Continued)											
VIII-R-212H[(h)]	1/com	100	atm[(c)]	1×10^3	127/40	atm	-	-	5×10^5	60	-
VIII-R-213	void										
VIII-R-214	spare										
VIII-R-215	1/com	100[(f)]	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-216	1/com	100[(f)]	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-217	1/com	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-218	2	104	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-219H[(b)]	2	100	atm[(c)]	1×10^3	104/40	atm	-	-	1×10^6	60	-
VIII-R-220H[(b)]	2	104	atm[(c)]	1×10^3	105/40	atm	-	-	1×10^6	60	-
VIII-R-221H[(h)]	2	100	atm[(c)]	1×10^3	127/40	atm	-	-	5×10^5	60	-
VIII-R-222	2	100	atm[(c)]	1×10^3	120/40	atm	Sheet 12/B Sheet 13	Sheet 6/C	1×10^3	60	100
VIII-R-223[(b)]	2	100	atm[(c)]	1×10^3	120/40[(d)]	atm	-	-	1×10^3	60	-
VIII-R-224H	2	100	atm[(c)]	1×10^3	120/40	atm	Sheet 5	Sheet 6/A	1×10^3	60	100
VIII-R-225	2	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-226	2	100	atm[(c)]	1×10^3	107/40	atm	-	-	1×10^3	60	-
VIII-R-227	2	100	atm[(c)]	1×10^3	107/40	atm	-	-	1×10^3	60	-
VIII-R-228[(b)]	2	100	atm[(c)]	1×10^3	128/40	atm	154	-	1×10^3	60	-
VIII-R-229	2	100	atm[(c)]	1×10^3	116/40	atm	-	-	1×10^3	60	-
VIII-R-230	2	100	atm[(c)]	1×10^3	120/40	atm	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 48 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY BUILDING - LEVEL 3											
VIII-R-301	1/com	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-302	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-303	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-304	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-305	1/com	100[f]	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-306	2	100	atm[c]	1×10^3	120/40	atm	-	-	1×10^3	60	-
VIII-R-307	1	104	atm[c]	1×10^3	-	-	-	-	1×10^3	-	-
VIII-R-308	2	104	atm[c]	1×10^3	-	-	-	-	1×10^3	-	-

TABLE 3.11.B.1-1 (SHEET 49 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

- = No high-energy line or safety-related equipment located in the room.

b. Indicates rooms that are served by Class 1E environmental support systems.

c. Within negative pressure boundary.

d. Maximum abnormal room temperature value rounded up to 120°F per reference 6.2.2.3. Consult reference 6.2.2.3 for actual temperature.

e. Maximum room temperature is due to one train RHR cooldown.

f. Normal room temperature is an assumed value which is based on bulk building or surrounding area temperature. This room does not contain safety-related equipment/instrumentation.

g. Short spike <1 s. For remainder of transient, see figure 6/C.

h. Room is included in the area temperature monitoring program (reference FSAR section 16.3).

i. Sheet numbers refer to figure 3.11.B.1-1. Alpha designators refer to specific curves on these sheets.

AUXILIARY BUILDING
(Sheet 41 of 41)

TABLE 3.11.B.1-1 (SHEET 50 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[n]	Pressure[n]	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL C											
IX-R-C01	1/com	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-C02	1/com	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-C03	1/com	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-C04[b]	2	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-C05[b]	2	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-C006[b]	2	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-C07H	1/com	104	atm	5×10^5	120/55	-	Sheet 5	Sheet 6/A	1×10^8	60	100
IX-R-C08[b]	2	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	100
IX-R-C09	1/com	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
CONTROL BUILDING - LEVEL B											
IX-R-B01[b]	2	85	atm	1×10^3	100/65	-	-	-	5×10^3	60	-
IX-R-B02H[m]	2	100	atm	1×10^3	104/50	-	-	-	5×10^4	60	-
IX-R-B03	2	100	atm	1×10^3	103/55	-	-	-	1×10^4	60	-
IX-R-B04[b]	2	85	atm	1×10^3	100/65[c]	-	-	-	1×10^3	60	-
IX-R-B05[b]	2	100	atm	1×10^3	101/65	-	-	-	5×10^3	60	-
IX-R-B06[b]	2	100	atm	1×10^3	106/55	-	-	-	1×10^3	60	-
IX-R-B07	2	100	atm	1×10^3	124/55	-	-	-	1×10^3	60	-
IX-R-B08	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B09	2	100	atm	1×10^3	105/55	-	-	-	1×10^4	60	-

CONTROL BUILDING
(Sheet 1 of 22)

TABLE 3.11.B.1-1 (SHEET 51 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL B (Continued)											
IX-R-B10H[m]	2	100	atm	1×10^3	107/50	atm	-	-	5×10^4	60	-
IX-R-B11H	2	100	atm	1×10^3	103/50	atm	-	-	5×10^4	60	-
IX-R-B12	2	100	atm	1×10^3	120/55	-	-	-	1×10^4	60	-
IX-R-B13	2	100	atm	1×10^3	120/55	-	-	-	5×10^3	60	-
IX-R-B14	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B15	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B16[b]	2	90	atm	1×10^3	100/65	-	-	-	1×10^3	60	-
IX-R-B17[b]	2	90	atm	1×10^3	100/65	-	-	-	1×10^3	60	-
IX-R-B18[b]	2	85	atm	1×10^3	100/65[d]	-	-	-	5×10^4	60	-
IX-R-B19H[m]	2	100	atm	1×10^3	106/50	atm					
IX-R-B20	spare		atm	1×10^3			-	-	1×10^3	60	-
IX-R-B21	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B22	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B23	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B24	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B25[b]	2	75	atm	1×10^3	80/70	-	-	-	1×10^3	50	-
IX-R-B26[b]	2	100	atm	1×10^3	100/65	-	-	-	1×10^3	50	-
IX-R-B27[b]	2	75	atm	1×10^3	80/70	-	-	-	1×10^3	50	-
IX-R-B28[b]	2	80	atm	1×10^3	80/70	atm	-	-	1×10^3	60	-
IX-R-B29[b]	2	100	atm	1×10^3	100/65[e]	-	-	-	1×10^3	50	-

TABLE 3.11.B.1-1 (SHEET 52 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[(a)]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL B (Continued)											
IX-R-B30	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B31[(b)]	2	100	atm	1×10^3	100/55	-	-	-	1×10^3	50	-
IX-R-B32[(b)]	2	75	atm	1×10^3	80/70[(f)]	-	-	-	1×10^3	50	-
IX-R-B33	2	100	atm	1×10^3	123/65	-	-	-	1×10^3	60	-
IX-R-B34	2	100	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-B35	2	100	atm	1×10^3	120/65	-	-	-	1×10^3	60	-
IX-R-B36[(b)]	2	100	atm	1×10^3	100/55[(g)]	-	-	-	1×10^3	50	-
IX-R-B37[(b)]	2	75	atm	1×10^3	80/70	-	-	-	1×10^3	50	-
IX-R-B38	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B39	1/com	100	atm	1×10^3	104/65	-	-	-	1×10^3	60	-
IX-R-B40	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B41	1/com	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B42	1/com	100	atm	1×10^3	104/65	-	-	-	1×10^3	60	-
IX-R-B43	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B44[(b)]	1/com	75	atm	1×10^3	80/70	-	-	-	1×10^3	50	-
IX-R-B45	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B46	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B47[(b)]	1/com	85	atm	1×10^3	92/55[(g)]	-	-	-	1×10^3	50	-
IX-R-B48[(b)]	1/com	100	atm	1×10^3	100/55	-	-	-	1×10^3	50	-
IX-R-B49[(b)]	1/com	75	atm	1×10^3	80/70[(f)]	-	-	-	1×10^3	50	-
IX-R-B50	1/com	100	atm	1×10^3	124/65	-	-	-	1×10^3	60	-

CONTROL BUILDING
(Sheet 3 of 22)

TABLE 3.11.B.1-1 (SHEET 53 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[a]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL B (Continued)											
IX-R-B51	1/com	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B52[b]	1/com	85	atm	1×10^3	100/65[e]	-	-	-	1×10^3	50	-
IX-R-B53[b]	1/com	80	atm	1×10^3	80/70	atm	-	-	1×10^3	60	-
IX-R-B54[b]	1/com	75	atm	1×10^3	80/70	-	-	-	1×10^3	50	-
IX-R-B55[b]	1/com	100	atm	1×10^3	100/65	-	-	-	1×10^3	50	-
IX-R-B56[b]	1/com	75	atm	1×10^3	76/70	-	-	-	1×10^3	50	-
IX-R-B57	spare						-	-			
IX-R-B58	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B59	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B60[b]	1/com	90	atm	1×10^3	100/65	-	-	-	1×10^3	60	-
IX-R-B61[b]	1/com	85	atm	1×10^3	100/65[d]				1×10^3	60	-
IX-R-B62[b]	1/com	90	atm	1×10^3	100/65	-	-	-	1×10^3	60	-
IX-R-B63	1/com	100	atm	1×10^3	120/55	-	-	-	5×10^3	60	-
IX-R-B64	spare						-	-			
IX-R-B65H[m]	1/com	100	atm	1×10^3	106/50	atm	-	-	5×10^4	60	-
IX-R-B66	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^4	60	-
IX-R-B67H[b]	1/com	100	atm	1×10^3	103/50	atm	-	-	5×10^4	60	-
IX-R-B68	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B69	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B70	1/com	100	atm	1×10^3	106/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 54 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[n]	DBA/Post-DBA Pressure[n]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
CONTROL BUILDING - LEVEL B (Continued)											
IX-R-B71	1/com	100	atm	1×10^3	124/55	-	-	-	1×10^3	60	-
IX-R-B72	1/com	100	atm	1×10^3	120/65	-	-	-	1×10^3	60	-
IX-R-B73	1/com	100	atm	1×10^3	105/55	-	-	-	1×10^4	60	-
IX-R-B74H[m]	1/com	100	atm	1×10^3	107/50	atm	-	-	5×10^4	60	-
IX-R-B75	1/com	100	atm	1×10^3	101/65	-	-	-	5×10^3	60	-
IX-R-B76[b]	1/com	85	atm	1×10^3	86/65[c]	-	-	-	1×10^3	60	-
IX-R-B77	1/com	100	atm	1×10^3	103/55	atm	-	-	1×10^4	60	
IX-R-B78H[m]	1/com	100	atm	1×10^3	104/50	-	-	-	5×10^4	60	-
IX-R-B79[b]	1/com	85	atm	1×10^3	100/65	-	-	-	5×10^3	60	-
IX-R-B80	2	100	atm	1×10^3	120/55	-	-	-	5×10^3	60	-
IX-R-B81	1/com	100	atm	1×10^3	120/55	-	-	-	5×10^3	60	-
IX-R-B82	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B83	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B84[b]	1/com	85	atm	1×10^3	100/65	atm	-	-	1×10^3	60	-
IX-R-B85[b]	2	85	atm	1×10^3	100/65	atm	-	-	1×10^3	60	-
IX-R-B86	2	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B87	2	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B88	1/com	100	atm	1×10^3	106/50	-	-	-	1×10^3	60	-
IX-R-B89	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B90	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-B91	1	100	atm	1×10^3	104/50	-	-	-	5×10^3	60	-
IX-R-B92	2	100	atm	1×10^3	104/50	-	-	-	5×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 55 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)	DBA/Post-DBA Pressure	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
CONTROL BUILDING - LEVEL A											
IX-R-A01	2	100	atm	1×10^3	115/65	-	-	-	1×10^4	60	-
IX-R-A02H[m]	2	100	atm	1×10^3	107/50	atm	-	-	1×10^5	60	-
IX-R-A03	2	100	atm	1×10^3	101/65	-	-	-	5×10^3	60	-
IX-R-A04	2	100	atm	1×10^3	103/65	-	-	-	5×10^3	60	-
IX-R-A05	spare										
IX-R-A06	2	100	atm	1×10^3	101/65	-	-	-	1×10^3	60	-
IX-R-A07	2	100	atm	1×10^3	107/55	-	-	-	1×10^3	60	-
IX-R-A10H[m]	2	100	atm	1×10^3	106/50	atm	-	-	1×10^5	60	-
IX-R-A11	2	100	atm	1×10^3	114/65	-	-	-	1×10^4	60	-
IX-R-A12H	2	100	atm	1×10^3	106/50	atm	-	-	1×10^5	60	-
IX-R-A13	2	100	atm	1×10^3	120/55	-	-	-	1×10^4	60	-
IX-R-A14	2	100	atm	1×10^3	103/55	-	-	-	1×10^4	60	-
IX-R-A15[b]	2	100	atm	1×10^2	100/65[h]	-	-	-	1×10^3	60	-
IX-R-A16[b]	2	100	atm	1×10^2	100/65[l]	-	-	-	1×10^2	60	-
IX-R-A17	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A18	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A19	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A20	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A21	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A22[b]	2	100	atm	1×10^3	100/65[i]	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 56 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[a]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL A (Continued)											
IX-R-A23	2	100	atm	1×10^3	102/55	-	-	-	1×10^3	60	-
IX-R-A24[b]	2	75	atm	1×10^3	80/70	atm	-	-	1×10^3	60	-
IX-R-A25	spare										
IX-R-A26	spare										
IX-R-A27	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A28	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A29	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A30	2	75	atm	1×10^3	104/60	atm	-	-	1×10^3	50	-
IX-R-A31	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A32	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A33	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A34	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A35	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A36	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A37	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A38	1/com	75	atm	1×10^3	104/60	atm	-	-	1×10^3	50	-
IX-R-A39	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A40	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A41	spare										
IX-R-A42	spare										
IX-R-A43[b]	1/com	75	atm	1×10^3	80/70	atm	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 57 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL A (Continued)											
IX-R-A44	1/com	100	atm	1×10^3	102/55	-	-	-	1×10^3	60	-
IX-R-A45[(b)]	1/com	100	atm	1×10^3	100/65[(i)]	-	-	-	1×10^3	60	-
IX-R-A46	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A47	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A48[(b)]	1/com	100	atm	1×10^2	100/65[(l)]	-	-	-	1×10^2	60	-
IX-R-A49	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A50[(b)]	1/com	100	atm	1×10^2	100/65[(h)]	-	-	-	1×10^3	60	-
IX-R-A51	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^4	60	-
IX-R-A52	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^4	60	-
IX-R-A53H[(b)]	1/com	100	atm	1×10^3	106/60	atm	-	-	1×10^5	60	-
IX-R-A54	1/com	100	atm	1×10^3	114/65	-	-	-	1×10^4	60	-
IX-R-A55H[(m)]	1/com	100	atm	1×10^3	106/50	atm	-	-	1×10^5	60	-
IX-R-A58	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A59	1/com	100	atm	1×10^3	101/55	-	-	-	1×10^3	60	-
IX-R-A60	1/com	100	atm	1×10^3	101/65	-	-	-	1×10^3	60	-
IX-R-A61	1/com	100	atm	1×10^3	101/65	-	-	-	5×10^3	60	-
IX-R-A62	1/com	100	atm	1×10^3	103/55	-	-	-	5×10^3	60	-
IX-R-A63	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A64	1/com	100	atm	1×10^3	115/55	-	-	-	1×10^4	60	-
IX-R-65H[(b)]	1/com	100	atm	1×10^3	107/50	atm	-	-	1×10^5	60	-

TABLE 3.11.B.1-1 (SHEET 58 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[n]	DBA/Post-DBA Pressure[n]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
CONTROL BUILDING - LEVEL A (Continued)											
IX-R-A66	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A67	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A68	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A69	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A70	1/com	104	atm	1×10^3	113/55	-	-	-	1×10^3	60	-
IX-R-A71	1/com	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A72	2	104	atm	1×10^3	113/55	-	-	-	1×10^3	60	-
IX-R-A73	2	104	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A74	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-A75[b]	1/com	75	atm	1×10^3	76/70	atm	-	-	1×10^3	60	-
IX-R-A76[b]	2	75	atm	1×10^3	80/70	atm	-	-	1×10^3	60	-
IX-R-A77[b]	1/com	85	atm	1×10^3	100/65	-	-	-	5×10^3	60	-
IX-R-A78	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^4	60	-
IX-R-A79[b]	2	85	atm	1×10^3	100/65	-	-	-	5×10^3	60	-
IX-R-A80	2	100	atm	1×10^3	120/55	-	-	-	1×10^4	60	-
IX-R-A81[b]	2	100	atm	1×10^3	100/55	-	-	-	1×10^3	60	-
IX-R-A82[b]	1/com	100	atm	1×10^3	100/55	-	-	-	1×10^3	60	-
CONTROL BUILDING - LEVEL 1											
IX-R-101	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-102	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 59 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 1 (Continued)											
IX-R-103	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-104	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-105	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-106	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-107	1/com	75	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-108	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-109	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-110	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-111	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-112	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-113	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-114	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-115	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-116	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-117	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-118	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-119	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-120	1/com	80	atm	1×10^3	87/55	-	-	-	1×10^3	60	-
IX-R-121	1/com	80	atm	1×10^3	86/65	-	-	-	1×10^3	60	-
IX-R-122	1/com	80	atm	1×10^3	86/65	-	-	-	1×10^3	60	-
IX-R-123	1/com	80	atm	1×10^3	86/65	-	-	-	1×10^3	60	-
IX-R-124	1/com	80	atm	1×10^3	86/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 60 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 1 (Continued)											
IX-R-125	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-126	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-127	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-128	1/com	80	atm	1×10^3	87/55	-	-	-	1×10^3	60	-
IX-R-129	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-130	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-131	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-132	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-133	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-134	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-135	1/com	75	atm	1×10^3	82/70	-	-	-	1×10^3	60	-
IX-R-136	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-137	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-138	1/com	80	atm	1×10^3	87/65	-	-	-	1×10^3	60	-
IX-R-139	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-140	1/com	75	atm	1×10^3	83/70	atm	-	-	1×10^3	60	-
IX-R-141	1/com	80	atm	1×10^3	87/60	-	-	-	1×10^3	60	-
IX-R-142	1/com	80	atm	1×10^3	104/55	atm	-	-	1×10^3	60	-
IX-R-143	1/com	75	atm	1×10^3	83/65	atm	-	-	1×10^3	60	-
IX-R-144	1/com	75	atm	1×10^3	83/65	atm	-	-	1×10^3	60	-
IX-R-145	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-146	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 61 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA[c]			Relative Humidity (max)	
Environmental Designator[a]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[n]	Pressure[n]	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 1 (Continued)											
IX-R-147	1/com	80	atm	1×10^3	87/55	-	-	-	1×10^3	60	-
IX-R-148	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-149	1/com	80	atm	1×10^3	87/55	-	-	-	1×10^3	60	-
IX-R-150	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-151	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-152	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-153	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-154	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-155	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-156[b]	1/com	85	+1/8" WG	1×10^3	85/60	atm[k]	85	(k)	1×10^3	60	-
IX-R-157[b]	1/com	85	+1/8" WG	1×10^3	85/70	atm[k]	85	-	1×10^3	60	-
IX-R-158[b]	1/com	85	+1/8" WG	1×10^3	85/70	atm[k]	85	-	1×10^3	60	-
IX-R-159	1/com	80	atm	1×10^3	104/55	atm	-	-	1×10^3	60	-
IX-R-160[b]	1/com	85	+1/8" WG	1×10^3	93/70	atm[k]	93	(k)	1×10^3	60	-
IX-R-161[b]	1/com	85	+1/8" WG	1×10^3	85/70	atm[k]	85	(k)	1×10^3	60	-
IX-R-162[b]	1/com	85	+1/8" WG	1×10^3	85/70	atm[k]	85	(k)	1×10^3	60	-
IX-R-163[b]	1/com	85	+1/8" WG	1×10^3	85/70[j]	atm[k]	85	(k)	1×10^3	50	-
IX-R-164[b]	2	85	+1/8" WG	1×10^3	85/70[j]	atm[k]	85	(k)	1×10^3	50	-
IX-R-165	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-166	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-167	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 62 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[(a)]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 1 (Continued)											
IX-R-168	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-169	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-170	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-171	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-172	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-173	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-174	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-175	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-176	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-177	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-178	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-179	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-180	1/com	100	+1/8" WG	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-181	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-182	1/com	100	+1/8" WG	1×10^3	120/55	atm	-	-	1×10^3	60	-
IX-R-183	1/com	75	+1/8" WG	1×10^3	104/60	atm	-	-	1×10^3	60	-
IX-R-184	1/com	75	+1/8" WG	1×10^3	104/55	atm	-	-	1×10^3	60	-
IX-R-185	1/com	75	+1/8" WG	1×10^3	104/60	atm	-	-	1×10^3	60	-
IX-R-186	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-187	1/com	75	+1/8" WG	1×10^3	104/55	atm	-	-	1×10^3	60	-
IX-R-188	1/com	75	+1/8" WG	1×10^3	104/55	atm	-	-	1×10^3	60	-
IX-R-189	1/com	75	+1/8" WG	1×10^3	104/55	atm	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 63 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[n]	Pressure[n]	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 1 (Continued)											
IX-R-190	1/com	75	+1/8" WG	1×10^3	120/55	atm	-	-	1×10^3	60	-
IX-R-191	1/com	80	+1/8" WG	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-192	1/com	100	+1/8" WG	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-193	1/com	75	+1/8" WG	1×10^3	104/55	atm	-	-	1×10^3	60	-
IX-R-194	1/com	100	+1/8" WG	1×10^3	120/55	atm	-	-	1×10^3	60	-
IX-R-195	1/com	100	+1/8" WG	1×10^3	120/55	atm	-	-	1×10^3	60	-
IX-R-196	1/com	100	+1/8" WG	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-197	1/com	100	+1/8" WG	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-198	1/com	100	+1/8" WG	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-199	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-199A	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-199B	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
Sample chase H	1/com	130	atm	5×10^5	130/40	-	Sheet 5	Sheet 6/A	1×10^8	60	100
CONTROL BUILDING - LEVEL 2											
IX-R-201	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-202	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-203	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-204	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-205	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-206	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-207	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-208	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 64 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 2 (Continued)											
IX-R-209	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-210	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-211	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-212	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-213	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-214	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-215	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-216	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-217	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-218	2	80	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-219	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-220	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-221	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-222	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-223[(b)]	2	100	atm	1×10^3	100/65	-	-	-	1×10^3	60	-
IX-R-224	2	100	atm	1×10^3	102/55	-	-	-	1×10^3	60	-
IX-R-225	1/com	100	atm	1×10^3	102/55	-	-	-	1×10^3	60	-
IX-R-226[(b)]	1/com	100	atm	1×10^3	100/65	-	-	-	1×10^3	60	-
IX-R-227	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-228	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-229	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 65 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 2 (Continued)											
IX-R-230	1/com	80	atm	1×10^3	90/65	-	-	-	1×10^3	60	-
IX-R-231	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-232	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-233	1/com	100	atm	1×10^3	101/65	-	-	-	1×10^3	60	-
IX-R-234	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-235	spare										
IX-R-236	spare										
IX-R-237	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-238	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-239	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-240	spare										
IX-R-241	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-242	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-243	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-244	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-245	1/com	80	atm	1×10^3	90/65	-	-	-	1×10^3	60	-
IX-R-246	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-247	1/com	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-248	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-249	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-250	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-251	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-

CONTROL BUILDING
(Sheet 16 of 22)

TABLE 3.11.B.1-1 (SHEET 66 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[(n)]	Pressure[(n)]	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 2 (Continued)											
IX-R-252	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-253	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-254	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-255	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-256	spare										
IX-R-257	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-258	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-259	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-260	spare										
IX-R-261	spare										
IX-R-262	spare										
IX-R-263	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-264	2	80	atm	1×10^3	90/65	-	-	-	1×10^3	60	-
IX-R-265	spare										
IX-R-266	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-267	1/com	75	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-268	1/com	75	atm	1×10^3	104/60	-	-	-	1×10^3	60	-
IX-R-269	2	80	atm	1×10^3	104/55	-	-	-	1×10^3	60	-
IX-R-270	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-271	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-272	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-273	1/com	100	atm	1×10^3	120/55	-	-		1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 67 OF 92)
ENVIRONMENTAL CONDITIONS

		Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
Environmental Designator[(a)]	Unit	Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[(n)]	Pressure[(n)]	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 2 (Continued)											
IX-R-274	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-275	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-276	2	130	atm	1×10^3	130/40	-	-	-	1×10^3	60	-
IX-R-277	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-UC201H	1/com	130	atm	5×10^5	130/40	-	Sheet 5	Sheet 6/A	1×10^8	60	100
IX-UC202H	2	130	atm	5×10^5	130/40	-	Sheet 5	Sheet 6/A	1×10^8	60	100
CONTROL BUILDING - LEVEL 3											
IX-R-301	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-302	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-303	1/com	100	atm	1×10^3	102/55	-	-	-	1×10^3	60	-
IX-R-304	1/com	100	atm	1×10^3	107/65	-	-	-	1×10^8	60	-
IX-R-305[(b)]	1/com	104	atm	1×10^3	110/65	-	-	-	5×10^3	60	-
IX-R-306	spare										
IX-R-307	2	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-308[(b)]	1/com	104	atm	1×10^3	110/65	-	-	-	1×10^3	60	-
IX-R-309	1/com	100	atm	1×10^3	110/55	-	-	-	1×10^3	60	-
IX-R-310[(b)]	1/com	104	atm	1×10^3	110/65	-	-	-	1×10^3	60	-
IX-R-311[(b)]	1/com	104	atm	1×10^3	110/65	-	-	-	5×10^3	60	-
IX-R-312[(b)]	1/com	104	atm	1×10^3	110/65	-	-	-	5×10^3	60	-
IX-R-313[(b)]	1/com	104	atm	1×10^3	110/65	-	-	-	1×10^3	60	-
IX-R-314	void										
IX-R-315	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-316	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-317	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 68 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[n]	Pressure[n]	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 3 (Continued)											
IX-R-318	spare										
IX-R-319	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-320[b]	1/com	104	atm	1×10^3	110/65	-	-	-	1×10^3	60	-
IX-R-321[b]	1/com	104	atm	1×10^3	110/65	-	-	-	1×10^3	60	-
IX-R-322[b]	1/com	100	atm	1×10^3	100/55	-	-	-	1×10^3	60	-
IX-R-323	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-324	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-325[b]	1/com	100	atm	1×10^3	118/55	-	-	-	1×10^3	60	-
IX-R-326	spare										
IX-R-327	spare										
IX-R-328	1/com	100	atm	1×10^3	120/40	-	-	-	1×10^3	100	-
IX-R-329	1/com	100	atm	1×10^3	120/40	-	-	-	1×10^3	60	-
IX-R-330	1/com	100	atm	1×10^3	120/40	-	-	-	1×10^3	60	-
CONTROL BUILDING - LEVEL 4											
IX-R-401	2	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	60	-
IX-R-402	2	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	60	-
IX-R-403	1/com	100	atm	1×10^3	120/55	11.7 psia	-	-	1×10^3	60	-
IX-R-404	1/com	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	60	-
IX-R-405	1/com	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	60	-
IX-R-406	1/com	80	atm	1×10^3	104/60	11.7 psia	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 69 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
CONTROL BUILDING - LEVEL 4 (Continued)											
IX-R-407	1/com	100	atm	1×10^3	120/55	11.7 psia	-	-	1×10^3	60	-
IX-R-408	1/com	100	atm	1×10^3	120/55	11.7 psia	-	-	1×10^3	60	-
IX-R-409	1/com	80	atm	1×10^3	104/60	11.7 psia	-	-	1×10^3	60	-
IX-R-410	1/com	100	atm	1×10^3	120/55	11.7 psia	-	-	1×10^3	60	-
CONTROL BUILDING - LEVEL 5											
IX-R-501	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-
IX-R-502	1/com	100	atm	1×10^3	120/55	-	-	-	1×10^3	60	-

TABLE 3.11.B.1-1 (SHEET 70 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

\- = No high-energy line or safety-related equipment located in the room.

b. Indicates rooms that are served by Class 1E environmental support systems.

c. Maximum room temperature 30 min after loss of all cooling is 112.7°F.

d. Maximum room temperature 30 min after loss of all cooling is 121.0°F.

e. Maximum room temperature 30 min after loss of all cooling is 123.7°F.

f. Maximum room temperature 30 min after loss of all cooling is 82.3°F.

g. Maximum room temperature 30 min after loss of all cooling is 122.0°F.

h. Maximum room temperature 30 min after loss of all cooling is 106.7°F.

i. Maximum room temperature 30 min after loss of all cooling is 106.0°F.

j. Maximum room temperature 30 min after loss of all cooling is 92.8°F.

CONTROL BUILDING
(Sheet 21 of 22)

TABLE 3.11.B.1-1 (SHEET 71 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES (Continued)

k. Indicates rooms which have an abnormal and DBA/post-DBA pressure of at least +1/8 in. WG with respect to adjacent areas when Class 1E environment support system is operated in emergency mode and atmospheric pressure when operated in isolation mode.

l. Maximum room temperature 30 min after loss of all cooling is 109.0°F.

m. Room is included in the area temperature monitoring program (FSAR 16.3).

n. Sheet numbers refer to figure 3.11.B.1-1. Alpha designators refer to specific curves on these sheets.

CONTROL BUILDING
(Sheet 22 of 22)

TABLE 3.11.B.1-1 (SHEET 72 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[g]	Pressure[g]	Int Dose (rads)	Normal (%)	DBA (%)
FUEL HANDLING BUILDING - LEVEL C											
VII-R-C01H[b]	2	104	atm[c]	5×10^5	110/40	atm	Sheet 16	-	1×10^7	60	-
VII-R-C02H[b]	2	104	atm[c]	5×10^5	117/40	atm	Sheet 16	-	5×10^7	60	-
VII-R-C03H[b]	2	104	atm[c]	5×10^5	120/40[d]	atm	Sheet 16	-	5×10^6	60	-
VII-R-C04H	2	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-C05H	1/com	104	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VII-R-C06H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-C07H[b]	1/com	104	atm[c]	5×10^5	110/40	atm	Sheet 16	-	1×10^7	60	-
VII-R-C08H[b]	1/com	104	atm[c]	5×10^5	117/40	atm	Sheet 16	-	5×10^7	60	-
VII-R-C09H[b]	1/com	104	atm[c]	5×10^5	120/40[d]	atm	Sheet 16	-	5×10^6	60	-
FUEL HANDLING BUILDING - LEVEL B											
VII-R-B01H[b]	2	104	atm[c]	5×10^5	120/40[e]	atm	Sheet 14	-	1×10^6	60	-
VII-R-B02H	2	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-B03H[b]	2	104	atm[c]	5×10^5	119/40	atm	Sheet 14	-	5×10^7	60	-
VII-R-B04H	1/com	104	atm[c]	5×10^5	120/40	atm	-	-	1×10^6	60	-
VII-R-B05H	1/com	104	atm[c]	5×10^5	120/40	atm	-	-	5×10^7	60	-
VII-R-B06H[f]	2	104	atm[c]	1×10^4	107/40	atm	-	-	5×10^5	60	-
VII-R-B07H	1/com	104	atm[c]	5×10^5	120/40	atm	-	-	1×10^6	60	-
VII-R-B08H[f]	1/com	104	atm[c]	1×10^4	107/40	atm	-	-	5×10^5	60	-

FUEL HANDLING BUILDING
(Sheet 1 of 4)

TABLE 3.11.B.1-1 (SHEET 73 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[g]	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
FUEL HANDLING BUILDING - LEVEL B (Continued)											
VII-R-B09H	1/com	104	atm[c]	5×10^5	120/40	atm	-	-	5×10^7	60	-
VII-R-B10H	1/com	104	atm[c]	5×10^5	120/40	atm	-	-	1×10^6	60	-
VII-R-B11H[b]	1/com	100	atm[c]	5×10^5	120/40[e]	atm	Sheet 14	-	1×10^6	60	-
VII-R-B12H[b]	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-B13H[b]	1/com	104	atm[c]	5×10^5	119/40	atm	Sheet 14	-	5×10^7	60	-
FUEL HANDLING BUILDING - LEVEL A											
VII-R-A01H[b]	2	104	atm[c]	5×10^5	120/40[e]	atm	Sheet 16	-	4×10^6	60	-
VII-R-A02H	2	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^6	60	-
VII-R-A03H	2	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^6	60	-
VII-R-A04H[b]	2	104	atm[c]	1×10^4	108/40	atm	-	-	1×10^6	60	-
VII-R-A05H	2	104	atm[c]	1×10^4	105/40	atm	-	-	5×10^5	60	-
VII-R-A06H	1/com	104	atm[c]	1×10^4	105/40	atm	-	-	5×10^5	60	-
VII-R-A07H[b]	1/com	104	atm[c]	1×10^4	108/40	atm	-	-	1×10^6	60	-
VII-R-A08H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^6	60	-
VII-R-A09H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^6	60	-
VII-R-A10H[b]	1/com	104	atm[c]	5×10^5	120/40[e]	atm	Sheet 16	-	4×10^6	60	-
VII-R-A11H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-A12H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-

FUEL HANDLING BUILDING
(Sheet 2 of 4)

TABLE 3.11.B.1-1 (SHEET 74 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal Temp (°F)	Normal Pressure	Normal Int Dose (rads)	Abnormal/Test Temp (°F) (max/min)	Abnormal/Test Pressure	DBA/Post-DBA Temp (°F)[g]	DBA/Post-DBA Pressure[g]	DBA/Post-DBA Int Dose (rads)	Relative Humidity (max) Normal (%)	Relative Humidity (max) DBA (%)
FUEL HANDLING BUILDING - LEVEL A (Continued)											
VII-R-A13H	2	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^8	60	-
VII-R-A14H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^8	60	-
FUEL HANDLING BUILDING - LEVEL 1											
VII-R-101H	2	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-102H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-103H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-104H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-105[b]	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^4	60	-
VII-R-106H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-107H	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	5×10^5	60	-
VII-R-108[b]	1/com	104	atm[c]	1×10^4	120/40	atm	-	-	1×10^4	60	-
VII-R-109H	1/com	104	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
VII-R-110H	2	104	atm[c]	5×10^5	120/40	atm	Sheet 5	Sheet 6/A	1×10^8	60	100
FUEL HANDLING BUILDING - LEVEL 3											
VII-R-301[b]	1/com	104	atm[c]	1×10^4	114/40	atm	-	-	1×10^4	60	-
VII-R-302H	1/com	104	atm[c]	1×10^4	131/40	atm	-	-	5×10^5	60	-
VII-R-303[b]	1/com	104	atm[c]	2×10^3	111/40	atm	-	-	3×10^3	60	-
VII-R-304[b]	1/com	104	atm[c]	2×10^3	112/40	atm	-	-	3×10^3	60	-
VII-R-305H	1/com	104	atm[c]	2×10^3	107/40	atm	-	-	3×10^3	60	-

FUEL HANDLING BUILDING
(Sheet 3 of 4)

TABLE 3.11.B.1-1 (SHEET 75 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

- = No high-energy line or safety-related equipment located in the room.

b. Indicates rooms that are served by Class 1E environmental support systems.

c. Within negative pressure boundary.

d. Maximum abnormal room temperature value rounded up to 120°F per reference 6.2.2.3. Consult reference 6.2.2.3 for actual temperature.

e. Maximum temperature is due to one train RHR cooldown.

f. Room is included in the area temperature monitoring program (FSAR 16.3).

g. Sheet numbers refer to figure 3.11.B.1-1. Alpha designators refer to specific curves on these sheets.

FUEL HANDLING BUILDING
(Sheet 4 of 4)

TABLE 3.11.B.1-1 (SHEET 76 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
AUXILIARY FEEDWATER PUMPHOUSE											
VI-R-101[b]	1	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
VI-R-102[b]	1	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
VI-R-103	1	(c)	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
VI-R-104H	1	104	atm	1×10^3	120/40	11.7 psia	(e)	(e)	1×10^3	90	-
VI-R-105H	1	104	atm	1×10^3	120/40	11.7 psia	(e)	(e)	1×10^3	90	-
VI-R-106H[b]	1	104[d]	atm	1×10^3	122/40	11.7 psia	(e)	(e)	1×10^3	90	-
VI-R-101[b]	2	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
VI-R-102[b]	2	100	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
VI-R-103	2	(c)	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
VI-R-104H	2	104	atm	1×10^3	120/40	11.7 psia	(e)	(e)	1×10^3	90	-
VI-R-105H	2	104[c]	atm	1×10^3	120/40	11.7 psia	(e)	(e)	1×10^3	90	-
VI-R-106H[b]	2	104[d]	atm	1×10^3	120/40	11.7 psia	(e)	(e)	1×10^3	90	-
VI-R-201	1	(d)	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-
VI-R-201	2	(d)	atm	1×10^3	120/40	11.7 psia	-	-	1×10^3	90	-

AUXILIARY FEEDWATER PUMPHOUSE
(Sheet 1 of 2)

TABLE 3.11.B.1-1 (SHEET 77 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

    - = No high-energy line or safety-related equipment located in the room.

b. Indicates rooms that are served by Class 1E environmental support systems.

c. No ventilation provided.

d. Temperature in these rooms reaches 122°F when the turbine-driven AFW pump is in operation.

e. Postulated pipe breaks in the steam supply line to the turbine-driven pump would impact only safety-related equipment associated with the pump. These postulated breaks would disable the pump; therefore, this equipment would not be required to function and qualification of this equipment to the pipe break environment is not required.

AUXILIARY FEEDWATER PUMPHOUSE
(Sheet 2 of 2)

TABLE 3.11.B.1-1 (SHEET 78 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
DIESEL GENERATOR BUILDING - LEVEL 1											
IV-R-101[(b)]	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-102	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-103[(b)]	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-104	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
DIESEL GENERATOR BUILDING - LEVEL 2											
IV-R-201	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-202	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-203[(b)]	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-204[(b)]	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-205	1/com	120	atm	1×10^3	120/17	11.7 psia	-	-	1×10^3	90	-
IV-R-206	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-207	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-208[(b)]	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-209[(b)]	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-210	1/com	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
DIESEL GENERATOR BUILDING - LEVEL 1											
IV-R-101[(b)]	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-102	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-103[(b)]	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-104	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-

DIESEL GENERATOR BUILDING
(Sheet 1 of 3)

TABLE 3.11.B.1-1 (SHEET 79 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
DIESEL GENERATOR BUILDING - LEVEL 2											
IV-R-201	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-202	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-203[b]	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-204[b]	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-205	2	120	atm	1×10^3	120/17	11.7 psia	-	-	1×10^3	90	-
IV-R-206	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-207	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-208[b]	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-209[b]	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-210	2	120	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-

DIESEL GENERATOR BUILDING
(Sheet 2 of 3)

TABLE 3.11.B.1-1 (SHEET 80 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

\- = No high-energy line or safety-related equipment located in the room.

b. Indicates rooms that are served by Class 1E environmental support systems.

DIESEL GENERATOR BUILDING
(Sheet 3 of 3)

TABLE 3.11.B.1-1 (SHEET 81 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[(a)]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
DIESEL FUEL OIL STORAGE TANK VALVE BOX - LEVEL A											
IV-R-A01	1/com	amb	atm	1×10^3	94/30	11.7 psia	-	-	1×10^3	90	-
IV-R-A02	1/com	amb	atm	1×10^3	94/30	11.7 psia	-	-	1×10^3	90	-
IV-R-A03	1/com	amb	atm	1×10^3	94/30	11.7 psia	-	-	1×10^3	90	-
IV-R-A01	2	amb	atm	1×10^3	94/30	11.7 psia	-	-	1×10^3	90	-
IV-R-A02	2	amb	atm	1×10^3	94/30	11.7 psia	-	-	1×10^3	90	-
IV-R-A03	2	amb	atm	1×10^3	94/30	11.7 psia	-	-	1×10^3	90	-
DIESEL FUEL OIL STORAGE TANK VALVE BOX - LEVEL 1											
IV-R-101	1/com	amb	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-
IV-R-101	2	amb	atm	1×10^3	120/50	11.7 psia	-	-	1×10^3	90	-

DIESEL FUEL OIL STORAGE TANK VALVE BOX
(Sheet 1 of 2)

TABLE 3.11.B.1-1 (SHEET 82 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

\- = No high-energy line or safety-related equipment located in the room.

DIESEL FUEL OIL STORAGE TANK VALVE BOX
(Sheet 2 of 2)

TABLE 3.11.B.1-1 (SHEET 83 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
NSCW PUMPHOUSE - TRAIN A											
XII-R-101	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-102	1/com	amb[c]	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-103	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-104	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-105	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-106	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-107	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-108	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-109	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-110	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-111	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
NSCW PUMPHOUSE - TRAIN B											
XII-R-201	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-202	1/com	amb[c]	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-203	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-204	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-205	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-206	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-207	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-208	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-209	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-

NSCW PUMPHOUSE
(Sheet 1 of 4)

TABLE 3.11.B.1-1 (SHEET 84 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
NSCW PUMPHOUSE - TRAIN B (Cont.)											
XII-R-210	1/com	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-211	1/com	amb amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-301	2	amb[b][c]	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-302	2	amb amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-303	2	amb amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-304	2	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-305	2	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-306	2		atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-307	2	amb[b]	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-308	2	amb[c]	atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-309	2		atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-310	2		atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
XII-R-311	2		atm	1 X 10^3	104/5	11.7 psia	-	-	1 x 10^3	90	-
NSCW PUMPHOUSE - TRAIN A											
XII-R-401	2	amb[b]	atm	1 X 10^3	104/5	11.7 psia	-	-	1 X 10^3	90	-
XII-R-402	2	amb[c]	atm	1 X 10^3	104/5	11.7 psia	-	-	1 X 10^3	90	-
XII-R-403	2	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 X 10^3	90	-
XII-R-404	2	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 X 10^3	90	-
XII-R-405	2	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 X 10^3	90	-
XII-R-406	2	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 X 10^3	90	-
XII-R-407	2	amb	atm	1 X 10^3	104/5	11.7 psia	-	-	1 X 10^3	90	-

TABLE 3.11.B.1-1 (SHEET 85 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
NSCW PUMPHOUSE - TRAIN A (Cont.)											
XII-R-408	2	amb amb	atm	1×10^3	104/5	11.7 psia	-	-	1×10^3	90	-
XII-R-409	2	amb	atm	1×10^3	104/5	11.7 psia	-	-	1×10^3	90	-
XII-R-410	2	amb[b][c]	atm	1×10^3	104/5	11.7 psia	-	-	1×10^3	90	-
XII-R-411	2		atm	1×10^3	104/5	11.7 psia	-	-	1×10^3	90	-

TABLE 3.11.B.1-1 (SHEET 86 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

- = No high-energy line or safety-related equipment located in the room.

b. No ventilation provided.

c. Minimum room temperature is 40°F.

NSCW PUMPHOUSE
(Sheet 4 of 4)

TABLE 3.11.B.1-1 (SHEET 87 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[f]	Pressure[f]	Int Dose (rads)	Normal (%)	DBA (%)
OUTSIDE AREAS - MAIN STEAM TUNNELS											
III-1T1H	1/com	amb	atm	1×10^3	-	11.7 psia	370[d]	56 psia[e]	1×10^3	-	100
III-2T1H	2	amb	atm	1×10^3	-	11.7 psia	370[d]	56 psia[e]	1×10^3	-	100
OUTSIDE AREAS - NSCW PIPING TUNNELS											
III-1T2A	1/com	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-1T2B	1/com	amb[c]	tm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-2T2A	2	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-2T2B	2	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
OUTSIDE AREAS - DIESEL GENERATOR COOLING TUNNEL											
III-1T3A	1/com	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-1T3B	1/com	amb	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-2T3A	2	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-2T3B	2	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
OUTSIDE AREAS - DIESEL GENERATOR ELECTRICAL TUNNELS											
III-1T4A[b]	1/com	amb	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
III-1T4B[b]	1/com	amb	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
III-2T4A[b]	2	amb[c]	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
III-2T4B[b]	2	amb[c]	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
III-1T8A	1/com	amb	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-1T8BH	1/com	amb	atm	1×10^3	-	11.7 psia	Sheet 2	Sheet 6/A	1×10^3	-	100

OUTSIDE AREAS
(Sheet 1 of 4)

TABLE 3.11.B.1-1 (SHEET 88 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
OUTSIDE AREAS - NSCW TOWER CABLE TUNNELS											
III-1T5A[b]	1/com	amb	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
III-1T5B[b]	1/com	amb	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
III-2T5A[b]	2	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
III-2T5B[b]	2	amb[c]	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
OUTSIDE AREAS - AUXILIARY FEEDWATER TUNNELS											
III-1T6AH	1/com	amb	atm	1×10^3	-	11.7 psia	304[d]	31.0 psia[e]	1×10^3	-	-
III-1T6BH	1/com	amb	atm	1×10^3	-	11.7 psia	326[d]	37.0 psia[e]	1×10^3	-	-
III-2T6AH	2	amb	atm	1×10^3	-	11.7 psia	304[d]	31.0 psia[e]	1×10^3	-	-
III-2T6BH	2	amb	atm	1×10^3	-	11.7 psia	326[d]	37.0 psia[e]	1×10^3	-	-
OUTSIDE AREAS - RWST AND RMWST TUNNELS											
III-RST	1/com	amb	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
III-RST	2	amb	atm	1×10^3	104/40	11.7 psia	-	-	1×10^3	-	-
OUTSIDE AREAS - ELECTRIC STEAM BOILER TUNNEL											
III-ESBT	1/com	amb	atm	1×10^3	-	11.7 psia	-	-	1×10^3	-	-
OUTSIDE AREAS - REACTOR MAKEUP WATER STORAGE TANK											
III-R-101	1/com	amb[c]	atm	1×10^3	atm	11.7 psia	-	-	1×10^3	-	-
III-R-101	2	amb[c]	atm	1×10^3	atm	11.7 psia	-	-	1×10^3	-	-

TABLE 3.11.B.1-1 (SHEET 89 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
OUTSIDE AREAS - REFUELING WATER STORAGE TANK											
III-R-101	1/com	amb[c]	atm	1×10^3	atm	11.7 psia	-	-	1×10^3	-	-
III-R-101	2	amb[c]	atm	1×10^3	atm	11.7 psia	-	-	1×10^3	-	-

OUTSIDE AREAS
(Sheet 3 of 4)

TABLE 3.11.B.1-1 (SHEET 90 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

- = No high-energy line or safety-related equipment located in the room.

b. Indicates rooms that are served by Class 1E environmental support systems.

c. Minimum room temperature is 40°F.

d. Maximum room temperature.

e. Maximum room pressure.

f. Sheet numbers refer to figure 3.11.B.1-1. Alpha designators refer to specific curves on these sheets.

OUTSIDE AREAS
(Sheet 4 of 4)

TABLE 3.11.B.1-1 (SHEET 91 OF 92)
ENVIRONMENTAL CONDITIONS

Environmental Designator[a]	Unit	Normal			Abnormal/Test		DBA/Post-DBA			Relative Humidity (max)	
		Temp (°F)	Pressure	Int Dose (rads)	Temp (°F) (max/min)	Pressure	Temp (°F)[b]	Pressure	Int Dose (rads)	Normal (%)	DBA (%)
TURBINE BUILDING											
X-R-Level 1H	1/2	104	atm	1×10^3	104	11.7 psia	Sheet 18	17.7 psia	1×10^3	60	100
X-R-Level 2H	1/2	104	atm	1×10^3	104	11.7 psia	Sheet 18	17.7 psia	1×10^3	60	100

TURBINE BUILDING
(Sheet 1 of 2)

TABLE 3.11.B.1-1 (SHEET 92 OF 92)
ENVIRONMENTAL CONDITIONS

NOTES

a. H = Harsh environment due to existence of either of the following conditions:

- Temperature increases due to the pipe break.
- TID > 1 x 10^4 rad.

\- = No high-energy line or safety-related equipment located in the room.

b. Sheet numbers refer to figure 3.11.B.1-1. Alpha designators refer to specific curves on these sheets.

TURBINE BUILDING
(Sheet 2 of 2)

TABLE 3.11.B.3-1 (SHEET 1 OF 14)

CONFORMANCE TO THE REQUIREMENTS OF NUREG-0588

Requirements per NUREG-0588	VEGP Conformance
1. Establishment of the Qualification Parameters for Design Basis Events (DBEs)	
1.1 Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)	
(1) The time-dependent temperature and pressure established for the design of the containment structure and found acceptable by the staff may be used for environmental qualification of equipment.	(1) The time-dependent temperature and pressure profile used for qualifying VEGP equipment located inside containment was developed specifically for equipment qualification purposes. The methodology incorporated in the development of this profile is more conservative than those used in the structural design calculations.
(2) Acceptable methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are summarized below. Acceptable methods for calculating mass and energy release rates are summarized in Appendix A. Pressurized Water Reactors (PWRs) <u>Dry containment</u> - Calculate LOCA containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Standard Review Plan (SRP), Section 6.2.1.1.A, NUREG-75/087.	(2) The containment pressure and temperature values used to qualify the equipment were derived from the Bechtel standard computer program COPATTA. This program is used for VEGP and has been reviewed and found to be acceptable by the NRC for other Bechtel projects.
(3) Not applicable.	
(4) The test profiles included in Appendix A to Institute of Electrical and Electronics Engineers (IEEE) Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4) IEEE Std. 323-1974 profiles, when used, are verified to envelop VEGP specific requirements.

TABLE 3.11.B.3-1 (SHEET 2 OF 14)

Requirements per NUREG-0588	VEGP Conformance
1.2 Temperature and Pressure Conditions Inside Containment - Main Steam Line Break (MSLB)	
(1) The environmental parameters used for equipment qualification should be calculated with a plant-specific model reviewed and approved by the staff.	(1) Conform. Refer to 1.1(2) response
(2) Models that are acceptable for calculating containment parameters are listed in section 1.1(2).	(2) Conform. Refer to 1.1(2) response.
(3) Not applicable.	
(4) The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4) VEGP is comparing the plant requirement with the test results to verify the adequacy.
(5) Where qualification has been completed but only LOCA conditions were considered, it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:	(5) VEGP containment profile includes both LOCA/MSLB requirements for the qualification effort.
(a) Calculate the peak temperature envelope from an MSLB using a model based on the staff's approved assumptions defined in section 1.1(2).	Conforms when employed.
(b) Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in item 2 of Appendix B.	Conforms when employed.
(c) If the calculated surface temperature exceeds the qualification temperature, the staff requires requalification testing be performed with appropriate margins, or qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature.	

TABLE 3.11.B.3-1 (SHEET 3 OF 14)

Requirements per NUREG-0588	VEGP Conformance
1.3 Effects of Chemical Spray	
The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by the equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, item (e) for caustic spray solution guidelines.	Chemical spray is addressed by requiring the most severe concentration that will be present. Vendor qualification is required to meet or exceed the required chemical spray concentration.
1.4 Radiation Conditions Inside and Outside Containment	
The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following that which equipment must remain functional. It should be assumed that the DBA-related environmental conditions occur at the end of the equipment qualified life.	The normal total integrated dose and DBA dose are considered in the qualification of safety-related equipment. Operability after irradiation to the calculated dose plus margin is verified.
The sample calculations in Appendix D and the following positions provide an acceptable approach for establishing radiation limits for qualifications. Additional radiation margins identified in Section 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used.	
(1) The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products.	(1) Conform.

TABLE 3.11.B.3-1 (SHEET 4 OF 14)

Requirements per NUREG-0588	VEGP Conformance
For all other non-LOCA DBA conditions, a source term involving an instantaneous release from the fuel to the atmosphere of 10 percent of the noble gases (except Kr-85 for which a release of 30 percent should be assumed) and 10 percent of the iodine is acceptable.	
(2) The calculation of the radiation environment associated with DBAs should take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures.	(2) Conform. Calculations are on file.
(3) The initial distribution of activity within the containment should be based on a mechanistically rational assumption. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate.	(3) Conform.
(4) Effects of engineered safety features systems, such as containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 (NUREG-75/087) and the related sections referenced in the appendices to that section.	(4) Conform.
(5) Natural deposition (i.e., plateout) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50-percent instantaneous plateout of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment.	(5) Conform.

TABLE 3.11.B.3-1 (SHEET 5 OF 14)

Requirements per NUREG-0588	VEGP Conformance
(6) For unshielded equipment located in the containment, the gamma dose and dose rate should be equal to the dose and dose rate at the centerpoint of the containment plus the contribution from location dependent sources such as the sump water and plateout, unless it can be shown by analyses that location and shielding of the equipment reduces the dose and dose rate.	(6) Conform.
(7) For unshielded equipment, the beta doses at the surface of the equipment should be the sum of the airborne and plateout sources. The airborne beta dose should be taken as the beta dose circulated for a point at the containment center.	(7) Airborne beta doses are calculated for a hemispherical geometry.
(8) Shielded components need be qualified only to the gamma radiation levels required, provided an analysis or test shows that the sensitive portions of the component or equipment are not exposed to beta radiation or that the effects of beta radiation heating and ionization have no deleterious effects on component performance.	(8) Conform.
(9) Cables arranged in cable trays in the containment should be assumed to be exposed to half the beta radiation dose calculated for a point at the center of the containment plus the gamma ray dose calculated in accordance with Section 1.4(6). This reduction in beta dose is allowed because of the localized shielding by other cables plus the cable tray itself.	(9) Beta is accounted for by testing with a gamma dose to irradiate cables inside containment to the calculated total integrated dose plus margin.
(10) Paints and coatings should be assumed to be exposed to both beta and gamma rays in assessing their resistance to radiation. Plateout activity should be assumed to remain on the equipment surface unless the effects of the removal mechanisms, such as spray washoff or stream condensate flow, can be justified and quantified by analysis or experiment.	(10) Plateout with no washoff is included in the conservative qualification doses used for VEGP. Paints and coatings used for inside containment and decontaminable areas meet American National Standards Institute (ANSI) 101.2 and 101.4 requirements.

TABLE 3.11.B.3-1 (SHEET 6 OF 14)

Requirements per NUREG-0588	VEGP Conformance
(11) Components of the emergency core cooling system (ECCS) located outside containment (e.g., pumps, valves, seals, and electrical equipment) should be qualified to withstand the radiation equivalent to that penetrating the containment plus the exposure from the sump fluid using assumptions consistent with the requirements stated in Appendix K to 10 CFR 50.	(11) Conform. BOP equipment is qualified based on this requirement. NSSS equipment supplied by Westinghouse, including its qualification, is reviewed and verified to meet this requirement.
(12) Equipment that may be exposed to radiation doses below 10^4 rads should not be considered to be exempt from radiation qualification unless analysis supported by test data is provided to verify that these levels will not degrade the operability of the equipment below acceptable values.	(12) For low levels of radiation exposure, the literature search conducted by EPRI-NP-2129, November 1981, for organic material indicates the presence of a radiation threshold level of 10^4 rads for electrical equipment and 10^3 rads for electronic equipment. Equipment is evaluated for all levels of radiation exposure.
(13) The staff will accept a given component to be qualified provided it can be shown that the component has been qualified to integrated beta and gamma doses which are equal to or higher than those levels resulting from an analysis similar in nature and scope to that included in Appendix D (which uses the source term given in item (1) above) and that the component incorporates appropriate factors pertinent to the plant design and operating characteristics, as given in these general guidelines.	(13) Conform. Test reports are reviewed to assure that the radiation levels are met and no degradation will affect the performance of the equipment.
(14) When a conservative analysis has not been provided by the applicant for staff review, the staff will use the radiation environment guidelines contained in Appendix D, suitably corrected for differences in reactor power level, type, containment, size, and other appropriate factors.	(14) Conform. Analysis used was conservative.
1.5 Environmental Conditions for Outside Containment	
(1) Equipment located outside containment that could be subjected to high-energy pipe breaks should be qualified to the conditions resulting from the accident for the duration required.	(1) Conform. Equipment is qualified to the effects of high-energy line breaks. Postulated temperature and radiation levels have been calculated. The applicable environmental designator for each piece of safety-related equipment is used to qualify test equipment.

TABLE 3.11.B.3-1 (SHEET 7 OF 14)

Requirements per NUREG-0588	VEGP Conformance
The techniques to calculate the environmental parameters described in section 1.1 through 1.4 above should be applied.	
(2) Equipment located in general plant areas outside containment where equipment is not subjected to a DBA environment should be qualified to the normal and abnormal range of environmental conditions postulated to occur at the equipment location.	(2) Normal and abnormal conditions for all plant areas were established. These conditions were used to qualify the equipment located in each area. However, per the VEGP revised mild EQ position, environmental qualification shall not apply to safety-related equipment located in mild environments.
(3) Equipment not served by Class 1E environmental support systems, or served by Class 1E support systems that may be secured during plant operation or shutdown, should be qualified to the limiting environmental conditions that are postulated for the location, assuming a loss of the environmental support system.	(3) Equipment essential for safe shutdown or accident mitigation is qualified to the limiting environmental conditions that are postulated assuming loss of the environmental support systems.
2. Qualification Methods	
2.1 Selection of Methods	
(1) Qualification methods should conform to the requirements defined in IEEE Std. 323-1974.	(1) The VEGP program for safety-related equipment located in harsh environments conforms to IEEE Std. 323-1974. Subsections of IEEE Std. 323-1974 Section 6 regarding "Aging" or "Qualified Life" do not apply to safety-related equipment located in mild environments.
(2) The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached.	(2) Type testing in accordance to IEEE Std. 323-1974 is the preferred method of qualification. Vendors perform partial testing as a basis for other methodology used, such as analysis.

TABLE 3.11.B.3-1 (SHEET 8 OF 14)

Requirements per NUREG-0588	VEGP Conformance
(3) The environmental qualification of equipment exposed to DBA environments should conform to the following positions. The bases should be provided for the time interval required for operability of this equipment. The operability and failure criteria should be specified and the safety margins defined.	(3) The VEGP program addresses operability of the equipment, including margins necessary to meet specified criteria.
(a) Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.	(a) The VEGP program allows several qualification methods for safety-related equipment. However, testing is the preferred method. Mechanical equipment is qualified by analyses or a combination of test and analyses.
(b) Any equipment (safety-related or nonsafety-related) that need not function in order to mitigate any accident but that must not fail in a manner detrimental to plant safety should be qualified by test to demonstrate its capability to withstand any accident environment for the time during which it must not fail.	(b) See (a) above. Nonsafety-related equipment which must maintain its structural integrity in order not to fail in a manner detrimental to plant safety is designed to withstand the accident environment.
(c) Equipment that need not function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety need only be qualified for its nonaccident service environment.	(c) Conform.
Although actual type testing is preferred, other methods when justified may be found acceptable. The bases should be provided for concluding that such equipment is not required to function in order to mitigate any accident and that its failure in any mode in any accident environment is not detrimental to plant safety.	Testing is the preferred method. Other methods are acceptable on a case-by-case basis with justification.
(4) For environmental qualification of equipment subject to events other than a DBA which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history or any applicable combination thereof coupled with partial type test data may be found acceptable, subject to the applicability and detail of information provided.	(4) Conform.

TABLE 3.11.B.3-1 (SHEET 9 OF 14)

Requirements per NUREG-0588	VEGP Conformance
2.2 Qualification by Test	
(1) The failure criteria should be established prior to testing.	(1) Qualification test plans and procedures are reviewed to assure that acceptance criteria are established prior to that test.
(2) Test results should demonstrate that the equipment can perform its required function for all service conditions postulated (with margin) during its installed life.	(2) Conform.
(3) The items described in Section 6.3 of IEEE Std. 323-1974 supplemented by items (4) through (12) below constitute acceptable guidelines for establishing test procedures.	(3) Conform.
(4) When establishing the simulated environmental profile for qualifying equipment located inside containment, it is preferred that a single profile be used that envelops the environmental conditions resulting from any DBE during any mode of plant operation (e.g., a profile that envelops the conditions produced by the MSLB and LOCA).	(4) VEGP uses a single profile for the accident conditions for equipment inside containment. On a case-by-case basis, the equipment may be tested to another profile if and only if the effects on the equipment envelop the effects resulting from the VEGP profile.
(5) Equipment should be located above flood level or protected against submergence by locating the equipment in qualified watertight enclosures. Where equipment is located in watertight enclosures, qualification by test or analysis should be used to demonstrate the adequacy of such protection. Where equipment could be submerged, it should be identified and demonstrated to be qualified by test for the duration required.	(5) The location of safety-related components was compared to the calculated flood level in the surrounding area. Those items identified as being below the flood level were reviewed for acceptability. In the majority of cases, it was confirmed that the submerged component was not required to be functional for the event that could cause the flooding and that failure of the component would be acceptable. In the remaining cases, it was shown that the component would perform its required function (e.g., input to an alarm) prior to the time it would become submerged. In such cases, it was also confirmed that subsequent submergence would not result in unacceptable consequences. In those cases in which failure of the component would be unacceptable, corrective actions were made to preclude submergence.

TABLE 3.11.B.3-1 (SHEET 10 OF 14)

Requirements per NUREG-0588	VEGP Conformance
	The circuitry for submerged components was reviewed for electrical interfaces with other components. It was judged that the circuit breakers have been properly coordinated, so that submergence of the component would not result in the failure of other connected devices.
(6) The temperature to which equipment is qualified, when exposed to the simulated accident environment, should be defined by thermocouple readings on or as close as practical to the surface of the component being qualified.	(6) The test chamber temperature simulating the DBA conditions is measured and monitored on or as close as practical to the surface of the component being qualified.
(7) Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability.	(7) Baseline functional tests and performance tests after a test sequence and periodic functional checks are performed during the test sequence to demonstrate operability.
(8) Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the onsite spray systems actuate.	(8) The DBE is simulated by test, which includes the chemical spray concentration and pH requirements. Caustic spray effects during the DBE are also addressed by analysis with justification.
(9) The operability status of equipment should be monitored continuously during testing. For long-term testing, however, monitoring at discrete intervals should be justified if used.	(9) The VEGP program requires monitoring of the test equipment based on its function. Intermittent monitoring is acceptable depending upon the function being simulated.
(10) Expected extremes in power supply voltage range and frequency should be applied during simulated event environmental testing.	(10) The VEGP program addresses operability at environmental extremes, including voltage.
(11) Dust environments should be addressed when establishing qualification service conditions.	(11) Conform.
(12) Cobalt-60 is an acceptable gamma radiation source for environmental qualification.	(12) Cobalt 60 is the source used for qualification
2.3 Test Sequence	
(1) The test sequence should conform fully to the guidelines established in Section 6.3.2 of IEEE Std. 323-1974. The test procedures should ensure that the same piece of equipment is used throughout the test sequence to and that the test simulates as closely as practicable the postulated accident environment	(1) Qualification reports identify the test sequence used in accordance to IEEE-325-1974 or daughter standards. Vendors are required to provide justification if other test sequences are used. If separate specimens are used to eliminate parts of the sequence, justification is required.

TABLE 3.11.B.3-1 (SHEET 11 OF 14)

Requirements per NUREG-0588	VEGP Conformance
2.4 Other Qualification Methods	
Qualification by analysis or operating experience implemented, as described in IEEE Std. 323-1974 and other ancillary standards, may be found acceptable. The adequacy of these methods will be evaluated on the basis of the quality and detail of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified. It is required that, when these methods are employed, some partial type tests on vital components of the equipment be provided in support of these methods.	Conform.
3. Margins	
(1) Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any margins (conservatism) applied during the derivation of the specified plant parameters.	(1) Margins are in accordance to IEEE Std. 323-1974. Qualification reports are reviewed, and margins are quantified to assure that adequate conservatism exists.
(2) In lieu of other proposed margins that may be found acceptable, the suggested values indicated in IEEE Std. 823-1974, Section 6.3.1.5, should be used as a guide. (Note exceptions stated in Section 1.4.)	(2) Conform.
(3) When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for un-certainties such as production errors and errors associated with defining satisfactory performance (e.g., when only a small number of units are tested).	(3) This is not applicable to VEGP. Appendix C is for boiling water reactors and condenser containments.

TABLE 3.11.B.3-1 (SHEET 12 OF 14)

Requirements per NUREG-0588	VEGP Conformance
(4) Some equipment may be required by the design to only perform its safety function within a short time period into the event (i.e., within seconds or minutes); and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be required to perform a safety function but must not fail within a short time period into the event and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories is required to remain functional in the accident environment for a period of at least 1 h in excess of the time assumed in the accident analysis. For all other equipment (e.g., post-accident monitoring, recombiners, etc.), the 10-percent time margin identified in Section 6.3.1.5 of IEEE Std. 323-1974 may be used.	(4) Equipment is qualified for the period before, during, and after the simulated DBA. For fast-acting equipment, functional operability is monitored at various times during the test. Acceptability of failures after the initial function is complete is reviewed on the case-by-case basis depending on the specific safety-related function of the equipment and the level of margin attained.
4. Aging	
(1) Aging effects on all equipment, regardless of its location in the plant, should be considered and included in the qualification program.	(1) The effect of aging on all environmentally degradable parts of safety-related equipment located in harsh environments is considered. Subsections of paragraph 6.3.2 of IEEE Std. 323-1974 regarding "Aging" or "Qualified Life" do not apply to safety-related equipment located in mild environments.
(2) The degrading influences discussed in Section 6.3.3, 6.3.4, and 6.3.5 of IEEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of equipment should be considered and included as part of the aging progress.	(2) Mechanical and/or electrical cycling simulating the expected number of operational cycles for a reasonable period of component life is included in the program.
(3) Synergistic effects should be considered in the accelerated aging programs. Investigation should be performed to assure that no known synergistic effects have been identified on materials that are included in the materials that are included in the equipment being qualified. Where synergistic effects have been identified, they should be accounted for in the qualification programs. Refer to NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), Qualification Testing Evaluation Quarterly Reports, for additional information.	(3) The only known synergistic effect that has been identified is for the cable.

TABLE 3.11.B.3-1 (SHEET 13 OF 14)

Requirements per NUREG-0588	VEGP Conformance
(4) The Arrhenius methodology is considered an acceptable method of addressing accelerated aging. Other aging methods that can be supported by type tests will be evaluated on a case-by-case basis.	(4) Arrhenius methodology is an acceptable method. Other methods are accepted on a case-by-case basis.
(5) Known material phase changes and reactions should be defined to ensure that no known changes occur within the extrapolation limits.	(5) Extrapolations are within known material physical limits.
(6) The aging acceleration rate used during qualification testing and the basis upon which the rate was established should be described and justified.	(6) Conform.
(7) Periodic surveillance testing under normal service conditions is not considered an acceptable method for ongoing qualification, unless the plant design includes provisions for subjecting the equipment to the limiting service environment conditions (specified in Section 3(7) of IEEE Std. 279-1971) during such testing.	(7) Ongoing qualification is not used for equipment in a harsh environment.
(8) Effects of relative humidity need not be considered in the aging of electrical cable insulation.	(8) Conform.
(9) The qualified life of the equipment (and/or component as applicable) and the basis for its selection should be defined.	(9) Conform.
(10) Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and in other methods that may be reasonably assumed, coupled with good engineering judgment.	(10) Conform.

TABLE 3.11.B.3-1 (SHEET 14 OF 14)

Requirements per NUREG-0588	VEGP Conformance
5. Qualification Documentation	
(1) The staff endorses the requirements stated in IEEE Std. 323-1974 that, "The qualification documentation shall verify that each type of electrical equipment is qualified for its application and meets its specified performance requirements. The basis of qualification shall be explained to show the relationship of all facets of proof needed to support adequacy of the complete equipment. Data used to demonstrate the qualification of the equipment shall be pertinent to the application and organized in an auditable form."	(1) VEGP documentation of safety-related equipment will be in an auditable file.
(2) The guidelines for documentation in IEEE Std. 323-1974 when fully implemented are acceptable. The documentation should include sufficient information to address the required information identified in Appendix E. A certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program.	(2) Conform.

TABLE 3.11.N.3-1 (SHEET 1 OF 16)

COMPARISON OF NSSS ENVIRONMENTAL QUALIFICATION PROGRAM TO NUREG-0588

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
1.0	Establishment of the Qualification Parameters for Design Basis Events	
1.1	Temperature and Pressure Conditions Inside Containment - Loss of Coolant	
1.1.1	The time-dependent temperature and pressure established for the design of the containment structure and found acceptable by the staff may be used for environmental qualification of equipment.	The containment structural design has been based on the results of an analysis performed by Westinghouse employing methodology described below. The results of this analysis are reported in section 6.2.
1.1.2	Acceptable methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are summarized below. Acceptable methods for calculating mass and energy release rates are summarized in Appendix A.	Westinghouse employs the methodology described in WCAP-8312A for calculating the LOCA mass and energy release. Appendix A to NUREG-0588 indicates that this methodology is acceptable to the staff.
	Pressurized Water Reactors (PWRs)	
	Dry containment - Calculate LOCA containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Standard Review Plan (SRP), Section 6.2.1.1.A, NUREG-75/087.	Westinghouse employs the COCO model described in WCAP-8327 to establish the containment pressure and temperature time-dependent variations following LOCA.
	Ice condenser containment – Calculate LOCA containment environment using LOTIC or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.B, NUREG-75/087.	NA
1.1.3	In lieu of using the plant-specific containment temperature and pressure design profiles for boiling water reactor (BWR) and ice condenser type plants, the generic envelope shown in Appendix C may be used for qualification testing.	NA

TABLE 3.11.N.3-1 (SHEET 2 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
1.1.4	The test profiles include in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	Westinghouse does not employ the test profiles included in Appendix A to IEEE Std. 323-1974.
1.2	Temperature and Pressure Conditions Inside Containment - Main Steam Line Break (MSLB)	
1.2.1	The environmental parameters used for equipment qualification should be calculated with a plant-specific model reviewed and approved by the staff.	Westinghouse employs the methodology described in WCAP-8822 for calculating the mass and energy release following an MSLB. Westinghouse has completed the mass and energy release calculations for VEGP assuming entrainment.
1.2.2	Models that are acceptable for calculating containment parameters are listed in Section 1.1(2).	Westinghouse employs the COCO model described in WCAP-8327 and WCAP-8936 to establish the containment pressure and temperature time-dependent variations following an MSLB. Westinghouse submitted these reports for generic staff review. The staff has accepted these methods on specific applications. Westinghouse will continue to use these models for containment analysis.
1.2.3	In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser plants, the generic envelope shown in Appendix C may be used.	NA

TABLE 3.11.N.3-1 (SHEET 3 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
1.2.4	The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles, unless plant-specific analysis is provided to verify the adequacy of those profiles.	Westinghouse does not employ the test profiles included in Appendix A to IEEE Std. 323-1974.
1.2.5	Where qualification has been completed but only LOCA conditions were considered, then it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:	
1.2.5.a	Calculate the peak temperature from an MSLB using a model based on the staff's approved assumptions defined in Section 1.1.(2).	
1.2.5.b	Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in item 2 of Appendix B.	Westinghouse has established a single target qualification envelope for equipment that is required to perform safety functions in a hostile environment resulting from a primary or secondary side break. This envelope has been selected to envelop, with margin, the anticipated range of LOCA and MSLB transients. As a consequence, there is no intent in the design of the program described in WCAP-8587 to employ this type of analysis to justify qualification for equipment required to perform a safety function in a hostile environment.
1.2.5.c	If the calculated surface temperature exceeds the qualification temperature, the staff requires that (a) requalification testing be performed with appropriate margins, or (b) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature. For the plants that are currently being reviewed or will be submitted for an operating license review within 6 months from issue date of this report, compliance with items (a) or (b) above may represent a substantial impact. For those plants, the staff will consider additional information submitted by the applicant to demonstrate that the equipment can maintain its functional operability if its surface temperature rises to the value calculated.	

TABLE 3.11.N.3-1 (SHEET 4 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
1.3	Effects of Chemical Spray	The maximum concentration of boron employed for containment spray is 2500 ppm, and the maximum permitted pH of the initial spray solution is 10.5. For qualification testing, Westinghouse specifies a chemical spray of 2500 ppm boron buffered with 0.9 wt% dissolved sodium hydroxide to maintain a pH of approximately 10.7, starting at time zero and terminating after 24 h. This spray concentration results in an increase in alkalinity of at least 10 percent compared to the maximum concentration defined by the specification and significantly exceeds the range of sump pH values permitted long-term by the Technical Specifications for VEGP.
1.3.1	The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system.	
1.3.2	If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, item (e), for caustic spray solution guidelines.	In the Westinghouse-designed containment spray system, no single failure can be postulated which would cause the specified test pH of 10.7 to be nonconservative for simulating long-term spray operations post-accident.

TABLE 3.11.N.3-1 (SHEET 5 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
1.4	Radiation Conditions Inside and Outside Containment Subsections 1, 2, 3, 4, 5, 6, 7, 9, 10, 11, 13, 14.	See table (B section).
1.4.8	Shielded components need be qualified only to the gamma radiation levels required, provided an analysis or test shows that the sensitive portions of the component or equipment are not exposed to beta radiation or that the effects of beta radiation heating and ionization have no deleterious effects on component performance.	Any potential concern resulting from exposure to beta radiation is limited to equipment located inside containment that is required to mitigate a high-energy line break (HELB) inside containment and only to any organic materials exposed to the in-containment environment. Qualification programs address this issue by analysis, test, or combination of both.
1.4.12	Equipment that may be exposed to radiation doses below 10^4 rads should not be considered to be exempt from radiation qualification unless analysis supported by test data is provided to verify that these levels will not degrade the operability of the equipment below acceptable values.	For safety-related electrical equipment that is not required to operate in a HELB environment and for which the anticipated qualified life integrated radiation dose is 10^4 rads or less, Westinghouse does not include a radiation aging simulation as part of any qualification testing. For such equipment and components, Appendix C to WCAP-8587 demonstrates, based on available test information on materials and components that up to approximately 10^4 rads there is no detectable effect on the structural characteristics of materials and components that would affect the capability of equipment to perform during a seismic event.
1.5	Environmental Conditions for Outside Containment Subsections 1, 2, 3	See table (B section).

TABLE 3.11.N.3-1 (SHEET 6 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
2.0	Qualification Methods	
2.1	Selection of Methods	
2.1.1	Qualification methods should conform to the requirements defined in IEEE Std. 323-1974.	The methodology employed by Westinghouse to qualify safety-related electrical equipment is described in WCAP-8587 and Std. 323-1974.
2.1.2	The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached.	Westinghouse qualifies equipment that is required to perform a safety function in a HELB environment by test.
2.1.3	The environmental qualification of equipment exposed to DBA environments should conform to the following positions:	The required operability time is discussed in paragraph 3.11.N.1.3.
	The basis should be provided for the time interval required for operability of this equipment.	
	The operability and failure criteria should be specified and the safety margins defined.	The primary purpose of equipment qualification is to reduce the potential for common-mode failures due to postulated environmental conditions. A test unit will therefore be considered to have failed the test if the performance requirements identified in WCAP-8587 cannot be met, unless an investigation can establish that the failure mechanism is not a common-mode origin or that plant specific analyses can demonstrate that the reduced capability is acceptable.

TABLE 3.11.N.3-1 (SHEET 7 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
		Margins are discussed under item 3.
2.1.3.a	Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.	When Westinghouse employs testing to qualify electrical equipment that must function in order to mitigate any accident, the acceptance criterion for the test is that the safety-related function must be demonstrated for the specified duration while the equipment is exposed to the simulated environmental conditions resulting from the accident.
2.1.3.b	Any equipment (safety-related or nonsafety-related) that need not function in order to mitigate any accident, but that must not fail in a manner detrimental to plant safety should be qualified by test to demonstrate its capability to withstand any accident environment for the time during which it must not fail.	This is discussed in subsections 15.1.4, 15.1.5, 15.2.7, and 15.4.1.
2.1.3.c	Equipment that need not function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety need only be qualified for its nonaccident service environment. Although actual type testing is preferred, other methods when justified may be found acceptable. The bases should be provided for concluding that such equipment is not required to function in order to mitigate any accident, and that its failure in any mode in any accident environment is not detrimental to plant safety.	Where Westinghouse supplies an item of safety-related electrical equipment that is located in an area where it can experience the environment resulting from a HELB but is not required to perform any safety function, Westinghouse has verified that any consequential failure of such equipment, due to the adverse environment, does not prejudice the safety-related functions of other equipment claimed in the accident analysis.

TABLE 3.11.N.3-1 (SHEET 8 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
2.1.4	For environmental qualification of equipment subject to events other than a DBA which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history, or any applicable combination thereof, coupled with partial type test data may be found acceptable, subject to the applicability and detail of information provided.	Potential abnormal environments are associated with recirculating operations post-accident and with a loss of non-Class 1E HVAC systems. Recirculation operations: Equipment supplied by Westinghouse that is required to perform a post-accident safety function as part of the recirculation loop is qualified by testing for anticipated increased radiation levels resulting from recirculation of radioactive fluid. Loss of non-Class 1E HVAC. For equipment located outside containments that is not required to be qualified for the effects of a HELB environment, Westinghouse has completed performance tests in accordance with Section 6.3.2(3) of IEEE 323-1974 at the anticipated environmental extremes, including loss of non-Class 1E HVAC where applicable.
2.2	Qualification by Test	
2.2.1	The failure criteria should be established prior to testing.	In Supplement 1 to WCAP-8587 Westinghouse has identified, for each item of safety-related equipment, the safety functions to be performed for all normal, abnormal, or accident conditions during or after which the equipment is required to provide a protective function. As stated in the response to staff position 2.1.(3), a test unit will be considered to have failed the test if the safety-related performance requirements cannot be met, unless an investigation can establish that the failure mechanism is not of common-mode origin or that plant specific analyses can demonstrate that the reduced capability is acceptable.
2.2.2	Test results should demonstrate that the equipment can perform its required function for all service conditions postulated (with margin) during its installed life.	Westinghouse qualification programs are designed to demonstrate that the equipment can perform its required safety function(s) for all postulated service conditions.

TABLE 3.11.N.3-1 (SHEET 9 OF 16)

NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
	Via the treatment of aging, as described in the response to staff position 4.0, a generic qualified life is established by Westinghouse for each item of equipment. This generic qualified life may be extended on a plant-specific basis by employing less conservative plant-specific assumptions concerning the plant normal operating environmental conditions. The qualified life established for the equipment on a specific plant will ultimately define the permitted installed life of the equipment. The subject of margin is discussed in the response to staff position 3.0.
2.2.3 The items described in Section 6.3 of IEEE Std. 323-1974 supplemented by items (4) through (12) below constitute acceptable guidelines for establishing test procedures.	When testing is the selected methodology for qualifying equipment, Westinghouse has established the test program in conformance with Section 6.3 of IEEE 323-1974.
2.2.4 When establishing the simulated environmental profile for qualifying equipment located inside containment, it is preferred that a single profile be used that envelops the environmental conditions resulting from any design basis event during any mode of plant operation (e.g., a profile that envelops the conditions produced by the MSLB and LOCAs).	Westinghouse prefers to use a single profile, enveloping both MSLB and LOCA, for qualification of equipment located inside containment which is required to perform a safety function to mitigate both HELBs. This approach is optimum in terms of schedule, manpower, and materials. However, there is no technical justification for making this a requirement for all equipment inside containment.
2.2.5 Equipment should be located above flood level or protected against submergence by locating the equipment in qualified watertight enclosures. Where equipment is located in watertight enclosures, qualification by test or analysis should be used to demonstrate the adequacy of such protection. Where equipment could be submerged, it should be identified and demonstrated to be qualified by test for the duration required.	NA

TABLE 3.11.N.3-1 (SHEET 10 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
2.2.6	The temperature to which equipment is qualified, when exposed to the simulated accident environment, should be defined by thermocouple readings on or as close as practical to the surface of the component being qualified.	In performing qualification tests for HELB environments, Westinghouse requires that the external environment temperature be measured as close to the test unit surface as practicable.
2.2.7	Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability.	Where the safety-related function of the equipment requires operation during the HELB, Westinghouse verifies the equipment performance before, during, and after the simulated event and verifies that the safety-related function is demonstrated for the specified required duration of the function.
2.2.8	Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the onsite spray systems actuate.	The response to item 1.3(1) is applicable for equipment located inside containment and qualified by test to operate in a HELB environment.
2.2.9	The operability status of equipment should be monitored continuously during testing. For long-term testing, however, monitoring at discrete intervals should be justified if used.	The response to item 2.2(7) is applicable.
2.2.10	Expected extremes in power supply voltage range and frequency should be applied during simulated event environmental testing.	The Class 1E instrumentation and control equipment qualified in Westinghouse scope is supplied by a guaranteed stabilized power supply. As a consequence, the range of electrical parameters employed is extremely small, and variations within the permitted range are considered insignificant. Westinghouse also addresses extremes in power supply voltage and frequency as required in the design and sizing of equipment to provide margins for all conditions.

TABLE 3.11.N.3-1 (SHEET 11 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
2.2.11	Dust environments should be addressed when establishing qualification service conditions.	Dust environments have not been established as a required and/or significant qualification parameters.
2.2.12	Cobalt-60 is an acceptable gamma radiation source for environmental qualification.	Westinghouse employs Cobalt-60 sources to simulate the effects of gamma and, in some cases, beta radiation for equipment qualified by test to operate in a HELB environment.
2.3	Test Sequence	
2.3.1	The test sequence should conform fully to the guidelines established in Section 6.3.2 of IEEE Std. 323-1974. The test procedures should ensure that the same piece of equipment is used throughout the test sequence and that the test simulates as closely as practicable the postulated accident environment.	Section 6.3.2 of IEEE 323-1974 neither mandates a single unique test sequence nor requires that the same piece of equipment be used through the test sequence: - The standard identifies a test sequence that is thought to be the most conservative for most equipment; however, alternative sequences are clearly permitted with justification. - Section 6.3.2(3) specifically permits the performance test at extremes of the normal ambient to be performed on another, essentially similar, piece of equipment.

TABLE 3.11.N.3-1 (SHEET 12 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
2.4	Other Qualification Methods	
	Qualification by analysis or operating experience implemented, as described in IEEE Std. 323-1974 and other ancillary standards, may be found acceptable. The adequacy of these methods will be evaluated on the basis of the quality and detail of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified. It is required that, when these methods are employed, some partial type tests on vital components of the equipment be provided in support of these methods.	Westinghouse does not employ analysis or operating experience as a prime method to establish the environmental qualification of safety-related electrical equipment. However, analysis or operating experience may be included in support of environmental qualification by test.
3.0	Margins	
3.1.1	Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any margins (conservatism) applied during the derivation of the specified plant parameters.	In general, Westinghouse has applied margin with respect to the design postulated accident conditions defined for each item of equipment in Section 1 of the corresponding EQDP. However, since Westinghouse has conducted generic qualification tests, the design environments selected by Westinghouse will already contain significant margin with respect to most plant-specific environmental parameters.
3.1.2	In lieu of other proposed margins that may be found acceptable, the suggested values indicated in IEEE Std. 323-1974, Section 6.3.1.5, should be used as a guide. (Note exceptions stated in Section 1.4.)	Westinghouse has applied specific margin to design parameters, in deriving type test parameters, as described in Section 7.1 of WCAP-8587.(1) This method of applying margin is in accordance with Section 6.3.1.5 of IEEE 323-1974, which recognizes increasing test levels, number of test cycles, and test duration as methods of incorporating margin in the test plan.

TABLE 3.11.N.3-1 (SHEET 13 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
3.1.3	When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for uncertainties such as production errors and errors associated with defining satisfactory performance (e.g., when only a small number of units are tested).	NA
3.1.4.a	Some equipment may be required by the design to only perform its safety function within a short time period into the event (i.e., within seconds or minutes); and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be required to perform a safety function but must not fail within a short time period into the event, and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories is required to remain functional in the accident environment for a period of at least 1 h in excess of the time assumed in the accident analysis.	Some equipment (e.g., transmitters) was not specified to maintain trip function accuracy requirements for longer than 5 min post-accident. The operability time was conservatively established based on the reactor trip ESF function performed by each equipment item considering what consequences failure of the device would have on the operator and the mitigation of the event. Margins for trip function requirements are contained in the HELB envelopes which encompass a full spectrum of break sizes and also justified by the fact that the signal generated by the sensor is "locked-in" by the protection system and will not reset should the sensor fail after the designated trip time requirement. Most of the equipment was also specified and qualified for much longer post-accident monitoring function times to slightly reduced accuracy requirements.
3.1.4b	For all other equipment (e.g., post-accident monitoring, recombiners, etc.), the 10-percent time margin identified in Section 6.3.1.5 of IEEE Std. 323-1974 may be used.	In qualifying equipment, Westinghouse did not always include any systematic margin on the specified duration of the safety function. Margin is included in radiation dose calculations and by a conservative selection of parameters for aging simulation.

TABLE 3.11.N.3-1 (SHEET 14 OF 16)

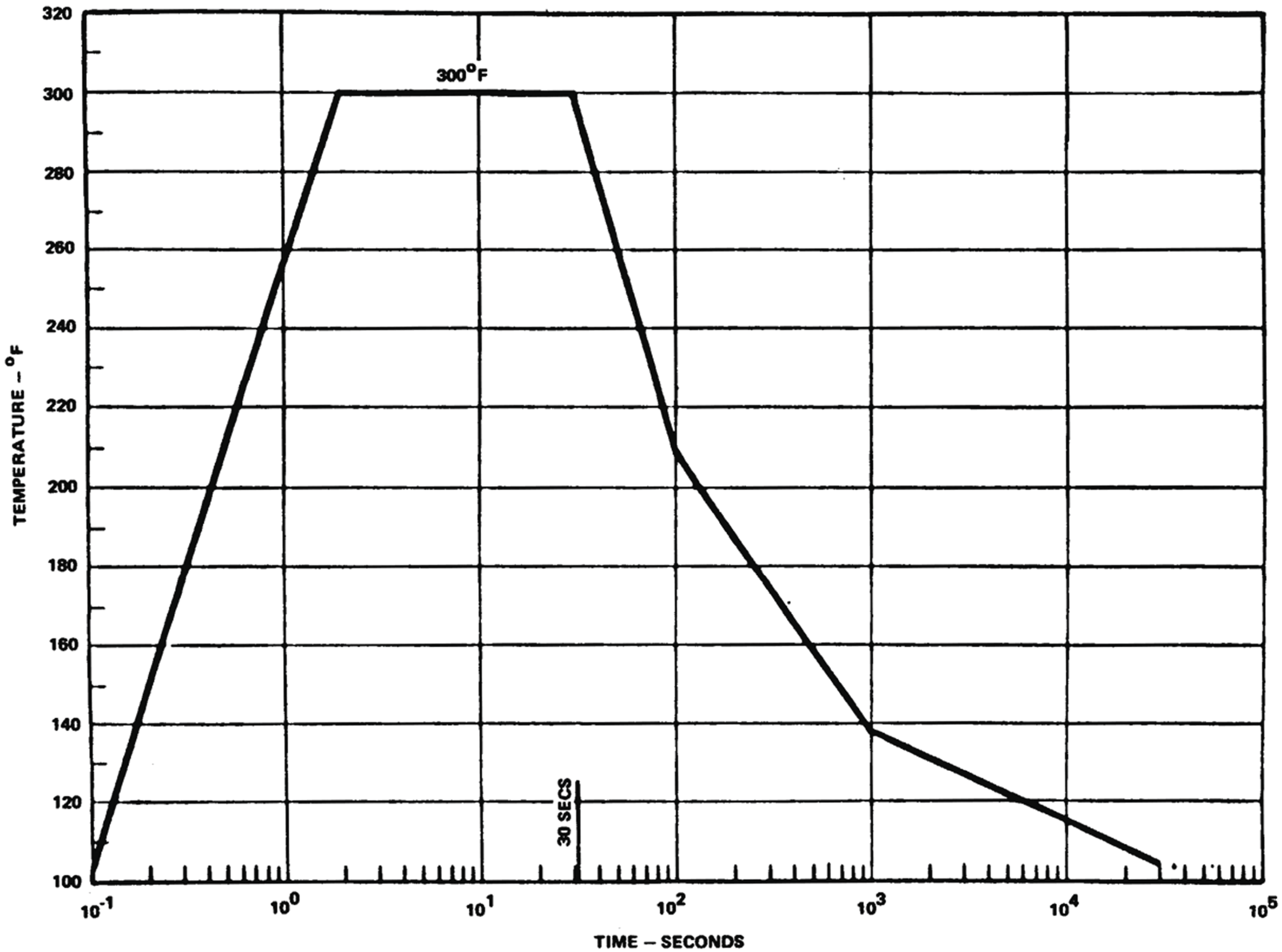
	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
4.0	Aging	
4.1	Aging effects on all equipment, regardless of its location in the plant, should be considered and included in the qualification program.	Westinghouse considers and includes in the equipment qualification programs the effects of aging, as applicable, irrespective of the location of the equipment in the plant.
4.2	The degrading influences discussed in Sections 6.3.3, 6.3.4, and 6.3.5 of IEEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of included as part of the aging programs.	Appendix B to WCAP-8587 describes the aging mechanisms to be considered and the methodology to be employed in simulating the aging effects. The mechanisms to be considered in the Westinghouse program in Sections 6.3.3, 6.3.4, and 6.3.5 of IEEE 323-1974 including electrical and mechanical stresses associated with cyclic operation of equipment.
4.3	Synergistic effects should be considered in the accelerated aging programs. Investigation should be performed to assure that no known synergistic effects have been identified on materials that are included in the equipment being qualified. Where synergistic effects have been identified, they should be accounted for in the qualification programs. Refer to NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), Qualification Testing Evaluation Quarterly Reports, for additional information.	Westinghouse has not identified any synergistic effects involving the materials and components comprising the equipment to be qualified in this program.
4.4	The Arrhenius methodology is considered an acceptable method of addressing accelerated aging. Other aging methods that can be supported by type tests will be evaluated on a case-by-case basis.	The Arrhenius equation is solely employed by Westinghouse in calculating appropriate temperature and duration parameters to accelerate the effects of thermal aging.
4.5	Known material phase changes and reactions should be defined to ensure that no known changes occur within the extrapolation limits.	No such mechanisms are known.

TABLE 3.11.N.3-1 (SHEET 15 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
4.6	The aging acceleration rate used during qualification testing and the basis upon which the rate was established should be described and justified.	Appendix D to WCAP-8587 justifies the acceleration parameters and rates to be employed for the aging program described in Appendix B to WCAP-8587.
4.7	Periodic surveillance testing under normal service conditions is not considered an acceptable method for ongoing qualification unless the plant design includes provisions for subjecting the equipment to the limiting service environment conditions (specified in Section 3(7) of IEEE Std. 279-1971) during such testing.	Westinghouse does not employ periodic surveillance testing or any form of ongoing qualification in the program described in WCAP-8587.
4.8	Effects of relative humidity need not be considered in the aging of electrical cable insulation.	For equipment subjected to HELB environments, the aging effects due to humidity during normal operation are judged to be insignificant compared to the effects of the high-temperature steam accident simulation; therefore, no additional humidity aging simulation is required. For equipment not subjected to HELB environments, the use of materials and components known to be significantly affected by humidity are avoided.
4.9	The qualified life of the equipment (and/or component as applicable) and the basis for its selection should be defined.	The EQDP identifies the demonstrated qualified life and justifies the value selected based on the aging mechanisms that have been simulated.
4.10	Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and in other methods that may be reasonably assumed, coupled with good engineering judgment.	Westinghouse conforms to this position in establishing the demonstrated qualified life of WRD-supplied equipment.

TABLE 3.11.N.3-1 (SHEET 16 OF 16)

	NUREG-0588 CATEGORY 1 POSITION	NSSS PROGRAM
5.0	Qualification Documentation	
5.1	The staff endorses the requirements stated in IEEE Std. 323-1974: "The qualification documentation shall verify that each type of electrical equipment is qualified for its application and meets its specified performance requirements. The basis of qualification shall be explained to show the relationship of all facets of proof needed to support adequacy of the complete equipment. Data used to demonstrate the qualification of the equipment shall be pertinent to the application and organized in an auditable form."	The documentation system established for a specific plant application should be described by the applicant. In support of the documentation effort Westinghouse has supplied to the utilities: - Identification of the limits to any environmental qualification. - Copies of all referenced Westinghouse topical reports. In addition, Westinghouse will maintain the available raw test data which supports the referenced qualification tests on equipment subject to HELB and available information concerning the performance testing of production units.
5.2	The guidelines for documentation in IEEE Std. 323-1974 are acceptable. The documentation should include sufficient information to address the required information identified in Appendix E. A certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program.	The Westinghouse qualification test reports meet the requirements of Section 5 to IEEE 323-1974 by providing the following essential information as a minimum: - Safety-related functional requirements to be demonstrated. - Range of applicable environmental parameters to be considered. - Identification of the test unit. - Description of the test facility and monitoring instrumentation. - Description of test unit mounting and interfaces. - Summary of the test procedures. - Summary of the test results.



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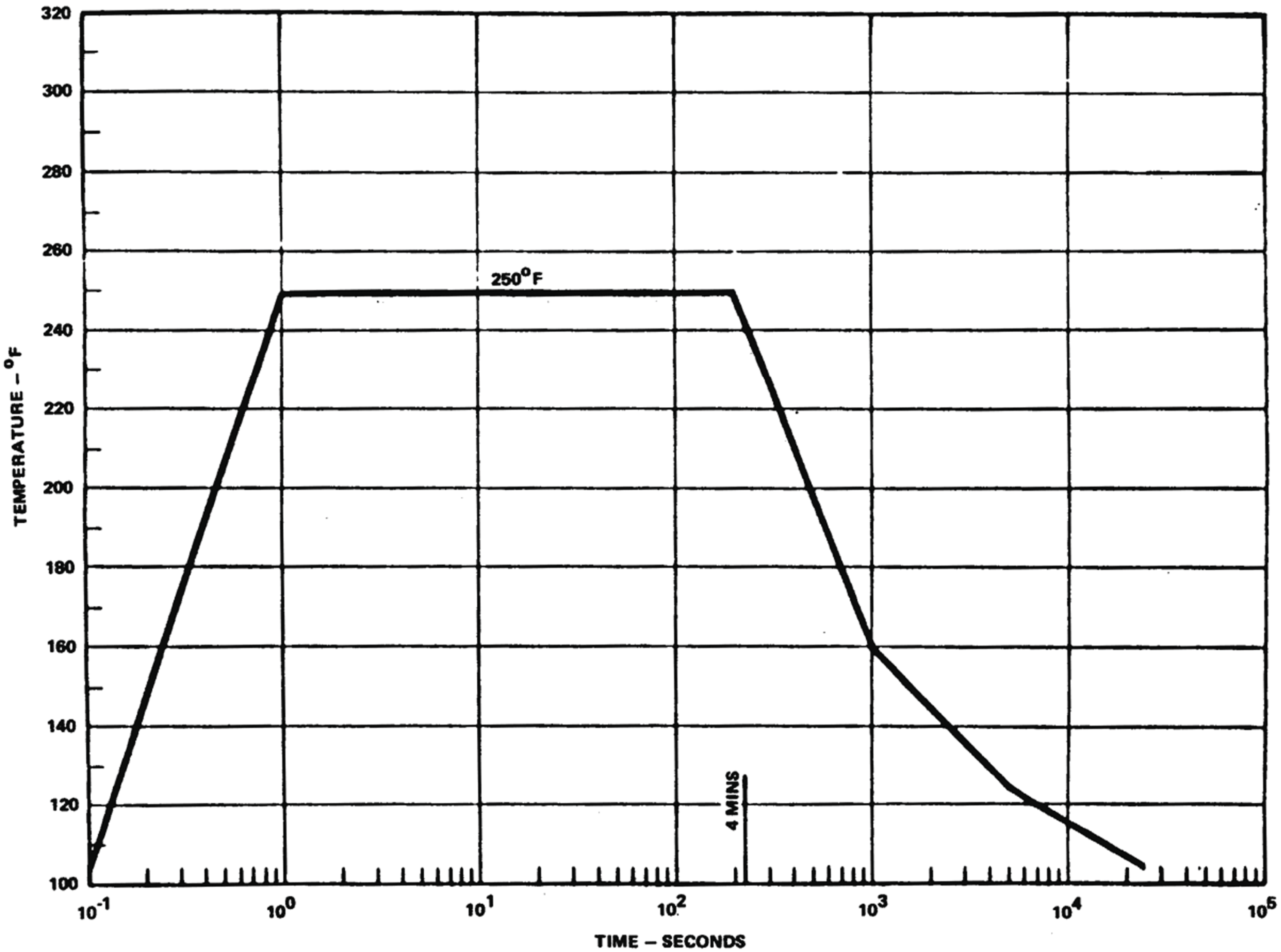
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEMPERATURE CONDITIONS FOR EQUIPMENT QUALIFICATION IN THE AUXILIARY BUILDING

FIGURE 3.11.B.1-1 (SHEET 1 OF 18)



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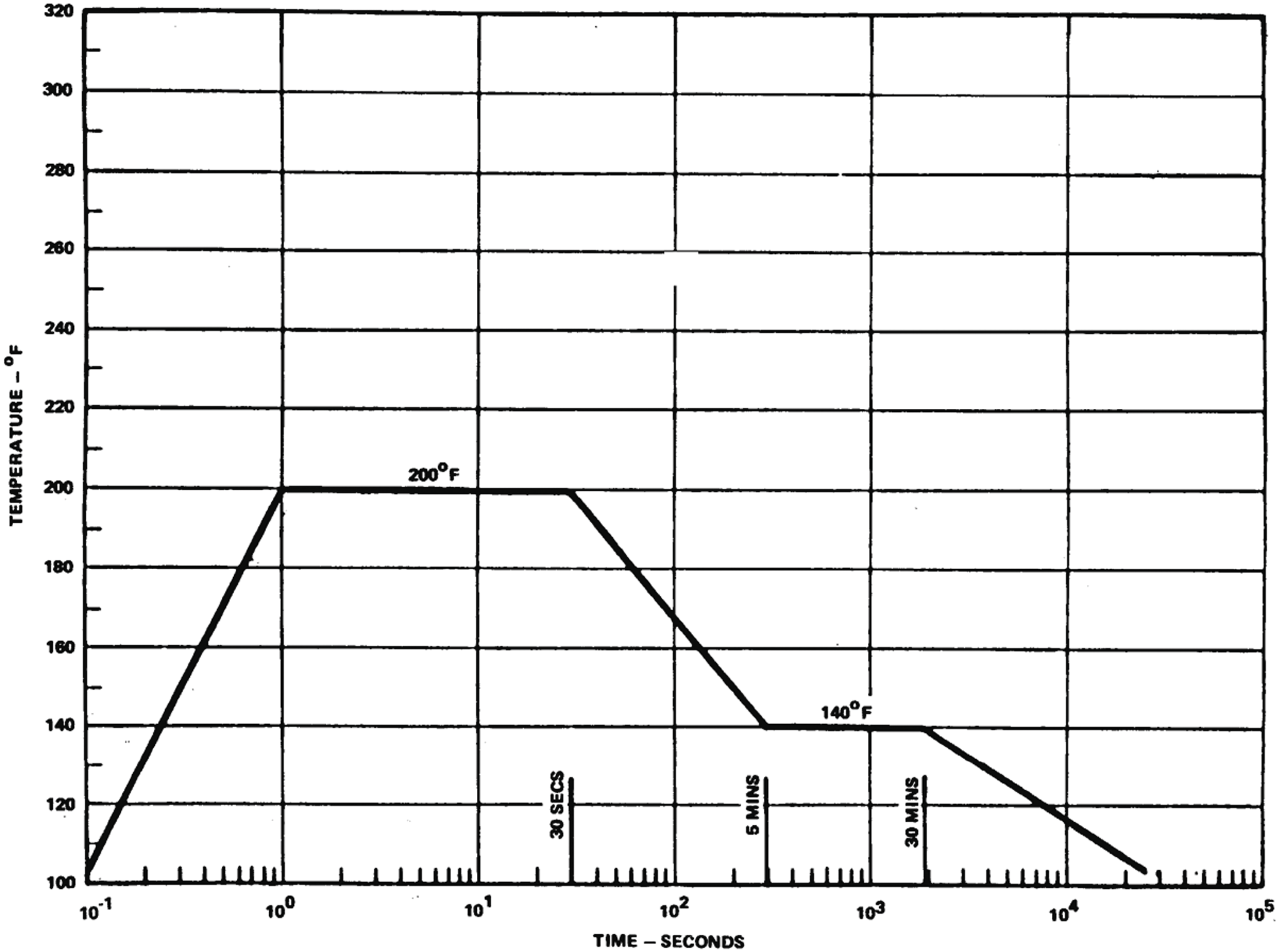
TEMPERATURE CONDITIONS FOR
EQUIPMENT QUALIFICATION IN
THE AUXILIARY BUILDING

FIGURE 3.11.B.1–1 (SHEET 2 OF 18)



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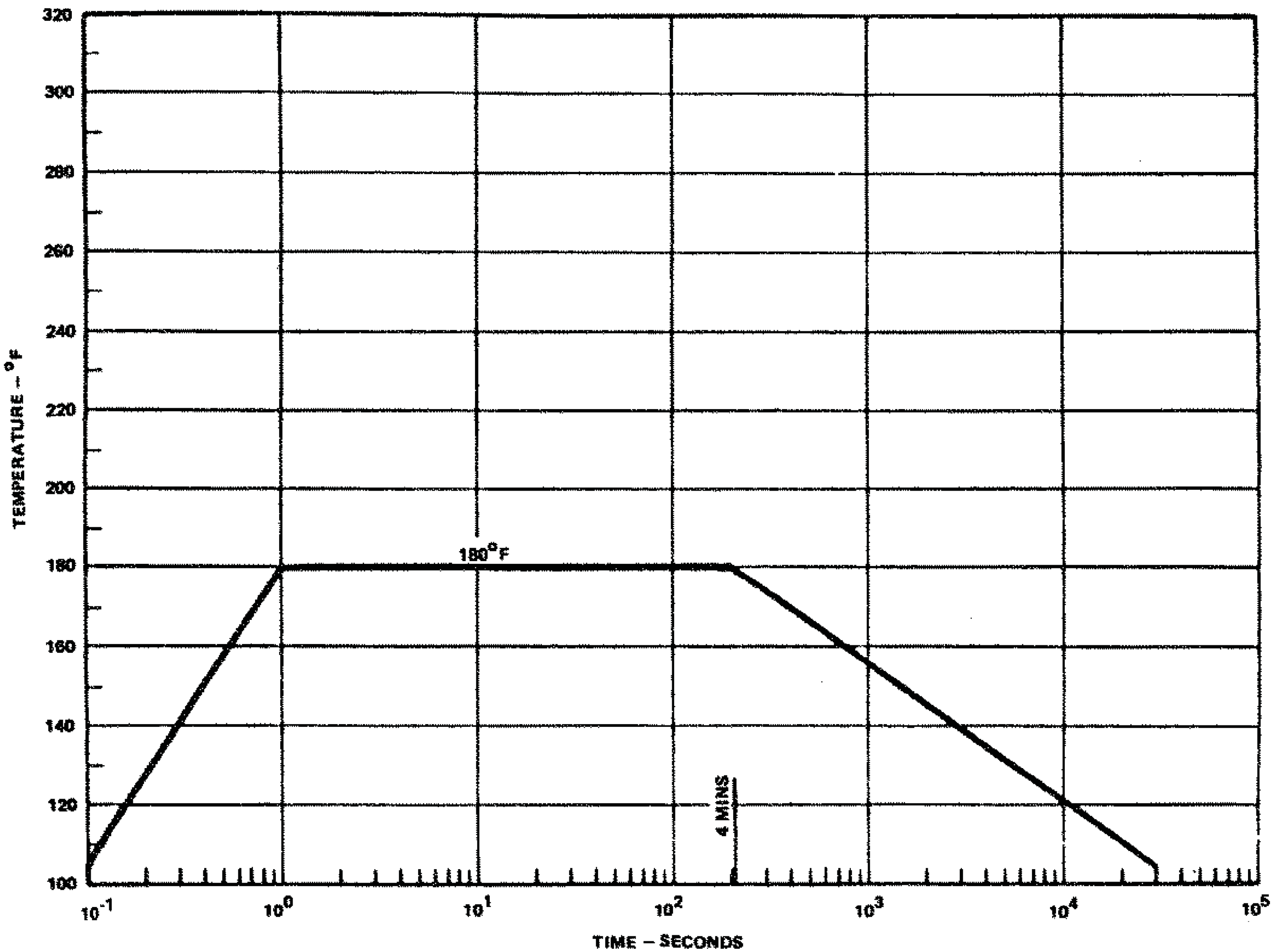
TEMPERATURE CONDITIONS FOR
EQUIPMENT QUALIFICATION IN
THE AUXILIARY BUILDING

FIGURE 3.11.B.1–1 (SHEET 3 OF 18)



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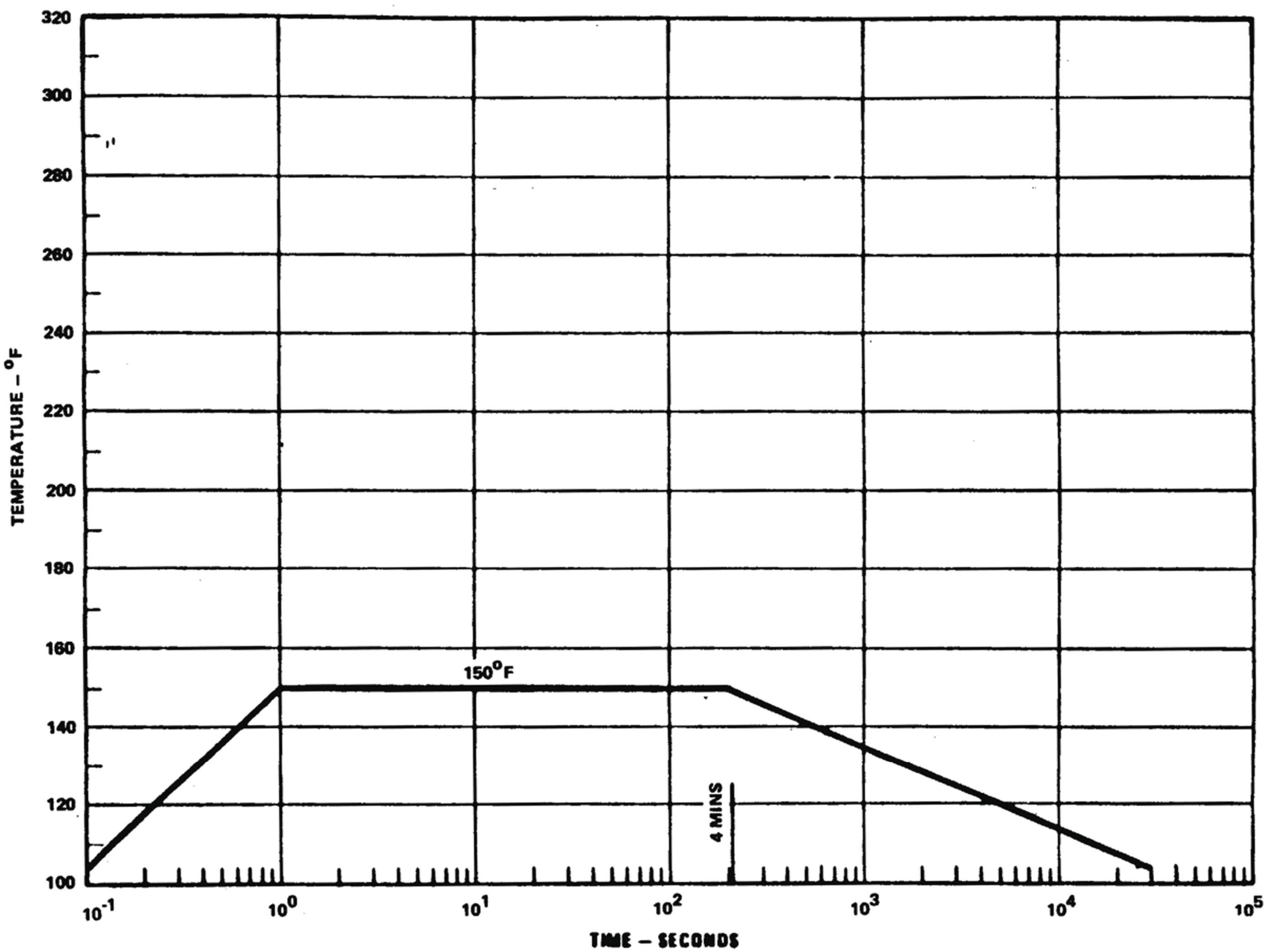
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TEMPERATURE CONDITIONS FOR EQUIPMENT QUALIFICATION IN THE AUXILIARY BUILDING
		FIGURE 3.11.B.1–1 (SHEET 4 OF 18)



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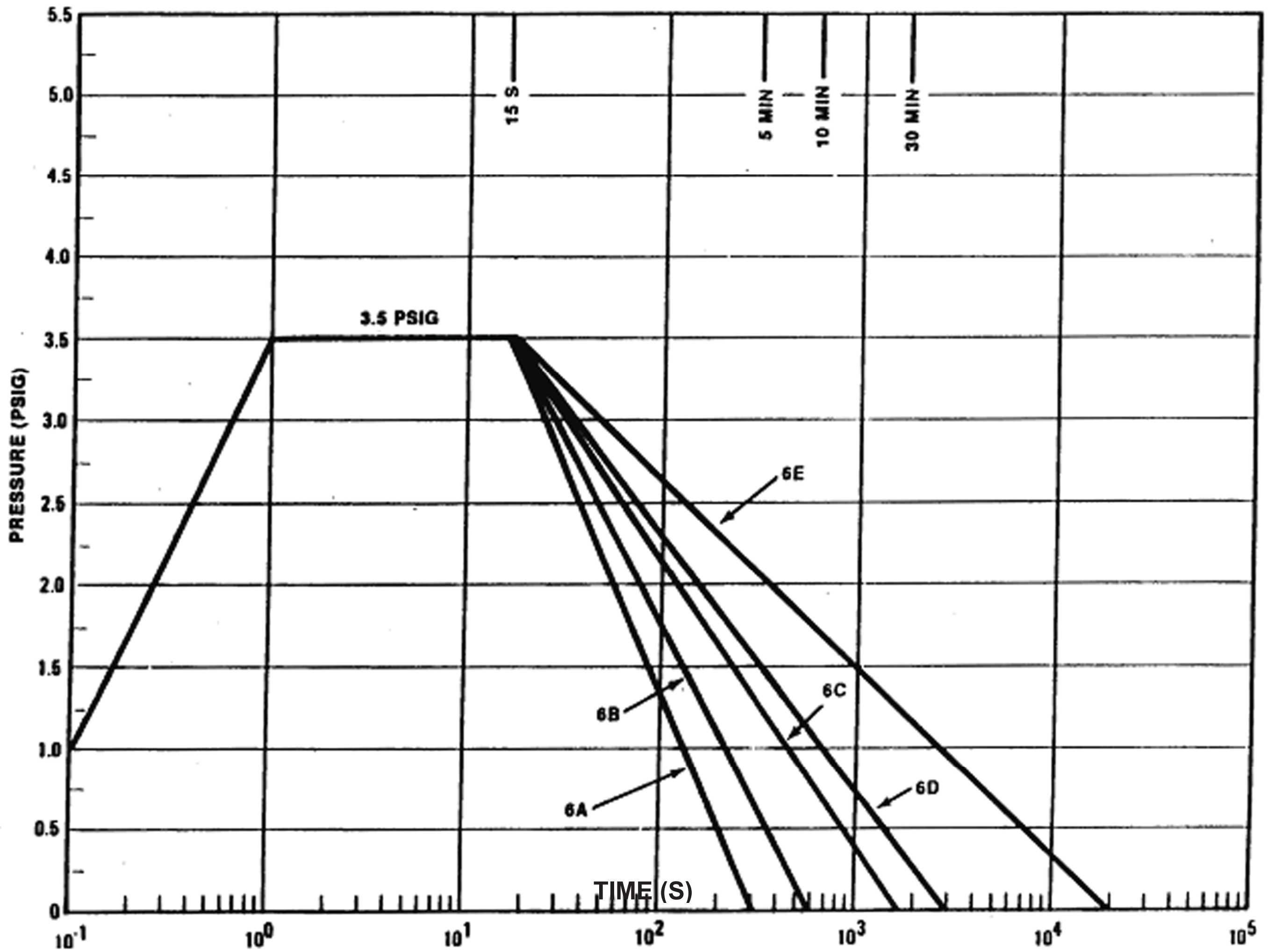
UNIT 1 AND UNIT 2

TEMPERATURE CONDITIONS FOR EQUIPMENT QUALIFICATION IN THE AUXILIARY BUILDING

FIGURE 3.11.B.1–1 (SHEET 5 OF 18)



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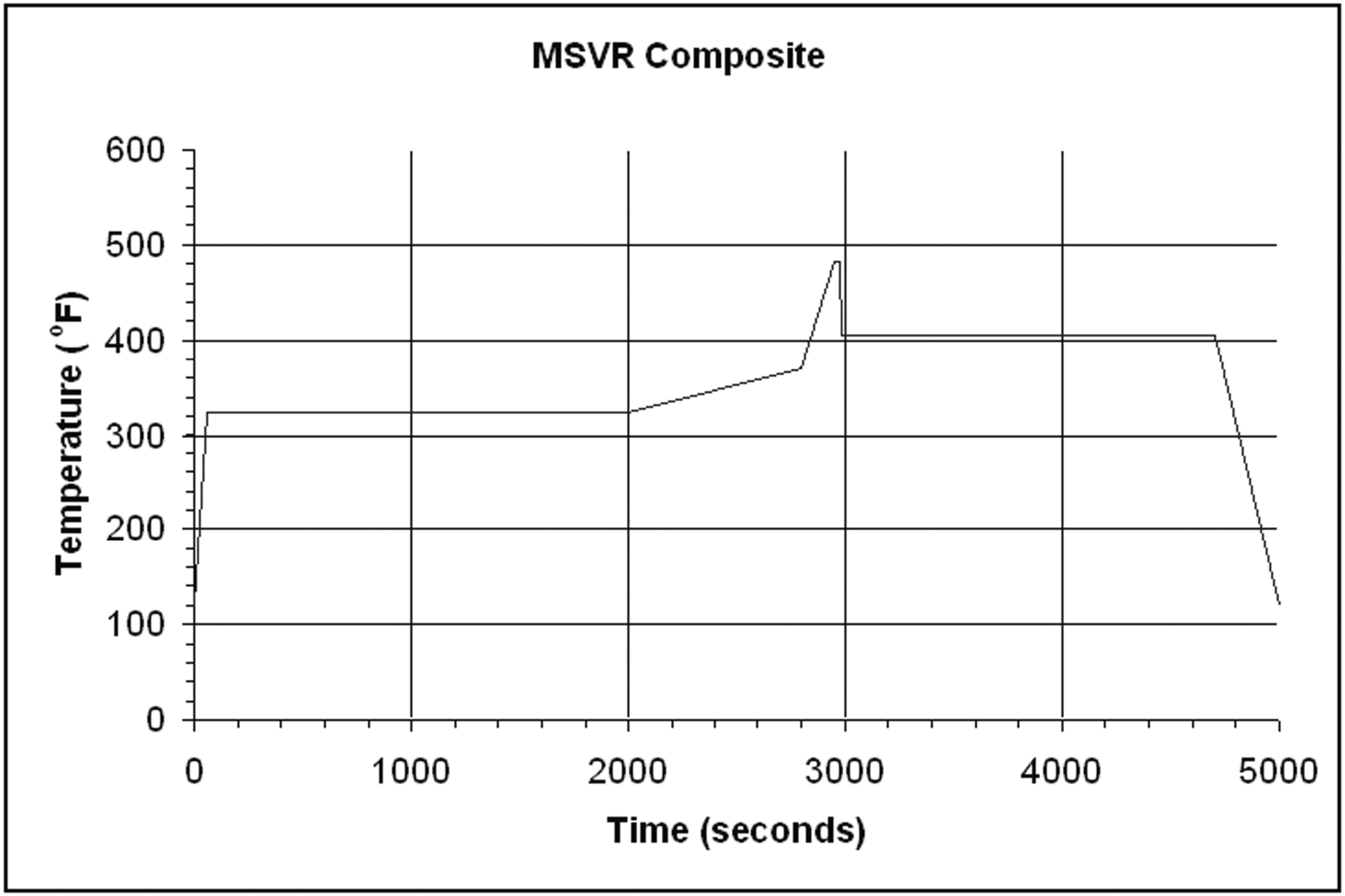
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRESSURE CONDITIONS FOR EQUIPMENT QUALIFICATION IN THE AUXILIARY BUILDING
	FIGURE 3.11.B.1–1 (SHEET 6 OF 18)



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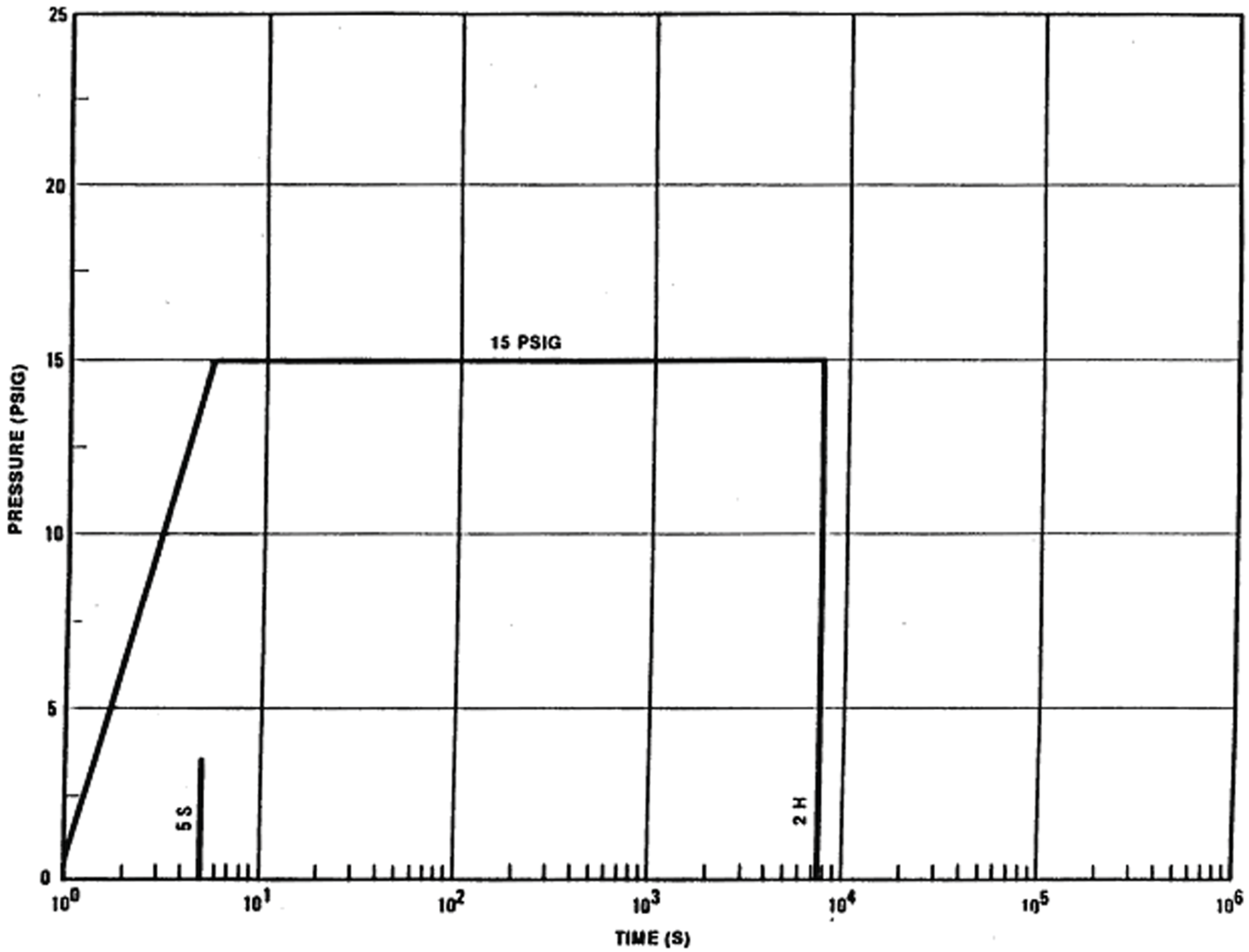
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEMPERATURE CONDITIONS FOR
EQUIPMENT QUALIFICATION
INSIDE MSIV OR MFIV AREA

FIGURE 3.11.B.1–1 (SHEET 7 OF 18)



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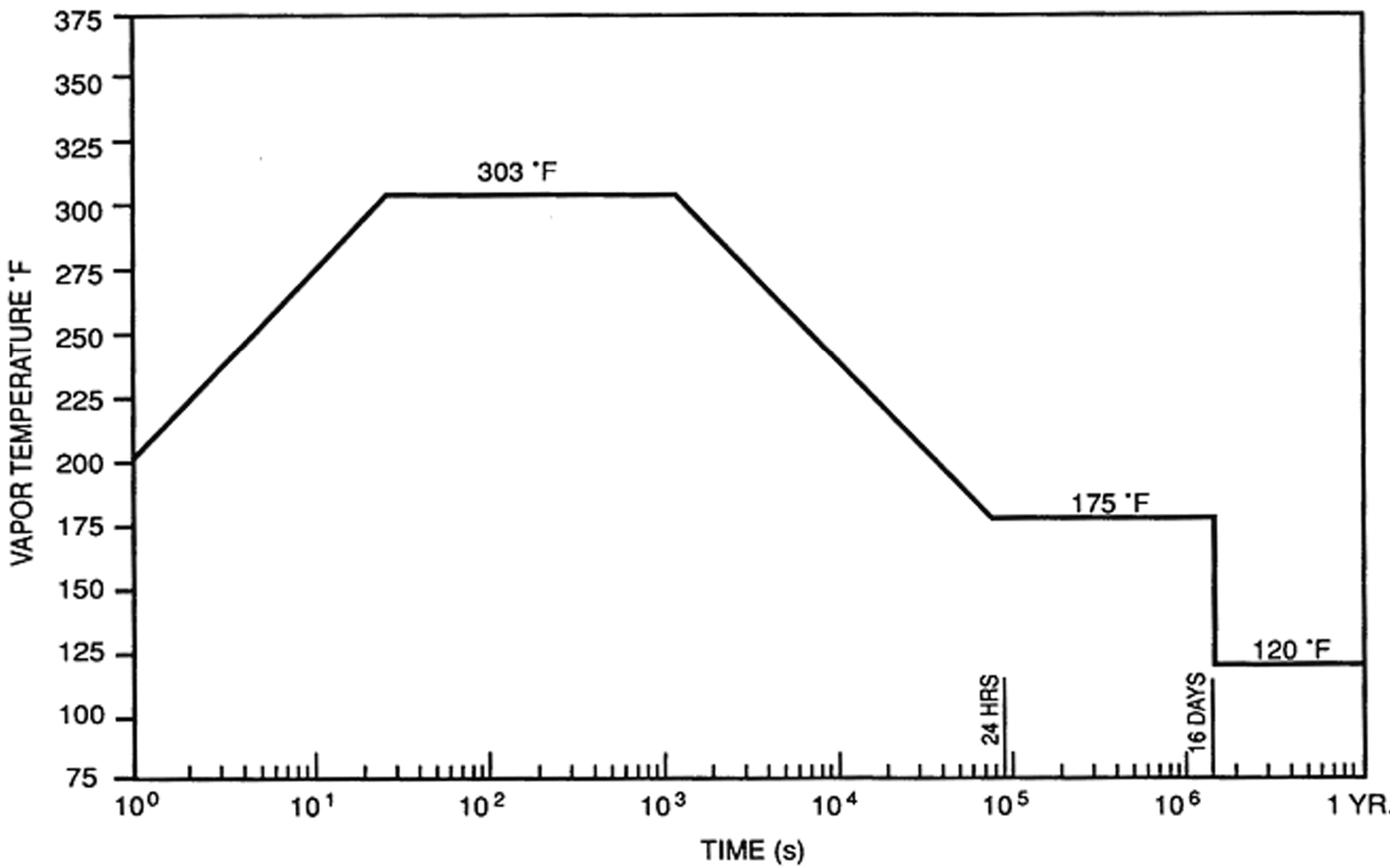
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRESSURE CONDITIONS FOR EQUIPMENT QUALIFICATION INSIDE MSIV OR MFIV AREA FIGURE 3.11.B.1–1 (SHEET 8 OF 18)



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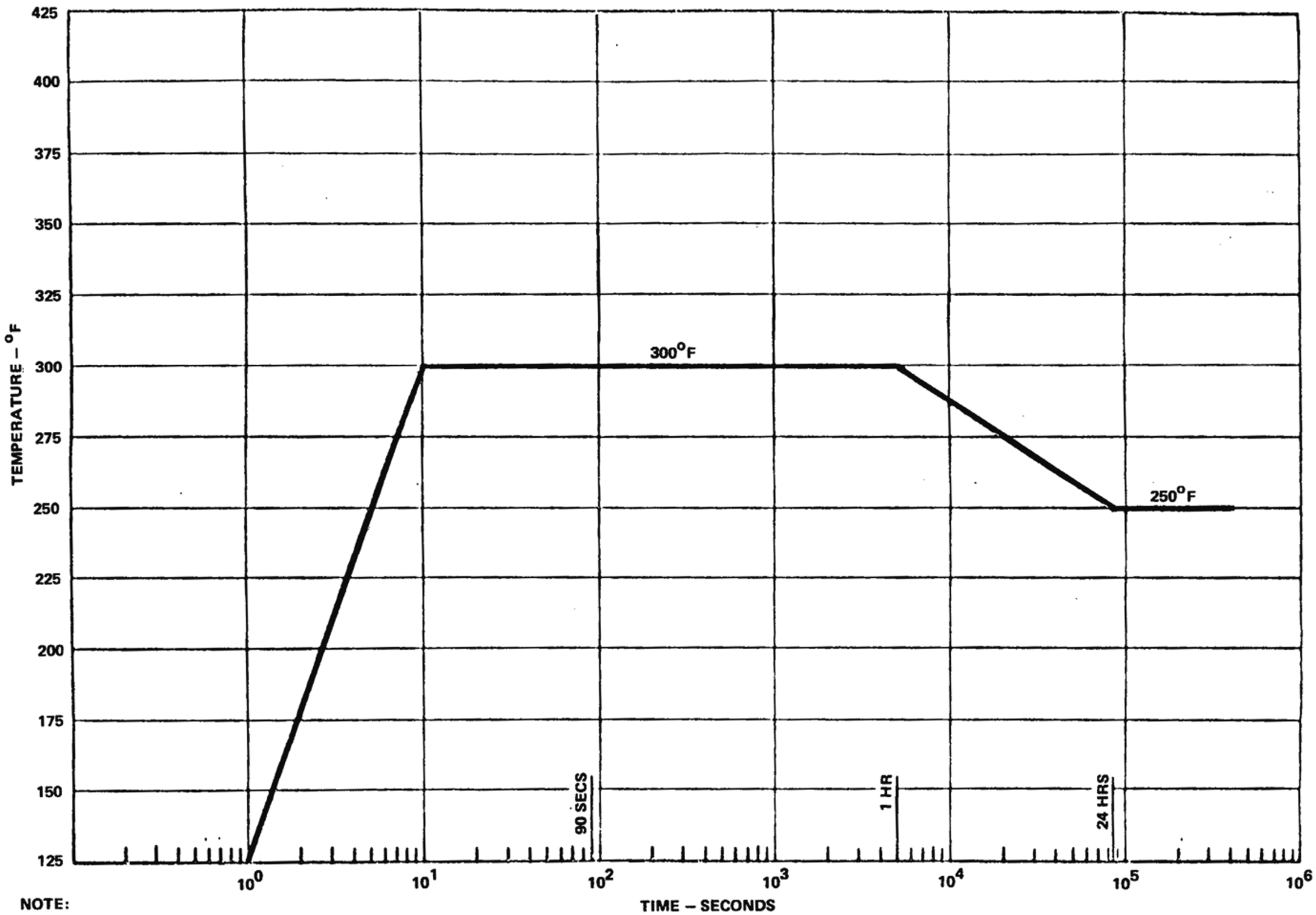
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TEMPERATURE CONDITIONS FOR ENVIROMENTAL QUALIFICATION INSIDE CONTAINMENT (VAPOR REGION)
		FIGURE 3.11.B.1–1 (SHEET 9 OF 18)



NOTE:

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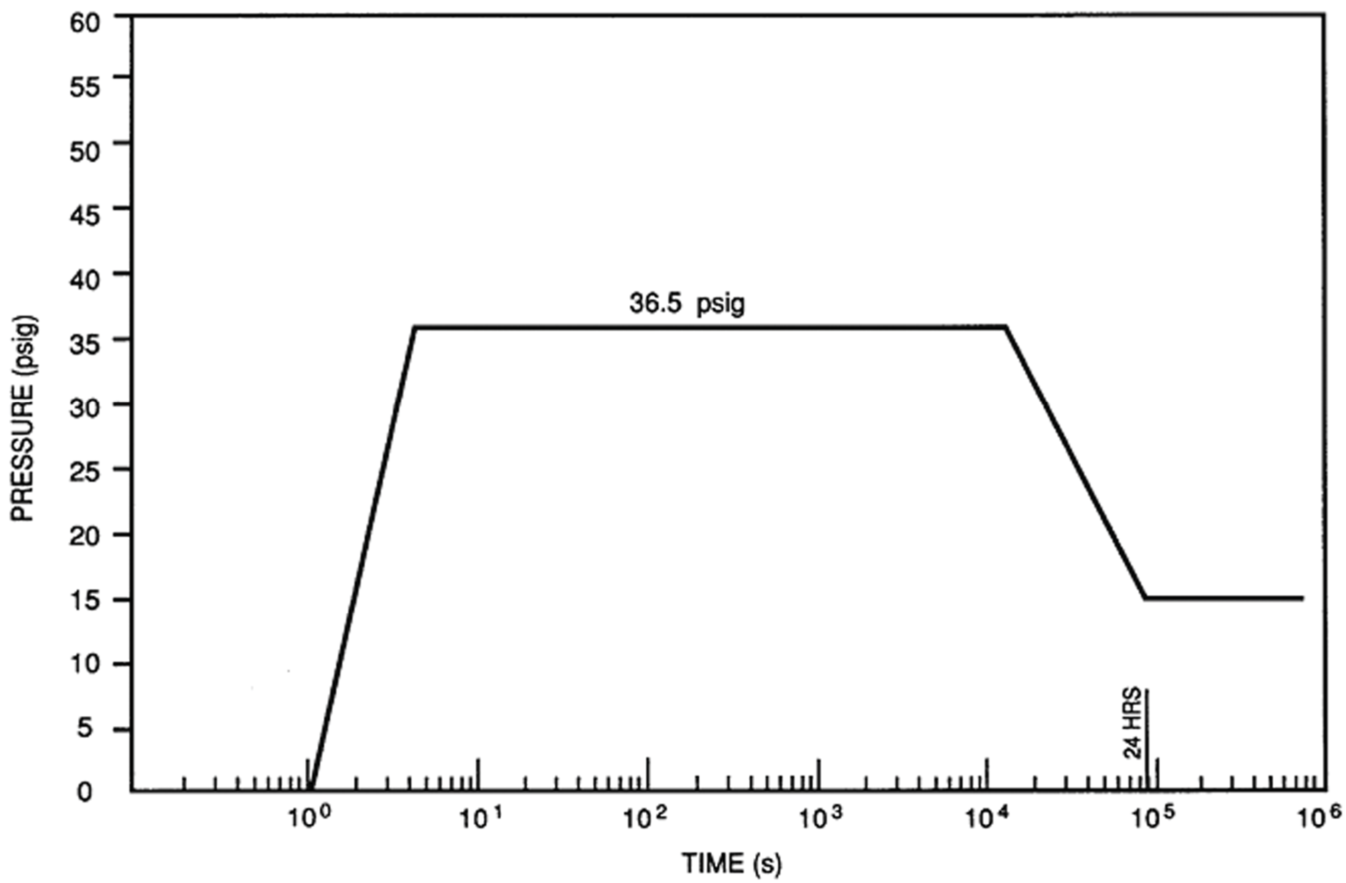
UNIT 1 AND UNIT 2

TEMPERATURE CONDITIONS FOR
EQUIPMENT QUALIFICATION
INSIDE CONTAINMENT (SUMP REGION)

FIGURE 3.11.B.1–1 (SHEET 10 OF 18)



NOTE:
COMBINED LOCA & MSLB
ENVELOPE CURVE SHOWN

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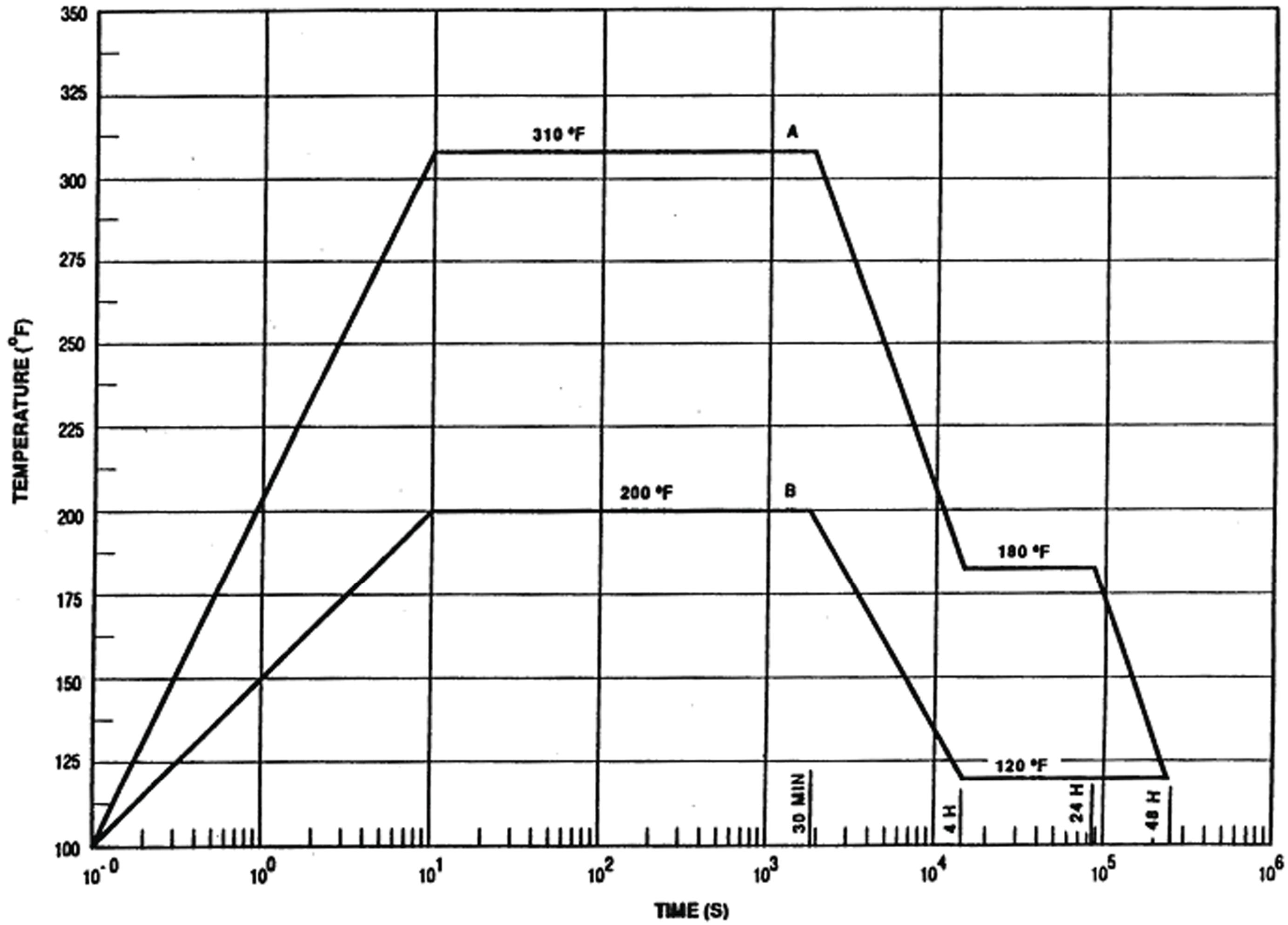
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRESSURE CONDITIONS FOR EQUIPMENT QUALIFICATION INSIDE CONTAINMENT FIGURE 3.11.B.1–1 (SHEET 11 OF 18)



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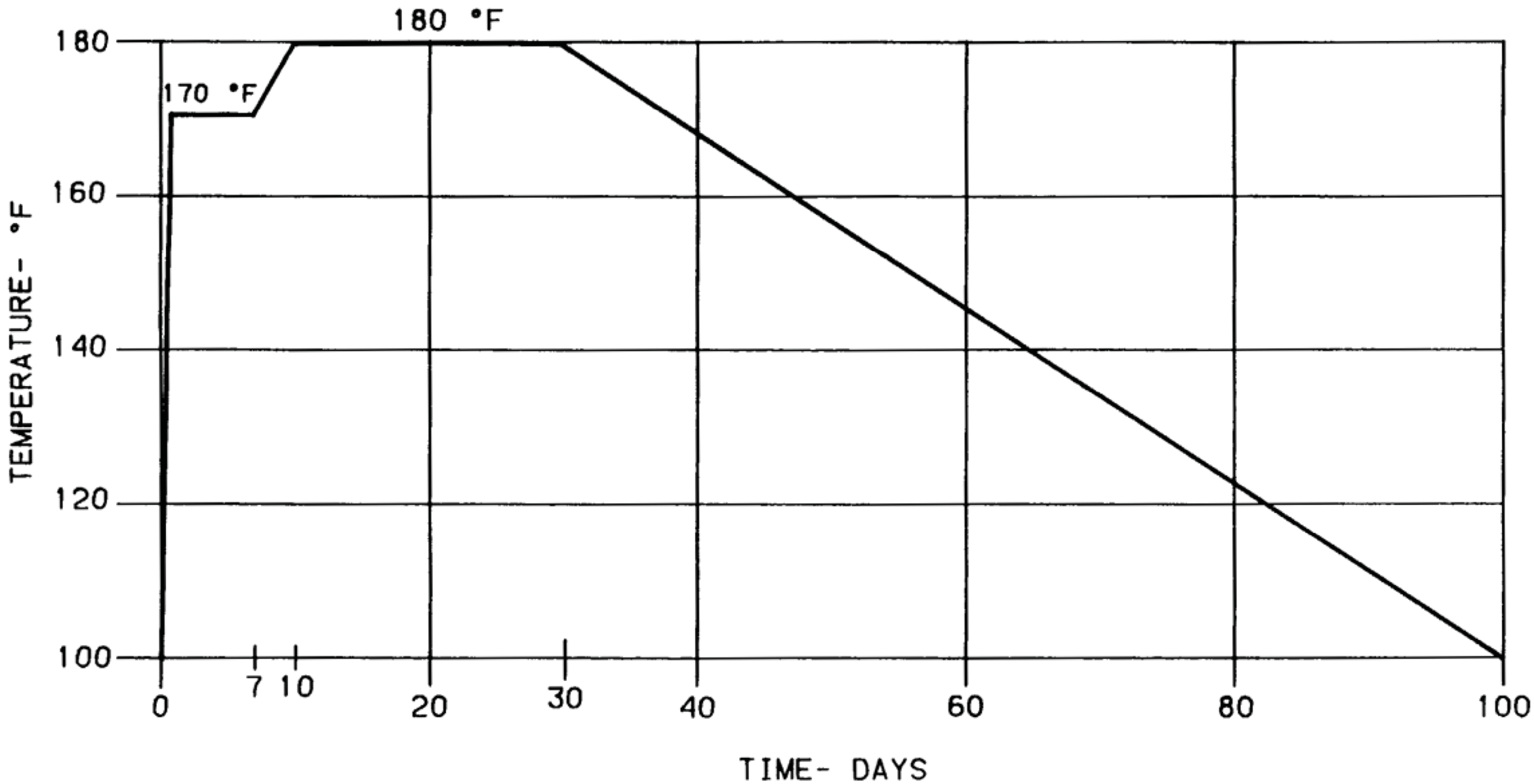
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEMPERATURE CONDITIONS FOR EQUIPMENT QUALIFICATION IN THE AUXILIARY BUILDING

FIGURE 3.11.B.1–1 (SHEET 12 OF 18)



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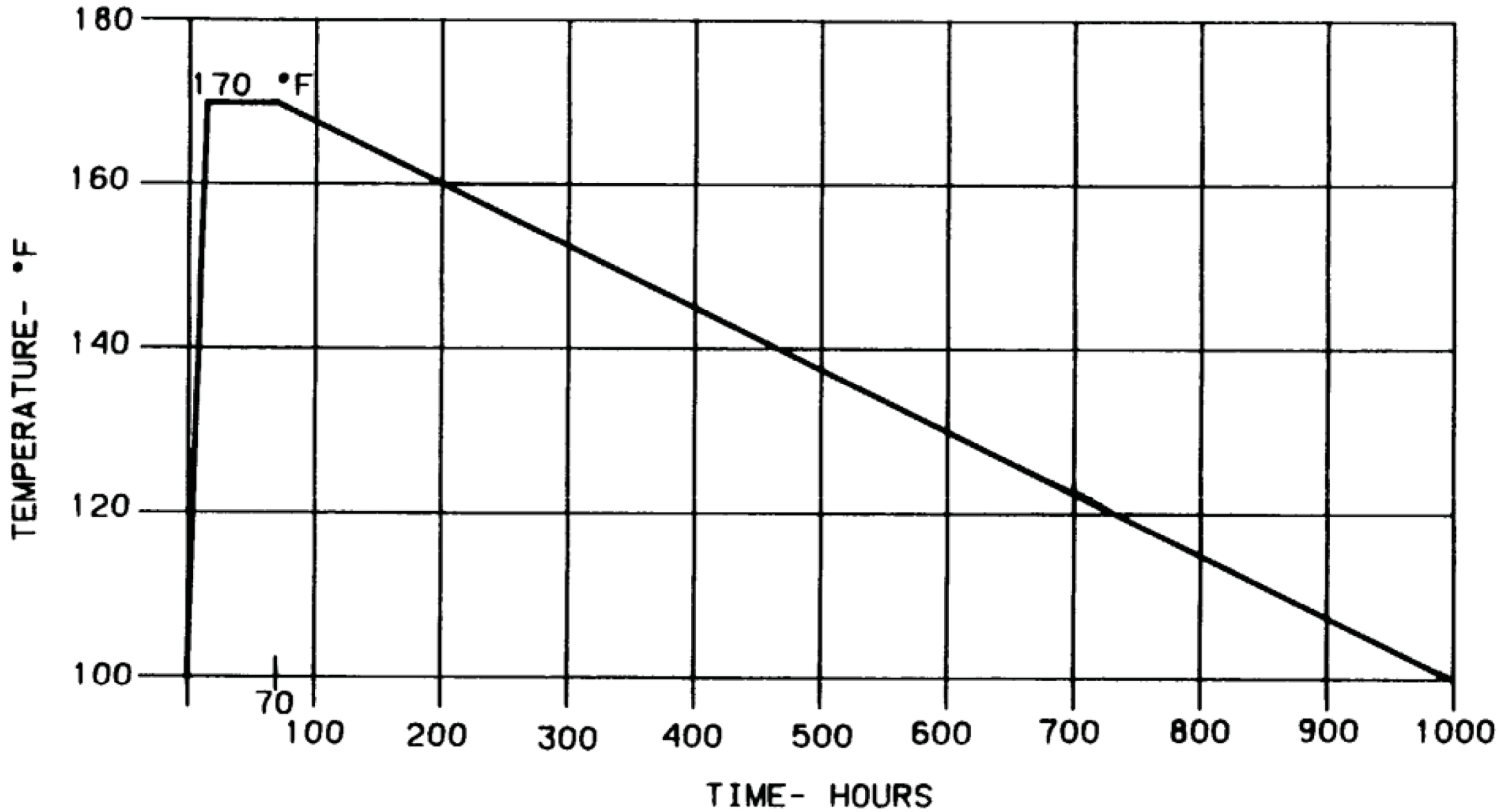
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

LOCA TEMPERATURE CONDITIONS
FOR EQUIPMENT QUALIFICATION

FIGURE 3.11.B.1–1 (SHEET 13 OF 18)



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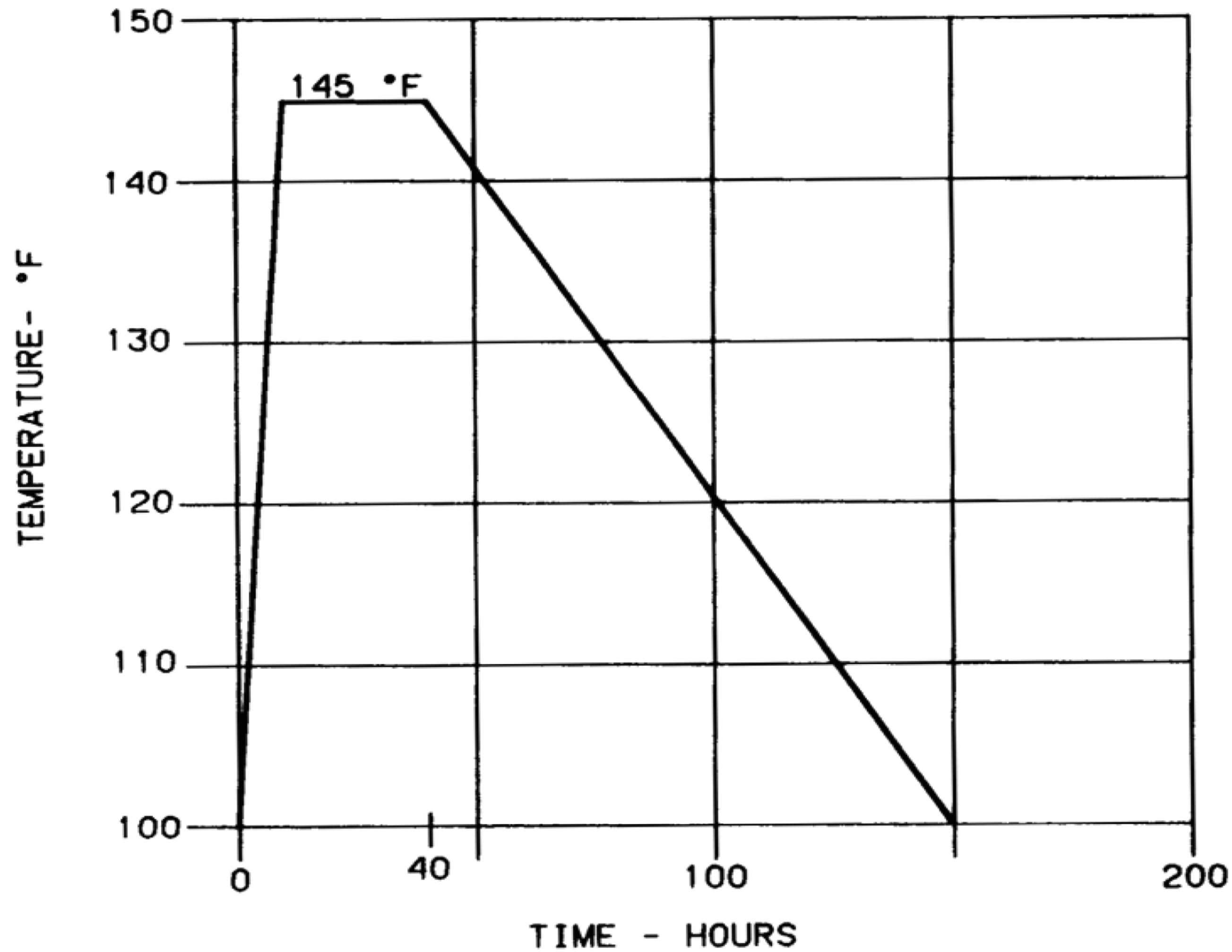
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

LOCA TEMPERATURE CONDITIONS
FOR EQUIPMENT QUALIFICATION

FIGURE 3.11.B.1–1 (SHEET 14 OF 18)



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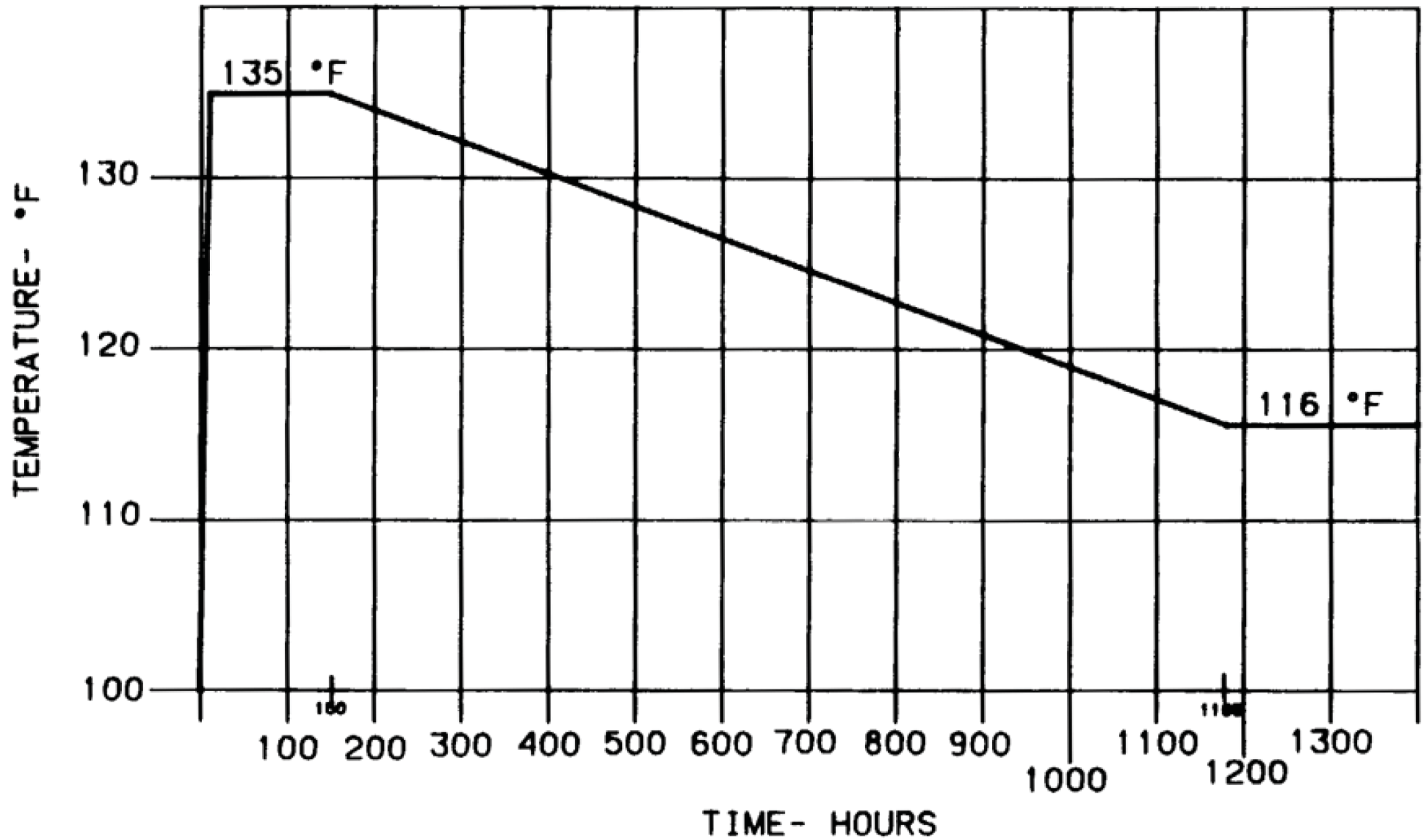
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

LOCA TEMPERATURE CONDITIONS
FOR EQUIPMENT QUALIFICATION

FIGURE 3.11.B.1–1 (SHEET 15 OF 18)



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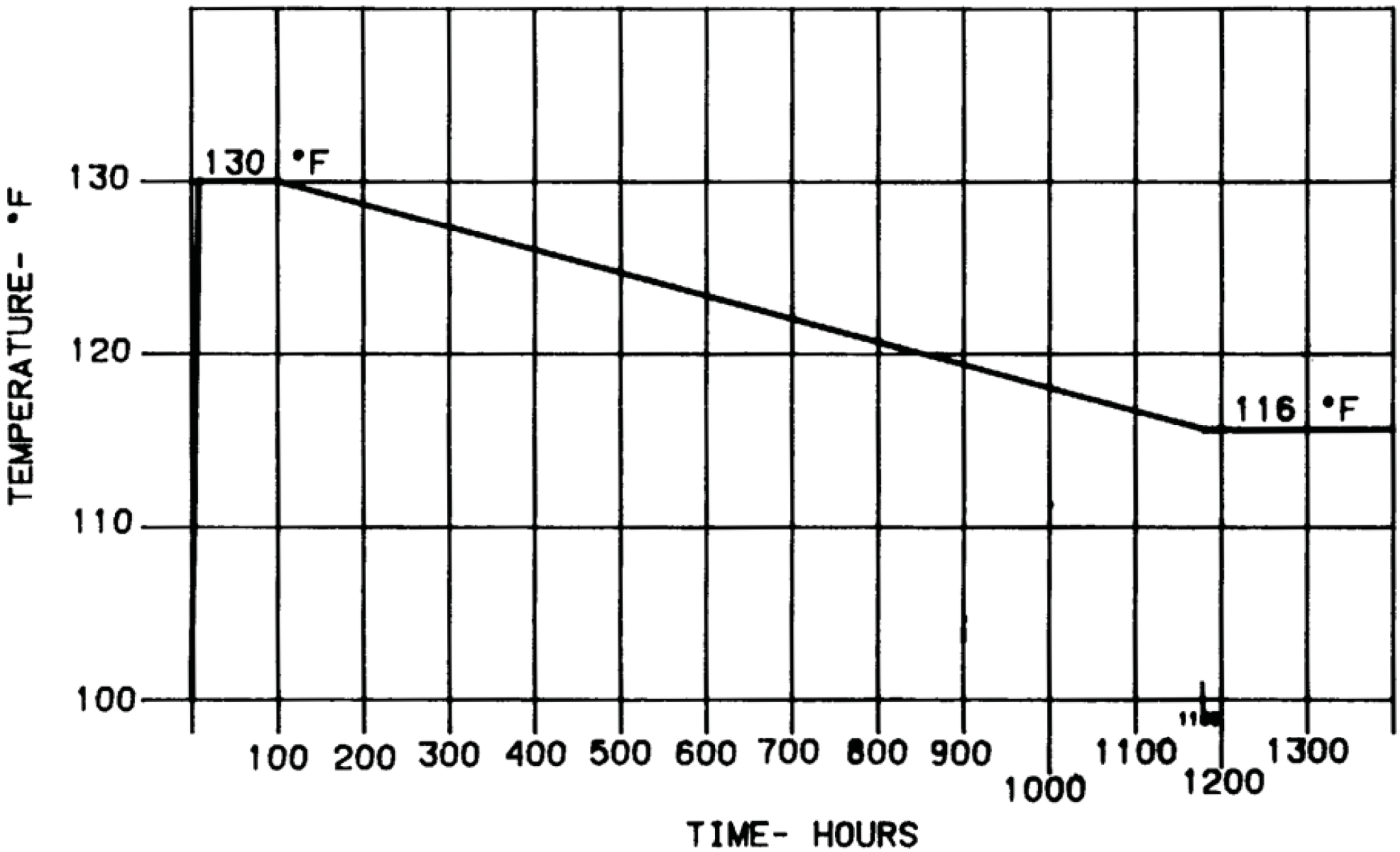
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

LOCA TEMPERATURE CONDITIONS
FOR EQUIPMENT QUALIFICATION

FIGURE 3.11.B.1–1 (SHEET 16 OF 18)



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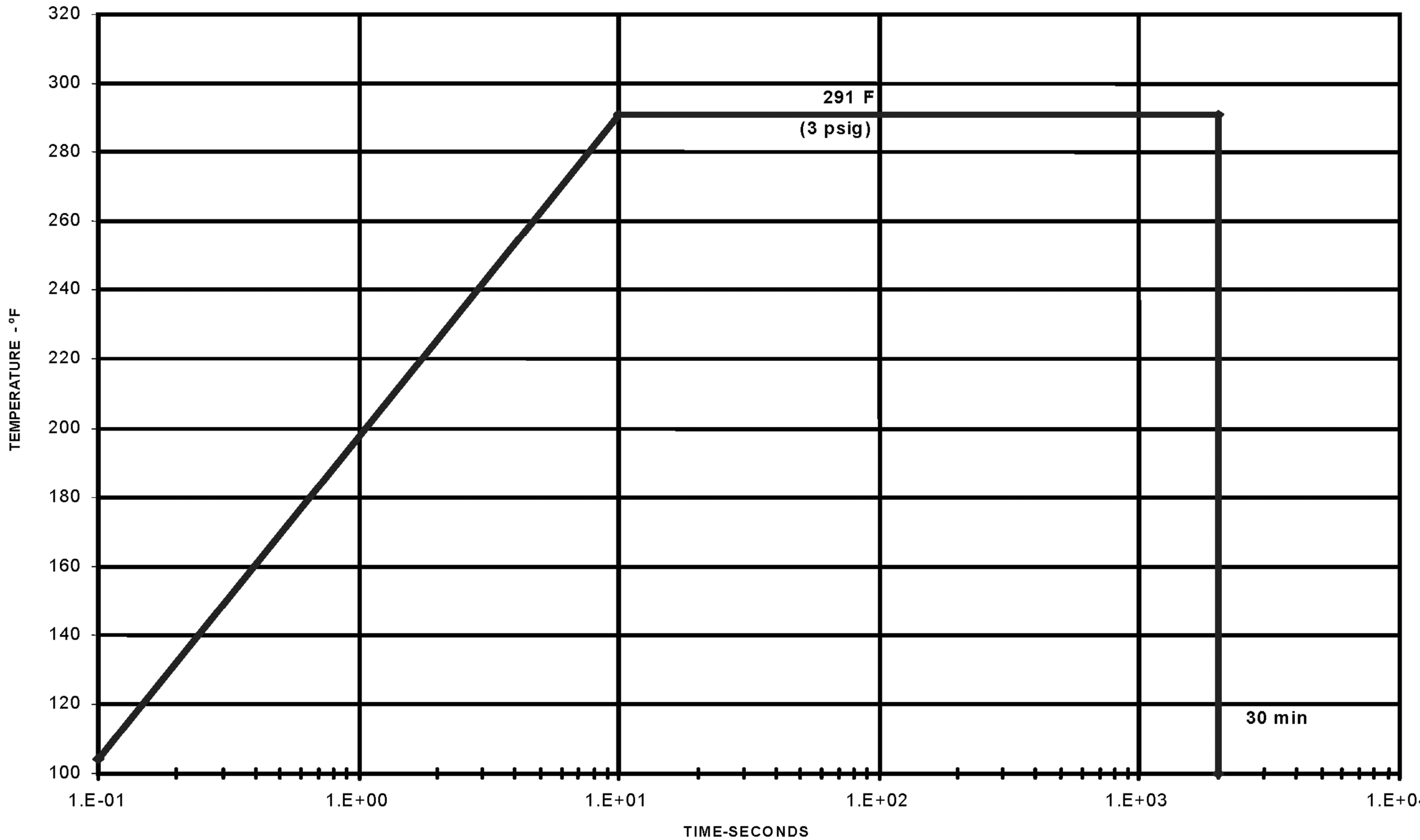
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	LOCA TEMPERATURE CONDITIONS FOR EQUIPMENT QUALIFICATION FIGURE 3.11.B.1–1 (SHEET 17 OF 18)



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UNIT 1 AND UNIT 2

TEMPERATURE CONDITIONS FOR
EQUIPMENT QUALIFICATION IN
THE TURBINE BUILDING

FIGURE 3.11.B.1-1 (SHEET 18 OF 18)

APPENDIX 3A

CONFORMANCE WITH REGULATORY GUIDES

This appendix is not applicable to the VEGP. Refer to section 1.9, Conformance to NRC Regulatory Guides.

APPENDIX 3B

COMPUTER PROGRAMS USED FOR STRUCTURAL, SEISMIC, AND GEOTECHNICAL ANALYSES

3B. COMPUTER PROGRAMS USED FOR STRUCTURAL AND SEISMIC ANALYSES BY BECHTEL POWER CORPORATION

Computer programs are updated under quality control procedures to enhance capabilities and to extend their applicability. As such, earlier versions of these programs, also verified, may have been used during earlier stages of the design effort.

3B.1 BECHTEL CE 201(CE 217), BECHTEL STRUCTURAL ANALYSIS PROGRAM, POST PROCESSOR (BSAP-POST)

A. Description

The Bechtel Structural Analysis Program, BSAP-POST CE 201 (CE 217) is a general-purpose, post-processor program for the BSAP (CE 800) finite-element analysis program. The BSAP-POST program can take the output from BSAP and display this data (graphically and/or on a line printer) or perform additional calculations. In addition, some of the capabilities of BSAP-POST can be used independently. For example, the concrete design module, OPTCON, can have design loads obtained from BSAP output or from punched cards.

The BSAP-POST program consists of a number of modules that can be used independently or sequentially to display or modify the contents of a data base under the control of an executive supervisor program. The data base consists of the contents of a file (TAPE 27) created by a BSAP analysis problem. The executive supervisor ensures that each module in BSAP-POST is compatible with every other module and initiates the execution of each module when required by input data supplied by the user.

B. Validation

The BSAP-POST program has been prepared by Bechtel and has a complete set of documentation, including a user's manual, a verification report, and a theoretical manual. These documents are on file with Bechtel data processing.

C. Extent of Application

The program is used in the analysis and design of structures.

3B.1.1 BECHTEL CE 207 BSAP-DYNAM

A. Description

The BSAP-DYNAM program analyzes soil-structure interaction problems.

B. Validation

The BSAP-DYNAM program has a complete set of documentation, including a user's manual and a verification manual. These documents are on file with Bechtel data processing.

C. Extent of Application

The program is used to determine the modal damping involved with soil-structure interaction. This program has problem-solving capabilities formerly executed by CE 251. (See subsection 3.B.1.4.)

3B.1.2 BECHTEL CE 239 HEMISPHERICAL DOME TENDON ANALYSIS (TENDON)

A. Description

The dome tendon computer program calculates forces and pressures on a hemispherical dome of a prestressed, three-buttress concrete containment building, resulting from prestress by two orthogonal groups of vertical dome tendons and one group of horizontal hoop tendons. One group of vertical dome tendons is located in parallel, vertical planes normal to the x-axis, extending from 90° to 180° azimuth angle. The second group is located in vertical planes normal to the y-axis and extends from zero to 90° azimuth. The third group is located in horizontal planes normal to the z-axis. Each of the vertical dome tendons (the first two groups) has equal areas and equal spacing measured along the spring line. They are anchored at the base of the containment building. The hoop dome tendons have equal areas, but the spacing may either be constant or may vary linearly with the latitude. The hoop tendons extend from the spring line into the dome region up to 45° latitude. Each hoop tendon is anchored at buttresses 240° apart. Successive hoop tendons are anchored at alternate buttresses.

In the analysis, the dome is subdivided into a grid pattern specified by the user. The program calculates the total pressure due to tendon forces at each grid node in the radial direction normal to the dome surface and in the circumferential (hoop or azimuth) and meridional directions. Nodal forces in the hoop and meridional directions are calculated at each node point. The pressures and forces calculated by this program are intended for use as input to a finite-element computer program to determine the stress distribution in the dome.

B. Validation

The TENDON program has a complete set of documentation, including a user's manual, and verification report. The theoretical background is contained in the user's manual. These documents are on file with Bechtel data processing.

C. Extent of Application

The program is used to determine effects of prestressing the tendons in the containment dome.

3B.1.3 BECHTEL CE 251 3D COMPOSITE MODAL DAMPING (GEMD)

A. Description

The GEMD program determines the approximate composite modal damping of a soil-structure interaction system.

B. Validation

The GEMD program has a complete set of documentation, including a user's manual, a verification manual, and a theory manual. These documents are on file with Bechtel data processing.

C. Extent of Application

The program determines the composite modal damping used in the impedance method for seismic analysis.

3B.1.4 BECHTEL CE 450, TURBINE MISSILE PROBABILITY (TURMIS)

A. Description

The TURMIS program computes the damage probability of a nuclear power plant subject to the impact of missiles generated by turbine blades or disks when they fail. The program combines the damage probabilities of all inputted targets to yield a total damage probability. The damage criteria are specified as scabbing or perforation of the concrete barrier, which is predicted by a formula or formulas that can be selected as input.

B. Validation

The TURMIS program has a complete set of documentation including a user's manual, a verification manual, and a theory manual.

Verification is on file with Bechtel data processing.

C. Extent of Application

The program is used to determine the probability of both low- and high-trajectory turbine missile strikes within the power block.

3B.1.5 BECHTEL CE 800, BECHTEL STRUCTURAL ANALYSIS PROGRAM (BSAP)

A. Description

The program performs the static and dynamic analyses of linear, elastic, three-dimensional structures using the finite-element method. The finite-element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved for either applied loads or temperature distributions.

Concentrated loads, pressures, or gravity loads may be applied.

Dynamic response routines are available for solving arbitrary dynamic loads or seismic excitations, using modal superposition. The program can also perform response spectrum and time-history analyses.

B. Validation

The solutions to test problems have been demonstrated to be essentially identical to the results obtained using the following recognized public-domain computer programs:

- EASE — Elastic Analysis Corporation
- STARDYNE — Mechanics Research Incorporated
- MARC/CDC — MARC Analysis Corporation
- ICES/STRUDL — McDonnell-Douglas Automation
- ASKA — Institute fur Statik and Dynamik, Stuttgart, Prof. A. J. Argyris

Agreement has also been established between BSAP program results and the results presented in the ASME Library of Benchmark Computer problems and solutions[1] and in the recognized technical journals. A complete set of documentation, including a user's manual, verification report, and theoretical manual, is on file with Bechtel data processing.

C. Extent of Application

The program is used to perform structural analyses for the majority of steel and concrete structures.

3B.1.6 BECHTEL CE 802, RESPONSE SPECTRA ANALYSIS (SPECTRA)

A. Description

The program computes the response spectra from an acceleration record digitized at equal time intervals. These spectra are plots of the maximum response of a simple oscillator over a range of natural periods and dampings.

The numerical method for computing the spectral values is based on the exact analytical solution of the governing differential equation. It is assumed that the accelerogram varies linearly between the time- history points. The response spectra are constructed by monitoring the maximum values of response parameters of each step of integration. The computed spectra are then widened to account for the effect of structural frequency variations.

B. Validation

The solutions of the program have been verified to be substantially identical with the closed formed analytical solutions of the following three tests problems:

1. Undamped system with a triangular load pulse.
2. Undamped system with a sinusoidal forcing function.
3. Damped system with a sinusoidal forcing function.

A program user's manual, a verification report, and a theoretical manual are on file with Bechtel data processing.

C. Extent of Application

The program is used to develop response spectra curves for all Seismic Category 1 structures.

3B.1.7 BECHTEL CE 982, CONTINUUM LINEAR ANALYSIS FOR SOIL STRUCTURE INTERACTION (CLASSI)

A. Description

The program is capable of evaluating the seismic response of a linear three-dimensional soil-structure interaction model. The analysis capability includes the effects of interaction of each structure and the soil as well as the interaction through the soil among adjacent structures. The method is based on a specialized form of substructuring where the elements of response of the superstructure, foundation, and soil are obtained independently and then are combined to satisfy the interaction conditions.

B. Validation

The CLASSI program has a complete set of documentation, including a user's manual, a verification report, and a theoretical manual. These documents are on file with Bechtel data processing.

C. Extent of Application

The program was used to compute the impedance functions of a layered medium.

3B.1.8 BECHTEL CE 915, A COMPUTER PROGRAM FOR EARTHQUAKE RESPONSE ANALYSIS OF HORIZONTALLY LAYERED SITES (SHAKE)

A. Description

The program computes the responses in a system of homogeneous, viscoelastic layers of infinite horizontal extent subjected to vertically traveling shear waves. The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent linear soil properties, using an iterative procedure to obtain values for modulus and damping compatible with the effective strains in each layer. The program handles systems with variation in both moduli and damping and takes into account the effect of the elastic base.

B. Validation

A complete set of documentation, including a user's manual, verification report, and theoretical manual, is on file with Bechtel data processing.

C. Extent of Application

The program was used to increase the time interval of the Bechtel synthetic time-history accelerograms from 0.005 to 0.01 s.

3B.1.9 BECHTEL ME 351, PIPE FORCE AND WHIP ANALYSIS (PRTHRUST/PIPERUP)

A. Description

The PRTHRUST/PIPERUP program performs a nonlinear elasto-plastic analysis of three-dimensional piping systems subjected to concentrated static or dynamic time-history forcing function. These forces may result from fluid jet thrust at the location of a postulated rupture of high-energy piping. The program is an adaptation of the finite element method to the specific requirements of pipe rupture analysis. Straight and curved-beam (elbow) elements are used to mathematically represent the piping, and axial and rotational springs are used to represent restraints. The stiffness characteristics of piping and restraints can reflect elastic/linear strain hardening material properties, and gaps between piping and restraints can be modeled.

B. Validation

Verification is on file with Bechtel data processing.

C. Extent of Application

The program was used to perform analysis of piping systems to obtain loads used in design of pipe whip restraints and their backup structures.

3B.1.10 BECHTEL TE 301, TWO-DIMENSIONAL STRUCTURAL ANALYSIS (MFRAME)

A. Description

The MFRAME program performs the analysis of two-dimensional framed structures using the direct stiffness method. It can process structures with beam elements and pin-jointed truss elements or a combination of these. The output consists of joint displacements, member end forces, joint loads, and reactions. The output joint loads are calculated internally from member end forces and therefore serve as a check for the validity of the solution.

B. Validation

The program has been verified, and appropriate documentation is on file with Bechtel data processing.

C. Extent of Application

The program was used for general beam, rigid frame, or truss analysis.

3B.1.11 FLUSH (CONTROL DATA CORP. (CDC) VERSION)

A. Description

This finite-element program uses two-dimensional soil and three-dimensional structure modeling techniques to compute the seismic response of a soil-structure interaction system that accounts for embedment and structure-to-structure effects.

B. Validation

Verification of CDC's version of FLUSH has been performed and appropriate documentation, as defined by CDC policy, is maintained by CDC's Utilities Service Center.

C. Extent of Application

The program is used to seismically analyze deeply embedded Seismic Category 1 structures.

3B.1.12 THE STRUCTURAL DESIGN LANGUAGE (ICES-STRUDL-II MCDONNELL-DOUGLAS AUTOMATION VERSION)

A. Description

The program performs structural analysis. Frame members can be used in conjunction with finite elements. Some special features include a built-in table for rolled steel wide flange shapes, a member selection procedure based upon the American Institute of Steel Construction Specification, a reinforced concrete member design and checking capability, and a dynamic analysis capability.

B. Validation

The program has been verified, and document traceability is available at McDonnell-Douglas Automation.

C. Extent of Application

The program is used to perform structural analysis for the basemat of the auxiliary building and preliminary structural analysis of the nuclear service cooling towers.

3B.1.13 ICES-LEASE (MCDONNELL-DOUGLAS AUTOMATION VERSION)

A. Description

The LEASE (limiting equilibrium analysis in soil engineering) program is a subsystem of ICES which performs stability analysis of arbitrary slopes by the method of slices.

B. Validation

The program has been verified, and document traceability is available at McDonnell-Douglas Automation.

C. Extent of Application

The program was used to determine the factor of safety against sliding of excavated slopes.

3B.1.14 ICES-SEPOL (MCDONNELL-DOUGLAS AUTOMATION VERSION)

A. Description

The SEPOL program is a subsystem of ICES which computes stress and strains in a layered soil system.

B. Validation

The program has been verified, and document traceability is available at McDonnell-Douglas Automation.

C. Extent of Application

The program was used to estimate the settlements of power block structures.

3B.1.15 BECHTEL CE 212 BSAP-PRE

A. Description

BSAP-PRE (CE-212) is an interactive program used to create and edit input data for the BSAP (CE 800) program. The finite element model data can be entered directly by the user or obtained from an external file.

B. Validation

The BSAP-PRE program has been prepared by Bechtel and has a complete set documentation, including a user's manual and validation report. These documents are on file with Bechtel data processing.

C. Extent of Application

The program is used to create input data files for BSAP (CE 800).

3B.1.16 OTHER COMPUTER PROGRAMS USED IN STRUCTURAL ANALYSES

In the course of generating structural design calculations, several programs are used to assist design efforts. These programs are limited in scope and are developed solely to assist the designer in making lengthy, repetitious calculations, thereby saving design efforts. The programs are listed and checked in the project design calculations. These programs are not itemized here because of their simplicity and nature of use.

3B.1.17 REFERENCES

1. Pressure Vessel and Piping - 1972 Computer Programs
2. Verification, ASME Committee on Computer Technology, Pressure Vessel and Piping Division.
3. Wilson, E. L., "SAP-IV - A Structural Analysis Program for Static and Dynamic Response of Linear Systems," University of California, Berkeley, EERC Report No. 73-11, June 1973.

APPENDIX 3C

DESIGN OF STRUCTURES FOR TORNADO MISSILE IMPACT

3C.1 INTRODUCTION

This appendix contains methods and procedures for analysis and design of steel and reinforced concrete structures and structural elements subject to tornado-generated missile impact effects. Structures subject to missile impact, postulated missiles, and other concurrent loading conditions are identified in section 3.5.

Missile impact effects are assessed in terms of local damage and structural response. Local damage (damage that occurs in the immediate vicinity of the impact area) is assessed in terms of perforation and scabbing.

Evaluation of local effects is essential to ensure that protected items would not be damaged directly by a missile perforating a protective barrier or by scab particles. Empirical formulas are used to assess local damage.

Evaluation of structural response is essential to ensure that protected items are not damaged or functionally impaired by deformation or collapse of the impacted structure.

Structural response is assessed in terms of deformation limits, strain energy capacity, structural integrity, and structural stability. Structural dynamics principles are used to predict structural response.

3C.1.1 PROCEDURES

The general procedures for analysis and design of structures or structural elements for missile impact effects include:

A. Defining the missile properties (such as type, material, deformation characteristics, geometry, mass, trajectory, strike orientation, and velocity).

B. Determining impact location, material strength, and thickness required to preclude local failure (such as perforation for steel targets and scabbing for reinforced concrete targets).

C. Defining the structure and its properties (such as geometry, section strength, deformation limits, strain energy absorption capacity, stability characteristics, and dynamic response characteristics).

D. Determining structural response considering other concurrent loading conditions.

E. Checking adequacy of structural design (stability, integrity, deformation limits, etc.) to verify that local damage and structural response (maximum deformation) will not impair the function of safety- related items.

3C.2 LOCAL EFFECTS

Evaluation of local effects consists of estimating the extent of local damage and characterization of the interface force-time function used to predict structural response. Local damage is confined to the immediate vicinity of the impact location on the struck element and consists of missile deformation, penetration of the missile into the element, possible perforation of the element, and, in the case of reinforced concrete, dislodging of concrete particles from the back face of the element (scabbing).

Because of the complex physical processes associated with missile impact, local effects are evaluated primarily by application of empirical relationships based on missile impact test results. Unless otherwise noted, these formulas are applied considering a normal incidence of strike with the long axis of the missile parallel to the line of flight.

3C.2.1 REINFORCED CONCRETE ELEMENTS

Where concrete exterior walls and roofs are used as barriers to offer missile protection, such walls have a 24-in. minimum thickness, while the roofs are at least 21 in. thick. The concrete has a 28-day compressive strength of at least 4000 psi (91-day strength for concrete containing pozzolan). Where the interior walls and slabs having concrete compressive strength of 5000 psi are used as missile barriers, such walls have an 18-in. minimum thickness, while the slabs are at least 14 in. thick. Therefore, the walls and roofs of these structures are resistant to perforation and scabbing by the postulated missiles discussed in paragraph 3.5.1.4.

3C.2.2 STEEL ELEMENTS

Steel barriers subjected to missile impact are designed to preclude perforation. An estimate of the steel element thickness for threshold of perforation for nondeformable missiles is provided by equation 2-1 (which is a more convenient form of the Ballistic Research Laboratory (BRL) equation for perforation of steel plates with material constant taken as unity[1]).

where:

$$T_p = \frac{(E_k)^{2/3}}{672D} \qquad E_k = \frac{M_m V_s^2}{2} \qquad (2\text{-}1)$$

T_p = steel plate thickness for threshold of perforation (in.).
E_k = missile kinetic energy (ft-lb).
M_m = mass of the missile (lb-s^2/ft).
V_s = missile striking velocity (ft/s).
D_p = missile diameter (in.).[1]

The design thickness to prevent perforation, t_p, must be greater than the predicted threshold value. The threshold value is increased by 25 percent to obtain the design thickness.

$$t_p = 1.25\, T_p \qquad (2\text{-}2)$$

where:

t_p = design thickness to preclude perforation (in.).

[1] For irregularly shaped missiles, an equivalent diameter is used. The equivalent diameter is taken as the diameter of a circle with an area equal to the circumscribed contact, or projected frontal area, of the noncylindrical missile. For pipe missiles, D is the outside diameter of the pipe.

3C.3 STRUCTURAL RESPONSE DUE TO MISSILE IMPACT LOADING

When a missile strikes a structure, large forces develop at the missile-structure interface, which decelerate the missile and accelerate the structure. The response of the structure depends on the dynamic properties of the structure and the time-dependent nature of the applied loading (interface force- time function). The force-time function is, in turn, dependent on the type of impact (elastic or plastic) and the nature and extent of local damage.

3C.3.1 GENERAL

In an elastic impact, the missile and the structure deform elastically, remain in contact for a short period of time (duration of impact), and subsequently disengage due to the action of elastic interface restoring forces.

In a plastic impact, the missile or the structure or both may deform plastically or sustain permanent deformation or damage (local damage). Elastic restoring forces are small, and the missile and the structure tend to remain in contact after impact. Plastic impact is much more common in nuclear plant design than elastic impact, which is rarely encountered. For example, test data indicate that the impact from all postulated tornado-generated missiles can be characterized as a plastic collision.

If the interface forcing function can be defined or conservatively idealized (from empirical relationships or from theoretical considerations), the structure can be modeled mathematically, and conventional analytical or numerical techniques can be used to predict structural response. If the interface forcing function cannot be defined, the same mathematical model of the structure can be used to determine structural response by application of conservation of momentum and energy balance techniques with due consideration for type of impact (elastic or plastic).

In either case, in lieu of a more rigorous analysis, a conservative estimate of structural response can be obtained by first determining the response of the impacted structural element and then applying its reaction forces to the supporting structure. The predicted structural response enables assessment of structural design adequacy in terms of strain energy capacity, deformation limits, stability, and structural integrity.

Three different procedures are given for determining structural response: the force-time solution, the response chart solution, and the energy balance solution. The force-time solution involves numerical integration of the equation(s) of motion and is the most general method applicable for any pulse shape and resistance function. The response chart solution can be used with comparable results, provided the idealized pulse shape (interface forcing function) and the resistance function are compatible with the response chart. The energy balance solution is used in cases where the interface forcing function cannot be defined or where an upper limit check on structural response is desired. This method will consistently overestimate structural response, since the resisting spring forces during impact are neglected.

In defining the mass-spring model, consideration is given to local damage that could affect the response of the element. For concrete slab elements, the beneficial effect of formation of a fracture plane which propagates from the impact zone to the back of the slab (back face fracture plane) just prior to scabbing (reference 2) is neglected. The formation of this fracture plane limits the forces transferred to the surrounding slab and significantly reduces overall structural response. Since scabbing is to be precluded in the design, the structural response check is made assuming the fracture plane is not formed. It is recognized, however, that should the missile velocity exceed that for threshold of scabbing, structural response would be limited by this mechanism.

Therefore, the structural response is conservatively evaluated ignoring formation of the fracture plane and any reduction in response.

3C.3.2 STRUCTURAL ASSESSMENT

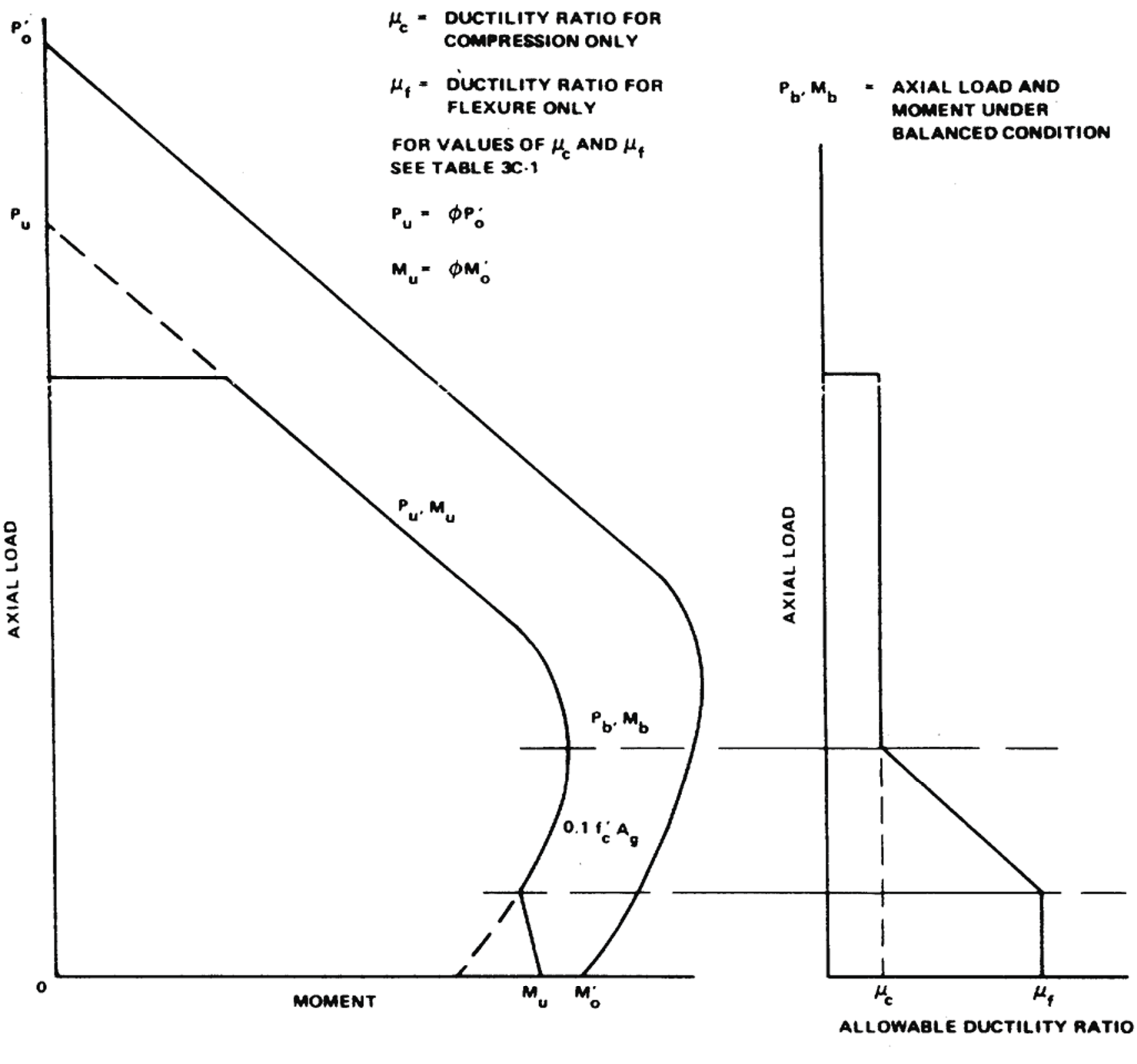
The predicted structural response enables assessment of design adequacy in terms of strain energy capacity, deformation limits, stability, and structural integrity.

For structures allowed to displace beyond yield (elasto-plastic response), a check is made to ensure that deformation limits would not be exceeded, by comparing calculated displacements or required ductility ratios with allowable values (such as those contained in table 3C-1).

3C.4 REFERENCES

1. Gwaltney, R. C., "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants," ORNL NSIC-22, Oak Ridge National Laboratory, Oak Ridge, Tennessee, for the USAEC, September 1968.

2. Rotz, J. V., "Results of Missile Impact Tests on Reinforced Concrete Panels," Vol 1A, pp 720-738, Second Specialty Conference on Structural Design of Nuclear Power Plant Facilities, New Orleans, Louisiana, December 1975.



(A) REINFORCED CONCRETE INTERACTION DIAGRAM (P VS M)

(B) ALLOWABLE DUCTILITY RATIO μ VS P

REV 14 10/07

SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

MAXIMUM ALLOWABLE DUCTILITY RATIO FOR REINFORCED CONCRETE SECTON WITH BEAM–COLUMN ACTION

FIGURE 3C–1

3D.1 SEISMIC RESPONSE SPECTRA

The VEGP seismic response spectra at selected levels of major Category 1 structures are provided in figures 3D-1 through 3D-78.

3D.2 SEISMIC ANALYSIS METHODS

The NRC, in its letter to GPC dated March 27, 1978, accepted the seismic design methods proposed in Preliminary Safety Analysis Report supplements 3 and 4 with the additional information provided in the GPC letter of February 20, 1978, on the scaling factor, subject to the completion of a confirmatory study and sensitivity study.

The NRC also requested that the seismic analysis include consideration of torsional moment no less than that required by the Uniform Building Code (to account for the seismic wave propagation effects), in addition to the effects resulting from the eccentricity between the center of mass and the center of rigidity at each level.

The reports on the confirmatory study and the sensitivity study together with the description of the methodology to account for torsion caused by the seismic wave propagation effects were submitted to the NRC in the GPC letter dated November 13, 1978. The reports on the confirmatory study and the sensitivity study and the write-up on torsion are reproduced in this section as requested during the NRC structural audit on December 4-6, 1984.

3D.2.1 CONFIRMATORY STUDY

3D.2.1.1 Introduction

In Supplement 3 (S3), dated November 1977 and Supplement 4 (S4), dated January 25, 1978, to the Vogtle PSAR, revisions to the seismic analysis methods were proposed by the Georgia Power Company. The NRC, in their letter dated March 27, 1978, stated that the proposed methods were acceptable and requested the Georgia Power Company to perform a study in which the calculated response spectra (in the reactor building and in one other safety-related structure) per Supplements 3 and 4 to the VEGP PSAR together with a scaling factor of 1.5 be compared with the response spectra obtained from the analyses using the lumped parameter representation of the soil and structure. In this report, such a comparison is presented. For this study, the containment (reactor building) and the control building are considered.

3D.2.1.2 Soil-Structure Interaction Analyses Per S3 and S4 Method

The details of the soil-structure interaction analysis procedure for deeply embedded structures can be found in the VEGP PSAR Supplement 3 and Supplement 4. The value of the scaling factor that is used to multiply the "envelop in-structure response spectra" generated for each deeply embedded seismic Category I structure has been documented as 1.5 by Georgia Power Company in their letter dated February 20, 1978, to the NRC. The response spectra from these analyses will be referred to as the "response spectra from the S3 and S4 method."

The important elements involved in the analysis are as follows:

A. 0.2 g (SSE) Regulatory Guide 1.60 design response spectra.

B. Regulatory Guide 1.61 damping values for the structure.

C. The design response spectra are defined for the free field and applied at the finished grade level (el 220 ft 0 in.).

D. The computer program FLUSH is used to perform the soil-structure interaction analyses using a finite element representation of the soil.

E. Strain dependent shear moduli (Figures 3.7-22B, 22C, and 22D of Supplement 3) and strain dependent damping values (Figures 3.7-21, 22, and 22A of Supplement 3) are used for the soil properties.

F. Soil property variation considered. The analyses are performed using the mean values of shear moduli, 1.5 times the mean values and the mean values divided by 1.5. The response spectra obtained from these three analyses are enveloped.

G. The response due to the three components of earthquake is combined using the SRSS technique.

H. The "envelop in-structure response spectra" is multiplied by a scaling factor of 1.5 in order to meet the 60 percent criteria of the Standard Review Plan, Section 3.7.1 - Subsection II.2.

I. The "envelop in-structure response spectra" is then broadened to account for uncertainties in modeling and analysis techniques and smoothed to arrive at the final in-structure response spectra.

3D.2.1.3 Soil-Structure Interaction Analysis Using the Impedance Method

The basic approach used in the impedance method is specified in Chapter 3 of Bechtel Topical Report BC-TOP-4A, "Seismic Analyses of Structures and Equipment for Nuclear Power Plants," Revision 3, dated November 1974. In following the intent of the BC-TOP-4A impedance method, the impedances used in the analyses were computed more appropriately by considering the actual shape of the foundation and the layering of the finished site. The response spectra from these analyses will be referred to as the "response spectra from the impedance method."

The important elements involved in the impedance method of analysis are identified below:

A. 0.2 g (SSE), Regulatory Guide 1.60 design response spectra.

B. Regulatory Guide 1.61 damping values for the structure.

C. The design response spectra are defined for the free field at the foundation level of the structure.

D. To account for the strain dependent nature of shear modulus, the impedance were computed using the iterated shear moduli obtained from the free field column study based on the "mean" soil properties. Based on the fundamental frequency of the soil-structure system, the corresponding impedance coefficients were selected from the frequency dependent impedance functions.

   For the vertical analyses, the impedances were computed using a P-wave velocity of 5,000 ft/s for soil layers below the water table. This is compatible with the assumptions made in the FLUSH analysis, that in saturated soils, the P-wave would travel with the P-wave velocity of the soil medium or the P-wave velocity of water, whichever is greater. The P-wave velocity of water has been assumed to be 5000 ft/s.

E. The embedment effects and structure-to-structure interaction effects are not considered. Neglect of embedment effects would tend to increase the magnitude of responses within the structure.

F. The response due to the three components of earthquake are combined using the SRSS technique.

G. The in-structure response spectra is broadened to account for uncertainties in modeling and analysis techniques and smoothed to arrive at the final in-structure response spectra.

3D.2.1.4 Comparison of Response Spectra from the S3 and S4 Method and the Impedance Method

A. Containment NRC in their letter dated March 27, 1978, requested that comparison be made at certain levels of the containment. These levels with their corresponding elevations are identified below:

- Top of base mat — el 169 ft 0 in.
- Operating floor level — el 220 ft 0 in.
- Reactor support level — el 179 ft 8 1/2 in.
- Steam generator support level — Vertical - el 171 ft 9 in.
Lower Lateral - el 194 ft 11 in.
Upper Lateral - el 217 ft 4 3/4 in.
- Polar crane girder — el 323 ft 0 in.

The response spectra at elevations 169 ft 0 in., 195 ft 0 in. (internal structure), 220 ft 0 in. (internal structure), and 323 ft 0 in. (exterior shell) would cover all of the above locations. The comparisons between the response spectra curves obtained for these levels from the S3 and S4 method and the impedance method are shown in figures 3D-79 through 3D-90.

B. Control Building. In the control building, the following levels are selected for comparison purpose.

- Top of base mat — el 180 ft 0 in.
- Control room floor — el 220 ft 0 in.
- Mechanical equipment room floor — el 260 ft 0 in.

The comparison between the response spectra curves obtained for these levels from the S3 and S4 method and the impedance method are shown in figures 3D-91 through 3D-99.

C. The comparison of the response spectra obtained from the S3 and S4 method, in which a scaling factor of 1.5 had been used, with those obtained from the impedance method, shows that the spectra from the S3 and S4 method, in general, envelop the spectra from the impedance method. The differences between the response spectra obtained from the two different methods could be attributed to the following.

1. Embedment effects

   In the calculations for the impedance method, embedment effects are neglected even though the structures are embedded in the soil to a certain depth. In the S3 and S4 method, where the FLUSH program is used to model the soil as an assemblage of finite elements, the structure is modeled in its embedded condition. As a result, the response spectra from FLUSH indicate a higher system frequency and lesser amplification due to the increased stiffness and damping caused by the embedment.

2. Structure to structure interaction effects

   In the S3 and S4 method, adjacent structures are modeled together in a FLUSH model and as such the resulting response spectra account for possible structure-soil-structure interaction effects. In the impedance method, the analysis is performed on the isolated structure and hence the effect of the adjacent structures is not considered. The effects of the structure-to-structure interaction are implicitly reflected in the FLUSH response spectra in terms of modifications of response and system frequencies, if such effects are significant.

3. Input motion

   In the S3 and S4 method, the design ground motion, (0.20 g, Regulatory Guide 1.60 spectra) is defined in the free field at the finished grade level, whereas in the impedance method, the design motion is assumed to be the free field motion at the base of the soil springs. Further, in the S3 and S4 method the resulting response spectra from FLUSH are multiplied by a scaling factor of 1.5.

4. Frequency independent impedances

   In the impedance method, the frequency independent impedances are selected based on the strain dependent shear moduli obtained from the soil column study in which the design motion is specified at the finished grade level.

5. Methodology

   In addition to the above, there are differences in the assumptions made in the finite element approach and the impedance approach to solving the soil-structure interaction problem.

3D.2.1.5 Conclusions

The comparison of the response spectra obtained for the containment and control building using the S3 and S4 method and the impedance method shows that, in general, the response spectra from the S3 and S4 method and those from the impedance method show similar characteristics despite the fact that there are differences in the methods of modeling and analysis. It is therefore concluded that the seismic analysis performed in accordance with S3 and S4 would form a conservative design basis.

3D.2.2 SENSITIVITY STUDY

3D.2.2.1 Introduction

In accordance with your request, a series of ground response studies have been performed to determine the computed variation of ground motion with depth at VEGP site. The analyses were made for two conditions:

A. A series of seven analyses were made in which recorded rock motions were used to establish bedrock motions, which were then propagated upward through the site soils from the underlying bedrock.

B. A series of eight analyses were made in which recorded motions on soil deposits with generally similar characteristics to those at the VEGP site were deconvolved through the soil profile.

The computed responses from the two procedures were compared with each other and with the results of the seismic motions proposed for use in the design of the plant. In all cases, the ground surface motions were scaled to a peak ground surface acceleration of 0.2 g to facilitate the comparisons outlined above.

3D.2.2.2 Soil Profile

The soil profile at the VEGP site is shown in figure 3D-100. After excavation and backfilling to plant grade, the soils will consist of about 100 ft of sand, underlain by about 65 ft of marl and then further sand extending down to very large depths. The actual depth of bedrock is not known but for the purposes of this study it was considered to be at about 600 ft below the ground surface and to have a shear wave velocity of about 3000 ft/s. Test data have shown that the shear wave velocity in the lower sands achieves a value of 1800 ft/s at a depth of 200 to 300 ft and assuming that the nature of the sand remains the same, these results would indicate that it would be likely to increase to a value of about 2600 ft/s at a depth of 600 ft due to the increase in overburden pressure alone. On this basis, a depth to rock of about 600 ft was considered reasonable. Increases in depth of 100 or 200 ft above the selected value would have very little influence on the results of the analysis in the upper few hundred ft of the deposit.

Soil properties assigned to the various layers shown in figure 3D-100 were those previously established by tests for the various layers and used in previous VEGP reports.

3D.2.2.3 Ground Motions

For the purpose of the analyses, eight accelerograms recorded on deep soil sites similar to the VEGP site were chosen for deconvolution studies. These motions were also chosen because the peak ground accelerations were reasonably close to the SSE acceleration of 0.2 g selected for the VEGP site. The eight records selected for these studies are listed as Analyses D-1 to D-8 in table 3D-1. They include records from the El Centro site in the El Centro earthquakes of

1934 and 1941, records from the Athenaeum site at the California Institute of Technology and the Holiday Inn site on Orion Boulevard in the San Fernando earthquake of 1971, two records recorded at the Olympia Highway Laboratory in different earthquakes, and records from Ferndale and Humboldt Bay. Details of the recording stations and the particular earthquakes producing the records are given in table 3D-1. In all cases, the record selected was scaled to have a peak acceleration of 0.2 g and used as a surface control motion for a deconvolution analysis on the basis of vertical shear wave propagation. Spectra for the ground surface motions obtained in this way are shown in figures 3D-101 through 3D-108.

The other seven analyses were made using seven different recorded rock motions to determine base rock excitation and the resulting response of the deposit due to upward wave propagation. For this purpose, the rock motions were considered to develop at an outcrop close to the site, the motions were then used to determine the motions at the base of the 600 ft soil deposit, and the motions throughout the deposit were computed for these conditions. The computer program SHAKE was used for all analyses.

The rock motions used for these studies were selected from three different earthquakes and are listed as analyses U-1 to U-7 in table 3D-2. They include records from the Temblor Station and San Luis Obispo in the Parkfield earthquake of 1966, three records from the C.I.T. Seismological Laboratory, Lake Hughes Station No. 4, and Griffith Park in the San Fernando earthquake of 1971, a record from the Taft Station in the Kern County, California earthquake of 1952, and a record from the Castaic Station in the San Fernando earthquake of 1971.

The original records from these stations were initially scaled to a peak rock acceleration of 0.2 g and the resulting ground surface motions computed at representative levels throughout the soil profile. Values of peak ground surface accelerations obtained in this way ranged from 0.2 g to 0.28 g. The computed motions were therefore scaled proportionally to produce a peak ground surface acceleration of 0.2 g. The low scaling factors used in this latter operation were not considered to have any significant influence on the results which might have been obtained if repeated trials using different levels of input rock accelerations had been used until each analysis had produced a computed peak ground surface acceleration of 0.2 g.

Response spectra for the scaled rock motions (producing 0.2 g peak acceleration at the ground surface as described above) are shown in figures 3D-109 through 3D-115. It was recognized that two of the motions used (the Taft and Castaic records) were not truly rock records but they were recorded on shallow depths of stiff soil and have essentially the same characteristics of rock records. It was also thought that these two records might simulate motions in the base rock if the actual depth of the soil profile were somewhat deeper than the value of 600 ft used in the analyses.

3D.2.2.4 Variation of Peak Ground Acceleration with Depth

The computed variations of peak ground acceleration with depth determined by the eight deconvolution analyses described above are shown in table 3D-3 and those for the upward wave propagation analyses from rock in table 3D-4. As described above, all analyses were adjusted to produce a peak ground surface acceleration of 0.2 g, providing a consistent basis for comparison. The results show some variations of peak acceleration with depth within the soil profile but overall no major differences. A comparison of the mean values determined by the deconvolution analyses with those determined by the upward wave propagation analyses is shown in figure 3D-123. Again it may be seen that the variations of mean peak acceleration with depth for the two separate studies are generally similar.

3D.2.2.5 Comparison of Ground Surface Motions with those Developed at 76 ft Depth

For each of the 15 analyses, comparisons of the response spectrum for the ground surface motions with that for the computed motions at a depth of 76 ft in the soil profile (the foundation level for the containment building) are shown in figures 3D-101 through 3D-108 and figures 3D-116 through 3D-122. Figures 3D-101 through 3D-108 show the results of the deconvolution studies while figures 3D-116 through 3D-122 show the results of the upward propagation studies. It is readily apparent that all of the analyses show a substantial reduction of the intensity of shaking developed at a depth of 76 ft and that the reduction is generally comparable for each of the analyses.

To provide a collective basis for assessing the significance of these results, computations were made to determine the mean plus 1 standard deviation spectral shape for the suite of motions determined by the eight deconvolution analyses and separately for the seven upward wave propagation analyses for (1) the ground surface motions, (2) the motions computed for a depth of 40 ft below the ground surface and (3) the motions computed for a depth of 76 ft below the ground surface. The results of these studies are shown in figures 3D-124 and 3D-125. It may be noted that the spectra for the ground surface motions are reasonably uniform over the period range from about 0.2 to 0.5 s but the spectra for the motions at depths of 40 and 76 ft contain significant valleys at periods of about 0.18 and 0.3 s in both cases. This reflects the influence of wave propagation effects at the ground surface.

It is interesting to note that the mean + 1σ spectrum for the suite of eight scaled ground surface motions and the seven computed ground surface motions determined in this sensitivity study are reasonably similar to that specified by Regulatory Guide 1.60, as evidenced by the comparison in figure 3D-126. However both spectra are somewhat lower than the Regulatory Guide 1.60 spectra for frequencies higher than about 5 Hz reflecting the filtering effects of high frequencies which tends to occur in deep soil deposits. As a result the use of the Regulatory Guide 1.60 spectrum as a design basis may be considered conservative in such cases. Finally, as shown by the comparative spectra in figure 3D-127, the mean + 1σ spectrum determined for the two groups of computed motions at 76 ft depth in this analytical study, i.e., eight deconvolution analyses of suitably scaled ground surface motion records and seven ground response analyses using suitably scaled rock motions) are very similar in shape to each other and to the computed spectrum at 76 ft depth obtained in previous design studies for the VEGP obtained by deconvolution of an artificial accelerogram representative of the Regulatory Guide 1.60 spectrum.

3D.2.2.6 Conclusion

The good agreement between the results of the ground motion sensitivity studies described above and the results obtained by deconvolution of the Regulatory Guide 1.60 spectrum supports the use of the latter spectrum and motions obtained by deconvolution of it for the design of the VEGP. On this basis it seems reasonable to conclude that the seismic design procedures previously proposed by Bechtel for the design of the VEGP provide a suitability conservative basis for design.

3D.2.3 METHODOLOGY TO ACCOUNT FOR TORSION CAUSED BY THE SEISMIC WAVE PROPAGATION EFFECTS

3D.2.3.1 Introduction

The NRC in their letter dated March 27, 1978, to Georgia Power Company requested that "all safety-related structures, systems and components be designed to resist a static seismic torsional moment which is not less than that required by the latest edition of the Uniform Building Code." Further, the NRC stated that this seismic torsional moment be included as a separate moment which is independent of that resulting from the eccentricity between the center of mass and the center of rigidity of the safety-related structures.

3D.2.3.2 Safety-Related Structures

The seismic analyses of the structures are performed on the three dimensional structure models that account for the eccentricities between the centers of mass and the centers of rigidity of the structures. The accelerations obtained from these analyses at all levels are first calculated. In the design, then the actual eccentricity is increased by 5 percent of the maximum plan dimension at that level and the design static seismic torsional moment is computed as the product of the augmented eccentricity and the story shear.

3D.2.3.3 Safety-Related Equipment, Systems, and Components

Since the intent of the NRC's request on the 'additional torsional requirements' is to account for the torsional motion imparted to the structure due to the effects of seismic wave propagation, it is to be noted that this would also impact the horizontal in-structure response spectra used for equipment qualification. The procedure used to obtain the effect of this torsional ground motion is described below.

A three dimensional lumped parameter model of the structure with "soil springs" is utilized to compute the torsional spectra. The structure model accounts for the eccentricities between the centers of mass and the centers of rigidity of the structure. The translational as well as the rotational (including torsional) stiffness and inertial characteristics are modeled. The foundation impedances consist of three translational (two horizontal and the vertical) and three rotational (two rocking and the torsional) springs, and are based on the "mean" soil properties.

The model is analyzed for the design horizontal ground motion time history conforming to the Regulatory Guide 1.60 horizontal response spectra applied in the free-field at the foundation level of the structure. The base shear computed from this analysis when multiplied by 5 percent of the maximum plan dimension at the foundation level yields the incremental static torsional moment (T_s) at that level.

Then a torsional ground motion time history conforming to the Regulatory Guide 1.60 horizontal response spectra is applied again in the free field at the foundation level of the structure. The maximum dynamic torsional moment (T_d) at the base of the structure is computed from this dynamic analysis.

The magnitude of the torsional ground motion is adjusted such that T_d at the base of the structure resulting from the torsional ground motion analysis is equal to the T_s resulting from the 5-percent eccentricity. The resulting time history response from the torsional degree of freedom of the base node would then represent the torsional response of the basemat. Multiplying this by the distance along the N-S/E-W direction of the extreme point in the building to the lumped mass node would give the maximum possible additional E-W/N-S horizontal time history response of the floor. From this, the "additional horizontal in-structure response spectra" can be computed.

Now that the magnitude of the torsional ground motion has been established, the torsional responses of the nodes at different levels of the building from the torsional ground motion analysis are used with the respective "extreme point distances" along the N-S/E-W direction to compute the "additional horizontal in-structure response spectra" at these levels.

The amplification of the torsional response of the structure as a function of height tends to be smaller than the amplification of the horizontal response of the structure. Therefore, as an added conservatism, the torsional input ground motion is increased such that the ratio between the maximum torsional acceleration at a given node (caused by the torsional ground motion) to the maximum horizontal acceleration at that node (caused by the horizontal ground motion) is maintained the same as at the foundation level of the structure.

The computed "additional horizontal in-structure response spectra" to account for the torsional ground motion effects are added to the horizontal in-structure response spectra obtained using the methods specified in VEGP PSAR Supplements 3 and 4, before the broadening of the peaks and smoothing of the curves are done. The peaks of the response spectra resulting from the addition of these two spectra are then broadened and the curves smoothed to arrive at the "final design in-structure response spectra" for the horizontal direction.

In the computation of the "additional horizontal in-structure response spectra" to be used for equipment mounted on a specific location, the actual distance of this location from the lumped mass node may be used instead of the "extreme point distance" at that level.

3D.3 SEISMIC RESPONSE SPECTRA COMPARISON

This section is provided in response to the request made by the NRC during their Structural Audit and Design Review on December 4, 1984. Specifically, the NRC requested that a comparison of the VEGP design in-structure response spectra with the impedance method response spectra provided in the Confirmatory Study (section 3D.2) be made.

The basis for the generation of curves identified as impedance method is described in the confirmatory study. The basis for the generation of the design in-structure response spectra is described in subsection 3.7.B. The horizontal design response spectra are an envelop of the E-W and N-S horizontal response spectra. A comparison of these response spectra is provided in figure 3D-128 through 3D-148.

The comparison of the design in-structure response spectra with the response spectra obtained using the impedance method shows that, in general, the design response spectra envelope those from the impedance method. Specifically, the following observations are made:

- The comparison of the horizontal response spectra curves shows that in the frequency range of 2 cps and higher, the design spectra acceleration values are significantly higher than the values obtained from the impedance method demonstrating the added conservatism in the design of structures, equipment, and systems. The slight difference in the frequency range below 2 cps is attributed to the fact that embedment effects were not considered in the impedance method (see section 3D.2). Embedment effects increase the fundamental soil-structure system frequency and, due to the increased damping, lower the response. Thus, it is expected that inclusion of the embedment effects in the impedance method would result in spectra that will be enveloped by the design criteria.

- The comparison of the vertical response spectra curves shows that overall, the design spectra acceleration values are significantly higher than the values obtained from the impedance method. The minor difference in certain frequency range exhibited in figure 3D-130 becomes insignificant if the comparison is made with the impedance method spectra before peak broadening. It should be noted that peak broadening need only be performed for the development of design spectra to account for uncertainties in modeling techniques.

In conclusion, the seismic analysis performed in accordance with VEGP methodology has resulted in conservative design of structures, equipment, and systems.

TABLE 3D-1
RECORDS USED FOR DECONVOLUTION ANALYSES

Analysis No.	Recording Station	Earthquake
D-1	C.I.T. Athenaeum	San Fernando, 1971
D-2	El Centro	El Centro, 1934
D-3	El Centro	El Centro, 1940
D-4	Holiday Inn (Orion Blvd)	San Fernando, 1971
D-5	Humboldt Bay	Ferndale, 1975
D-6	Ferndale City Hall	Ferndale, 1954
D-7	Olympia Highway Lab.	Olympia, 1949
D-8	Olympia Highway Lab.	Pacific N.W., 1965

TABLE 3D-2

RECORDS USED FOR UPWARD WAVE PROPAGATION ANALYSES

Analysis No.	Recording Station	Earthquake
U-1	Temblor	Parkfield, 1966
U-2	C.I.T. Seismological Lab.	San Fernando, 1971
U-3	Lake Hughes, No. 4	San Fernando, 1971
U-4	Griffith Park	San Fernando, 1971
U-5	San Luis Obispo	Parkfield, 1966
U-6	Taft	Kern County, 1952
U-7	Castaic	San Fernando, 1971

TABLE 3D-3

COMPUTED VARIATIONS OF PEAK ACCELERATIONS (g)
WITH DEPTH FROM DECONVOLUTION ANALYSES

	Analysis No.								
Depth (ft)	D-1	D-2	D-3	D-4	D-5	D-6	D-7	D-8	Average
0	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20
10	0.18	0.18	0.19	0.19	0.20	0.19	0.18	0.17	0.18
20	0.15	0.15	0.19	0.18	0.19	0.19	0.16	0.13	0.17
30	0.15	0.14	0.18	0.17	0.18	0.18	0.15	0.12	0.16
40	0.14	0.12	0.16	0.16	0.19	0.18	0.13	0.11	0.15
51	0.12	0.11	0.15	0.15	0.16	0.17	0.13	0.10	0.14
62	0.10	0.09	0.13	0.14	0.14	0.16	0.12	0.10	0.12
76	0.09	0.09	0.12	0.13	0.12	0.15	0.11	0.11	0.12
92	0.10	0.09	0.13	0.13	0.10	0.14	0.10	0.11	0.13
110	0.10	0.09	0.14	0.12	0.09	0.14	0.08	0.10	0.11
120	0.10	0.09	0.14	0.11	0.08	0.14	0.08	0.10	0.11
140	0.09	0.09	0.13	0.10	0.08	0.13	0.08	0.09	0.10
162	0.10	0.08	0.13	0.10	0.09	0.10	0.07	0.10	0.10
267	0.13	0.15	0.13	0.10	0.10	0.10	0.10	0.09	0.11
600	0.18	0.32	0.52	0.25	0.13	0.17	0.14	0.13	

TABLE 3D-4

COMPUTED VARIATIONS OF PEAK ACCELERATIONS (g)
WITH DEPTH FROM UPWARD WAVE PROPAGATION ANALYSES

	Analysis No.							
Depth (ft)	U-1	U-2	U-3	U-4	U-5	U-6	U-7	Average
0	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20
10	0.19	0.19	0.18	0.19	0.18	0.19	0.18	0.19
20	0.16	0.18	0.16	0.18	0.17	0.17	0.15	0.17
30	0.14	0.17	0.12	0.17	0.16	0.16	0.12	0.15
40	0.11	0.15	0.10	0.15	0.15	0.14	0.10	0.13
51	0.07	0.13	0.11	0.13	0.14	0.13	0.08	0.11
62	0.06	0.11	0.12	0.11	0.13	0.12	0.07	0.10
76	0.07	0.09	0.14	0.09	0.11	0.11	0.06	0.10
92	0.08	0.10	0.14	0.09	0.09	0.11	0.06	0.10
110	0.08	0.10	0.14	0.09	0.11	0.11	0.06	0.10
120	0.08	0.10	0.13	0.10	0.11	0.16	0.07	0.11
140	0.09	0.09	0.11	0.10	0.11	0.16	0.07	0.12
162	0.09	0.10	0.08	0.09	0.11	0.11	0.08	0.09
267	0.11	0.10	0.11	0.10	0.13	0.11	0.10	0.11
600	0.12	0.12	0.13	0.13	0.13	0.13	0.09	0.12

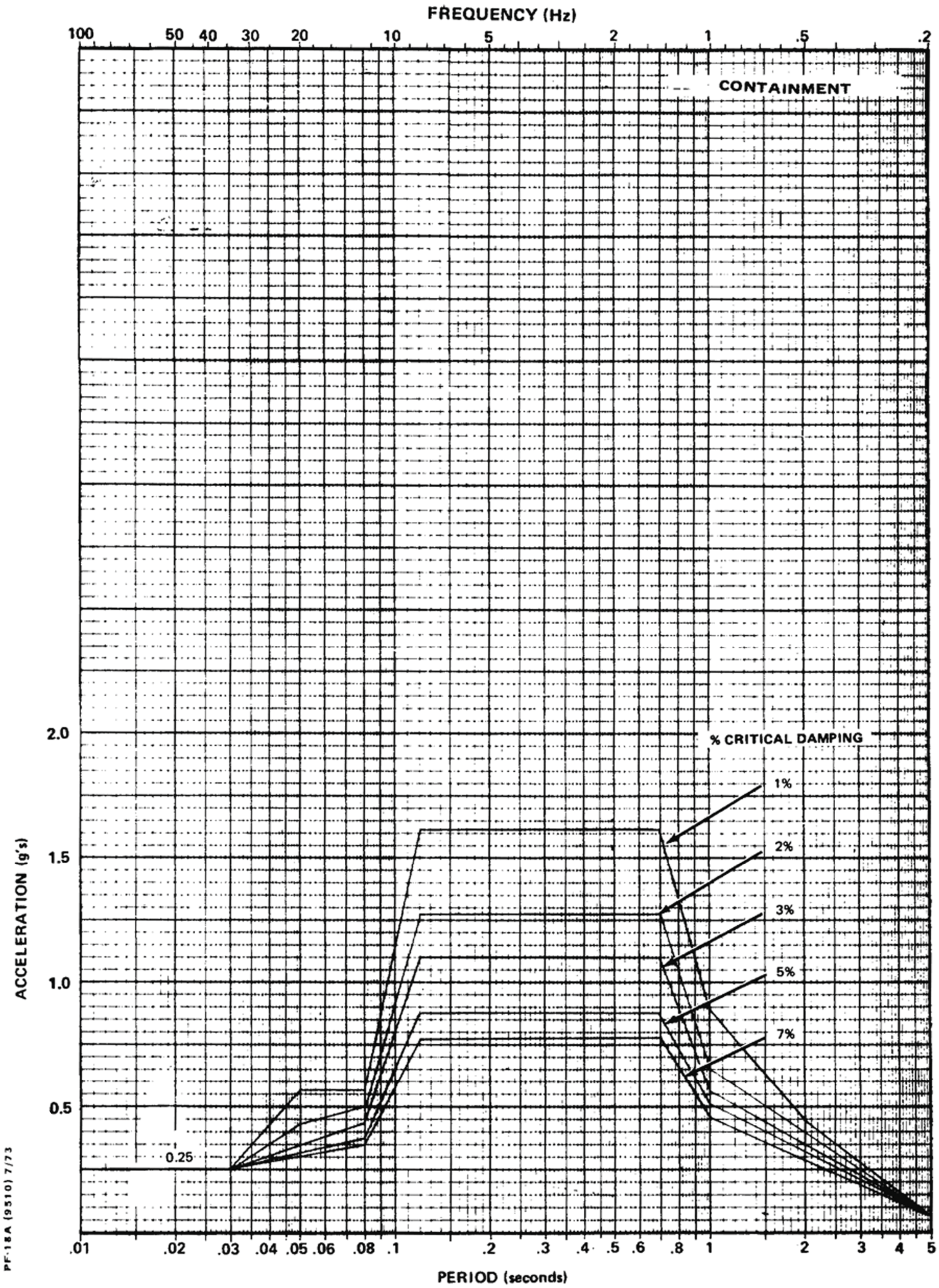


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FIGURE 3D–1

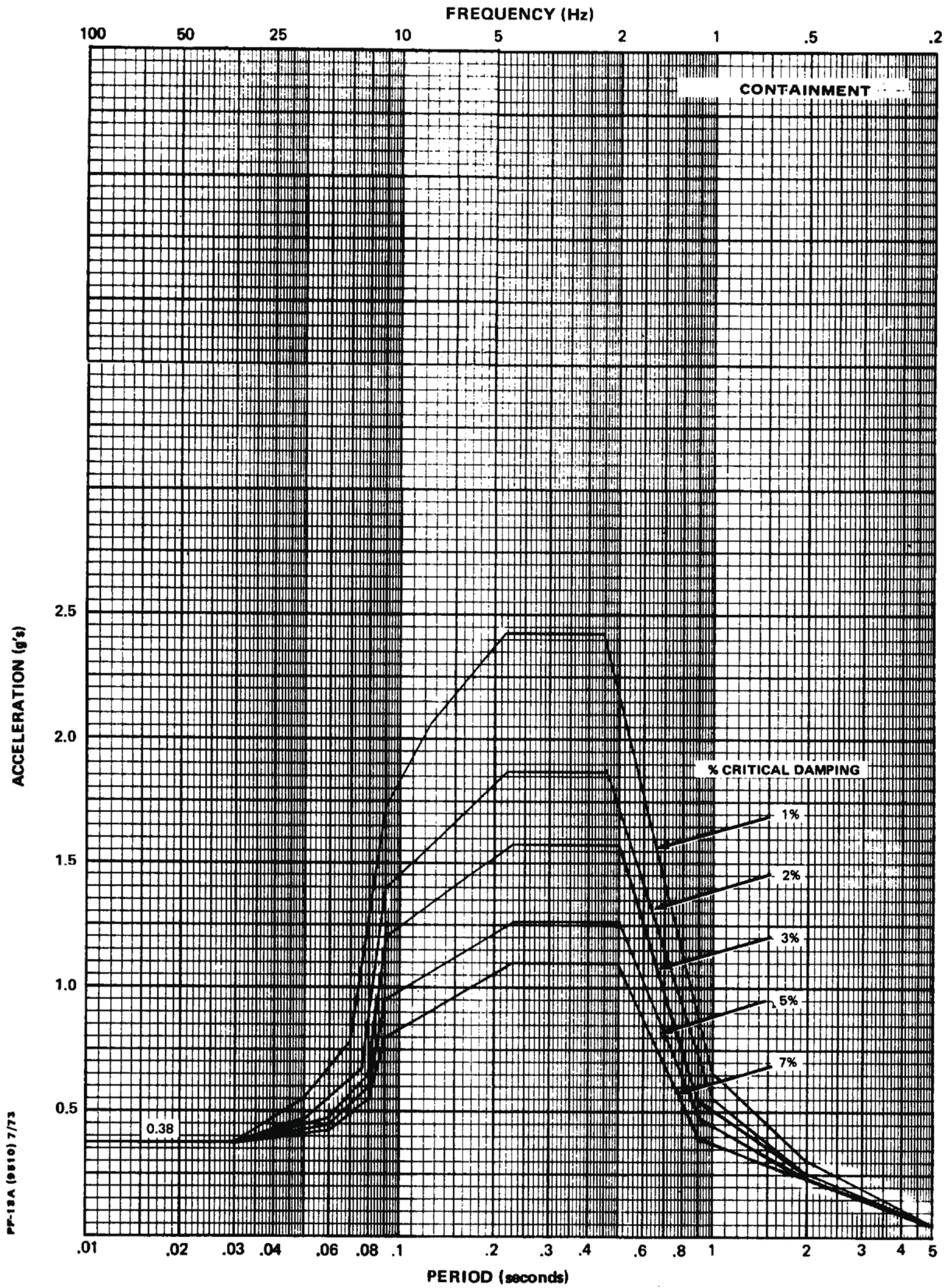

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FIGURE 3D–2

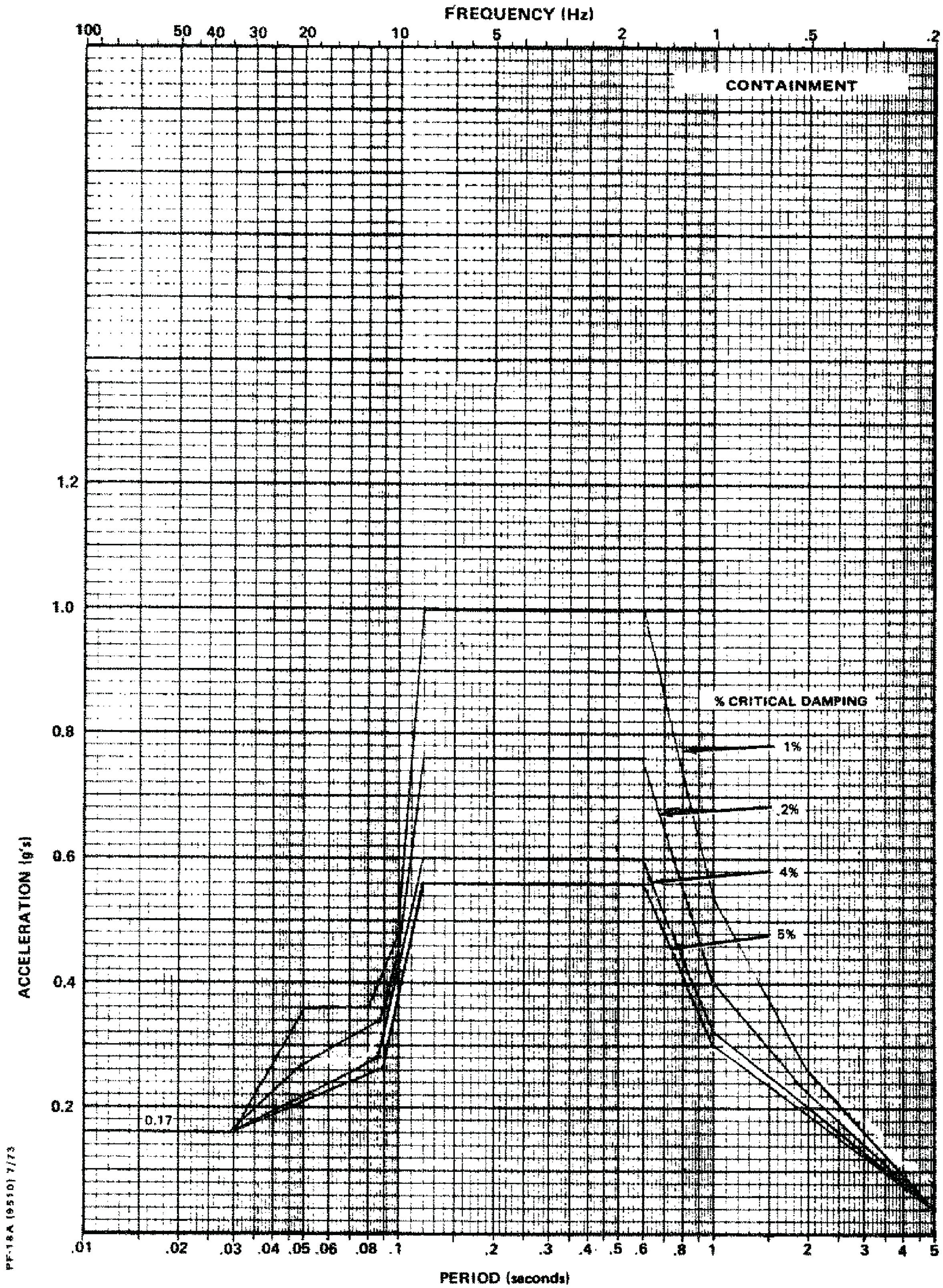

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FIGURE 3D–3

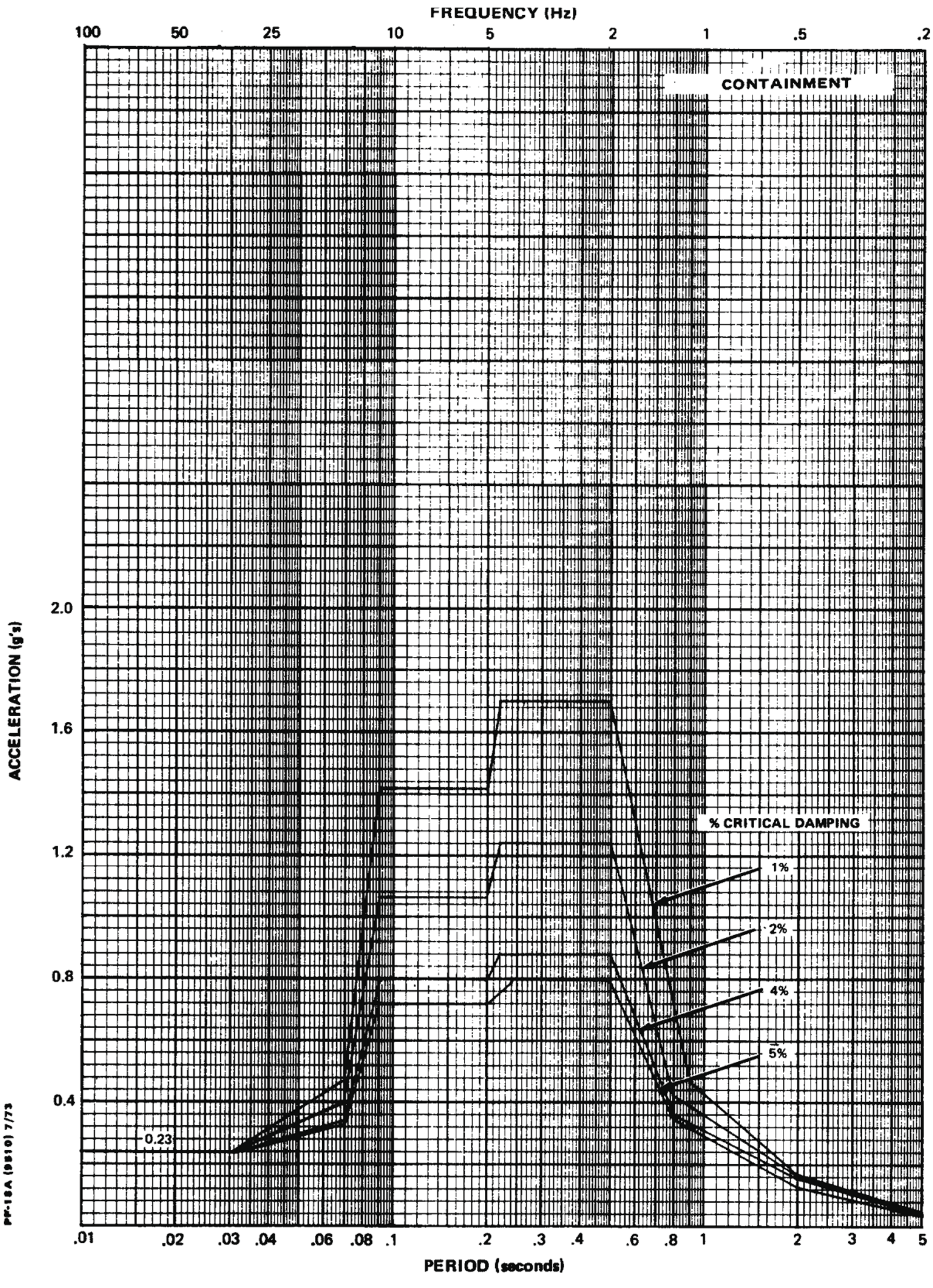

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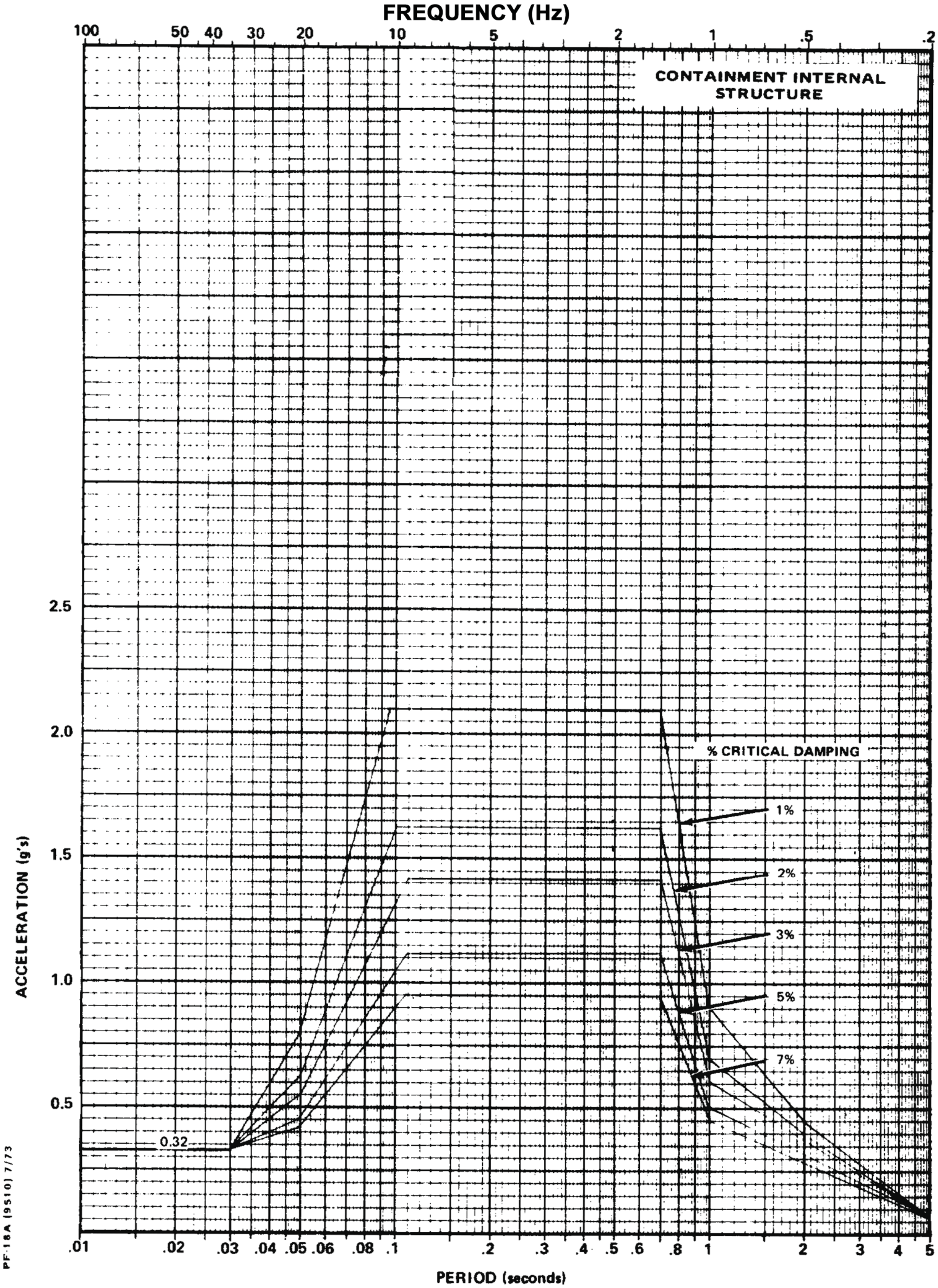
FIGURE 3D–4



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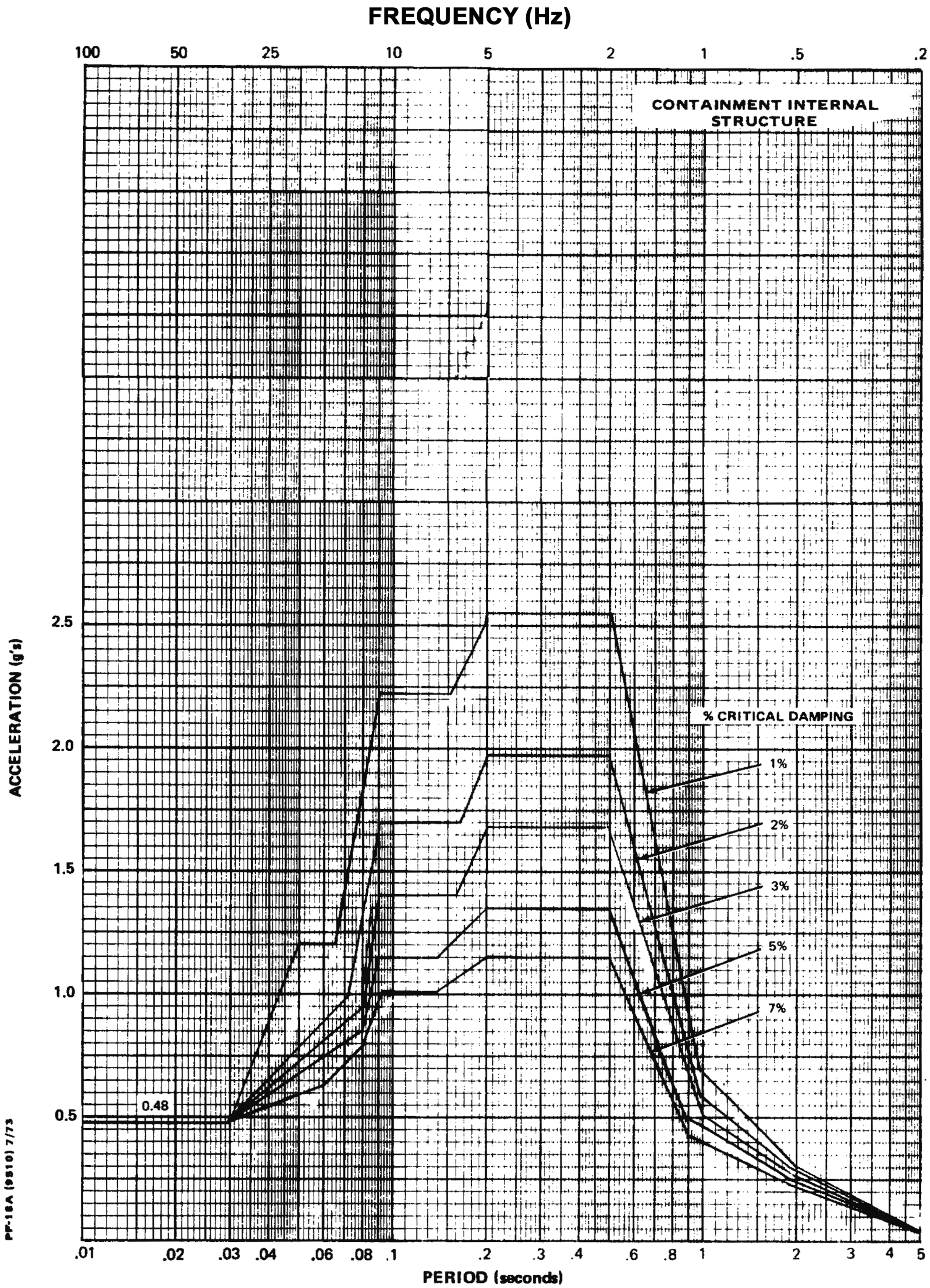

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SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	SAFE SHUTDOWN EARTHQUAKE VERTICAL ACCEL. RESPONSE SPECTRA EL. 220 FT 0 IN. FIGURE 3D–6

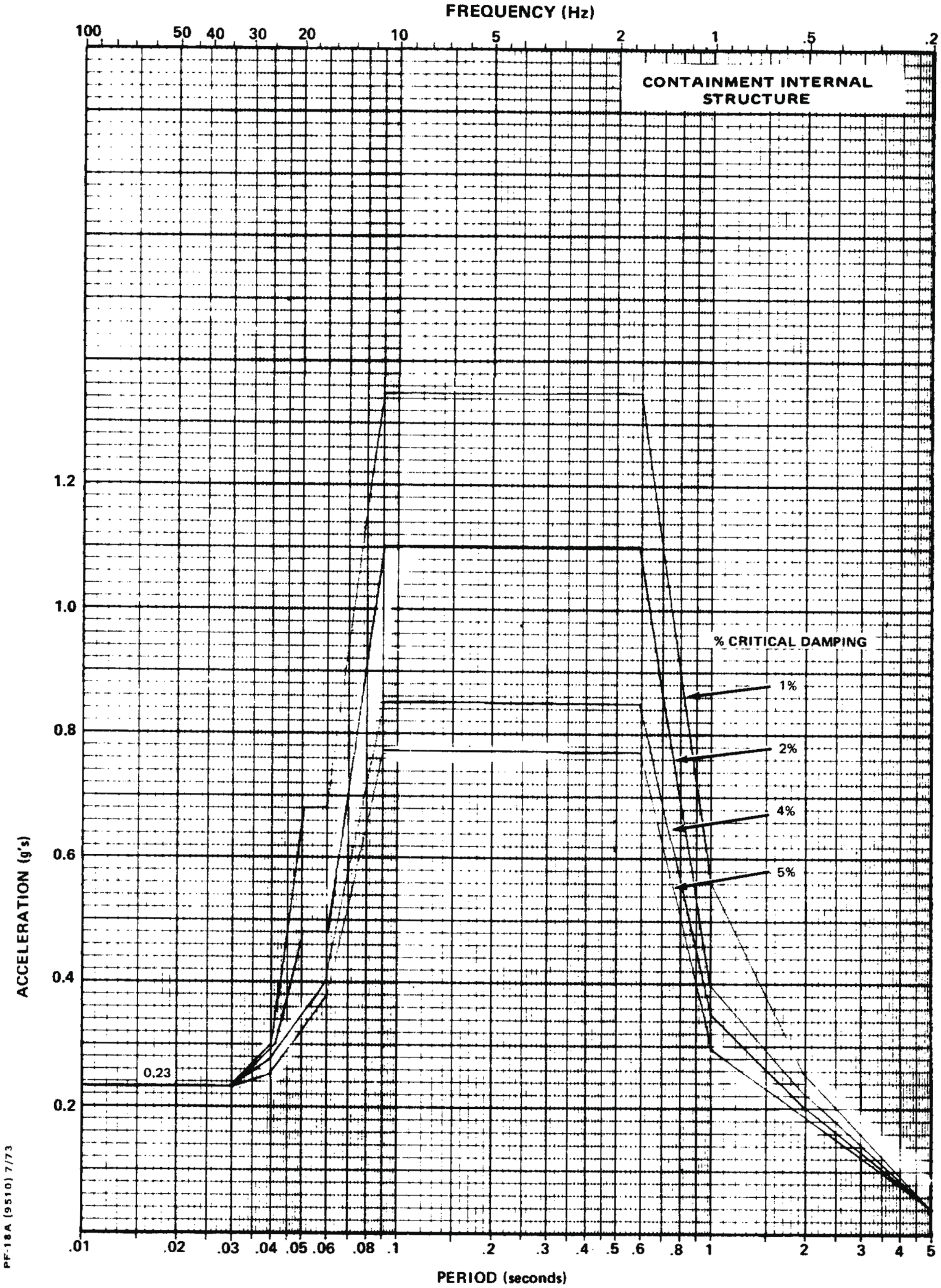

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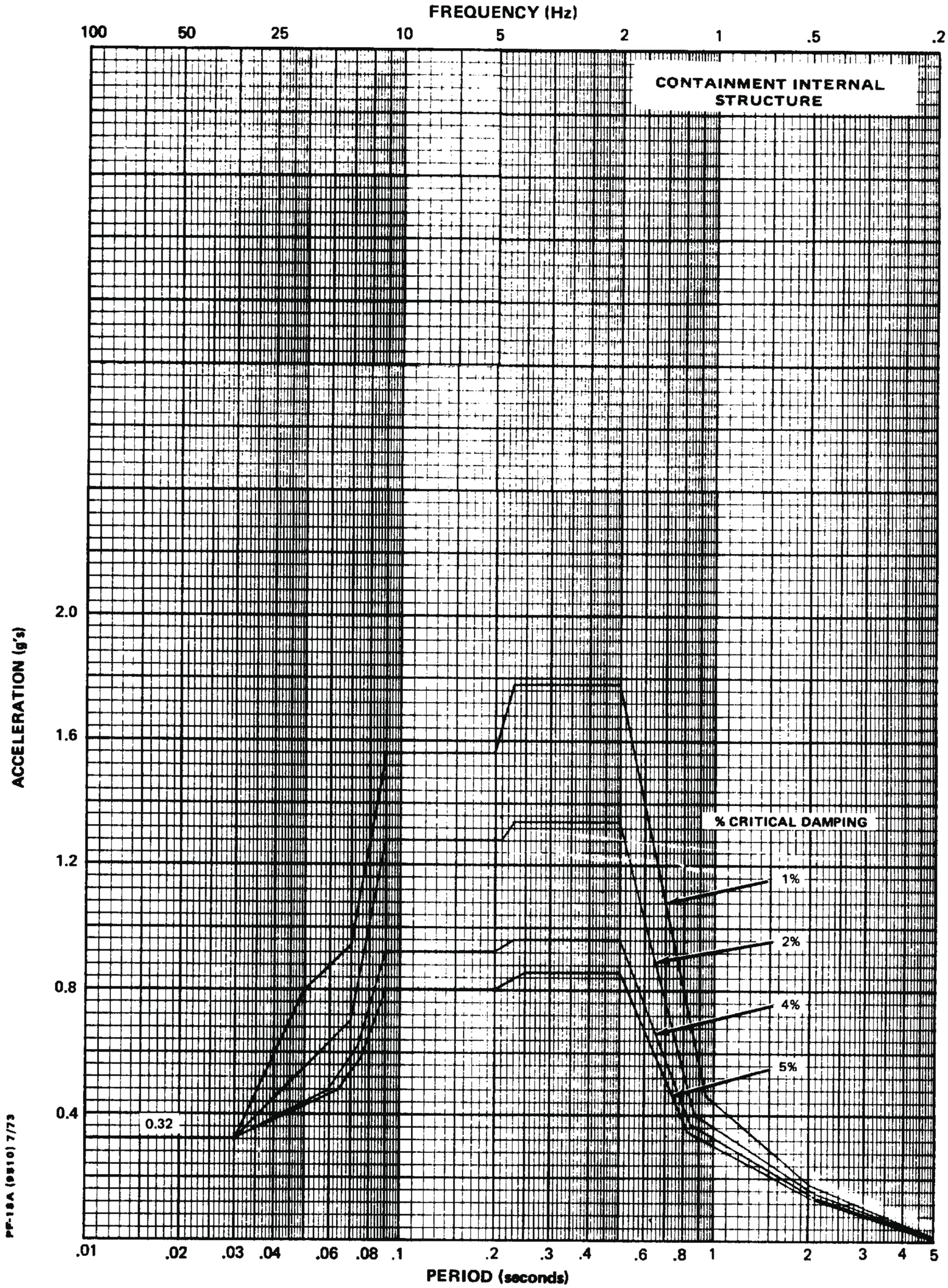
FIGURE 3D–7



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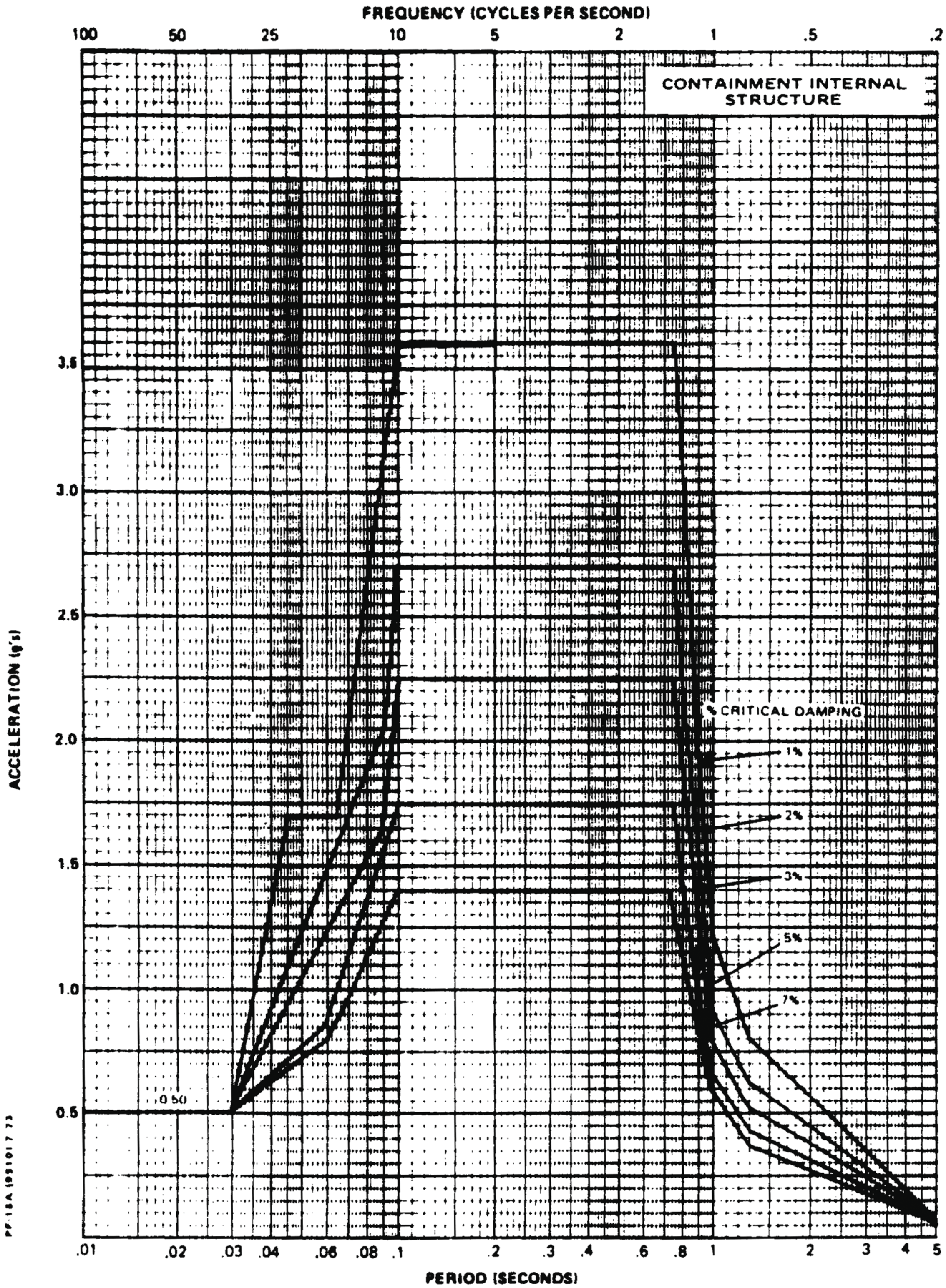

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FIGURE 3D–9

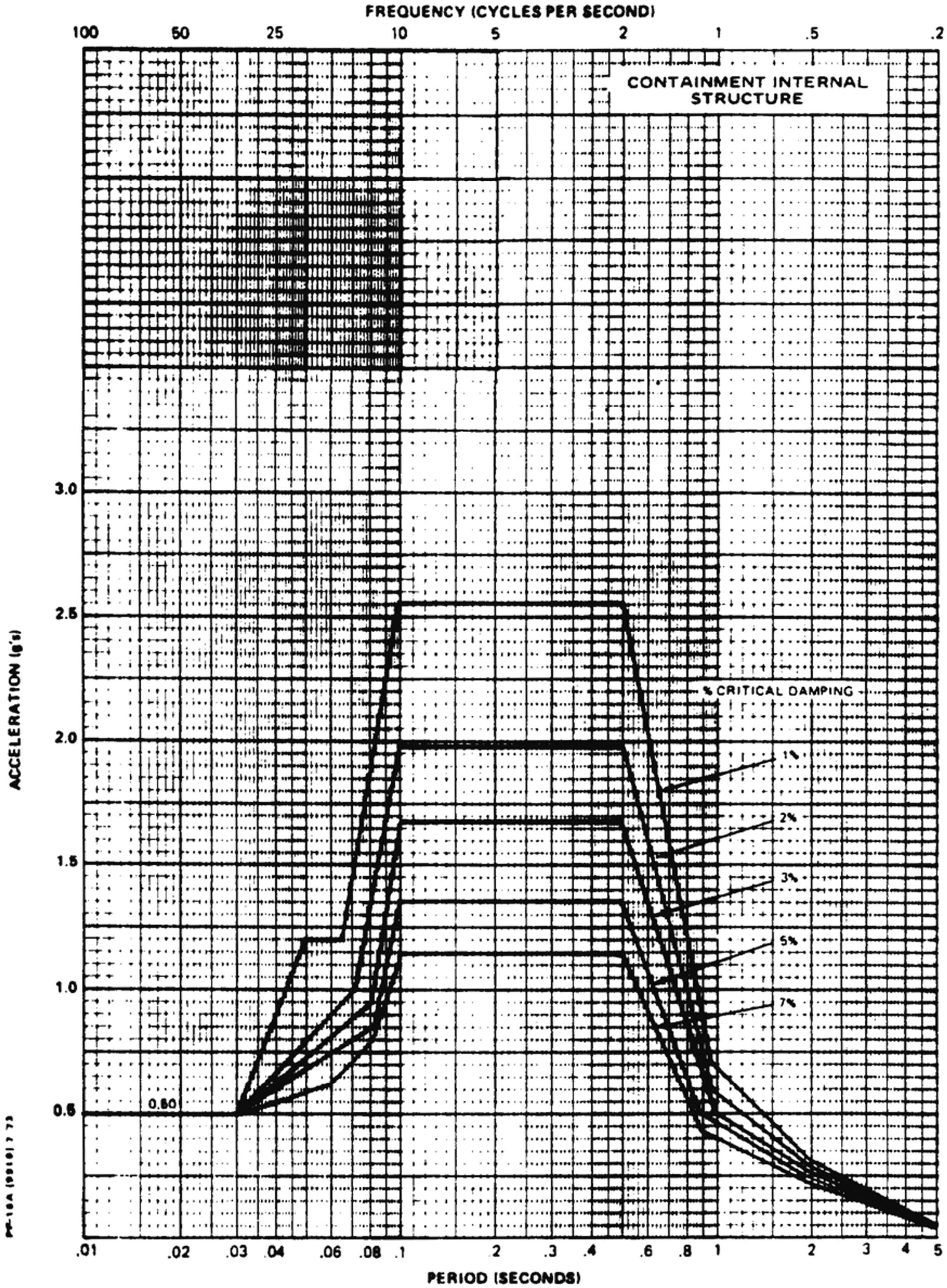

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		FIGURE 3D–10

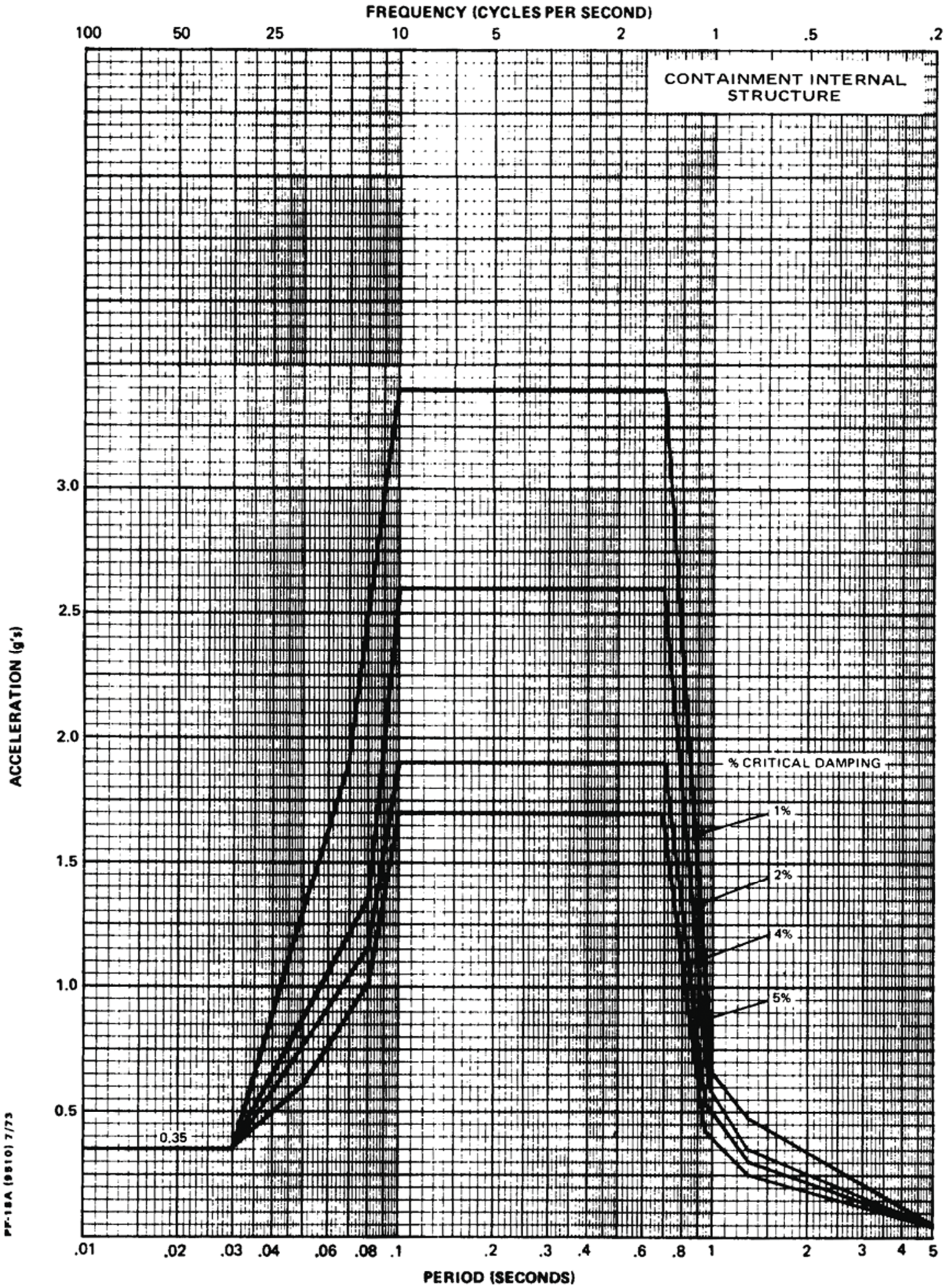

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FIGURE 3D–11

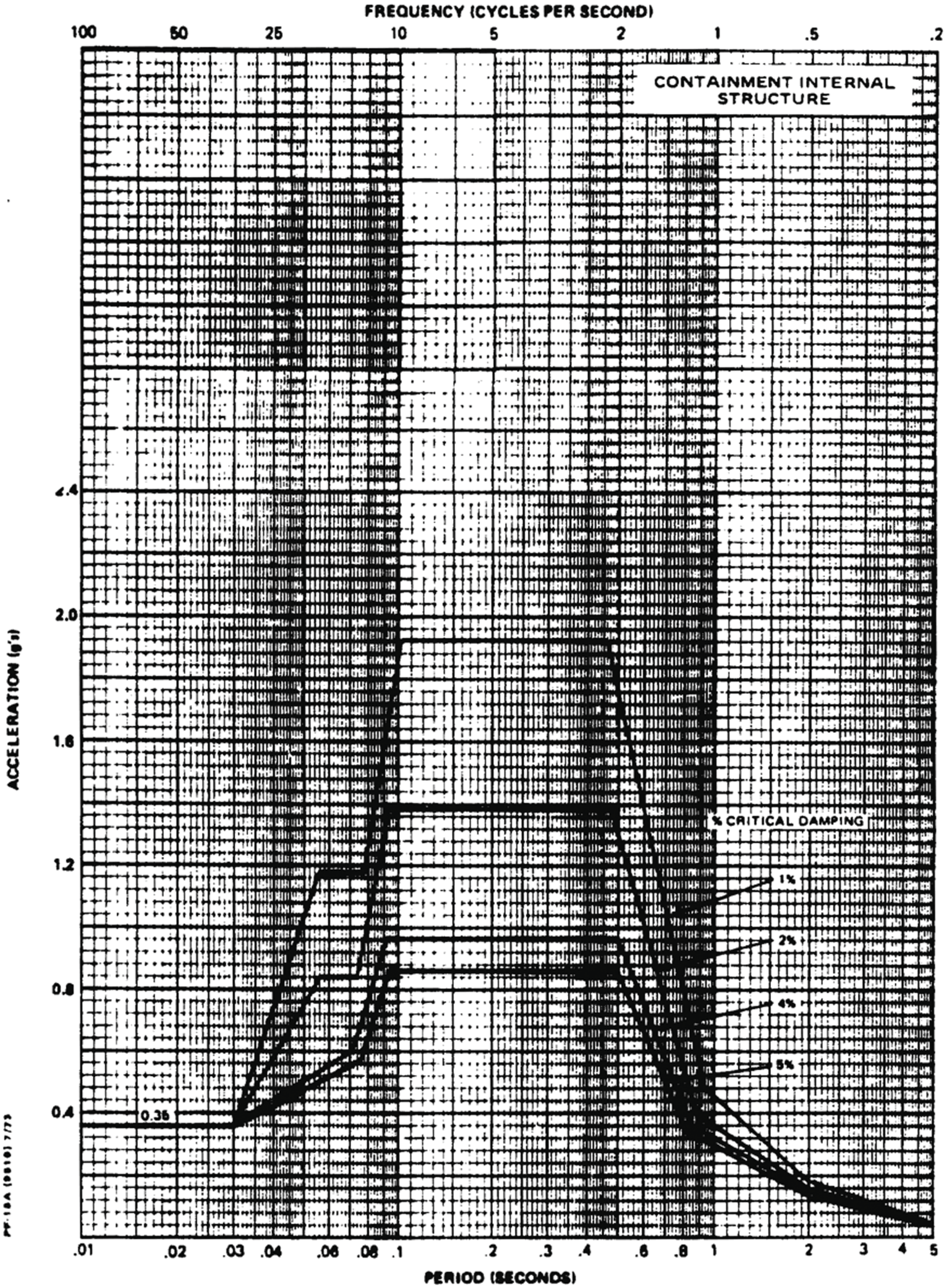

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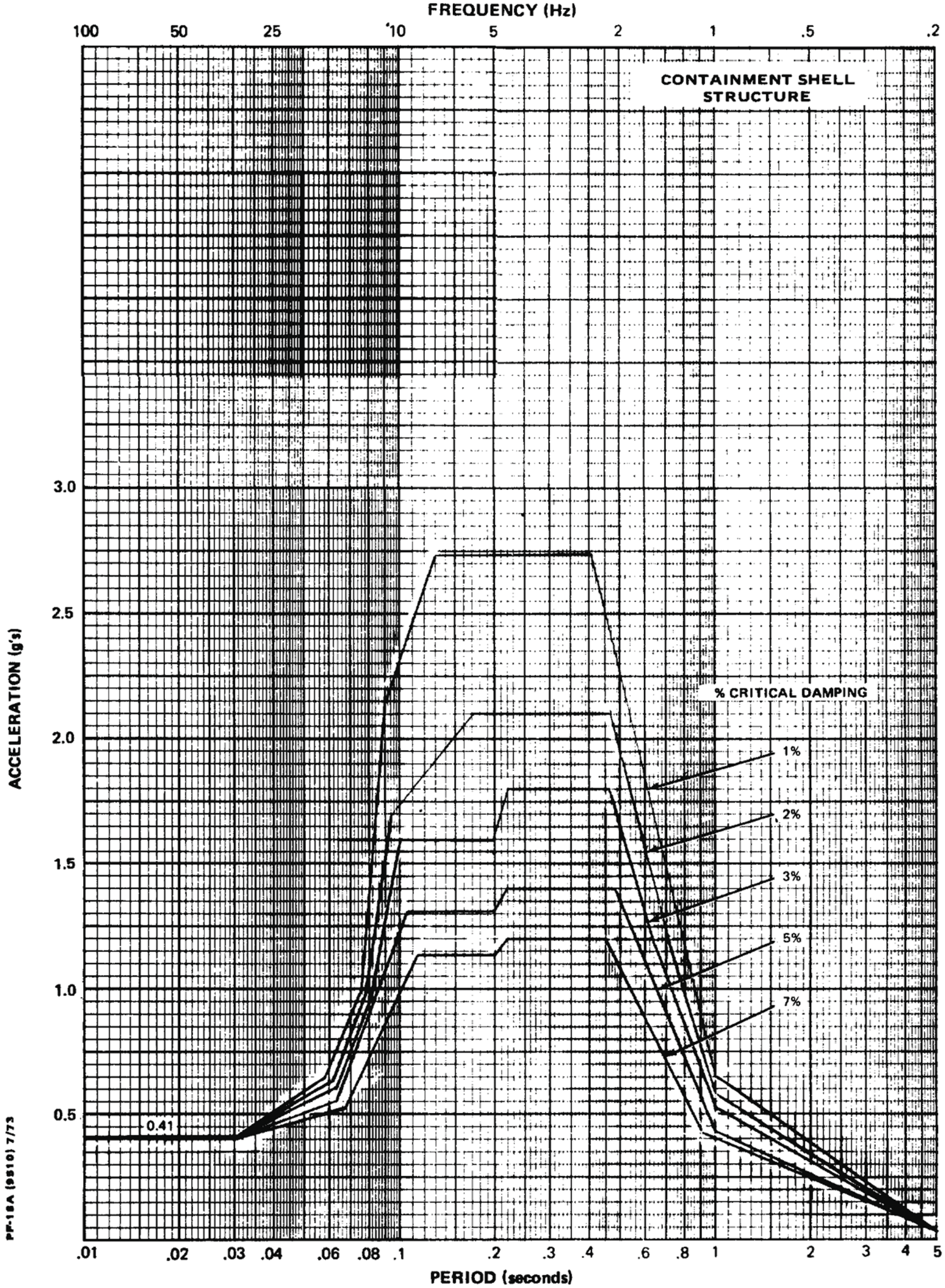
FIGURE 3D–12



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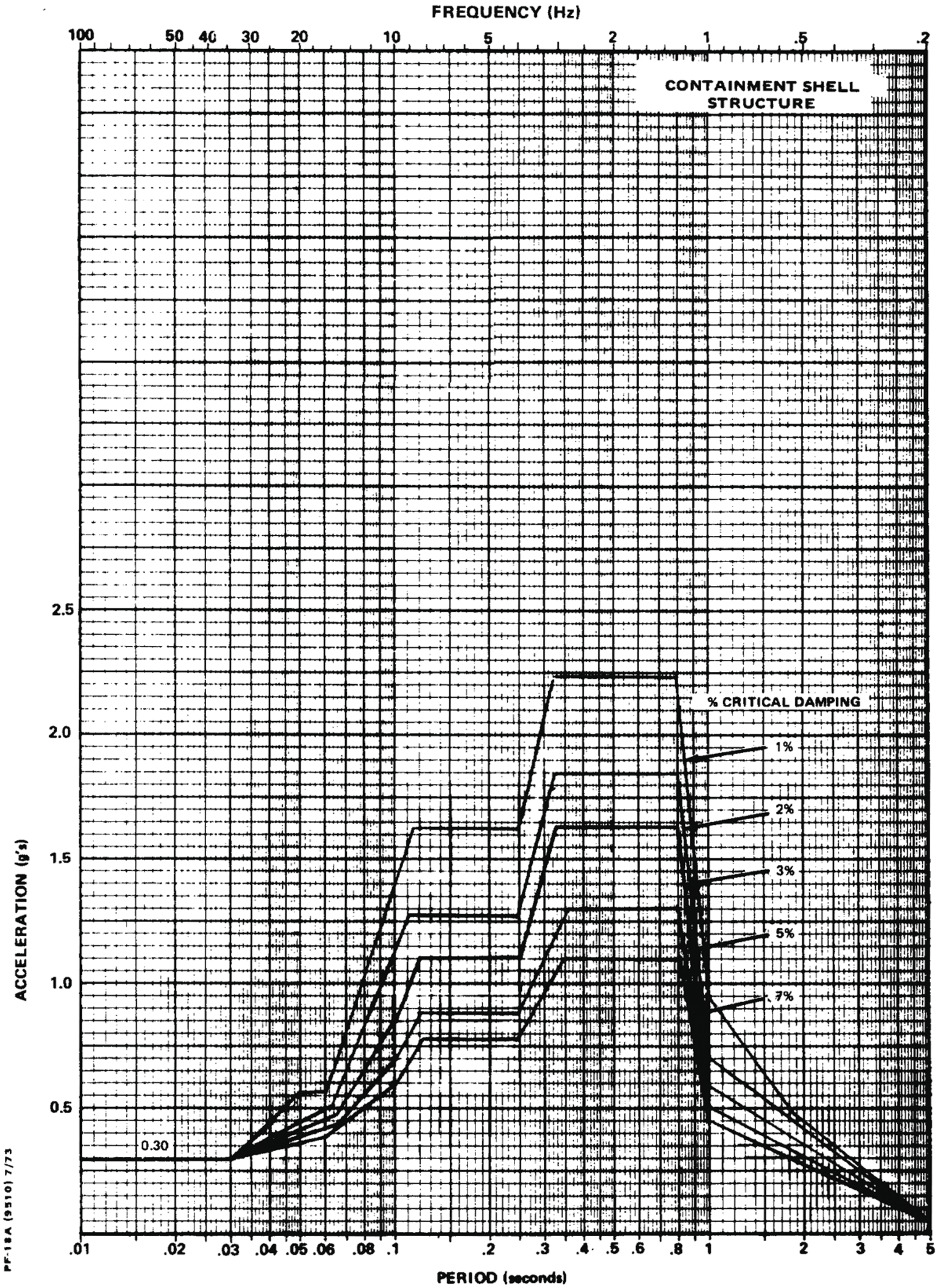

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FIGURE 3D–14

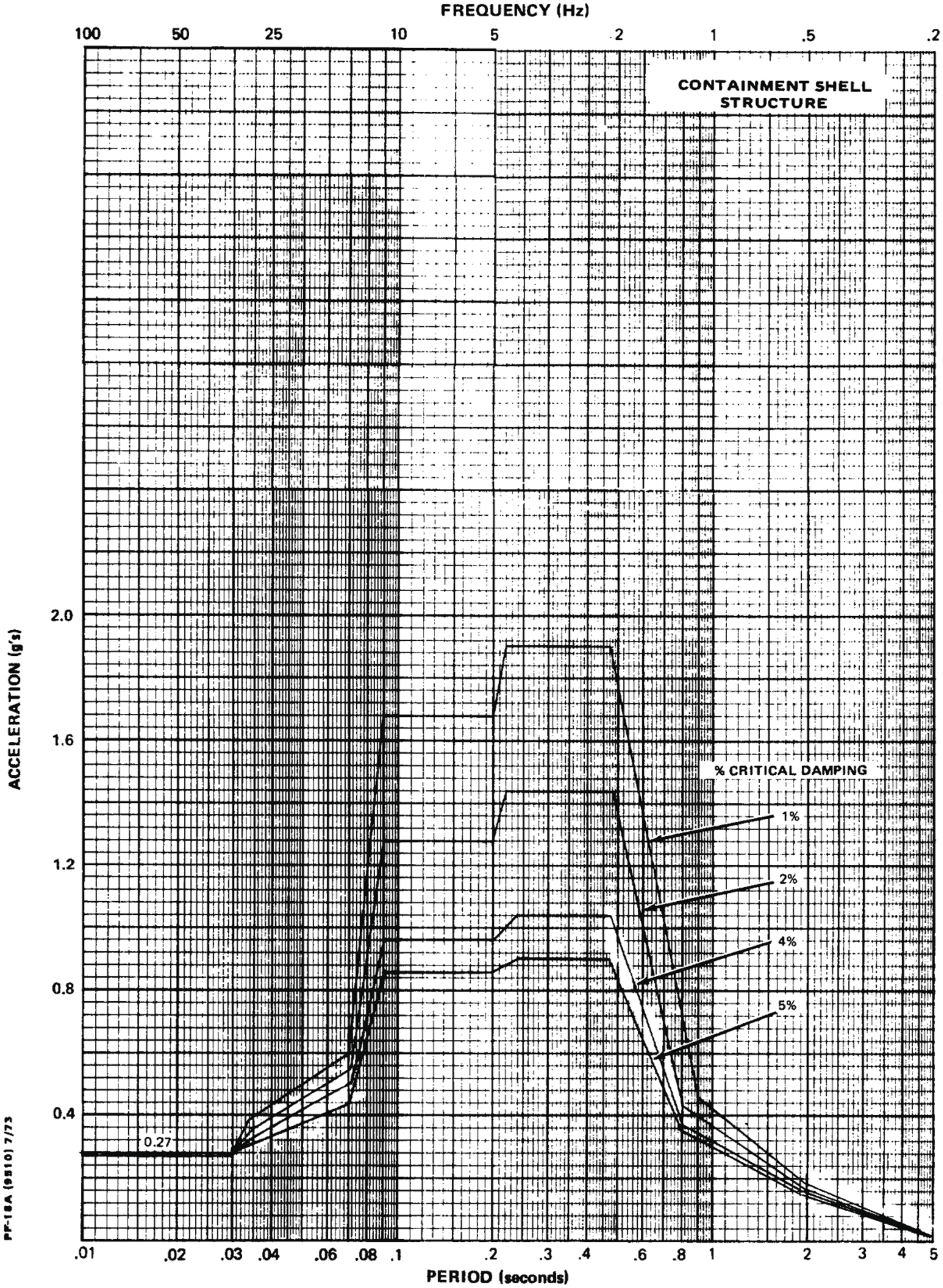

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FIGURE 3D–15

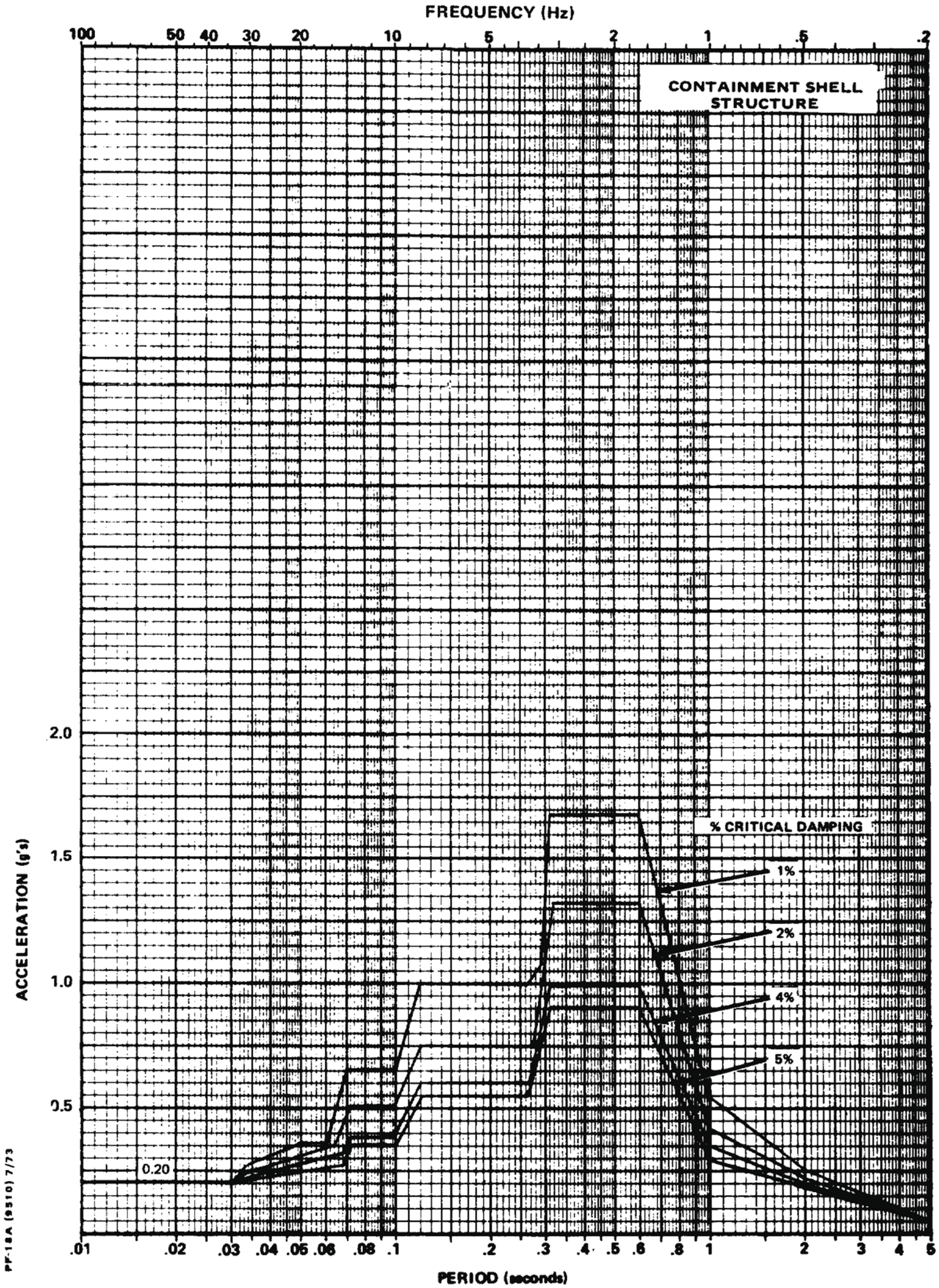

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FIGURE 3D–16

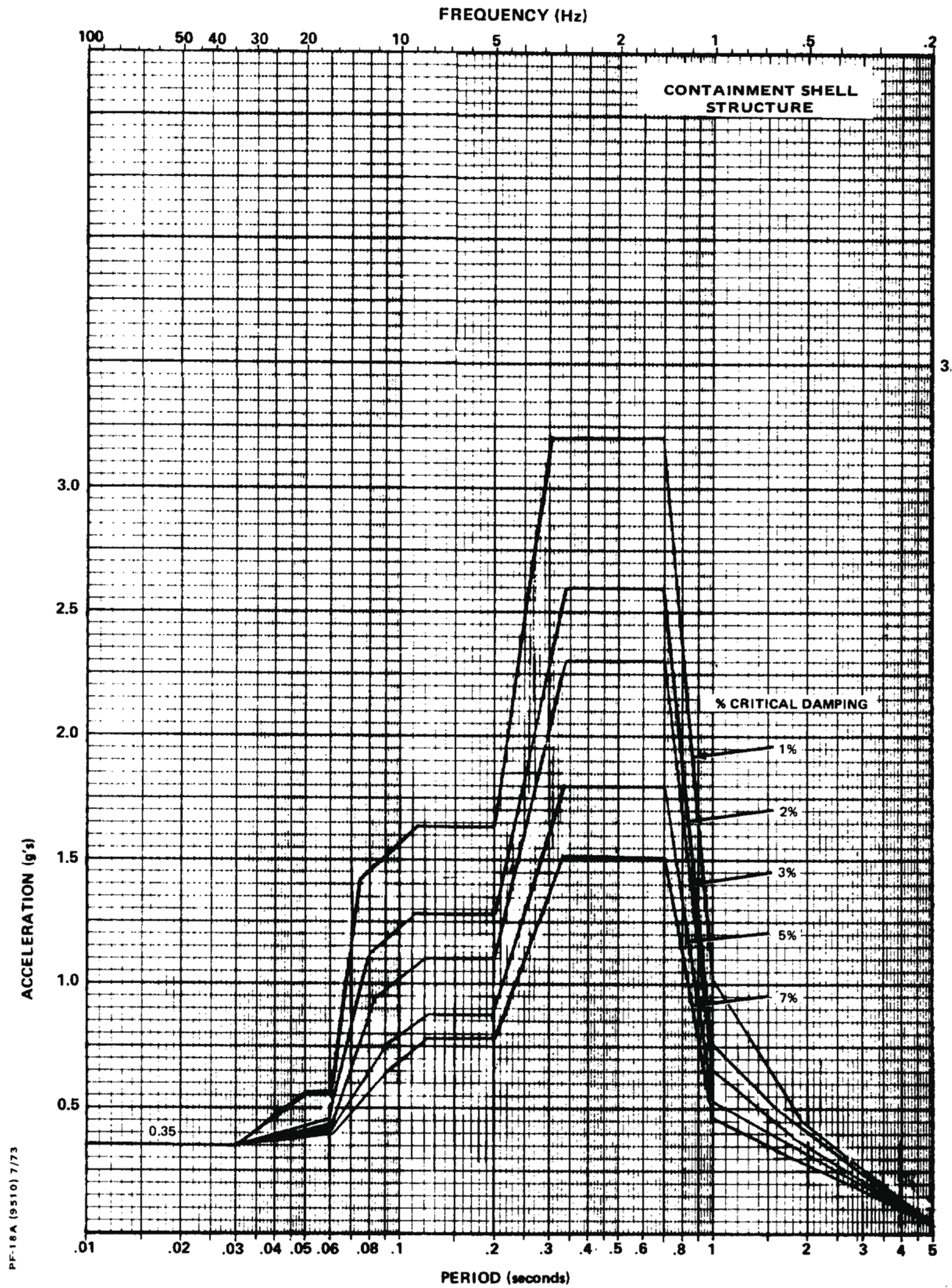

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FIGURE 3D–17