APPENDIX C

STRUCTURAL LOADING CRITERIA

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1.0 SCOPE

This appendix provides additional information pertinent to the preceding sections concerning the structural loading criteria applied to Class I structures and components. Station structures and components are classified according to service and location. Class I Seismic structures and components are listed in USAR Section XII-2.1.2.

The loads, loading combinations and allowable limits described in this appendix apply only to Class I structures and components as defined in USAR Section XII-2.1.1. The criteria in this appendix are intended to supplement applicable industry design codes where necessary. Compliance with these criteria is intended to provide design safety margins which are appropriate to extremely reliable structural components when account is taken of rare event potentialities associated with a Safe Shutdown Earthquake or postulated loss-of-coolant accident or a combination thereof.

Class I components may not always be designed to satisfy the criteria using analytical techniques; alternately, the design of some components may be based upon test results, empirical evidence, or experience, including the use of earthquake experience and test data according to the SQUG Generic Implementation Procedure (GIP). This method was used in the resolution of USI A-46 and may be used for the design and modification of Class I equipment within the scope of GIP-3. GIP-3 is further described in this Appendix.

2.0 CONCRETE AND STEEL STRUCTURES

2.1 Description

The Class I concrete and steel structures are designed considering three interrelated primary functions for the design loading combinations described in Subsection C-2.3. The first consideration is to provide structural strength equal to or greater than that required to sustain the combination of design loads and provide protection to other vital Class I structures and components. The second consideration is to maintain structural deformations within limits such that Class I components will not experience loss of function. The third consideration is to preclude excessive leakage by preventing excessive deformation and cracking, when containment integrity is required.

In general, the load combinations considered and their allowable limits are formed on a "quasi-probabilistic" basis. This means that the higher the probability that a given set of conditions could occur, the lower the allowable limits. This also forms the basis for neglecting some combinations of loads. In general, only one highly improbable event is considered in any load combination, since the probability of two unrelated, highly improbable events occurring simultaneously is vanishingly small.

2.2 Loads

The loads considered in the design of Class I concrete and steel structures include the following:

D = Dead load of the structure and related equipment plus any other permanent loads contributing stresses, such as soil or hydrostatic loads; live loads expected to be present when the station is operating; and the loads due to thermal expansion under normal operating conditions. This load takes into account any deviations from normal operating conditions which are reasonably expected to occur during the design lifetime of the station.

R = Loads resulting from jet forces and pressure and temperature transients associated with rupture of a single pipe within the primary containment.

R’ = Loads due to a High Energy Line Break (HELB) outside containment. This includes the effects of pressure and temperature transients, pipe whip impact forces, and reactions from pipe anchors.[25]

E = Loads due to the Operating Basis Earthquake (OBE) (0.10g horizontal ground acceleration; 2/3[1] of horizontal ground acceleration applied simultaneously for vertical seismic acceleration). These loads are also known as Maximum Probable Design Earthquake loads.

E' = Loads due to the Safe Shutdown Earthquake (SSE) (0.20g horizontal ground acceleration; 2/3[1] of horizontal ground acceleration simultaneously applied for vertical seismic acceleration). These loads are also known as Hypothetical Maximum Possible Design Earthquake loads.

Flood = Loads due to flooding the Drywell up to 23 feet above the Drywell flange level.

W = Design wind loading conditions. Refer to USAR Section XII-2.3.3.1.

T = Loads due to the effects of a tornado. Refer to USAR Section XII-2.3.3.2.

Primary Containment System components associated with the Mark I program are evaluated for the loads as described in the CNS Plant Unique Analysis Report and in accordance with NUREG 0661.

2.3 Load Combinations and Allowable Limits

The loads combinations and allowable limits considered in the design of Class I concrete and steel structures include the following (also see USAR Table XII-2-1).

Load Combination	Limits
D+E	Stresses remain within normal code allowable stresses (AISC 6th Edition for structural steel, ACI 318-63 for reinforced concrete, ASME B&PV Code Section III [Class B], S67 for Primary Containment). The customary increase in design stress for earthquake loadings is not permitted.
D+W	Maximum allowable stresses may be increased one-third above normal code allowable stresses.
D+R+E	Stresses remain within normal code allowable stresses. The customary increase in design stress for earthquake loadings is not permitted. In the case of jet impingement loading on Primary Containment, where it is backed up by concrete, it may be assumed that local yielding may take place but it shall be established that a rupture will not occur. For jet impingement loading on Primary Containment (including containment penetration assemblies) where Primary Containment is not backed up by concrete, the primary stresses must not exceed 90% of the yield strength of the material at 300°F.
D+E+Flood	Local membrane stresses in Primary Containment may exceed the yield point, but with a margin against rupture.

Load Combination	Limits
D+T	Maximum allowable stresses are as follows: Steel - AISC 6th Edition allowable yield stresses with a maximum of 90% of the allowable yield stress. Concrete - ACI 318-63 "Working Stress Design" method with allowable stresses in concrete of 0.85 f'_c, and allowable stresses in reinforcing steel of 0.9 f_y. Alternatively, when the "Ultimate Strength Design" method is used a load factor of 1.0 is applied to this load combination with appropriate reduction factors as described in ACI 318-63.
D+R+E'	Maximum allowable stresses are as follows: Steel - AISC 6th Edition allowable yield stresses with a maximum of 90% of the allowable yield stress. In the case of jet impingement loading on the primary containment, where it is backed up by concrete, it may be assumed that local yielding may take place but it shall be established that a rupture will not occur. For jet impingement loading on the primary containment (including containment penetration assemblies) where the primary containment is not backed up by concrete, the primary stresses must not exceed 90% of the yield strength of the material at 300°F. Concrete - ACI 318-63 "Working Stress Design" method with allowable stresses in concrete of 0.85 f'_c, and allowable stresses in reinforcing steel of 0.9 f_y. Alternatively, when the "Ultimate Strength Design" method is used a load factor of 1.0 is applied to this load combination with appropriate reduction factors as described in ACI 318-63.
D+R'+E'	The allowable stresses are generally limited to 90% of their ultimate values, i.e., flexural stresses in concrete are limited to 0.85 f'_c, and stresses in reinforcing steel are limited to 0.90 fy (note in certain instances, stresses in reinforcing steel up to approximately 1.0 f_y were considered acceptable for this extreme load combination).[25]

Primary Containment System components associated with the Mark I program were evaluated for the load combinations and allowable limits as described in the CNS Plant Unique Analysis Report, and in accordance with NUREG 0661. Allowable stresses for these components are based on the requirements of the ASME B&PV Code, Section III, S77.

Governing Codes

This appendix identifies the governing codes and supplementary requirements for the design and installation of structures. Repairs, replacements or modifications of structures may be performed to these requirements or to the requirements of later editions of the construction codes provided the safety design bases described in the USAR are maintained. The following codes are the codes of record for Class I structures:

For structural steel, the governing code is the American Institute of Steel Construction Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, adopted April 17, 1963 (referred to herein as AISC 6th Edition).

For concrete structures, the governing code is the American Concrete Institute Standard ACI 318-63, Building Code Requirements for Reinforced Concrete, as amended March 6, 1963 (referred to herein as ACI 318-63).

For the original design for Primary Containment, the governing code is the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels, Subsection B (for Class B Vessels), 1965 Edition using addenda up to and including the Summer 1967 Addenda (referred to herein as ASME B&PV Code, Section III [Class B], S67).

For the reevaluation of the portions of the Primary Containment System associated with the Mark I program, the governing code is the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, 1977 Edition using addenda up to and including Summer 1977 Addenda (referred to herein as ASME B&PV Code, Section III, S77).

Allowable Stresses

The allowable stresses for Class I structures are generalized in USAR Table XII-2-1. The definition of f'_c is the specified compressive strength of concrete in pounds per square inch at a specified age (normally at 28 days). At CNS the predominant concrete mix is specified to have a minimum 28 day strength of 3,000 psi using Type II cement.[4]} The actual strengths used are listed in the following paragraphs.

The concrete mix specified for the following structural parts of buildings has a minimum 28 day strength of 4,000 psi: Reactor Building foundation mat and substructure, Control Building foundation and substructure, Radwaste Building foundation mat and substructure, substructure of the Intake Structure.

The concrete mix specified for the following structural parts of buildings has a minimum 28 day strength of 3,000 psi: Reactor Building superstructure, Reactor Pedestal, Control Building superstructure, Radwaste Building superstructure, superstructure of the Intake Structure, all parts of the Diesel Generator Building and Control Corridor.

The concrete mix specified for the Containment Pedestal and leveling slabs of all buildings have a minimum 28 day strength of 2,000 psi.

Allowables for load combinations involving E’, T, R’ and certain combinations involving R are based on a criterion for "no loss of function" such that safe shutdown can be achieved. The criterion for "no loss of function" for structures is that stresses will remain in the elastic range.[45] Concrete compressive stresses due to bending are limited to $0.85f'_c$, reinforcing steel stresses are limited to 90% of yield, and tensile and bending stresses in structural steel members are limited to 90% of their respective yield stresses.[6]

Structural deformations are not the controlling criteria for safe shutdown structures (see USAR Section XII-2.1 for a list of safe shutdown structures). Concrete crack control is maintained by limiting reinforcing steel stresses to those given in Table XII-2-1.

The book "Design of Concrete Structures,"[2] shows stress-strain curves for a member under axial compression. Figure 2.2 of this reference shows that the concrete stress-strain curve for fast rate of loading (typical for earthquake or tornado) does not deviate appreciably from a straight line below a stress level of $0.85f'_c$ (2,550 psi).

In addition, concrete gains additional strength with age. Figure 22 from the "Concrete Manual"[3] shows the effect of age on compression strength. A mix having a minimum 28 day compressive test strength (f'_c) of 3,000 psi will approximate minimum test strengths of 3,700 psi at 90 days, 3,900 psi at six months, 4,100 psi at one year, and 4,200 psi at two years.

At the time of fuel loading of the reactor the concrete had attained a minimum age of at least nine months, and at full operation more than 14 months. Thus, the allowable design stress of $0.85f'_c$ (2,550 psi) is in actuality only approximately 64% to 67% of the ultimate concrete strength.

Based on the above reasons, it is concluded that stress distribution is linear, and the use of $0.85f'_c$ for the allowable concrete stress for the working stress design method is a valid approach.[4]

All structures, except the Drywell Biological Shield Wall, are designed considering uni-axial stress conditions. As such, the allowable stresses for shear, as a measure of diagonal tension, for bond and anchorage are as given in the ACI 318-63 Code.[5]

2.4 Method of Analysis

A dynamic seismic analysis was performed for Class I structures shown in Burns and Roe Drawings 2050, 2051, 2052, 2053, 2054, 2056, 2059, 2060, 2061, 2062, 2063, 2064, 2065, 2066, 2067, 2068, and 2069. The structural analysis is performed using various calculational methods and techniques. Much of the structural design is performed utilizing the "Working Stress Design" method as defined in ACI 318-63 and in the AISC 6th Edition. Some portions of the Class I structures are designed by the "Ultimate Strength Design" method described in Chapters 15 through 19 of ACI 318-63.

Load combinations and allowable limits on stresses are as shown in the USAR Section C-2.3. Structural ductility of concrete members is provided by proportioning the structure so that a calculated value of 0.9 f_y for the main tension reinforcement is the limiting stress value at the critical cross section in flexure.

No methods of inelastic analysis have been used on any of the components discussed in the appendix.[7] Only uni-dimensional stress conditions are considered to exist in all Class I steel structures other than the steel containment vessel.[8] The allowable stresses appropriate to these uni-dimensional analyses are as generalized in USAR Table XII-2-1.

USAR Section II-5.2 describes the bases for the selection of maximum horizontal and vertical ground accelerations associated with the Safe Shutdown Earthquake and the Operating Basis Earthquake. The original vertical seismic acceleration component was specified as one-half of the horizontal ground acceleration. This design requirement was increased to two-thirds of the horizontal ground acceleration, which is the current Licensing/Design Basis.[1] Evaluations were performed to confirm that the increase in vertical seismic acceleration component was accommodated by existing margins and conservatisms in the original design analyses of the Principal Class I Structures.[67]

As described in USAR Section XII-2.3.5.2.1, the dynamic analysis consisted of 1) developing a mathematical model, 2) performing the analysis,

3) obtaining a structural response, and 4) plotting the response spectra. For example, mathematical models of the Reactor and Control Buildings are shown in Figures C-2-1 and C-2-2, respectively.

2.5 Implementation of Criteria

2.5.1 Reactor Building Foundation (See Table C-2-1)

The soil bearing stress under the Reactor Building is derived from the application of all dead and uniformly distributed live loads on each main floor to allow for vertical loads during station operation and refueling. Several load cases are examined depending on how much live load and water is considered to be present. Seismic loads for the OBE and SSE are also considered.

The primary objective is to establish whether soil bearing stresses will be within the allowable stresses, that there is no possibility of excessive uneven settlement, and that there will be no possibility of soil liquefaction when subject to seismic vibrations.

The analysis shows that (1) maximum stresses are below allowable values, (2) relative settlement will be small, and (3) that liquefaction will not occur.

2.5.2 Reactor Building Floor System (See Table C-2-2)

Reactor Building floor systems are designed for the loadings tabulated in Subsection XII-2.3.2. The conservative assumption is made that the Operating Basis Earthquake (E) or the Safe Shutdown Earthquake (E’) vertical load could occur simultaneously with other vertical live loads. Wind (W) and tornado (T) loads do not apply. Accident loads (R) are included where applicable.

The resulting stresses are within the Code allowable values, therefore, the floor systems are structurally adequate without excessive deflections or cracking of concrete.

2.5.3 Reactor Building Concrete Walls (See Table C-2-3)

The concrete walls of the Reactor Building are analyzed for the design loads (D) combined with seismic loads due to the Operating Basis Earthquake (E) or Safe Shutdown Earthquake (E'). Accident loads (R) are included where applicable. Tornado loads (T) were checked and found governing for the tornado generated missile criteria. Stresses due to seismic moments and shears were calculated, and combined together with the gravity stresses.

Jet forces are postulated to occur only in the Main Steam Line Tunnel (see USAR Section C-2.5.8).[12] Equipment reactions are taken as forces applied to the supporting structure and the structure is designed accordingly. The Reactor Building exterior walls were investigated for penetration by the postulated tornado generated missiles as described in USAR Section XII-2.3.3.2.

TABLE C-2-1
REACTOR BUILDING FOUNDATION

Description/Criteria	Method of Analysis	Load Combination*	Max. Allowable Stress-ksf
Design load (D) includes all dead and equipment loads plus 25% of the full live loads.	Max. net soil stress = $\frac{D + \Delta P}{A} + \frac{M}{S}$ (KSF) - $(\alpha D_a+\gamma_{sat}+D_b)$	D+E	$1.5q_a = 18$ q_a = net avg. contact pressure
Vertical seismic load (ΔP) is assumed to be produced by a vertical acceleration equal to 2/3 of the ground horizontal seismic acceleration, both acting simultaneously** Overturning moment (M) is produced by the horizontal seismic acceleration and all design load eccentricities	where: D = Design load, kips ΔP = vert seismic load, kips M = horiz. seismic overturning moment kip-ft A = foundation mat area, sq-ft S = section modulus of foundation mat ft^3 γ = unit weight of soil above water table γ_{sat} = saturated unit wt. of soil D_a = depth of soil above water table D_b = depth of saturated soil above bottom of foundation	D+E' E = Loads due to OBE E' = Loads due to SSE	$1.5q_a = 18$

* Floor design load (D) includes dead loads, equipment loads, plus 25% of the full live loads (appropriate for operating conditions) for combination with seismic or tornado loads.
Floor design is also checked for maximum live loads (100 to 1,000 psf) not in combination with seismic or tornado loads.

** Vertical = 1/2 horizontal per PSAR; 2/3 was used for conservatism.

TABLE C-2-2
REACTOR BUILDING FLOOR SYSTEM

Description/Criteria	Method of Analysis	Load Combination*	Max. Allowable Stress-ksi
Floor system is designed for the loadings tabulated in Subsection XII-2.3.2.	Working Stress Design Method	D + E	F_c = 1.35 F_v = 0.060 F_t = 20.0
Materials conform as follows: Concrete f'_c = 3,000 psi at 28 days max. strength per ACI 318-63; Reinforcing ASTM Designation: A615 Grade 40 per ACI 318-63; Structural Steel ASTM Designation: A36-67 per AISC Manual, 1963	Same as for D + E'	D + E'	F_c = 2.55 F_v = 0.093 F_t = 36.0
Vertical seismic load is assumed to be produced by a vertical acceleration equal to 2/3 of horizontal ground acceleration**			
Max. allowable stresses for D+E' load combination are: Concrete Fc = 0.85 f'_c Reinforcing Ft = 0.90 f_y Structural Steel = 0.90 f_y			

* Maximum allowable Stresses for D + E combination are not increased above code allowable values, i.e., customary 1/3 increase not used.

** Vertical = 1/2 horizontal per PSAR; 2/3 was used for conservatism.

TABLE C-2-3
REACTOR BUILDING CONCRETE WALLS

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
Design Load (D) includes all dead and equipment loads, plus 25% of the full live loads.	Working Stress Design of ACI 318-63	D + E	F_t = 20.0 F_c = 1.35 F_y = 0.060
Materials: Concrete f'_c = 3,000 psi at 28 days max. strength per ACI 318-63	Working Stress Design Method of ACI 318-63	D + E'	F_t = 36.0 F_c = 2.55 F_v = 0.093
Reinforcing ASTM Designation: A615 Grade 40 per ACI 318-63			
Maximum allowable stresses for D+E' load Concrete $F_c = 0.85f'_c$			
Reinforcing $F_t = 0.90\ f_y$			
Maximum allowable stresses for D+E load combination are not increased above code allowable values, i.e., customary 1/3 increase not used.			

Note: F_t and f_t are tension in reinforcing
F_c and f'_c are compression in concrete
F_v and f_v are shear in concrete

The resulting stresses are within the Code allowable values, therefore, the Reactor Building walls are structurally adequate without excessive deflections or cracking of concrete.

As described in USAR Section XII-2.3.5.1.5, removable block walls are provided in the Reactor Building to satisfy equipment access requirements. These walls are shown in Burns and Roe Drawing 4215, and their horizontal seismic acceleration curve is shown in Figure C-2-12.

2.5.4 Reactor Building Steel Superstructure (See Tables C-2-4 and C-2-5)

The Reactor Building structural steel superstructure (truss frame and columns supporting roof and siding) are evaluated for various combinations of design loads (D), earthquake loads (E and E'), wind loads (W), and tornado loads (T). Design loads include dead loads, live loads and crane loads. While crane live loads are not considered as a controlling load combination with lateral loading in the Burns and Roe analysis, subsequent evaluation has demonstrated the 108 ton live load combined with earthquake loading is still bounded by the tornado load without crane live loads.[74] Two cases of tornado loading are considered. Case A considers the siding to be intact and subject to a 75 psf pressure, and Case B assumes the siding to have blown off and the remaining superstructure to be subjected to full force tornado loads (270 psf due to 300 mph winds). For Case A, wind loads are taken as 45 psf on the walls, and as a 34 psf uplift on the roof due to 100 mph winds. For Case B, the roof is considered to remain intact and it is assumed that no pressure differential exists between the top and bottom of the roof. Tornado loads were found to govern over Safe Shutdown Earthquake (SSE) loads. D + W was considered and found not to be governing when compared to D + E or D + T. The analysis method is the Allowable Stress Design Method of the AISC 6th Edition considering unsymmetrical beam loading where applicable. For design loads in combination with wind or Operating Basis Earthquake (OBE) loads, the resultant stresses calculated are below the Code allowable stresses; therefore, the integrity of secondary containment will be assured. For design loads combined with SSE loads, it was obvious by inspection that the increase in stresses caused by the SSE would be small because of the relatively small masses involved; therefore, secondary containment integrity will be assured.

The siding will blow off when the design wind velocity of 100 mph is appreciably exceeded (75 psf in or out).[12] This is assured by the use of necked-down steel "control release fasteners" between the girt system and the supporting columns. A rigorous test and quality assurance program was instituted to provide assurance that these fasteners will fail at a maximum 75 psf. In addition, the barometric pressure drop that precedes the tornado need only be about ½ psi before the siding blows out rather than inward.

The exposed structural steel frame of the Reactor Building superstructure has been designed to withstand winds of 300 mph intensity with stresses limited to 90% of yield. The 300 mph tornado forces have also been applied to the surfaces of the Reactor Building crane. Reactor Building metal siding is designed for normal 100 mph wind loading. The steel superstructure is not designed for tornado missiles.

TABLE C-2-4
REACTOR BUILDING STRUCTURAL STEEL COLUMNS

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
Material: ASTM Designation: A36 per AISC Manual, 1963	Working Stress Design Method considering unsymmetrical beam loading where applicable	D + LL + Crane Load	F_A = 18.3 per AISC F_B = 22.0 (Upper section crane columns)
Structural Steel Column designed for all dead and live loads as follows: Wheel reactions from 108 ton crane in combination with roof DL plus 20 psf LL plus wind @ 45 psf on building Vertical seismic load is assumed to be produced by a vertical acceleration equal to ½ of horizontal ground acceleration plus normal DL partial LL crane DL			F_A = 17.4 F_B = 22.0 (Lower section crane columns) F_A = 24.3 F_B = 29.2 (Upper section crane columns) F_A = 23.2 F_B = 29.2 (Lower section crane columns) F_A = 17.4 F_B = 22.0 (Lower section crane columns)
Tornado load - building siding remains intact until 75 psf loading is reached at which time it is considered removed and full tornado load is applied to the structural steel members	Same as Above	DL + Tornado on structural steel only	F_A = 27 F_B = 32.4 (Upper section crane columns) F_A = 27 F_B = 32.4 (Lower section crane columns)

TABLE C-2-5
REACTOR BUILDING STEEL ROOF TRUSS-FRAME ABOVE EL. 117'

Description/Criteria	Method of Analysis	Load Combination*	Max. Allowable Stress-ksi
Material: Structural steel ASTM Designation: A36 per AISC Manual, 1963	Working Stress Design	D + E	F_A = 17.62 comp. (Top chord)
		D + T	F_A = 26.6 comp. (Top chord)
Design wind loading: 45 psf on walls due to 100 mph wind: uplift 34 psf on roof due to 100 mph wind: dead weight of roof deck and built-up roofing 8 psf		D + E	F_A = 22.0 ten. (Bottom chord)
		D + T	F_A = 32.4 ten. (Bottom chord)
Tornado wind load (T) on walls averages a pressure of 270 psf on exposed surfaces after siding has blown off at ± 75 psf		D + E	F_A = 22.0 ten. (Diagonal member)
		D + T	F_A = 32.4 ten. (Diagonal member)
		D + E	F_A = 15.4 comp. (Truss vertical)
108 ton capacity crane on column bracket		D + T	F_A = 22.6 comp. (Truss vertical)
		D + E	F_A = 22.0 ten. (Lower chord bracing)
		D + T	F_A = 32.4 ten. (Lower chord bracing)
E' load disregarded because tornado is governing at similar allowable stresses of 0.9 F_y			

* D+W was considered and found not governing; for D+W the margins between the maximum allowable stresses and the calculated stresses were greater than for D+E or D+Tornado.

2.5.5 Drywell Biological Shield Wall (See Table C-2-6)

The comments regarding the Reactor Building concrete walls (see USAR Section C-2.5.3) generally apply also to the Drywell Biological Shield Wall. Drywell Safe Shutdown Earthquake loads control over tornado loads. The concrete surface is protected by the Drywell vessel which acts as a thermal shield, and by a 2 inch air gap which impedes the flow of heat. Thus, the transient itself has little effect on the concrete.[13]

The Drywell Biological Shield Wall structure is designed considering a bi-axial stress condition. There are vertical compressive stresses and tensile hoop stresses. The plane of shear resistance is a horizontal plane upon which compressive stresses act and is unaffected by the tensile hoop stresses. For the longitudinal or seismic shear the allowable stress is considered equal to the allowable shear stress, as a measure of diagonal tension, and as given in the ACI 318-63 Code. (The 1971 edition of the same code specifies a nominal permissible shear stress for nonprestressed concrete members equal in value with that prescribed for the nominal permissible shear stress in the plane of the shear walls: $Vc = 2(f'_c)^{0.5}$. The radial shear stress caused by discontinuities is considered to be diagonal tension and the allowable stress is given in the ACI 318-63 Code.

For the splices in the tensile hoop reinforcement in the Drywell Biological Shield Wall structure the allowable stress for bond is as given in the ACI 318-63 Code. These Code allowables are for a uni-axial stress condition. They should be conservative for this structure since the bi-axial compression which exists on the splices should improve the bond capability of the concrete.

The analysis shows that the stresses in the Drywell Biological Shield Wall will not exceed the allowable values.

No equipment or piping inside the drywell is anchored to the Drywell Biological Shield Wall. Drywell penetrations passing through the biological shield wall are not anchored to the biological shield wall annular sleeves. However, when required by a rupture analysis of the Drywell penetration nozzles, and/or when required to provide positive bellows seal protection, limit stops were utilized. For penetrations which have expansion joint bellows seals, limit stops were provided to assure that the bellows seals would not fail due to excessive torsional rotation, or due to axial collapse of the joint at the time of pipe rupture. Stress analysis indicated that some penetrations require only lateral limit stops and, therefore, no axial or torsional limit stops are required for this type of penetration. The limit stops are normally disengaged. The limit stops consist of sleeves, embedded in the biological shield wall and extend to the flued head fitting outside the shield wall, coaxially with the containment penetration. In the vicinity of the flued head fitting, a flange is attached to the sleeve extension. A gap is maintained between the sleeve and the penetration assembly so that no contact is made due to thermal differential movements. When rupture loading is applied to the containment penetration, the assembly deflects and contact is made between the flued head fitting and the limit stops at the flange. The flange, which is provided at this location, serves as a stiffening ring which transmits the ensuing rupture loading to the embedded sleeve, which in turn transmits the load to the biological shield wall.[13][23]

TABLE C-2-6
DRYWELL SHIELDING CONCRETE

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
Drywell shield acts as a structural wall carrying floors, Design load (D) consists of all dead loads, equipment loads, plus 25% of full live load	Working Stress Design. Thermal stresses are included in the analysis. Seismic forces are super- imposed on the results.	D + E + thermal gradient	F_t = 20 F_c = 1.35 F_v = 0.060
Seismic loads (E&E') are according to the response spectra for the reactor building Accident load (R) includes 60°F normal operational thermal gradient in winter and jet forces due to ruptured pipe Materials conform as follows: Concrete f'_c = 3,000 psi at 28 days minimum strength per ACI 318-63 Reinforcing ASTM Designation: A615 Grade 40 per ACI 318-63 Structural Steel ASTM Designation: A36-67 per AISC Manual, 1963 Maximum allowable stresses for D+E' load combination are: Concrete F_c = 0.85 F'_c Reinforcing F_t = 0.90 f_y		D + E' + R (Jet due to any pipe break within the spherical area of the drywell section + thermal gradient)	F_t = 36.0 F_c = 2.55 F_v = 0.113

The loading combination which is the controlling design case for these limit stops is D+R. To provide assurance that these limit stops can withstand this loading condition, the limit stops are designed for those forces which will produce a plastic hinge in the pipe at the flued head fitting (which is in the vicinity of the limit stop). In this manner an upper limit of capability is provided for by the limit stop. The allowable reinforcing steel stress for this condition is $0.90f_y$. The allowable concrete stress in the Drywell Biological Shield Wall for this condition is $0.85f'_c$.

For the loading condition when a jet reaction does not occur, the limit stops are disengaged and transmit no piping loadings to the Drywell Biological Shield Wall.

The design of the Drywell Biological Shield Wall in the vicinity of large openings was performed in accordance with the allowable stress criteria specified in USAR Table XII-2-1. The design considered stress concentration effects assuming that the Drywell Biological Shield Wall in the vicinity of the openings was a two dimensional system contained in the plane tangent to the mid-surface of the Drywell Biological Shield Wall in the area under consideration. Out-of-plane effects were considered separately. Stresses resulting from both effects were accounted for in the design.

2.5.6 Reactor Pedestal (See Table C-2-7)

The Reactor Pedestal is designed using the Working Stress Design Method of ACI 318-63. The thermal gradient through the shell is considered. The forces required to restrain a ruptured Reactor Recirculation pipe are also considered. The calculated maximum stresses are close to the normal Code allowable values indicating that excessive cracking will not occur.

The controlling load combinations include design loads (D) + temperature gradient loads (R) + Operating Basis Earthquake (OBE) loads (E) evaluated against normal Code allowables, and design loads (D) + temperature gradient loads (R) + Safe Shutdown Earthquake (SSE) loads (E') + jet loads (R) evaluated against 90% of ultimate allowables. Since the stress criteria (i.e., 90% of ultimate) is the same for the loading combination of design + accident (jet) + OBE + temperature effects and the loading combination of design + accident (jet) + SSE + temperature effects, the SSE case controls.[16]

Pedestal Temperature Differential[16]

During the steady state, the temperature differential between the inside and outside of the pedestal is very small and its effect may be neglected. This temperature condition is maintained through the use of the Drywell fan coil units which circulate air from the outside to the inside of the Reactor Pedestal.

During the transient state, or accident state, in which temperature builds up on the outside of the pedestal but in which the heat may pass through the openings in the pedestal to the inside of the pedestal, a temperature gradient may be built up across the wall. For accident conditions which include jet loads, higher stresses are permitted over the allowable ACI stresses. Since these stresses are below the ultimate capacity of the concrete and still within its elastic range the structures will function during and after an accident condition.

TABLE C-2-7
REACTOR CONCRETE PEDESTAL

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
The Reactor Vessel Pedestal consists of a 3'-3" thick x 26'-7" high cylindrical wall rising from a concrete base, flaring out to 5'-10 ¾" thick at the top. The base has an average thickness of 7' at wall and a spherical shaped bottom matching the sphere of drywell. The shears and moments are transferred to the drywell through a welded steel shear ring and the drywell and the concrete.	Working Stress Design Method. For seismic loads response spectra are used. Circumferential stresses due to temperature are in accordance with ACI 307.	D + R + E	F_c = 1.35 F_t = 20.0
		D + R + E (Horiz. Hoop stress)	F_c = 1.35 F_t = 20.0
		D + R + E' + Jet at pump restraint	F_c = 2.55 F_t = 36.0
Materials: Concrete f'_c = 3,000 psi at 28 days max. strength per ACI 318-63 Reinforcing ASTM Designation: A615 Grade 40 per ACI 318-63			
Maximum allowable stresses for D+R+E' load: Concrete $F_c = 0.85\ f'_c$ Reinforcing $F_t = 0.90\ f_y$			
Maximum allowable stresses for D+R+E load combination are not increased above code allowable values, i.e., customary 1/3 increase not used			
Accident load (R) includes jet forces (where noted) due to ruptured pipe and temperature gradient in concrete		D + R + E' + Jet (at reactor nozzle)	F_c = 2.55 F_t = 36.0 F_v = 0.16

The method used for determining the temperature stresses in the circumferential and vertical directions were those set forth in the ACI Standard Specification for the Design and Construction of Reinforced Concrete Chimneys (ACI 307-69).

Ring Girder/RPV Support Loading[16]

The ring girder is designed to transfer the vertical and horizontal loads of the Reactor Pressure Vessel (RPV) skirt flange to the top of the Reactor Pedestal.

The horizontal shears on the RPV skirt flange are transferred to the top flange of the ring girder by 60-A490 high strength bolts in the same friction-type connection as is described in the AISC Code.

The amount of frictional force available to resist horizontal shear is directly proportional to the normal pressure (proof load) between the RPV skirt flange and top flange of the ring girder. The total frictional force and the coefficient of sliding friction are independent of the areas in contact. The friction-type connection of the RPV skirt flange to the ring girder, in which some of the bolts lose a part of their clamping force (proof load) due to applied tension during an earthquake, suffers no overall loss of frictional shear resistance during an earthquake. The bolt tension produced by the moment is coupled with a compensating compressive force on the other side of the axis of bending. The total frictional force remains constant so long as the total pressure remains the same.

The total frictional force due to a coefficient of friction of .15 and a proof load of 313 kips per bolt is 2,820 kips or 6.9 times the Operating Basis Earthquake (OBE) shear load of 403 kips or 2.1 times the Safe Shutdown Earthquake (SSE) shear plus jet load of 1,307 kips. However, if the coefficient of friction is assumed zero, the bolts acting as bearing-type connections could resist a total horizontal shear of 15.0 (at AISC Code stresses) times the OBE shear load of 403 kips or 6.85 (at 90% yield stresses) times the SSE shear load plus jet of 1,307 kips. Therefore, the high strength bolt connections of the RPV skirt flange to the top flange of the ring girder, with or without friction, are more than adequate for the respective design loads.

The vertical loads on the RPV skirt flange are transferred to the top of the Reactor Pedestal by the ring girder acting as a bearing plate. The ring girder is designed according to AISC 6th Edition.

The load from the ring girder is transmitted to the Reactor Pedestal by direct bearing of the ring girder on the Reactor Pedestal. For the seismic condition, and the seismic plus jet condition, the overturning moment and the shear force are resisted by the anchor bolts connecting the ring girder to the Reactor Pedestal. Without considering friction the maximum tension of 14.0 ksi and the maximum shear of 3.6 ksi occur in the anchor bolts for the conditions of SSE plus jet force plus operating load. These stresses are below the working allowable stresses. The shear force for the condition of operating plus OBE is resisted by the anchor bolts with a concrete bearing factor of safety of 2.8. The shear force for the condition of operating plus SSE plus jet condition is resisted by the anchor bolts with a concrete bearing factor of safety 1.76.

The concrete shear which resists the anchor bolt shear force without considering friction is a peripheral or punching type shear. The ACI 318-63 allowable ultimate stress for peripheral shear when shear reinforcement is provided is $6\phi\ (f'_c)^{0.5}$. The resulting factor of safety for OBE shear load equals 3.5 and for SSE shear plus jet load equals 1.1.

For the case of OBE, the allowable shear stress at the base of the pedestal is $1.1\ (f'_c)^{0.5}$. The concrete takes most of this shear (60 psi) and the horizontal reinforcing in the Reactor Pedestal takes the excess shear, (2.3 psi) acting as stirrups. This stress represents a very low stress for the stirrup reinforcing.

For the case of SSE + jet + temperature the allowable shear stress is $3.5\ \phi(f'_c)^{0.5}$. The concrete has the capacity to take this shear load and no shear reinforcing is required since the total shear stress is 154 psi which is less than the allowable.

Drywell Shear Ring/Reactor Pedestal Loading[16]

The shear ring connected to the Drywell has the capacity to resist the shear delivered to it from the Reactor Pedestal. For the OBE condition, the bearing pressure of the concrete against the ring is 750 psi and the shear across the ring itself is 1,930 psi. For SSE + accident + temperature, the bearing pressure of the concrete against the ring is 2,300 psi and the shear across the ring itself is 4,670 psi. The stresses are below the allowables for concrete of 750 psi for OBE and 2,700 psi for SSE; and 8,750 psi for the ring material as per the ASME Code.

There are two safety factors relating to the action between the concrete foundation within the Drywell, and the Drywell shell. The factor of safety against overturning and the factor of safety against sliding, neglecting friction.

a - The maximum overturning moment for DBE + accident = 100,000 K-ft

The stabilizing moment for this condition = 185,000 K-ft

$$\text{Factor of safety (overturning)} = \frac{185{,}000}{100{,}000} = 1.85$$

b - The forces resisting sliding neglecting friction are discussed in the paragraph above, where the sliding forces resulted in stresses in the concrete which were below the allowable for OBE and for SSE + accident.

2.5.7 Primary Containment (See Tables C-2-8 through C-2-10)

The Primary Containment is a radioactive material barrier consisting of the Drywell, in which the Reactor Pressure Vessel is located, the Suppression Chamber, and process lines out to the first isolation valve outside the containment wall.

TABLE C-2-8
DRYWELL MEMBRANE STRESSES

Description/Criteria	Method of Analysis	Load Combination*	Max. Allowable Stress-ksi
The vessel is bulb shaped and houses the Reactor Pressure Vessel, the coolant recirculation lines, pumps, etc. In case of an operating accident the vessel must contain the steam released within the Drywell, and conduct this steam to the Suppression Chamber. Structural steel plate material is ASME SA516 fabricated to ASTM Designation A300. Minimum service temperature 30°F with Charpy impact requirements at maximum 0°F. Seismic design load includes load due to vertical acceleration equal to 2/3 of the horizontal ground acceleration.	ASME Section III including Code Cases 1330-1 and 1177-5 and addenda as of June 9, 1967, for Vessel Class "B". Stress intensities are defined per Code para. N-414 and their limits are per Code para. N-414. End conditions are found with methods described in the book Theory of Plates and Shells by Timoshenko.	D + R + E D + R + E	Primary General Membrane Sm = 17.5 @ 281°F Primary Local Membrane PL = 1.5 Sm = 26.25 @ 281°F
After an accident the Drywell may be flooded up to el. 1,000', stresses shall be below yield point (without seismic load), or may exceed yield but with a margin against rupture if seismic is considered.		D + R + E	Primary + Secondary + Bending Q = 3 Sm = 52.50 @ 281°F
Accident load (R) includes pressure and temperature in the primary containment.		D + E + Flood	Yield 38.0 @ ambient Ultimate 70.0 Critical Buckling 24.21 (meridional)

* The seismic load due to the OBE (E') was found by inspection to be not governing; for the OBE the margins between the maximum allowable stresses and the calculated stresses would be greater than for the SSE (E).

TABLE C-2-9
JET IMPINGEMENT FORCE STRESSES

Description/Criteria	Method of Analysis	Load Combination*	Max. Allowable Stress-ksi
A jet force is assumed to occur in any direction within the drywell.	Find maximum load on shell prior to breaking	D + R	30.33 (Equipment door)
	Compare with given jet load	D + R	26.84 (Jet deflector support at vent)
The force is calculated as 1,250 psi pressure acting on the area equal to the cross section of ruptured pipe.	Apply the smaller load of either of the above: calculate deformation, limiting stresses to prevent a progressive deformation or strain as follows:	D + R	90 (90% of yield of T-1 Steel at 300°F) (Jet deflector baffle plate)
		D + R	30.33(Top closure head)
The jet impingement force is considered to act coincidentally with the design internal pressure and 150°F shell temperature.		D + R	30.33 (Cylinder above flanges)
	(a) $P_m + P_b \leq 0.9$ yield (b) $P_m + P_b + Q \leq 2$ yield	Experimental test in 1964	CB&I experimental investigation proved that 3/4" thick plate can deform 3" without failure (Spherical shell)
Temperature of the shell and welds are assumed to be 300°F if hit directly by jet.	Pm = general primary membrane stress Pb = primary bending stress Q = secondary membrane + bending stress	D + R	30.33 (Cylindrical shell)
		D + R	27.90 (Upper spray header pipe)
Local thermal effects and dynamic jet effects are disregarded.	Yield taken from ASME Sec. III Table N-424 at pertinent temperature	D + R	21.20 (Upper spray header weld)
There is 2" air gap between drywell shell and backup concrete.	Assume shear type failure of weld, and its stress equal to 7/10 of parent material.	D + R	30.33 (Upper spray header support)
		D + R	27.90 (Upper spray header inlet pipe)
Material is ASME SA516 Grade 70 fabricated to ASTM Designation: A300 or T1 by USS where noted.		D + R	27.90 (Lower spray header pipe)

TABLE C-2-9 (Continued)
JET IMPINGEMENT FORCE STRESSES

Description/Criteria	Method of Analysis	Load Combination*	Max. Allowable Stress-ksi
The load combination D+R is a lesser case of D+R+E or D+R+E'. The effect of E or E' is insignificant when compared to the effect of the jet impingement force.		D + R	21.20 (Lower spray header weld)
		D + R	30.33 (Lower spray header support)

* The seismic load due to the SSE (E') was found by inspection to be not governing; for the SSE, the margins between the maximum allowable stresses and the calculated stresses would be greater than for the OBE (E).

References: Formulas for Stress & Strain by Roark (4th Edition)
"Analysis of Shells of Revolution" by A. Kalnin, Journal of Applied Mechanics, September, 1964
"Stresses from Radial Loads" by P. P. Bijlaard, Welding Journal, December, 1954
"Theory of Plates" by Timoshenko

TABLE C-2-10
DRYWELL STABILIZER SHEAR LUGS

Description/Criteria	Method of Analysis	Load Combination*	Max. Allowable Stresses-ksi
The stabilizer mechanism transfers into building the reaction due to seismic loads or seismic plus jet loads acting on the Drywell, reactor and shield, or seismic, plus flooding of the Drywell.	ASME Code Section III including addenda as of June 9, 1967, for vessels Class "B" Formulas for Stress and Strain by Roark, Case 22 for plate.	D + E	<u>Male Lug</u> σ_B=(plate) 22.8 σ_S=(plate) 13.8 σ_C=(weld) 15.8
	σ_C = combined stress =	D + R + E	σ_B=30.32 σ_S=18.35 σ_C=21.01
The geometry of the stabilizer allows for radial and vertical movements due to pressure and temperature.	$\sqrt{(\sigma_B + \sigma_T)^2 + \sigma_S}$ σ_B = bending stress σ_T = tensile stress σ_S = shear stress σ_C = F_b (AISC)	D + E + Flood	σ_B=30.32 σ_S=18.35 σ_C=21.01
Materials: Components attached to the drywell are ASME SA516 Grade 70 fabricated to ASTM Designation: A300, per ASME Code Section III: Components outside the drywell are ASTM Designation: A36 per AISC-1963		D + E	<u>Female Lug</u> σ_B=22.8 σ_S=13.8 σ_C=15.8
		D + R + E	σ_B=30.32 σ_S=18.35 σ_C=21.01
Stress increase by 1/3 is allowed for jet loading or flooding.		D + E + Flood	σ_B=30.32 σ_S=18.35 σ_C=21.01
Accident loads (R) considered include jet loads, temperature and pressure.			

* The seismic load due to the SSE (E') was found by inspection to be not governing; for the SSE, the margins between the maximum allowable stresses and the calculated stresses would be greater than for the OBE (E).

The original design, fabrication, erection, inspection, and testing of Primary Containment conforms to ASME Boiler & Pressure Vessel Code, Section III (for Class B Vessel) using addenda up to and including the Summer, 1967 Addenda, and Code Cases 1177-5 and 1330-1. Elements such as platforms and accessories conform to AISC 6th Edition. Piping conforms to USAS Pressure Piping Code B.31.1.0, 1967. Safety and construction requirements conform to the codes and regulations of the State of Nebraska.

The Drywell is analyzed for load combinations corresponding to several different plant conditions including testing, normal operating, refueling, accident, and flooding conditions. Accident load (R) includes pressure and temperature in the primary containment. Jet loads due to a pipe break inside containment are evaluated separately as discussed in the paragraph below. Stresses are calculated for Primary General Membrane, Primary Local Membrane, and Primary + Secondary + Bending stress conditions and are all within the appropriate Code allowables.

The Drywell shell, and the miscellaneous appurtenances (e.g., doors, jet deflectors, spray header, etc.) are evaluated for jet impingement stresses as a result of a pipe break inside containment. These components are evaluated for the load combination of design loads (D) + accident loads (R) (includes pressure and temperature transients as well as jet loads). The load combination D+R is a lesser case of D+R+E or D+R+E'. The effect of E or E' is insignificant when compared to the effect of the jet impingement force, therefore was not typically evaluated. Jet force analysis is described in more detail in USAR Section C-2.5.7.2.

The Drywell stabilizer shear lugs are analyzed for load combinations corresponding to several different plant conditions. The calculated stresses are all within the appropriate Code allowables.

The elements of the pressure suppression support system designed by ASME B and PV Code, Section III, and the structural elements designed by AISC specifications are listed below along with the sections of each code that applies.[17]

Torus Columns, Ties, Bracing Struts, and Seismic Ties

Designed per AISC Specification. Materials and inspection of welds to the torus shell all in accordance with ASME Section III. Stress analyses are described in the <u>CNS Plant Unique Analysis Report</u>.

Drywell Stabilizer Supports

Designed per ASME Section III with allowable stresses, material, and inspections in accordance with ASME Section III.

Torus Ring

Designed per ASME Section III with stress analysis as described in the <u>CNS Plant Unique Analysis Report</u>, material, and inspections in accordance with ASME Section III.

The original design analysis for Primary Containment was based on the following general references:

Formulas for Stress and Strain by Roark (4th Edition)
Theory of Plates and Shells by Timoshenko.
Woinowsky-Krieger (2nd Edition)
Beams on Elastic Foundations by Hetenyi (7th Printing)
University of Michigan Press
Welding Research Council Bulletin 107
Process Equipment Design by Brownell & Young (1959)
AISC Manual of Steel Construction (6th Edition)
Some problems were solved using other references.

As part of the Mark I Containment Program, the Primary Containment System was reevaluated to include hydrodynamic loads resulting from SRV discharges and postulated LOCA events. As a result of these evaluations, the Suppression Chamber support system was reinforced to enhance the response characteristics of the Suppression Chamber (also referred to as the torus) and to increase its load carrying capacity. The Mark I containment program is described in more detail below.

2.5.7.1 Mark I Containment Program

The Mark I Containment Program involved a complete reevaluation of the CNS Mark I Primary Containment to include hydrodynamic loads that had not been included in the original design. This reevaluation included extensive testing and analysis, and resulted in a series of modifications which restored the originally intended design safety margins. This section briefly describes the history of this program, these loads, and the loading combinations, allowable limits, analytical procedures used for structural evaluations, and the resulting Long Term Program modifications. Mark I piping (Torus Attached Piping) evaluations are described in USAR Section C-3.3.3.5.3.

2.5.7.1.1 Program History

The Mark I containment reevaluation program began in late 1974 with the discovery of internal damage in the suppression chambers of several foreign and domestic power plants.[47] In early 1975, the NRC transmitted letters to NPPD[52,53] relating to hydrodynamic loadings associated with SRV discharges and LOCA events which were not explicitly considered in the original design of the containment system. They also requested that these loads be quantified and an assessment be performed of the effects of these loads on the CNS containment components.

Recognizing that these evaluation efforts would be similar for all Mark I BWR plants, NPPD joined an ad hoc Mark I Owners Group with G.E. as the lead technical organization. The objectives of the Owners Group were to determine the magnitude and significance of these dynamic loads and to identify courses of action needed to resolve outstanding safety concerns. The Mark I Owners Group divided this task into two programs: a Short-Term Program (STP) for early assessment of critical components, and a Long-Term Program (LTP) for final resolution of the issues.

General studies on the new postulated loads were conducted by G.E., Bechtel Power Corporation, and others. NPPD retained Kaiser Engineers, Inc., and EDS Nuclear, Inc., to determine the impact of program conclusions on CNS and perform the plant-unique load definitions and structural evaluations to restore the originally intended design safety margins at CNS.

2.5.7.1.2 Short-Term Program

The objectives of the Short-Term Program (STP) were to verify that the Primary Containment System would maintain its integrity and functional capability when subjected to the most probable loads induced by a postulated design-basis LOCA, and to verify that continued plant operation was not inimical to the health and safety of the public. The STP justified interim plant operation while further tests and evaluations were conducted during the comprehensive Long term Program (LTP). During the STP review, structural safety margins were increased by the implementation of procedures to maintain a differential pressure of at least one pound per square inch between the drywell and the torus during reactor operation. Upon completion of the LTP, the requirement to maintain a differential pressure was recinded.[81] In addition, during the course of the STP review, NPPD performed modification to the Suppression Chamber support system to provide additional design safety margins.

2.5.7.1.3 Long-Term Program

The Long-Term Program (LTP) activities were initiated in June, 1976. The objectives of the LTP were to establish design basis loads that are appropriate for the life of each Mark I BWR facility, and to restore the originally intended design safety margins for each Mark I containment system. These objectives were satisfied through extensive testing and analytical programs that led to the development of generic methods for the definition of suppression pool hydrodynamic loading events and the associated structural assessment techniques. The program also included establishment of structural acceptance criteria, and evaluations of both load mitigation devices and system modifications to improve margins of safety. The results of these generic studies are available in numerous Topical Reports.

The results of the plant-unique analyses demonstrating compliance with Mark I Containment Program requirements for the CNS Primary Containment System and associated piping are documented in the CNS Plant Unique Analysis Report.

During Mark I LOCA tests at the full-scale test facility, unanticipated cycling of the vacuum breaker valve occurred which resulted in significant damage to the valve. As a result, plant-specific vacuum breaker structural evaluations were required to ensure that valve disc closing impacts, caused by Drywell vent pressure oscillations during the chugging phase of a LOCA, would not result in damage sufficient to prevent the vacuum breakers from performing their intended safety function. Maximum expected impact velocity for this event was determined by Continuum Dynamics, Inc. For the CNS-specific vacuum breaker evaluation, the maximum impact velocity is 5.821 radians/second at a 0.0 psid vacuum breaker setpoint. This is the bounding impact velocity since impact velocity decreases with increasing setpoint. The vacuum breaker structural evaluation determined that component stresses due to hydrodynamic impact loads are all within ASME Code allowable stress levels.[58][59]

2.5.7.1.4 Loads

The design loads used in performing the Mark I containment reevaluation were determined using the criteria established in the NRC Safety Evaluation Report.[32] The load definition procedures for hydrodynamic loads in the CNS containment reevaluations were taken from the Load Definition Report (LDR)[54], as modified by the NRC Acceptance Criteria.[55]

The additional loads which were not considered in the original design basis are briefly described below.

LOCA-Related Loads

In the event of a postulated LOCA, reactor steam and water would expand into the Drywell atmosphere. Depending upon the size of a postulated pipe break inside the Drywell, three LOCA categories are considered. These categories are the Design Basis Accident, Intermediate Break Accident, and Small Break Accident. The discharge of an air-steam mixture into the Suppression Chamber during a LOCA results in Suppression Chamber pressurization and heat-up, and hydrodynamic loads. These hydrodynamic loads are identified as follows:

1. Pool Swell - results from the air in the vent system being forced into the suppression pool at a sufficiently high rate that the upper water volume of the pool is displaced upward, later falling back to its original position;

2. Condensation Oscillation - results from steam or a steam-and-air mixture flowing through the vent system at a high rate, and forming discharge bubbles at the end of the downcomers which oscillate in size and pressure;

3. Chugging - is a result of intermittent flow of nearly pure steam through the downcomer exits and into the suppression pool, forming large bubbles which expand and then rapidly collapse.

SRV Discharge-Related Loads

CNS is equipped with SRVs to control primary system pressure transients. For these transients, the SRVs actuate to divert part or all of the generated steam to the suppression pool. Following an SRV actuation, steam enters the SRV discharge lines, compressing the air and expelling the water slug in the submerged portions into the suppression pool. Following water clearing, the compressed air is accelerated into the suppression pool and forms high-pressure bubbles. These bubbles expand and contract a number of times before they rise to the suppression pool surface. This is followed by injection of essentially pure steam into the pool. The loads on the Suppression Chamber, Suppression Chamber internal structures, and attached piping as a result of this discharge are referred to as SRV discharge-related loads.

2.5.7.1.5 Load Combinations and Allowable Limits

The structural and mechanical acceptance criteria and the general analysis techniques were obtained from the Mark I LTP Structural Acceptance Criteria Plant Unique Analysis Application Guide.[33] This Application Guide also defines the general categories of structures, the design load combinations, and the corresponding service level limits for all the structural components. Potentially bounding load combinations were identified for evaluations of the structural components.

In general, requirements of the ASME B&PV Code, Section III, S77 and Code Case N-197 were used to determine the allowable stress limits.[61]

2.5.7.1.6 Method of Analysis

Static and dynamic analysis procedures were used to evaluate the Suppression Chamber shell and supports, vent system, and structural components internal to the Suppression Chamber. Three-dimensional finite element models of segments of the Suppression Chamber and vent system were developed, taking advantage of symmetry conditions. The enclosed fluid in the Suppression Chamber was modeled using the added-mass formulation to consider fluid-structure interaction.

An additional stress analysis of the torus was performed to establish a general corrosion allowance. The model used in this analysis explicitly modeled the enclosed fluid using acoustic fluid elements. The details of this additional analysis are documented in the CNS Plant Unique Analysis Report.

Elastic analysis procedures were used for all components except the platform system components internal to the Suppression Chamber. Since this structure is classified as non-safety-related, its design was in accordance with the Limit Analysis Design rules of the ASME Code.

The ASME B&PV Code, Section III, S77, was generally used in demonstrating the margins of safety required for steel structures and piping. The combined state of stress for each of the structural components meets the allowable values from the ASME Code for all design load combinations. Fatigue usage was found to be within allowables at all critical locations. Stability against buckling was also verified for the Suppression Chamber shell and the vent pipes. In several cases, direct application of the LTP design requirements resulted in unusual hardship without a compensating increase in plant safety margins. Alternate analytical approaches or interpretations were used in these cases. These exceptions to design requirements are discussed in the CNS Plant Unique Analysis Report.

2.5.7.1.7 Modifications Completed

The Mark I containment reevaluation program resulted in numerous modifications to the containment system components which restored the originally intended design safety margins. Table C-2-11 lists the modifications that have been completed. Modifications to existing containment components and supports were designed, fabricated, and installed to the requirements of the ASME B&PV Code, Section III, S77. Modifications involving new structural components were also designed, fabricated, and installed to the requirements of ASME B&PV Code, Section III, S77. Modifications to existing structural components were designed, fabricated and installed to the requirements of the original code of record. This code of record was typically the latest edition of the AISC Code.

G.E. performed an evaluation of the LDR SRV Load Cases for CNS. Plant changes were implemented which mitigates SRV subsequent actuation-induced loads during postulated LOCA events.[56] The evaluation performed was NEDE-22197 (December, 1982) and installed under DC 83-001, Low-Low Set. The primary concerns were the potential high thrust loads on the discharge piping, and the high frequency pressure loading on the containment. G.E. concluded that delayed isolation achieved by means of a PCIS Group 1 (MSIV closure) low reactor water level trip setpoint at Level 1, combined with a 90 psi minimum low-low set relief logic, produced the maximum potential benefit. These design changes are discussed in USAR Section IV-4.5.[57]

TABLE C-2-11

SUMMARY OF MARK I CONTAINMENT AND PIPING MODIFICATIONS

COMPONENT NAME	NATURE OF MODIFICATION
STRUCTURAL COMPONENTS	
Suppression Chamber Shell and Supports	
Suppression Chamber Support Column	Plate reinforcement to column web and flanges
Column Anchorage	Installed anchor bolts, brackets, and box beam assemblies
Column-to-Supp. Chamber Connection	Additional full penetration weldment
Suppression Chamber Saddle	Full saddles connecting suppression chamber support columns
Ring Girder	Web stiffeners; local reinforcement of weld to shell
Vent System	
Vent Header/Downcomer Intersection	Reinforced 80 penetrations with stiffener plates and pads
Downcomers	Reduced downcomer submergence by truncation
Downcomer Ties	Installed tie bar and ring assembly at each downcomer pair
Vent Header Deflector	Installed deflector assembly in all suppression chamber bays
Vent Header Supports	Removed existing supports; resupported from girder above
DW/WW Vacuum Breakers	Reinforced 12 vacuum breaker penetrations
Miscellaneous Supp. Chamber Internals	
Monorail	Installed midbay supports in all suppression chamber bays
Service Platform	Replaced existing supports; added new supports, bracing and grating tie-down
Drywell Steel Framing	Reinforcement of beam set connections and framing members
MISCELLANEOUS SYSTEM MODIFICATIONS	
Drywell/Wetwell Pressure Differential	Installed Pump Around System
Supp. Chamber Temp. Monitoring System	Installed monitoring system and instrumentation
PIPING SYSTEMS	
S/RV Discharge Piping	
Wetwell Piping	Rerouted with stronger pipe; added 12 new supports
T-Quencher Discharge Device	Installed T-quencher device on each S/RV line
T-Quencher Support	Installed quencher support assembly in 8 bays
Quencher Support Bracing	Installed quencher support bracing in 8 bays
Vacuum Breakers	Installed two, 10-inch vacuum breakers on each line
Pipe Supports and Restraints	Installed 89 new or modified supports in Drywell
Torus Attached Piping	
Large Bore Supports	Installed 151 new or modified supports
Small Bore Supports	Installed 54 new supports
Small Bore Rerouting	Rerouted 5 lines
Branch Line Supports	Installed 25 new or modified supports
Suppression Chamber Penetrations	Reinforced three large bore penetrations
Valve Operator Supports	Reinforced 13 valve yolks
Pump Anchors	Modified anchorage of 4 RHR pumps
Torus Internal Piping	
HPCI Turbine Exhaust	Rerouted and resupported HPCI sparger
RCIC Turbine Exhaust	Rerouted and resupported RCIC sparger
Core Spray Return Test Line	Truncated test lines
RHR Return Test Line	Installed reducer, discharge elbow, and new supports
Spray Header	Reinforced existing supports
Vent Drain Line	Rerouted lines and installed supports

2.5.7.2 Pipe Whip and Jet Force Analysis

2.5.7.2.1 General

The measures used to assure that the Drywell shell and all essential systems and components (as defined in Sections 3.6.1 and 3.6.2 of NUREG-0800, "Standard Review Plan") within the containment (components of the primary and secondary coolant systems, engineered safety features, and equipment supports) have been adequately protected against blowdown jet forces are similar to measures used on other plants of a design similar to CNS. A brief discussion is presented here, but a more detailed description of these protection measures was provided in Quad Cities Units 1 and 2, Docket 50-254 and 50-265, FSAR Amendments 23 and 25; Vermont Yankee, Docket 50-271, FSAR Amendment 26; and Pilgrim, Docket 50-293.

As noted in USAR Section XII-2.3.6 and USAR Table XII-2-6, the Primary Containment System is designed to withstand all forces associated with a postulated Loss-Of-Coolant Accident, including forces resulting from impingement of steam and/or water from a pipe which has been postulated to be ruptured.[63]

2.5.7.2.2 Jet Force Analysis[49]

For the analysis of the Drywell shell in the vicinity of penetration nozzles, the loading combination of pressure plus dead and live load plus Operating Basis Earthquake (OBE) plus thermal expansion load plus jet force from a pipe break is categorized as an Emergency Condition. When Safe Shutdown Earthquake (SSE) load is substituted for OBE load, the loading combination is categorized as a Faulted Condition. The allowable membrane stresses are as specified in the ASME B&PV Code, Section III [Class B] The classification of stresses are in accordance with Table N-413 of Section III. The limits of stress intensities are in accordance with Figure N-414 of Section III (Summer 1970 Addenda) for the applicable condition. This includes the effect of longitudinal or circumferential type rupture of the penetrating pipe, as well as pressure impingement from an adjacent pipe. Although the "Maximum Seismic" combination is categorized as a Faulted Condition loading, containment vessel components were analyzed to Emergency Condition stress criteria, utilizing elastic analysis. No containment vessel components were designed to the faulted condition.[21,22,49]

The Main Steam and Feedwater piping penetrations are provided with expansion bellows seals between the flued head fittings and the penetration nozzles. As shown in Figure V-2-3, these seals are protected from pressure impingement by means of coaxial guardpipes and continuous rings at the inside end of the guardpipe, which serve the function of jet deflectors. These penetration assemblies are also furnished with axial, torsional and lateral limit stops at the flued head fittings, as well as pipe anchors on the piping outside of the outer isolation valves for protection of the penetration assemblies due to the effects of jet force loading.[49]

For protection of other penetration assemblies, when required to resist the effects of jet force loading, lateral limit stops have been furnished in the vicinity of the applicable flued head fittings.[49]

2.5.7.2.3 Pipe Whip Analysis[49,50,51]

The Reactor Recirculation System piping within the Primary Containment has been provided with a system of pipe restraints designed to protect the Drywell shell and adjacent essential piping systems and components from the effects of pipe whip. The maximum distance between restraint brackets is no greater than that distance between any one restraint and the Drywell shell plate.

The design criteria for the pipe restraints are as follows:

1. The pipe restraints are arranged and provide sufficient clearance so as not to interfere with normal system operation, including Operating Basis Earthquake Loads.

2. The magnitude of the pipe restraint design loads is established as the product of system operating pressure and pipe flow area. Fluid dynamics are not considered as the ruptured pipe is assumed to load the restraints in a slow even manner before fluid dynamics occur.

3. The allowable stresses for the restraint brackets is established as 150 percent of AISC Code allowable stresses for the materials used. The pump restraint cables are limited to 90% of the catalog advertised breaking strength.

4. The restraint brackets are fabricated from ASTM - A36 cold rolled steel and the cables are constructed from extra strength improved plow steel wire rope.

5. The physical separation provided in the arrangement of redundant engineered safety features provides spatial separation to preclude concurrent damage to more than one redundant safety feature by a single postulated pipe failure.

6. The pipe ruptures are postulated to occur anywhere in the Reactor Recirculation System and are assumed as either circumferential guillotine type breaks or longitudinal pipe splits. For piping systems other than the Reactor Recirculation piping system, the following criteria of postulated pipe rupture location are applied:[50]

   a. Stress Criteria

   Any intermediate points between terminal points, where the primary plus secondary stresses derived on an elastically calculated basis under loadings associated with normal and upset operating conditions (i.e., dead weight plus pressure plus Operating Basis Earthquake plus thermal loads) exceeds 2 S_H, where S_H is the hot allowable stress specified in USAS B31.1.0-1967.

   b. Criteria for Pipe Size and Type of Break

      (1) Longitudinal split in piping exceeding one (1) inch and up to four (4) inches nominal size.

      (2) Circumferential breaks in piping exceeding one (1) inch nominal size.

c. Pipe Pressure Criteria

Only that piping, which is directly exposed to RPV pressure during normal reactor operation, is postulated to rupture.

d. Terminal Point Criteria[51]

Piping breaks were postulated at all terminal points in each piping run or branch run in all piping within Primary Containment, except where:

(1) both of the following pipe system conditions are met:

a. the service temperature is less than 200°F, and
b. the design pressure is 275 psig or less, or

(2) the piping is physically separated (or isolated) from other piping or components by protective barriers, or restrained from whipping by plant design features, such as concrete encasement, or

(3) following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system or component important to safety, or

(4) the internal energy level associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

The critical piping systems were investigated originally for location and magnitude of maximum stress, as well as the stresses at terminal points, such as RPV nozzle attachments and branch connections to major piping runs.[50] The stresses are due to dead weight plus pressure plus Operating Basis Earthquake plus thermal loads.

The original summary of stresses is shown on Table C-2-12.

The original analyses as well as subsequent reanalyses show that all intermediate points between terminal points have primary plus secondary stresses less than the critical value 2 S_H. Summaries of the stresses associated with the subsequent analyses are documented in the existing calculations of record.

In order to provide additional assurance of protection against violation of containment, CNS installed a redundant safety system comprised of energy absorbing tornado siding to line the reactor containment in critical areas as described in USAR Section C-2.5.7.2.4.

TABLE C-2-12

ORIGINAL TABULATION OF CRITICAL PIPE STRESSES DUE TO DEAD WEIGHT PLUS PRESSURE, PLUS OPERATING BASIS EARTHQUAKE, PLUS THERMAL LOADS

System Description	$2\ S_h$ (PSI)	Maximum Allowable Stress (PSI)	Terminal Point $1.2\ S_h$ (PSI)	Terminal Point Calculated Stress (PSI)	Terminal Point Location
Main Steam	35,000	18,710	21,000	7,395	R.P.V. Nozzle
Core Spray Disch. Pump 1B	27,400	11,060	16,440	8,276	R.P.V. Nozzle
Core Spray Disch. Pump 1A	27,400	10,892	16,440	8,013	R.P.V. Nozzle
Clean-Up Recirc. Pump Suct.	28,900	24,360	17,350	17,217	20" RH-1D HDR.
10" MS-1 to HPCI Turb.	30,000	22,120	18,000	7,690	24" MS-1 HDR.
3" MS-1 to RCIC Turb.	30,000	10,957	18,000	6,128	24" MS-1 HDR.
Reactor Feed Water	27,400	12,300	16,440	11,340	N.E. & S.E. R.P.V. Nozzle
Reactor Feed Water			16,440	8,810	N.W. & S.W. R.P.V. Nozzle
RHR Pump Suct.	27,400	17,200	16,440	6,418	Conn. to Recirc. HDR
RHR Pump Disch.	30,000	13,528	18,000	10,718	South Conn. to Recirc. HDR
RHR Pump Disch.	30,000	14,371	18,000	11,111	North Conn. to Recirc. HDR
RHR Head Spray*	27,400	9,449	16,440	7,650	R.P.V. Nozzle

* RHR Head Spray was removed per DC 86-78.

NOTE: See existing calculations of record for updated stress values.

It is noted that no dynamic analysis for pipe whip was performed.[50]

Piping within the containment, such as the Core Spray and RHR systems has been designed such that the physical separation provided from one another will preclude concurrent damage and thus not affect the core cooling capacity.

A comparison of the analysis performed in Amendment 3 to the Fitzpatrick FSAR, Docket No. 50-333, with the operating characteristics of CNS has shown that the core heatup resulting from an additional single active failure, beyond the initial pipe rupture, would produce a peak clad temperature that does not exceed the 1,370°F noted in the Fitzpatrick response. This is based on a Main Steam line rupture with the failure of an adjacent Core Spray and Reactor Recirculation riser line.

The difference in design of support structures within the Drywell, between the Fitzpatrick and CNS facilities, preclude the addition of piping restraints on CNS such as those which were committed to in Amendment 3 of the Fitzpatrick FSAR. However, the special containment impact protection discussed in the next section provides the desired redundance in pipe whip protection.

2.5.7.2.4 Pipe Whip Energy Absorbing Material[50]

An investigation was conducted to determine the feasibility of selectively installing energy absorbing material inside the Drywell to reduce the potential mechanical effects to the interior of the Primary Containment resulting from the failure of a large, unrestrained pipe in the primary pressure boundary.

The piping in question is large pipe which is normally pressurized to reactor pressure (Main Steam, HPCI Steam Supply, Feedwater, and RHR). For this study, large pipes are assumed to rupture by instantaneous and complete severance at circumferential butt welds with the jet reaction force acting normal to the rupture surface and resulting in pipe movement around a plastic hinge.

In developing a solution to protect the reactor containment from pipe whip resulting from such a pipe rupture, the following guidelines were utilized:

1. The equipment or material substantially reduces the potential for reactor containment rupture.

2. The equipment or material installed does not weaken or endanger either structures, piping or components within the reactor containment during normal or accident conditions.

3. The equipment or material is adaptable for use with existing structures.

4. The equipment or material does not hinder access for inservice inspection.

Based on the above investigation, CNS installed tornado siding as manufactured by H. H. Robertson Company, to line the Drywell shell in critical areas, where impact from a whipping pipe in question might penetrate the containment (see Burns and Roe Drawing 4286). This tornado siding is made of corrugated high-strength steel with a steel plate backing. All material used in construction is 12 gage carbon steel with a yield point of 60,000 psi. The siding is capable of absorbing .93 x 10^6 ft-lbs of kinetic energy per square foot. The material is compatible with the containment environment under all normal or accident conditions. The siding is typically nine inches thick and is fabricated out of rectangular sections varying in size from 2' x 3' to 2' x 9'. The sections are connected to the Drywell shell. The use of small sections of siding permits the liner to follow the contour of the Drywell shell. The liner does not restrict access to piping, welds or components for inservice inspection. Special care has been taken to assure that proper and adequate Containment Spray distribution is not affected. The installation of the liner has a negligible affect on the free Drywell volume and has no affect on the accident or post-accident environment.

The use of this liner provides a high degree of confidence that the integrity of the containment barrier will be assured, to prevent an uncontrolled release of fission products.

2.5.8 Main Steam Tunnel

The Main Steam Tunnel is a biological enclosure, with 5'-0" thick walls, floors, and ceilings (except the floor of the main section is only 2’-0" thick), for the four Main Steam and the two Feedwater lines going from the Reactor Building to the Turbine Building.[15] The inside dimensions are 28'-0" wide by 48'-3" long by 25’-0" high in the main section, and 11’-0" high in the shallow section. The enclosure houses a structural steel anchor for resisting operating and jet loads from the Main Steam and Feedwater lines. This support is anchored to the sides and bottom of the tunnel. Also located within the enclosure are the second (outer) isolation valves for two systems. Blowout panels consisting primarily of light-weight cellular concrete are located in the pipe chase blockouts of the wall separating the Main Steam Tunnel and the Turbine Building.[41] These blowout panels are designed to rupture[42] in postulated events of a High Energy Line Break (HELB) to relieve the resulting pressure in the Main Steam Tunnel. These blowout panels also function as a secondary containment boundary.

The analysis of the Main Steam Tunnel considered, in addition to the loadings associated with a postulated pipe break accident (pressure buildup and impact forces due to pipe whip), temperature effects, reactions due to pipe anchors and seismic effects. The seismic effects were defined as equivalent static forces corresponding to a horizontal component of earthquake equal to 30% of gravity. This value is consistent with the seismic criteria established for the Reactor Building for the Safe Shutdown Earthquake. The allowable stresses for this combination of loads are described in USAR Section App.C-2.3. The analysis concludes that the structural integrity of the Main Steam Tunnel is maintained under the load conditions expected to occur at the time of the hypothesized pipe break accident.[25]

Local penetration, due to a whipping pipe, into the 5’-0" thick reinforced concrete wall is evaluated to be a maximum of 10.3", using analytical procedures and formula contained in: "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants," by Richard C. Gwaltney, Oak Ridge National Laboratory, Report #ORNL-NSIC-22. The computed penetration depth is less than 1/3 the total wall thickness. It is concluded, therefore, that the structural integrity of the wall will not be locally impaired.[25]

The time variation of steam pressure and temperature following a steam line break in the Main Steam Tunnel was initially calculated using the CONTEMPT/ESCAPE computer program, was recalculated using the EDSFLOW (a version of RELAP4/MOD5) computer program (EDS calculation 0840-002-5.0)[39] for evaluation of environmental effects, and was again recalculated using the GOTHIC computer program (NEDC 96-006)[40] to validate the results of the EDS calculation using a newer and more accurate analysis methodology. The pressure and temperature distribution curves from the EDS calculation indicate a peak temperature of 287°F and a peak pressure of approximately 15 psig after the pipe break. An investigation was made of vapor temperatures in the Main Steam Tunnel, and in the Reactor and Turbine buildings. The temperature differences in the two atmospheres do not create a temperature gradient in the tunnel wall because their durations are not long enough to allow a gradient to penetrate the full thickness of the wall. Hence; the temperature condition in the concrete is only a "skin" effect. Even though the pressure rise is a dynamic type of loading condition, since it occurs for a very short duration, the average pressure of 15 psig was conservatively used as a static load. The combination of these loadings will result in calculated stresses that will not exceed the yield stress in the reinforcing steel and 85% of the conservative ultimate strength (3000 psi) of the concrete as indicated in Amendment 25 of the FSAR.

The Main Steam line anchor (also referred to as the pipe tunnel anchor or piping anchor) is constructed of ASTM A-36 structural steel anchored to the wall by embedded anchor bolts. The structural steel under normal operating loads, is designed using the allowable stresses permitted by the AISC 6th Edition. In addition to the normally encountered piping loads, the anchor is designed for the jet loading resulting from the break of one Main Steam or one Feedwater line. The jet loading includes the effects of all feasible combinations of pipe rupture loads. Forces from this anchor are resisted by the 5'-0" thick Main Steam Tunnel wall. The forces encountered are transmitted from the pipe to the wall by caged trunnions which are attached to the pipe. These are in turn attached to a steel truss framework structure which transmits all postulated loadings to the tunnel walls by means of bearing plates and embedded anchor bolts.[14] For the loading condition of normal operating loads plus jet forces the stresses in the structural steel were permitted to reach 0.9 f_y. The allowable loads on the anchor bolts under normal operating loads were those recommended by the Uniform Building Code. Under normal operating plus jet loads the anchor bolt loads were kept considerably under the ultimate capacity in bearing. The concrete tunnel has been designed to withstand the jet force plus a postulated temperature and pressure rise within the confined area so that the allowable stresses in the reinforcing steel are not greater than 0.9 f_y, and the concrete stresses are limited to 0.85 f'_c.

2.5.9 Sacrificial Shield Wall[15]

The Sacrificial Shield Wall is a 47'-3¾" high cylindrical shell with an inside diameter of 21'-8". Its base sets on top of the Reactor Pedestal. The shell consists of 12 structural steel columns equally spaced around the circumference. The outside surface of the wall is formed by a 5/16 inch steel liner plate welded to the outboard flange on the columns, while the inside surface is formed by a ¼ inch steel liner plate welded to the outboard face of the column inboard flange.

A concrete fill is placed between these liner plates to act only as a radiological shield. The concrete is assumed not to have any structural significance. A heavier density shielding concrete fill, having a density of 210 lbs/ft^3 is used around the active core area. There is a removable section of steel shielding at each Reactor Pressure Vessel (RPV) penetration to facilitate inspection of the penetration and two removable sections near the base of the wall for inspection access at the base of the RPV. There are steel beams framing into the column webs to support platform floor beams and pipe hangers. The Sacrificial Shield Wall is anchored and transfers loading acting on it to the Reactor Pedestal through the column base anchor bolts and shear lugs. The RPV is laterally supported by stabilizers between the RPV and the Sacrificial Shield Wall at the top. There is a pipe truss between the Sacrificial Shield Wall and the Reactor Building to laterally support the wall.

The Sacrificial Shield Wall is designed, without considering the concrete for any structural purpose, to withstand seismic forces and to support normal pipe loading and pipe restraint loads in the event of a pipe rupture in the Drywell. Also considered in the design of the shield wall is the jet load of either a circumferential break of the Main Steam reactor vessel penetration or the Reactor Recirculation outlet penetration.

The AISC 6th Edition was used in the design of the steel in the Sacrificial Shield Wall for all the forces on the structure including the Operating Basis Earthquake without any increase in the allowable stresses. However, an analysis was made of the steel under all loading conditions including jet loads and the Safe Shutdown Earthquake loading using the allowable tension and bending stresses as 90% of the yield stress. This stress criterion is considered acceptable and adequate for the one time loading condition.

The stresses due to the loads described above do not exceed the applicable allowable stresses, previously outlined, for ASTM A-36 steel having a yield stress of 36,000 psi.

2.5.10 Concrete Attachments

A grid system of Richmond Screw Anchor Co. cast in place concrete inserts is provided on the underside of the Reactor Building floors and in walls for pipe supports and anchors. Shear and tension allowables for these inserts are the working load values recommended by Richmond Screw Anchor Co. The safe working load values for self-drilling anchors is based on the application of a factor of safety to the ultimate tension and shear values recommended by Phillips Drill Co., Inc. The safety factor used in tension is 6 to 1, and in shear is 5 to 1 except as utilized under the IE Bulletin 79-02 pipe support base plate design review. Where location permits, pipe supports and anchors are connected to supplementary structural steel which in turn is connected to the wall or ceiling with expansion bolts. For installations prior to approximately 1982, these expansion bolts are Phillips Red Head self-drilling anchors, snap-off type ranging from 1/4" diameter to 7/8" diameter and Phillips Red Head wedge anchors ranging from 3/8" diameter to 1-1/4" diameter. Each self-drilling anchor was installed with epoxy under rigidly controlled procedures.[14] For installations after approximately 1982, Hilti concrete anchors and Drillco Maxi-bolts have been used.[46] Installation of other types or brands of anchors is acceptable when approved under an Engineering Change.

In response to NRC IE Bulletin No. 79-02, an extensive concrete expansion anchor bolt verification program was undertaken in 1979. As a result, each Seismic Class IS pipe support secured by concrete expansion anchor bolts was analyzed taking into account base plate flexibility in the calculation of the anchor loads. Flexibility was also considered in cases where expansion anchor bolts were installed directly on structural members (i.e. angles, channels, etc.) instead of using baseplates. The allowable anchor loads are based on the ultimate anchor capacity based on vendor testing, using a factor of safety equal to 5 for shell type expansion anchors, and 4 for wedge and sleeve type expansion anchors. In addition, the bolt capacity takes into account the effects of shear-tension interaction (using a 4/3 elliptical interaction equation), minimum edge distance and proper bolt spacing. It was also noted that there were no cases of expansion anchor bolts being installed in concrete block (masonry) walls. Expansion anchor bolts installed after 1979 are installed in accordance with rigorous quality control procedures to ensure proper installation.[63,65]

Pipe support loads from thermal constraint of piping and from seismic excitation (including SSE), as well as dead weight loads were considered in the design of all piping systems. Anchor bolt design loads were chosen to accommodate all of these loads utilizing the appropriate factor of safety described above under static primary load conditions. All bolts covered under the IEB 79-02 program have been properly torqued to their required loads, as described above, and as such are designed to account for the effects of cyclic loads such that the state of stress on the bolts does not change throughout the life of the plant. Therefore, fatigue action and loosening of the expansion anchor bolt assembly are prevented, and the stress range above the torque values are kept to a minimum. The prescribed torque loadings have accounted for the design requirement to permit the expansion anchor assemblies to successfully support loads which are repetitive.[64]

The original construction test data indicated a maximum concrete design strength of 3,500 psi. Burns and Roe Report on Base Plates Design, dated October 7, 1980, drafted in response to NRC Bulletin 79-02, states that tests carried out on concrete samples taken from Class 1 areas shows the ultimate concrete strength over 5,000 psi. Statements from ITT Philips Drill Division Red Head Concrete Anchoring Handbook and Specifiers Guide, 1973, together with statements in the PCI Design, Precast and Prestressed Concrete, Prestressed Concrete Institute, 1971, indicate that anchor capacity increases as the concrete strength increases.

Concrete expansion anchors use the following ultimate concrete strengths based on installed location in Class 1 structures. These values are based on ACI 318-63 28 day concrete strength.

Building	Compressive Concrete Strength (fc’)
Intake Structure	5,000 psi
Control Corridor	5,000 psi
Diesel Generator Building	5,000 psi
Control Building Substructure	5,000 psi
Control Building 903'-6"	4,000 psi
Control Building 918'-0"	5,000 psi
Control Building 932'-6"	5,000 psi
Control Building Roof	4,750 psi
Reactor Building SW Quad Slab	4,200 psi
Reactor Building SE Quad Slab	5,000 psi
Reactor Building NW Quad Slab	5,000 psi

Reactor Building NE Quad Slab	5,000 psi
Reactor Building SW Quad Wall	5,000 psi
Reactor Building SE Quad Wall	5,000 psi
Reactor Building NW Quad Wall	4,690 psi
Reactor Building NE Quad Wall	5,000 psi
Reactor Building Closure Walls	5,000 psi
HPCI Room and Reactor Bents	5,000 psi
Containment Walls to 903'-6"	5,000 psi
Drywell, Quad Equipment Pads, & Stairwells to 903'-6"	3,890 psi
Reactor Building 903'-6"	4,420 psi
Reactor Building 931'-6"	4,900 psi
Reactor Building 958'-3"	5,000 psi
Reactor Building 976'-0"	4,780 psi
Reactor Building 1001'-0"	5,000 psi
Reactor Sacrificial Shield Wall	5,000 psi

The original design criteria for Philips Redhead-S Type anchor bolts associated with conduit/cable tray hangers was based upon the ICBO Codes. Using ultimate values from the manufacturer's test data the maximum allowable loading values were determined.

Event	Factor of Safety
OBE	4
SSE	3

NOTE: The factors of safety given in this paragraph only apply to original conduit/cable tray hangers. New hangers installed after January 1, 1986, are to use factors of safety as laid down in IE Bulletin 79-02 (see above paragraph). The reason for upgrading the factors of safety with regard to conduit/cable tray hanger anchor bolts is to ensure uniformity of anchor bolt design.

TABLE C-2-13
CONTROL BUILDING FOUNDATION

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress
Design load (D) includes all dead and equipment loads plus 25% of the full live loads	Maximum soil stress = $\frac{D + \Delta P}{A} = \frac{M}{S} - (\gamma D_a + \gamma_{sat} D_b)$	D + E	1.5 q_a = 18
Vertical seismic load (ΔP) is assumed to be produced by a vertical acceleration equal to ½ of the ground horizontal seismic acceleration, both acting simultaneously	where: D = Design load, kips ΔP = Vert. seismic load, ½ kips M = Horiz. seismic overturning moment, kip-ft	D + E'	1.5 q_a = 18
Overturning moment (M) is produced by the horizontal seismic acceleration and all design load eccentricities	A = Foundation mat area, sq-ft S = Section modulus of foundation mat, ft^3 γ = Unit wt. of soil above water table γ_{sat} = Saturated unit wt. of soil D_a = Depth of soil above water table D_b = Depth of saturated soil above bottom of foundation		

TABLE C-2-14
CONTROL BUILDING FLOOR SYSTEM

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
Floor system is designed for the loadings tabulated in Subsection XII-2.3.2	Working Stress Design Method - Slabs, beams, and girders	D + E	F_c = 1.35 F_v = 0.060 F_t = 20.0
Materials conform as follows: Concrete f'_c = 3,000 psi at 28 days max. strength per ACI 318-63. Reinforcing ASTM Designation: A615 Grade 40 per ACI 318-63			
Vertical seismic load is assumed to be produced by a vertical acceleration equal to ½ of horizontal ground acceleration		D + E'	F_c = 2.55 F_v = 0.093 F_t = 36.0
		D + T	F_c = 2.55 F_v = 0.093 F_t = 36.0
Maximum allowable stress for D+E' and D+T load combinations are: Concrete $F_c = 0.85\ f'_c$ Reinforcing $F_t = 0.90\ f_y$			
Maximum allowable stresses for D+E combination are not increased above code allowable values, i.e., customary 1/3 increase not used			
Tornado wind load (T) on roof averages a pressure of 270 PSF or a 3 psi suction press.			

TABLE C-2-15
CONTROL BUILDING CONCRETE WALLS

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
Design load (D) includes all dead and equipment loads plus 25% of the full live loads	Working Stress Design Bearing Walls	D + E	F_t = 20.0 F_c = 1.35 F_v = 0.060
Materials: Concrete f'_c = 3,000 psi at 28 days max. strength per ACI 318. Reinforcing ASTM A615 Grade 40 per ACI 318		D + E'	F_t = 36.0 F_c = 2.55 F_v = 0.093
		D + T	F_t = 36.0 F_c = 2.55 F_v = 0.093
Maximum allowable stresses for D+E' and D+T loads: Concrete F_c = 0.85 f'_c Reinforcing F_t - 0.90 f_t			
Maximum allowable stresses for D+E load combination are not increased above code allowable values, i.e., customary 1/3 increase not used			
Tornado wind load (T) on walls averages a pressure of 270 psf or a 3 psi suction pressure			

2.5.11 Control Building (See Tables C-2-13 through C-2-15)

The Control Building has been reviewed in a manner similar to that described above for the Reactor Building for applicable combinations of design loads. The stresses in this structure have been found to be within the appropriate allowable limits.

2.5.12 Elevated Release Point (See Table C-2-16)

The Elevated Release Point (ERP) is described in USAR Section XII-2.2.6. The ERP is designed for design loads (D), wind loads (W), and seismic loads (E, E’). The ERP is not designed for tornado loads as explained in USAR Section XII-2.2.6. All the resulting stresses are within the Code allowable values.

TABLE C-2-16
ELEVATED RELEASE POINT FOUNDATIONS AND TOWER

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
Design load (D) including all dead and equipment loads plus ½" radial ice on all members	Free standing or rigid towers foundation Working Stress Method	D + W	F_t = 20x1.33 = 26.7
Design wind loading (W) is 100 mph with wind pressures varying from 44 PSF at ground level to 73 PSF at top blowing against the projected area of members covered by 1/2" radial ice			F_c = 1.35x1.33 = 1.80 F_v = 0.110x1.33 = 0.146
Live loads on platforms: Platforms at top and 30' levels = 100 PSF Rest Platforms = 50 PSF			
Loading conditions D+E and D+E' were investigated and found not to be governing when compared with D+W			
Materials: Concrete f'_c = 3,000 psi at 28 days maximum strength per ACI 318-63. Reinforcing ASTM A615 Grade 40 per ACI 318-63			

TABLE C-2-16 (Continued)
ELEVATED RELEASE POINT FOUNDATIONS AND TOWER

Description/Criteria	Method of Analysis	Load Combination	Max. Allowable Stress-ksi
Tower Steel Leg members A440 (50,000 psi yield) Other members A36 (36,000 psi yield) Maximum allowable stresses for D+W load combinations are increased by 1/3 for foundation design. (Working stress method) Concrete $F_t = 1.33\ f_s$ $F_c = 1.33\ f'_c$ $F_v = 1.33\ v_c$	Tower Steel Ultimate design method based on formulae and provisions conforming to U.S. Bureau of Reclamation "Transmission Structures Design Standard No. 10"	D + W	F_a (leg) = 35.77 for $\frac{l}{r}$ = 57 F_a (brace) = 10.60 for $\frac{l}{r}$ = 174 F_t = 45.00 for A-440 F_t = 32.40 for A-36 F_b = 45.00 for A-440 (not used in bending) F_b = 32.40 for A-36 Soil pressure = 12.0 ksf maximum

* Max. actual stresses for load combination W+D or W-D (max uplift)

3.0 COMPONENTS

3.1 Intent and Scope

3.1.1 Components Designed By Rational Stress Analysis

These general design criteria are intended to apply to those ductile metallic structures or components which are normally designed using rational stress analysis techniques such as pressure vessels, reactor internal components, etc. The criteria may also be applied to those components or structures whose ultimate loading capability is determined by tests. These criteria are intended to supplement applicable industry design codes where necessary. Compliance with these criteria is intended to provide design safety margins which are appropriate to extremely reliable structural components when account is taken of rare event potentialities such as might be associated with a Safe Shutdown Earthquake or primary pressure boundary coolant pipe rupture, or a combination of events.

3.1.2 Components Designed Primarily By Empirical Methods

There are many important Class I components or equipment which are not normally designed or sized directly by stress analysis techniques. Simple stress analyses are sometimes used to augment the design of these components, but the primary design work does not depend upon detailed stress analysis. These components are usually designed by tests and empirical experience, which may include earthquake experience for the seismic qualification of Class I components. Complete detailed stress analysis is currently not meaningful nor practical for these components. Examples of such components are valves, pumps, electrical equipment, and mechanisms. Field experience and testing are used to support the design. Alternatively, use of the SQUG Generic Implementation Procedure (GIP-3) may be used for seismic qualification of electrical and mechanical equipment within the scope of the GIP. Where the structural or mechanical integrity of components is essential to safety, the components referred to in these criteria must be designed to accommodate the events of the Safe Shutdown Earthquake (SSE) or Operating Basis Earthquake (OBE), or a design basis pipe rupture, or a combination where appropriate. The reliability requirements of such components cannot be quantitatively described in a general criterion because of the varied nature of each component and its specific function in the system.

3.2 Loading Conditions and Allowable Limits

The loading conditions established herein are expressed in generic terms and are related in a probabilistic manner to the loads which are to be investigated for safety considerations. Related probabilistic definitions are used to determine an appropriate minimum safety factor which is used to establish structural design allowable limits and functional design allowable limits. Certain of the limits described in these criteria, i.e., deformation limit, and fatigue limit, are included for completeness, but do not necessarily require application to all components. Where it is clear to the designer that, based upon experience, fatigue or excess deformation are not of concern for a particular structure or component, a formal analysis with respect to that limit is not required.

3.2.1 Loading Conditions

The loading conditions may be divided into four categories; Normal, Upset, Emergency, and Faulted conditions. These categories are generically described as follows:

3.2.1.1 Normal Conditions

Any condition in the course of operation of the station under planned and anticipated conditions, in the absence of Upset, Emergency or Faulted Conditions.

3.2.1.2 Upset Conditions

Any deviations from Normal Conditions anticipated to occur often enough that design should include a capability to withstand those conditions. The Upset Conditions include abnormal operational transients caused by a fault in a system component requiring its isolation from the system, transients due to loss of load or power, and any system upset not resulting in a forced outage. The Upset Conditions may include the effect of the Operating Basis Earthquake.

3.2.1.3 Emergency Conditions

Any deviations from Normal Conditions which 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 specific damage developed in the system.

3.2.1.4 Faulted Conditions

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

Beginning with the 1977 Edition of the ASME Boiler and Pressure Vessel Code, these conditions are designated as Service Level A (Normal), Level B (Upset), Level C (Emergency), Level D (Faulted).

3.2.2 Allowable Limits

In addition to the generic definition of loading conditions in the preceding paragraphs, the meaning of these terms is expanded in quantitative probabilistic language. The purpose of this expansion is to clarify the classification of any hypothesized accident or sequence of loading events so that the appropriate limits or safety margins are applied. Knowledge of the event probability is necessary to establish meaningful and adequate safety factors for design. Table C-3-1 illustrates the quantitative event classifications. The probabilities of Table C-3-1 have been assigned to establish the appropriate structural design limits for the loading conditions in Subsection C-3.2.1. A summary of these limits is shown in the tables listed below.

DEFORMATION LIMIT	Table C-3-2
PRIMARY STRESS LIMIT	Table C-3-3
BUCKLING STABILITY LIMIT	Table C-3-4
FATIGUE LIMIT	Table C-3-5

There are many places where, through the exercise of designer judgment, it is unnecessary to actually carry out a formal analysis for each of these limits. A simple example consists of the case where two pieces of pipe of different wall thicknesses are joined at a butt weld. If they are both subjected to the same loading, only the thinner piece would require a formal analysis to demonstrate that the primary stress limit has been satisfied.

The term SF_{min} is defined as the minimum safety factor on load or deflection and is related to the event probability by the following equation:

$$SF_{min} = \frac{9}{3 - \log_{10} P_{40}}$$

where: $10^{-1} > P_{40} \geq 10^{-5}$

For event probabilities smaller than 10^{-5} or greater than 10^{-1}, the following apply:

Event Probability	Min. Safety Factor
$10^{-5} > p^{40} \geq 10^{-6}$	1.125
$1.0 > P_{40} \geq 10^{-1}$	2.25

These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SF_{min} values corresponding to the event probabilities are summarized in Table C-3-6.

The loadings which occur as a result of the conditions listed are factored into the design of the components in accordance with the requirements of the applicable design code, or to the requirements of these criteria. Where applicable design codes or acceptance criteria provide specific allowable limits for a given loading condition, these limits may be substituted for the criteria above on the basis that appropriate safety margins are provided in the limits. In particular, editions of the ASME Boiler and Pressure Vessel Code subsequent to 1977 specify allowable limits on primary stress for piping components under Emergency and Faulted Conditions. These limits are used whenever these ASME Code editions apply for design.

TABLE C-3-1

LOADING CONDITION PROBABILITIES

Upset (likely)	$1.0 > P_{40} \geq 10^{-1}$
Emergency (low probability)	$10^{-1} > P_{40} \geq 10^{-3}$
Faulted (extremely low probability)	$10^{-3} > P_{40} \geq 10^{-6}$

Where P_{40} = 40 year event encounter probability

TABLE C-3-2

DEFORMATION LIMIT

Either One of (Not Both)		General Limit
a.	Permissible Deformation, DP / Analyzed deformation causing loss of function, DL	$\leq \frac{0.9}{SF_{min}}$
b.	Permissible Deformation, DP / Experimental deformation causing loss of function, DE	$\leq \frac{1.0}{SF_{min}}$ *

Where:

DP = permissible deformation under stated conditions of normal, upset, emergency or fault

DL = analyzed deformation which would cause a system "loss of function"**

DE = experimentally determined deformation which would cause a system "loss of function"**

* Equation b was not applied unless supporting data had been submitted for AEC staff evaluation.[18]

** Note that "loss of function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they will be specifically delineated. From a practical viewpoint, it is convenient to interchange, with the loss of function condition, some deformation condition at which function is assured if the required safety margins from the functioning condition can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, core support deformation causing fuel disarrangement or excess leakage of any component.

TABLE C-3-3

PRIMARY STRESS LIMIT

	Any One of (No More than One Required)	General Limit
a.	$\frac{\text{Elastic Evaluated Primary Stresses, PE}}{\text{Permissible Primary Stresses, PN}}$	$\leq \frac{2.25}{SF_{min}}$
b.	$\frac{\text{Permissible Load, IP}}{\text{Largest Lower Bound Limit Load, CL}}$	$\leq \frac{1.5}{SF_{min}}$
c.	$\frac{\text{Elastic Evaluated Primary Stress, PE}}{\text{Conventional Ultimate Strength at Temperature, US}}$	$\leq \frac{0.75}{SF_{min}}$
d.	$\frac{\text{Elastic - Plastic Evaluated Nominal Primary Stress, EP}}{\text{Conventional Ultimate Strength at Temperature, US}}$	$\leq \frac{0.9}{SF_{min}}$
e.	$\frac{\text{Permissible Load, LP}}{\text{Plastic Instability Load, PL}}$	$\leq \frac{0.9}{SF_{min}}$ *
f.	$\frac{\text{Permissible Load, LP}}{\text{Ultimate Load from Fracture Analysis, UF}}$	$\leq \frac{0.9}{SF_{min}}$ *
g.	$\frac{\text{Permissible Load, LP}}{\text{Ultimate Load or Loss of Function Load from Test, LE}}$	$\leq \frac{1.0}{SF_{min}}$ *

* Equations e, f, g were not used unless supporting date had been submitted for NRC Staff evaluation.[18]

Where:

PE = Primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

PM = Permissible primary stress levels under normal or upset conditions under applicable industry code.

LP = Permissible load under stated conditions of emergency or fault.

TABLE C-3-3 (Continued)

PRIMARY STRESS LIMIT

CL = Lower bound limit load with yield point equal to 1.5 S_m where S_m is the tabulated value of allowable stress at temperature as contained in the ASME III Code or its equivalent. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the unaxial case.

US = Conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.

EP = Elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

PL = Plastic instability load. The "plastic instability load" is defined here as the load at which any load bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

UF = Ultimate load from fracture analyses. For components which involve sharp discontinuities (local theoretical stress concentration > 3) the use of a "fracture mechanics" analysis where applicable, utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where "fracture mechanics" may be applied are for fillet welds or end of fatigue life crack propagation.

LE = Ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall use adjusted values to account for material properties and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

TABLE C-3-4

BUCKLING STABILITY LIMIT

Any One of (No More than One Required)		General Limit
a.	$\frac{\text{Permissible Load, LP}}{\text{Code normal event permissible load, PN}}$	$\leq \frac{2.25}{SF_{min}}$
b.	$\frac{\text{Permissible Load, LP}}{\text{Stability Analysis Load, SL}}$	$\leq \frac{0.674}{SF_{min}}$
c.	$\frac{\text{Permissible Load, LP}}{\text{Ultimate Buckling Collapse Load from Test, SE}}$	$\leq \frac{1.0}{SF_{min}}$ *

* Equation c was not used unless supporting data had been submitted for AEC Staff review.[18]

Where:

LP = Permissible load under stated conditions of Normal, Upset, Emergency, or Faulted.

PN = Applicable code normal event permissible load.

SL = Stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity of column members.

SE = Ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

TABLE C-3-5

FATIGUE LIMIT

Summation of mean fatigue[1] damage usage including Emergency or Faulted events with design and operation loads following Miner's Hypotheses....either one (not both)	a.	Fatigue cycle usage from analysis	≤ .05[2]*
	b.	Fatigue cycle usage from test	≤ 0.33 *

* Equations a and b were not used unless supporting data had been submitted for AEC Staff review. Fatigue analysis was performed per footnote 2.[18]

(1) Fatigue failure is defined here as a 25% area reduction for a load carrying member which is required to function or excess leakage causing loss of function, whichever is more limiting. In the fatigue evaluation, the methods of linear elastic stress analysis may be used when the $3S_m$ range limit of ASME III has been met. If $3S_m$ is not met, account will be taken of (a) increases in local strain concentration, (b) strain ratcheting, (c) re-distribution of strain due to elastic-plastic effects. The January, 1969, draft of the USAS B31.7 Piping Code may be used where applicable or detailed elastic-plastic methods may be used. With elastic-plastic methods, strain hardening may be used not to exceed in stress for the same strain, the steady state cyclic strain hardening measured in a smooth low cycle fatigue specimen at the average temperature of interest.

(2) It is acceptable to use the ASME Section III Design Fatigue curves in conjunction with a cumulative usage factor of 1.0 (using Miner's Hypothesis) in lieu of using the mean fatigue data curves with a limit on fatigue usage of 0.05, since the two methods are approximately equivalent.

TABLE C-3-6

MINIMUM SAFETY FACTORS

Loading Conditions	Loads	P_{40}	SF_{min}
Upset	N and E	10^{-1}	2.25
	or N and U	10^{-1}	2.25
Emergency	N and R	10^{-3}	1.5
	N and E'	10^{-3}	1.5
	Other combinations in this probability range	$<10^{-1}$ to 10^{-3}	<2.25 to 1.5
Fault	N and E' and R	1.5×10^{-6}	1.125
	Other combinations in this probability range	$<10^{-3}$ to 10^{-6}	<1.5 to 1.125

Where: N = Normal loads

U = Upset loads (result in maximum system pressure)

E = Maximum probable earthquake

E' = Maximum possible earthquake

R = Loads resulting from jet forces and pressure and temperature transients associated with rupture of a single pipe within the primary containment. This load is considered as indicated in the tables.

The minimum safety factor decreases as the event probability diminishes and if the event is too improbable (incredible: $P_{40} < 10^{-6}$), then no safety factor is appropriate or required.

TABLE C-3-7
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS
STABILIZER BRACKET AND ADJACENT SHELL			
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sect. III defines primary membrane plus primary bending stress intensity limit for SA302 - Gr. B For normal and upset condition Stress limit = 1.5x26,700 = 40,000 psi For emergency condition Stress limit = 1.5x40,000 = 60,000 psi For faulted condition Stress limit = 2.0x26,700 = 53,400 psi	Normal and upset condition load 1. OBE (E) 2. Design pressure	Membrane plus bending	40,000 psi
	Emergency condition load 1. SSE(E') 2. Design pressure	Membrane plus bending	60,000 psi
	Faulted condition loads 1. SSE (E') 2. Jet reaction forces 3. Design pressure	Membrane	53,400 psi
VESSEL SUPPORT SKIRT			
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sect. III, defines stress limit for SA516 GR70 For normal and upset condition B = 12,000 psi For emergency condition S_{limit} = 1.5 B = 18,000 psi For faulted condition S_{limit} = 2.0 B = 24,000 psi	Normal and upset condition loads 1. Dead weight 2. OBE (E)	Compressive membrane	12,000 psi
	Emergency condition loads 1. Dead weight 2. SSE (E')	Compressive membrane	18,000 psi
	Faulted condition loads 1. Dead weight 2. SSE (E') 3. Jet reaction forces	Compressive membrane	24,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS
SHROUD SUPPORT GUSSETS			
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sect. III defines allowable primary membrane stress plus bending stress for SB168 material.	Normal and upset condition loads 1. OBE (E) 2. Pressure drop across shroud (normal) 3. Subtract dead weight	Membrane plus bending	34,950 psi
For normal and upset condition $S_A = 1.5\ S_M =$ 1.5 X 23.30 = 34.95 KSI	Emergency condition loads 1. SSE (E') 2. Pressure drop across shroud (normal) 3. Subtract dead weight	Membrane plus bending	52,430 psi
For emergency condition $S_{limit} = 1.5\ S_A =$ 1.5 X 34.95 = 52.43 KSI For faulted condition $A_{limit} = 2.0\ S_A =$ 2.0 X 34.95 = 69.90 KSI	Faulted condition loads 1. SSE (E') 2. Pressure drop across shroud during faulted condition 3. Subtract dead weight	Membrane plus bending	69,900 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS
TOP GUIDE-LONGEST BEAM			
Primary Stress Limit - The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for type 304 stainless steel plate.	Normal and upset condition loads 1. OBE (E) 2. Weight of structure 3. Weight of temporary control curtains.	General membrane plus bending	25,388 psi
For normal and upset condition Stress Intensity $S_A = 1.5\ S_M =$ 1.5x16,925 psi = 25,388 psi. For emergency condition $S_{limit} = 1.5\ S_A =$ 1.5x25,388 = 38,081 psi	Emergency condition loads 1. SSE (E') 2. Weight of structure 3. Weight of temporary control curtains.	General membrane plus bending	38,081 psi
For faulted condition $S_{limit} = 2\ S_A =$ 2x25,388 = 50,775 psi	Faulted condition loads (same as emergency condition)	General membrane plus bending	50,775 psi
TOP GUIDE BEAM END CONNECTIONS			
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sect. III, defines material stress limit for type 304 stainless steel.	Normal and upset condition loads 1. OBE (E) 2. Weight of structure 3. Weight of temporary control curtains.	Pure shear	10,155 psi
For normal and upset condition Stress Intensity $S_A = 0.6\ S_M =$ 0.6x16,925 psi = 10,155 psi For emergency condition $S_{limit} = 1.5\ S_A =$ 1.5x10,155 psi = 15,232 psi.	Emergency condition loads 1. SSE (E') 2. Weight of structure 3. Weight of temporary control curtains.	Pure shear	15,232 psi
For faulted condition $S_{limit} = 2\ S_A =$ 2x10,155 psi = 20,310 psi	Faulted condition loads (same as emergency condition)	Pure shear	20,310 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS
TOP GUIDE ALIGNERS			
Primary Stress Limit - The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for type 304 stainless steel plate.	Normal and upset condition loads 1. OBE (E) 2. Weight of structure 3. Weight of temporary control curtains.	General membrane plus bending	25,388 psi
For normal and upset condition Stress Intensity $S_A = 1.5\ S_M = 1.5x16,925$ psi = 25,388 psi	Emergency condition loads 1. SSE (E') 2. Weight of structure 3. Weight of temporary control curtains.	General membrane plus bending	38,081 psi
For emergency condition $S_{limit} = 1.5\ S_A = 1.5x25,388 =$ 38,081 psi	Faulted condition loads (same as emergency condition)	General membrane plus bending	50,775 psi
For faulted condition $S_{limit} = 2\ S_A = 2x38,081 =$ 50,775 psi			

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS
RPV STABILIZER			
Primary Stress Limit AISC 6th Edition specification for the construction, fabrication, and erection of structural steel for buildings	Upset condition 1. Spring preload 2. OBE (E)	Rod Bracket	127,000 psi 22,000 psi 14,000 psi
For normal & upset conditions AISC allowable stresses, but without the usual increase for earthquake loads	Emergency condition 1. Spring preload 2. SSE (E')	Bracket	33,000 psi 21,000 psi
For emergency conditions 1.5 x AISC allowable stresses For faulted conditions Material yield strength	Faulted condition 1. Spring preload 2. SSE (E') 3. Jet reaction load	Bracket	36,000 psi 21,500 psi
RPV SUPPORT (RING GIRDER)			
Primary Stress Limit AISC 6th Edition specification for the design, fabrication and erection of structural steel for buildings	Normal and upset condition 1. Dead loads 2. OBE (E) 3. Loads due to scram	Top Flange Bottom Flange Vessel to girder bolts	27,000 psi 27,000 psi 60,000 psi 18,000 psi
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads.	Emergency condition 1. Dead loads 2. SSE (E') 3. Loads due to scram	Top Flange Bottom Flange Vessel to girder bolts	40,500 psi 40,500 psi 90,000 psi 20,800 psi
For faulted conditions 1.67 X AISC allowable stresses for structural steel members Yield strength for high strength bolts (vessel to ring girder)	Faulted condition 1. Dead loads 2. SSE (E') 3. Jet reaction load	Top Flange Bottom Flange Vessel to girder bolts	45,000 psi 45,000 psi 125,000 psi 72,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS
CORE SUPPORT			
Primary Stress Limit - The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for type 304 stainless steel plate.	Normal and upset condition loads 1. Normal operation pressure drop 2. OBE (E)	General membrane plus bending	25,388 psi
For allowable stresses see top guide, longest beam, above	Emergency condition loads 1. Normal operation pressure drop 2. SSE (E')	General membrane plus bending	38,081 psi
	Faulted condition loads 1. Pressure drop after recirculation line rupture 2. SSE (E')	General membrane plus bending	50,775 psi
CORE SUPPORT ALIGNERS			
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sect. III, defines material stress limit for type 304 stainless steel.	Normal and upset condition load 1. OBE (E)	Pure shear	10,155 psi
For allowable shear stresses, see top guide beam end connections, above.	Emergency condition load 1. SSE (E')	Pure shear	15,232 psi
	Faulted condition load 1. SSE (E')	Pure shear	20,310 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	PRIMARY STRESS TYPE	MOMENT LIMIT accounting for pressure loads
FUEL CHANNELS			
Primary Stress Limit - Allowable stress S_M for Zircaloy or NSF determined according to methods recommended by ASME Boiler and Pressure Vessel Code, Sect. III.	Normal and upset condition load 1. OBE (E) 2. Normal pressure load	Membrane and bending	28,230* in. lbs.
Allowable moment determined by calculating limit moment using Table C-3-3, equation (b), then applying SF_{min} for applicable loading conditions.	Emergency condition load 1. SSE (E') 2. Normal pressure load	Membrane and bending	42,350* in. lbs.
(S_m = 9,270 psi; 1.5 S_m = 13,900 psi) Emergency limit load = 1.5 x Normal limit load calculated using 1.5 S_m = yield.	Faulted condition load 1. SSE (E') 2. Loss of cooling accident pressure	Membrane and bending	56,500* in. lbs.

* The moments shown above are applicable to the previous analysis (e.g., prior to MELLL and ICF). The current fuel design is consistent with the MELLL and ICF evaluation and has been qualified based on the pressure drops associated with MELLL and ICF.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS
CRD HOUSING SUPPORT			
Primary Stress Limit - AISC specification for the design, fabrication and erection of structural steel for buildings	Faulted condition loads 1. Dead weight 2. Impact force from failure of a CRD housing	Beams (top cord) Beams (bottom cord)	33,000 psi 33,000 psi 33,000 psi 33,000 psi
For normal and upset condition $F_a = 0.60\ F_y$ (tension) $F_b = 0.60\ F_y$ (bending) $F_v = 0.40\ F_y$ (shear)	(Dead weights and earthquake loads are very small as compared to jet force.)	Grid structure	41,500 psi 27,500 psi
For faulted conditions F_a limit = 1.5 F_a (tension) F_b limit = 1.5 F_b (bending) F_v limit = 1.5 F_v (shear) F_y = Material yield strength			
RECIRCULATING PIPE AND PUMP WHIP RESTRAINTS			
Primary Stress Limit Structural Steel: AISC specification for the design, fabrication and erection of structural steel for buildings.	Faulted condition loads 1. Jet force from a complete circumferential failure (break) of recirculation line.	Brackets on 28" pipe Cable on pump restraints	33,000 psi 90% of listed breaking strength
For normal or upset conditions $F_a = 0.60\ F_y$ (tension)			
For faulted conditions F_a limit = 1.5 F_a (tension) F_y = yield strength			
Cable (wire rope):			
For faulted conditions 90% of listed breaking strength			

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS
CONTROL ROD DRIVE HOUSING			
Primary Stress Limit - The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Sect. III, for Class A vessels, for type 304 stainless steel.	Normal and upset condition Loads 1. Design pressure 2. Stuck rod scram loads 3. OBE (E)	Maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing for normal, upset and emergency conditions.	15,800 psi
For normal and upset condition S_m = 15,800 psi @ 575°F			
For emergency conditions S_{limit} = 1.5 S_m = 1.5x15,800 = 23,700 psi	Emergency condition loads 1. Design pressure 2. Stuck rod scram loads 3. SSE (E')		23,700 psi
CONTROL ROD DRIVE			
Primary Stress Limit - The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for SA212 TP316 tubing. For normal and upset condition S_A = 1.5 S_m = 1.5x17,375 = 26,060 psi	Normal and upset condition Loads Maximum hydraulic pressure from the control rod drive supply pump. NOTE: Accident conditions do not increase this loading. Earthquake loads are negligible.	Maximum stress intensity occurs at a point on the Y-Y axis of the indicator tube.	26,060 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS
CONTROL ROD GUIDE TUBE			
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section III for Type 304 stainless steel tubing. For normal and upset conditions $S_A = 1.5\ S_m = 1.5 \times 15,800 =$ 23,700 psi For faulted condition $S_{limit} = 2.0\ S_A = 2.0 \times 23,700 =$ 47,400 psi	Faulted condition loads 1. Dead weight 2. Pressure drop across guide tube due to failure of recirculation line 3. SSE (E')	The maximum bending stress under faulted loading conditions occurs at the center of the guide tube	47,400 psi
INCORE HOUSING			
Primary Stress Limit - The allowable primary membrane stress is based on ASME Boiler and Pressure Vessel Code, Sect. III, for Class A vessels for Type 304 stainless steel. For normal and upset conditions $S_m = 15,800$ psi @ 575°F For emergency condition (N+E') $S_{limit} = 1.5\ S_m = 1.5 \times 15,800 =$ 23,700 psi	Emergency condition loads 1. Design pressure 2. SSE (E')	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration	23,700 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS
HYDRAULIC CONTROL UNIT PIPING			
From USAS B31.1.0 - 1967 Code for power pressure piping	Normal Condition Load Maximum normal hydraulic system pump pressure	3/4" drive withdraw piping	15,000 psi
For Normal Conditions:			
S_h = 15,000 psi			
For upset and emergency condition: When upset or emergency condition exists for less than 1% of the time, the code allows 20% increase in stress.	Upset condition load 1. Shut off pump pressure 2. OBE (E) (negligible load)	3/4" drive withdraw piping	18,000 psi
S_a = 1.2 S_h = 18,000 psi	Emergency condition 1. Shut off pump pressure 2. SSE (E') (negligible load)	3/4" drive withdraw piping	18,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS
SPENT FUEL STORAGE RACKS			
Stresses due to normal, upset, or emergency loading shall not cause the racks to fail so as to result in a critical fuel array.	Emergency condition "A" loads 1. Dead loads 2. Full fuel load in rack 3. SSE (E')	At column to base welds At base hold down lug (casting)	11,000 psi 20,000 psi
Primary Stress Limit - Paper numbers 3341 and 3342, Proceedings of the ASCE, Journal of the Structural Division, Dec., 1962 (task committee on light-weight alloys) (Aluminum)	Emergency condition "B" loads (see below)		
Emergency Conditions Stress limit = yield strength at 0.2% offset.			

(1) Load testing showed that the structure would not yield when subjected to simulated emergency condition "A" loads. Strain gages mounted on the welds showed that calculated stresses are conservative.

(2) Calculated stresses compare very well with test results.

EMERGENCY CONDITION "B"

Loading

In addition to the loading conditions given above, the racks were tested and analyzed to determine their capability to safely withstand the accidental, uncontrolled drop of the fuel grapple from its full retracted position into the weakest portion of the rack.

Method of Analysis

The displacement of the vertical columns at the ends of the racks was determined by considering the effect of the grapple kinetic energy on the upper structure. The energy absorbed shearing the rack longitudinal structural member welds was determined.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Reactor Vessel Internals and Associated Equipment (Cont'd)

Method of Analysis (continued)

The effect of the remaining energy on the vertical columns was analyzed. Equivalent static load tests were made on the structure to assure that the criteria were met.

Results of Analysis

All criteria are met.

Analysis shows that the grapple would shear the welds in the area where the impact occurred. The longitudinal structural member bends but does not fail in shear. Grapple penetration into the rack is not sufficient to cause the vertical columns to deflect the fuel into a critical array. Static load testing showed that forces in excess of those resulting from a grapple drop are required to cause the columns to deflect to the extent that the criteria is violated.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Piping

CRITERIA	LOADING	
Stress due to all normal and upset loadings must not exceed the limits of USAS B31.1.0.		"Note: Analyses of this piping system demonstrate compliance with the applicable Code and are documented in the existing calculations of record."
1. The sum of the longitudinal stresses due to pressure and dead weight must be less than the hot allowable stress.	a. Dead weight b. Pressure	
2. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of OBE(E) must be less than 1.2 times the hot allowable stress.	a. Dead weight b. Pressure c. OBE (E)	
3. The sum of the longitudinal stresses due to pressure and dead weight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion at stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. Thermal loads	
4. The sum of the longitudinal stresses due to pressure, dead weight, and OBE (E) plus the thermal expansion stress intensity range must be less than 1.2 times the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. OBE (E) d. Thermal loads	

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Piping (Cont'd)

CRITERIA	LOADING
For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limit:	
1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of SSE (E') must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40 year plant life is 10^{-3} and SF = 1.5.	a. Dead weight b. Pressure c. SSE (E')
2. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of OBE (E) must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40 year plant life is 10^{-2} and SF = 1.8.	a. Dead weight b. Maximum pressure c. OBE (E)
3. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of SSE (E') must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40 years plant life is $.25 \times 10^{-3}$ and SF = 1.36.	a. Dead weight b. Maximum pressure c. SSE (E')

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Recirculation Loop Piping

"Note: Reanalysis of this piping system was performed as part of the piping replacement program in 1985. The basis for design criteria, loading conditions and stress allowables are provided in Impell Corp. Report No. 01-0840-1268; "Stress Analysis of Reactor Recirculation, Core Spray, and Reactor Water Cleanup Drywell Systems for Cooper Nuclear Station.[34] Subsequent analyses based on the requirements of Reference 34 demonstrate compliance with the applicable Code and are documented in the existing calculations of records."

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS - Core Spray Discharge Piping CS-ID

"Note: Reanalysis of this piping system was performed as part of the piping replacement program in 1985. The basis for design criteria, loading conditions and stress allowables are provided in Impell Corp. Report No. 01-0840-1268; "Stress Analysis of Reactor Recirculation, Core Spray, and Reactor Water Cleanup Drywell Systems for Cooper Nuclear Station.[34] Subsequent analyses based on the requirements of Reference 34 demonstrate compliance with the applicable Code and are documented in the existing calculations of record."

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS - Clean-Up Recirculation Pump Suction Piping CU-IS

"Note: Reanalysis of this piping system was performed as part of the piping replacement program in 1985. The basis for design criteria, loading conditions and stress allowables are provided in Impell Corp. Report No. 01-0840-1268; "Stress Analysis of Reactor Recirculation, Core Spray, and Reactor Water Cleanup Drywell Systems for Cooper Nuclear Station.[34] Subsequent analyses based on the requirements of Reference 34 demonstrate compliance with the applicable Code and are documented in the existing calculations of record."

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS - Main Steam Piping to HPCI Turbine & Residual Heat Exchangers 1A & 1B

"Note: Analyses of this piping system demonstrate compliance with the applicable Code and are documented in the existing calculations of record."

CRITERIA	LOADING
Stress due to all normal and upset loadings must not exceed the limits of USAS B31.1.0.	
1. The sum of the longitudinal stresses due to pressure and dead weight must be less than the hot allowable stress.	a. Dead weight b. Pressure
2. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of OBE must be less than 1.2 times the hot allowable stress.	a. Dead weight b. Pressure c. OBE
3. The sum of the longitudinal stresses due to pressure and dead weight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion at stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. Thermal loads
4. The sum of the longitudinal stresses due to pressure, dead weight, and OBE plus the thermal expansion stress intensity range must be less than 1.2 times the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. OBE d. Thermal loads

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS - Main Steam Piping to HPCI Turbine & Residual Heat Exchangers 1A & 1B (Cont'd)

CRITERIA	LOADING
For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limits:	
1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of SSE must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40 year plant life is 10^{-3} and SF = 1.5.	a. Dead weight b. Pressure c. SSE
2. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of OBE must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40 years plant life is 10^{-2} and SF = 1.8.	a. Dead weight b. Maximum pressure c. OBE
3. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of SSE must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40 years plant life is $.25 \times 10^{-3}$ and SF = 1.36.	a. Dead weight b. Maximum pressure c. SSE

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS Reactor Feed Piping

"Note: Analyses of this piping system demonstrate compliance with the applicable Code and are documented in the existing calculations of record."

CRITERIA	LOADING
Stress due to all normal and upset loadings must not exceed the limits of USAS B31.1.0.	
1. The sum of the longitudinal stresses due to pressure and dead weight must be less than the hot allowable stress.	a. Dead weight b. Pressure
2. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of OBE must be less than 1.2 times the hot allowable stress.	a. Dead weight b. Pressure c. OBE
3. The sum of the longitudinal stresses due to pressure and dead weight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion at stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. Thermal loads
4. The sum of the longitudinal stresses due to pressure, dead weight, and OBE plus the thermal expansion stress intensity range must be less than 1.2 times the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. OBE d. Thermal loads

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS - Reactor Feed Piping (Cont'd)

CRITERIA	LOADING
For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limits:	
1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of SSE must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40 year plant life is 10^{-3} and SF = 1.5.	a. Dead weight b. Pressure c. SSE
2. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of OBE must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40 years plant life is 10^{-2} and SF = 1.8.	a. Dead weight b. Maximum pressure c. OBE
3. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of SSE must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40 years plant life is $.25 \times 10^{-3}$ and SF = 1.36.	a. Dead weight b. Maximum pressure c. SSE

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS Residual Heat Removal Pump Suction Piping

"Note: Analyses of this piping system demonstrate compliance with the applicable Code and are documented in the existing calculations of record."

CRITERIA	LOADING
Stress due to all normal and upset loadings must not exceed the limits of USAS B31.1.0.	
1. The sum of the longitudinal stresses due to pressure and dead weight must be less than the hot allowable stress.	a. Dead weight b. Pressure
2. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of OBE must be less than 1.2 times the hot allowable stress.	a. Dead weight b. Pressure c. OBE
3. The sum of the longitudinal stresses due to pressure and dead weight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion at stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. Thermal loads
4. The sum of the longitudinal stresses due to pressure, dead weight, and OBE plus the thermal expansion stress intensity range must be less than 1.2 times the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. OBE d. Thermal loads

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS - Residual Heat Removal Pump Suction Piping (Cont'd)

CRITERIA	LOADING
For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limits:	
1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of SSE must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40 year plant life is 10^{-3} and SF = 1.5.	a. Dead weight b. Pressure c. SSE
2. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of OBE must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40 years plant life is 10^{-2} and SF = 1.8.	a. Dead weight b. Maximum pressure c. OBE
3. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of SSE must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40 years plant life is $.25 \times 10^{-3}$ and SF = 1.36.	a. Dead weight b. Maximum pressure c. SSE

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS Residual Heat Removal Pump Discharge Piping

CRITERIA	LOADING
Stress due to all normal and upset loadings must not exceed the limits of USAS B31.1.0.	
1. The sum of the longitudinal stresses due to pressure and dead weight must be less than the hot allowable stress.	a. Dead weight b. Pressure
2. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of OBE must be less than 1.2 times the hot allowable stress.	a. Dead weight b. Pressure c. OBE
3. The sum of the longitudinal stresses due to pressure and dead weight plus the thermal expansion stress intensity range must be less than the sum of the allowable stress range for expansion at stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. Thermal loads
4. The sum of the longitudinal stresses due to pressure, dead weight, and OBE plus the thermal expansion stress intensity range must be less than 1.2 times the sum of the allowable stress range for expansion stresses plus the hot allowable stress.	a. Dead weight b. Pressure c. OBE d. Thermal loads

"Note: Analyses of this piping system demonstrate compliance with the applicable Code and are documented in the existing calculations of record."

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Class IN/IS - Residual Heat Removal Pump Discharge Piping (Cont'd)

CRITERIA	LOADING
For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limits:	
1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of SSE must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40 year plant life is 10^{-3} and SF = 1.5.	a. Dead weight b. Pressure c. SSE
2. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of OBE must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40 years plant life is 10^{-2} and SF = 1.8.	a. Dead weight b. Maximum pressure c. OBE
3. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of SSE must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40 years plant life is $.25 \times 10^{-3}$ and SF = 1.36.	a. Dead weight b. Maximum pressure c. SSE

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Recirculation Pumps

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR ACTUAL THICKNESS
1. Casing Minimum Wall Thickness Loads: Normal and upset condition Design pressure & temperature Primary membrane stress limit: Allowable working stress per ASME Section III, Class C	$t = \frac{PR}{SE-06P} + C$ where: t = minimum required thickness, in. P = design pressure, psig R = maximum internal radius, in. S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in.	3.00 in.
2. Casing Cover Minimum Thickness Loads: Normal and upset condition Design pressure & temperature Primary bending stress limit: 1.5 S_m per ASME code for Pumps and Valves for Nuclear Power Class I	$S_r = \frac{3w}{4t^2}\left[a^2 - 2b^2 + \frac{b^4(m-1)4b^4(m+1)1n\,a/b + a^2b^2(m+1)}{a^2(m-1) + b^2(m+1)}\right]$ $+ \frac{3w}{2\pi t^2}\left[1 - \frac{2mb^2 - 2b^2(m+1)1n\,a/b}{a^2(m-1) + b^2(m+1)}\right] = 14{,}950 \text{ psi}$ $S_t = -\frac{3w(m^2-1)}{4mt^2}\left[\frac{a^4 - b^4 - 4a^2b^2 1n\,a/b}{a^2(m-1) + b^2(m+1)}\right] +$ $\frac{3w}{2\pi mt^2}\left[1 + \frac{ma^2(m-1) - mb^2(m+1) - 2(m^2-1)a^2 1n\,a/b}{a^2(m-1) + b^2(m+1)}\right] = 14{,}950 \text{ psi}$	

where:

S_r = radial stress at outer edge, psi
S_t = tangential stress at inner edge, psi
W = pressure load, psi
W = uniform load along inner edge, lb.
t = disc thickness, in.
m = reciprocal of Poisson's ratio
a = radius of disc, in.
b = radius of disc hole, in.

TABLE C-3-7 (Cont'd) LOADING CRITERIA
Recirculation Pumps (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR ACTUAL THICKNESS
3. Cover and seal flange bolt areas Loads: Normal and upset condition Design pressure & temperature. Design gasket load Bolting Stress Limit: Allowable working stress per ASME Section III, Class C	Bolting loads, areas and stresses are calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Appendix II.	Cover Flange Bolts 27,000 psi Seal Flange Bolts 20,000 psi
4. Cover Clamp Flange Thickness Loads: Normal and upset condition Design pressure & temperature. Design gasket load. Design bolting load. Tangential Flange Stress Limit: Allowable working stress per ASME Section III, Class C	Flange thickness and stress are calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Appendix II.	Flange Thickness 8.25"
5. Pump Nozzle Membrane and Bending Stress Loads: Normal and upset condition Design pressure & temperature. Piping reactions during normal	$S_L = \frac{\pi}{4} \frac{D^2 P}{A} + \frac{M}{Z} + \frac{F}{A}$ $S_C = \frac{PD}{2t}$ $S_S = \frac{TR_0}{J}$	

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Recirculation Pumps

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR ACTUAL THICKNESS
5. Pump Nozzle Membrane and Bending Stress (Cont'd)		
Combined Stress Limit:		
1.5 S_m per ASME code for Pumps and Valves for Nuclear Power Class I.	$S = \frac{S_L + S_C}{2} + \left[\left(\frac{S_L \;\; S_C}{2}\right)^2 + S_S^2\right]^{1/2}$	
	where:	Pump Nozzle Stresses
	S_L = longitudinal stress, psi	
	S_C = circimferential stress, psi	
	S_S = torsional stress, psi	28,650 psi
	D = nozzle internal diameter, in.	
	P = design pressure, psi	28,650 psi
	A = nozzle cross section metal area, in^2	
	M = maximum bending moment, in. lb.	28,650 psi
	F = maximum longitudinal force, lb.	
	t = nozzle wall thickness, in.	28,650 psi
	J = polar moment of inertia, in^4	
	R_O = nozzle outside radius, in.	
	T = torsional moment	
6. Mounting Bracket Combined Stress		
Loads: Flooded weight SSE (E') Combined Stress Limit: Yield Stress	Bracket vertical loads are determined by summing the equipment and fluid weights and vertical seismic forces. Bracket horizontal loads are determined by applying the specified seismic force at mass center of pump-motor assembly (flooded). Horizontal and vertical loads are applied simultaneously to determine tensile, shear and bending stresses in the brackets. Tensile, shear and bending stresses are combined to determine maximum combined stresses.	Maximum Combined Stresses 15,600 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Recirculation Pumps (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR ACTUAL THICKNESS
7. Stresses Due to Seismic Loads Loads: Operating pressure and temperature SSE (E') Combined Stress Limit: Yield stress	The flooded pump-motor assembly is analyzed as a free body supported by constant support hangers from the pump brackets. Horizontal and vertical seismic forces are applied at mass center of assembly and equilibrium reactions are determined for the motor and pump brackets. Load, shear, and moment diagrams are constructed using live loads, dead loads, and calculated snubber reactions. Combined bending, tension and shear stresses are determined for each major component of the assembly including motor, motor support barrel, bolting and pump casing. The maximum combined tensile stress in the cover bolting is calculated using tensile stresses determined from loading diagram plus tensile stress from operating pressure.	Motor Bolt Tensile Stress: 40,000 psi Pump Cover Bolt Tensile Stress: 43,200 psi Motor Support Barrel Combined Stresses: 22,400 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RHR Pump

OBE (E) Coefficients	0.23 g horizontal	0.07 g vertical
SSE (E') Coefficients	0.46 g horizontal	0.14 g vertical

The RHR pump is a verticaly mounted, single stage pump driven by direct-coupled motors. The motor is mounted to and above the pump, and the pump is mounted to the foundation.

Statement of Criteria	Method of Analysis	Results
1. Closure bolting is designed to contain the internal design pressure of the pump casing without exceeding the allowable stress of the bolting material. Allowable stresses at design temperature are in accordance with ASME B&PV Code, Section VIII.	1. Bolting loads and stresses are calculated in accordance with the "Rules for Bolted Flange Connections", ASME Boiler and Pressure Vessel Code, Section VIII, Appendix II.	1. Allowable Stress 25,000 psi Pump Design Pressure 450 psig Max. Design Temperature 350°F
2. The minimum wall thickness of the pump limits stress to the allowable stress when subjected to design pressure and temperature. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	2. Stress in the pump casing is calculated at the point of maximum internal pump diameter by the formula $$S_c = \frac{P(D + .2t)}{2t}$$ where S_c = calculated stress, psi P = pump design pressure, psi D = maximum pump internal diameter t = actual min. metal thickness less corrosion allowance, 0.080 inches	2. Allowable Stress 14,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RHR Pump (Cont'd)

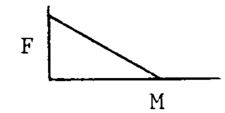
Statement of Criteria

3. Application of forces and moments by attaching pipe on pump nozzles under combined maximum thermal expansion and OBE loading (E) reaction plus load due to internal pressure does not produce an equivalent bending and torsional stress in the nozzles in excess of the allowable stress as defined by the ASME B&PV Code, Section VIII.

   For SSE (E'), less than 1.5 of allowable stress.

Method of Analysis

3. Stresses will not be excessive if the maximum force when taken with the maximum moment falls below the line.



		E	E'	
suction	$F_{intercept}$ (M=0)	77,007	124,988	lbs.
	$M_{intercept}$ (F=0)	725,709	1,385,100	in.lbs.
discharge	$F_{intercept}$ (M=0)	65,696	116,898	lbs.
	$M_{intercept}$ (F=0)	629,003	1,060,236	in.lbs.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Core Spray Pump

OBE (E) Coefficients	0.23 g horizontal	0.07 g vertical
SSE (E') Coefficients	0.46 g horizontal	0.14 g vertical

The Core Spray or RHR pump is a vertically mounted, single stage pump driven by direct coupled motors. The motor is mounted to and above the pump, and the pump is mounted to the foundation.

Statement of Criteria	Method of Analysis	Results
1. Closure bolting is designed to contain the internal design pressure of the pump casing without exceeding the allowable stress of the bolting material. Allowable stresses at design temperature are in accordance with ASME B&PV Code, Section VIII.	1. Bolting loads and stresses are calculated in accordance with the "Rules for Bolted Flange Connections", ASME Boiler and Pressure Vessel Code, Section VIII, Appendix II.	1. Allowable Stress 20,000 psi Pump Design Pressure 500 psi Max. Design Temperature 212°F
2. The minimum wall thickness of the pump limits stress to the allowable stress when subjected to design pressure and temperature. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	2. Stress in the pump casing is calculated at the point of maximum internal pump diameter by the formula $$S_c = \frac{P(D + .2t)}{2t}$$ where S_c = calculated stress, psi P = pump design pressure, psi D = maximum pump internal diameter t = actual min. metal thickness less corrosion allowance, 0.080 inches	2. Allowable Stress 14,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Core Spray Pump (Cont'd)

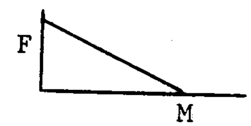
Statement of Criteria

3. Application of forces and moments by attaching pipe on pump nozzles under combined maximum thermal expansion and OBE loading reaction (E) plus load due to internal pressure does not produce an equivalent bending and torsional stress in the nozzles in excess of the allowable stress as defined by the ASME B&PV Code, Section VIII.

   For SSE (E') less than 1.5 of allowable stress.

Method of Analysis

3. Stresses will not be excessive if the maximum force when taken with the maximum moment falls below the line.



		E	E'	
Suction	^{F}intercept (M=0)	54,535	79,808	lbs.
	^{M}intercept (F=0)	402,803	623,427	in.lbs.
Discharge	^{F}intercept (M=0)	30,624	59,668	lbs.
	^{M}intercept (F=0)	293,562	426,519	in.lbs.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
HPCI Pump

OBE (E) Coefficients	0.23 g horizontal	0.07 g vertical
SSE (E') Coefficients	0.46 g horizontal	0.14 g vertical

The HPCI pumps are multi-stage, horizontally mounted, split case pumps driven by their respective steam turbines through couplings. The HPCI pump consists of a main pump and booster on a common baseplate separate from the turbine baseplate.

Statement of Criteria	Method of Analysis	Results Main Pump
1. Closure bolting is designed to contain the internal design pressure of the pump casing without exceeding the allowable working stress of the bolting material. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	1. Bolting loads and stresses are calculated in accordance with the "Rules for Bolted Flange Connections", ASME Boiler and Pressure Vessel Code, Section VIII, Appendix II.	1. Design Pressure 1,500 psig Allowable Stress 20,000 psi Results Boost Pump Design Pressure 450 psi Allowable Stress 20,000 psi
2. The minimum wall thickness of the pump limits stress to the allowable working stress when subjected to design pressure plus corrosion allowance. Allowable stresses are in accordance with ASME B&PV Code, Section III. The maximum stress in the pump casing when subjected to design pressure does not exceed the allowable working stress of the material. The allowable stress are in accordance with ASME B&PV Code, Section III.	2. Nozzle stress is calculated by the following formula $S_h = \frac{P(D + .2t)}{2Et}$ Volute stress is calculated by the following formula $S_v = \frac{Pb}{2_t}\left(\frac{R + a}{R}\right)$ Roark p. 307 Case 26 and R = a - 0.5b	2. Main Pump Design Pressure 1,500 psig Allowable Stress 14,000 psi Boost Pump Design Pressure 450 psig Allowable Stress 14,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
HPCI Pump (Cont'd)

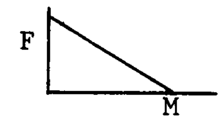
Statement of Criteria

3. Application of forces and moments by attaching pipe on pump nozzles under combined maximum thermal expansion and OBE loading reaction (E) plus load due to internal pressure does not produce an equivalent bending and torsional stress in the nozzles in excess of the allowable stress as defined by the ASME B&PV Code, Section VIII.

   For SSE (E') less than 1.5 of allowable stress.

Method of Analysis

3. Stresses will not be excessive if the maximum force when taken with the maximum moment falls below the line.



		E	E'	
Suction	^{F}intercept (M=0)	33,000	65,000	lbs.
	^{M}intercept (F=0)	430,000	700,000	in.lbs.
Discharge	^{F}intercept (M=0)	31,000	63,000	lbs.
	^{M}intercept	250,000	540,000	in.lbs.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RCIC Pump

OBE (E) Coefficients	0.23 g horizontal	0.07 g vertical
SSE (E') Coefficients	0.46 g horizontal	0.14 g vertical

The RCIC pump is a multi-stage, horizontally mounted, split case pump driven by their respective steam turbines through couplings. The RCIC pump and turbine are on a common baseplate.

Statement of Criteria	Method of Analysis	Results
1. Closure bolting is designed to contain the internal design pressure of the pump casing without exceeding the allowable working stress of the bolting material. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	1. Bolting loads and stresses are calculated in accordance with the "Rules for Bolted Flange Connections", ASME Boiler and Pressure Vessel Code, Section VIII, Appendix II.	1. Design Pressure 1,500 psig Allowable Stress 20,000 psi
2. The minimum wall thickness of the pump limits stress to the allowable working stress when subjected to design pressure plus corrosion allowance. Allowable stresses are in accordance with ASME B&PV Code, Section III.	2. Stress in the pump nozzle is calculated at the point of maximum internal pump diameter by the formula $S_c = \frac{P(D + .2t)}{2Et}$ $S_c = 0.8Sa$	2. Design pressure 1,500 psig Allowable Stress 14,000 psi
The maximum stress in the pump casing when subjected to design pressure does not exceed the allowable working stress of the material. The allowable stress is in accordance with ASME B&PV Code, Section III.	volute stress is computed by the following formula: $S_b = \frac{\beta P_b^2}{t^2}$ Roark p. 225 Case # 36 β = factor from Roark a = volute length b = volute width	Design Pressure 1,500 psig Allowable Stress 14,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RCIC Pump (Cont'd)

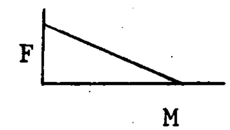
Statement of Criteria

3. Application of forces and moments by attaching pipe on pump nozzles under combined maximum thermal expansion and OBE loading reaction (E) plus load due to internal pressure does not produce an equivalent bending and torsional stress in the nozzles in excess of the allowable stress as defined by the ASME B&PV Code, Section VIII.

   For SSE less than 1.5 of allowable stress.

Method of Analysis

3. Stresses will not be excessive if the maximum force when taken with the maximum moment falls below the line.



		E	E'	
Suction	^{F}intercept (M=0)	6,200	11,800	lbs.
	^{M}intercept (F=0)	34,700	64,000	in.lbs.
Discharge	^{F}intercept (M=0)	2,800	7,200	lbs.
	^{M}intercept (F=0)	12,300	30,000	in.lbs.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Standby Liquid Control Pump

OBE (E) Coefficients	0.33 g horizontal	0.07 g vertical
SSE (E') Coefficients	0.66 g horizontal	0.14 g vertical

The standby liquid control pump is a horizontally mounted, multiple piston, positive displacement pump. It is driven by an ACgear motor through a coupling. The pump and motor are independently bolted to a common baseplate. The baseplate is anchored to its foundation.

Statement of Criteria	Method of Analysis	Results
1. Closure bolting is designed to contain the internal design pressure of the pump without exceeding the allowable working stress of the bolting material. Allowable stresses are in accordance with ASME B&PV Code.	1. Bolting loads and stresses are calculated in accordance with the "Rules for Bolted Flange Connections", ASME Boiler and Pressure Vessel Code, Section VIII, Appendix II.	1. Stuffing Box Bolts Allowable Stress 25,000 psi Cylinder Head Bolts Allowable Stress 25,000 psi
2. The maximum stress in the pump fluid cylinder when subjected to design pressure does not exceed the allowable working stress of the material. The allowable stress is in accordance with ASME B&PV Code, Section VIII.	2. Stress in the pump fluid cylinder is calculated at the point of maximum stress by the pressure area method.	2. Design Pressure 1,400 psig Allowable Stress 16,500 psi
3. The stresses in the motor mounting bolts when the motor is subjected to the SSE does not exceed 0.9 of yield stress and twice the allowable shear stress for bolting material in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII.	3. The seismic forces acting on the motor to subject the bolting to shear or tension are considered. The motor is isolated from the pump and nozzle forces by the flexible coupling.	3. The bolt stresses under (E') Tension Allowable Stress Shear Allowable Stress

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Standby Liquid Control Pump (Cont'd)

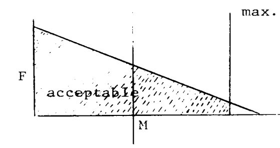
Statement of Criteria

4. The stresses in the pump mounting bolts due to the combination of OBE acting on the flooded pump plus the attaching pipe reactions under combined maximum thermal expansion plus OBE does not exceed the allowable shear and tensile stresses for the bolting material in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. The attaching pipe reaction plus the load due to internal pressure does not produce an equivalent bending and torsional stress in nozzles in excess of the allowable stress.

The stresses in the pump mounting bolts due to the combination of the SSE acting on the flooded pump plus the attaching pipe reaction under combined maximum thermal expansion plus SSE does not exceed 0.9 times the yield stress in tension and twice the allowable shear stress for the bolting material in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. The attaching pipe reaction plus the load due to internal pressure does not produce an equivalent bending and torsional stress in nozzles in excess of 1.5 times allowable stress.

Method of Analysis

4. The maximum force taken with the maximum moment shall fall below the line on the force-moment diagram:



E

Discharge	F =	342 lbs
	M =	.9450 in. lbs., but not to exceed 3400 in. lbs.
Suction	F =	711 lbs.
	M =	39,100 in. lbs., but not to exceed 16,600 in. lbs.

E'

Discharge	F =	684 lbs
	M =	18,900 in. lbs., but not to exceed 5,330 in. lbs.
Suction	F =	1,422 lbs
	M =	78,200 in. lbs., but not to exceed 24,700 in. lbs.

where M is maximum moment (in. lbs.) in any direction and F is maximum force (lbs.) in any direction.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
HPCI Turbine

OBE (E) Coefficients	0.75 g horizontal	0.07 g vertical
SSE (E') Coefficients	1.50 g horizontal	0.14 g vertical

The HPCI turbine is a horizontally mounted solid wheel turbine, having two stages (or wheels) in tandem. The turbine drives its pump through a flexible coupling. The turbine to baseplate mounting is accomplished by bolting and doweling at the exhaust end, and by a sliding pedestal (axial movement only) at the inlet end. The HPCI turbine is mounted on its own baseplate, independent of the pump mounting.

Statement of Criteria	Method of Analysis	Results	
1. Closure bolting is designed to contain the internal design pressure of the turbine casing without exceeding the allowable working stress of the bolting material. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	1. Bolting loads and stresses are calculated in accordance with the "Rules for Bolted Flange Connections", ASME Boiler and Pressure Vessel Code, Section VIII, Appendix II.	1. Allowable Stress	20,000 psi
2. The minimum wall thickness of the turbine casing limits stress to the allowable working stress when subjected to design pressure plus corrosion allowance. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	2. Stresses in the various pressure containing portions of the turbine casing are calculated according to the Rules of Part UG, Section VIII, of the ASME Boiler & Pressure Vessel Code.	2. Allowable Stress	17,500 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
HPCI Turbine (Cont'd)

Statement of Criteria	Method of Analysis
3. The forces and moments imposed by the attached piping on the turbine inlet and exhaust connections satisfy the following conditions: The resultant force and moment from the combination of dead weight and thermal expansion is less than that stipulated by the equipment vendor. The resultant force and moment from the combination of dead weight, thermal expansion and OBE (E) or SSE (E') is less than that demonstrated acceptable by detailed seismic analysis of the equipment.	3. The total resultant of the forces and the total resultant of the moments on both the inlet and the exhaust connections of the turbine satisfies the following conditions: For the combination of dead weight and maximum thermal expansion, Inlet-----F = 2,520 lbs. M = 7,570 ft. lbs. Exhaust---F = 3,310 lbs. M = 9,930 ft. lbs. For the combination of dead weight, maximum thermal expansion and OBE (E), Inlet-----F = 8,000 lbs, but not to exceed 5,000 lbs. M = 20,000 ft. lbs. Exhaust---F = 25,000lbs, but not to exceed 11,500 lbs. M = 20,000ft. lbs. For the combination of dead weight, maximum thermal expansion, and SSE (E'), Inlet-----F = 12,000 lbs, but not to exceed 7,500 lbs. M = 30,000 ft. lbs. Exhaust---F = 37,500 lbs, but not to exceed 17,250 lbs. M = 30,000 ft. lbs.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
HPCI Turbine (Cont'd)

Statement of Criteria

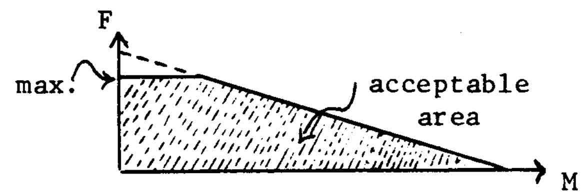
3. Continued

4. The stresses in the turbine anchor bolts (turbine to baseplate, and baseplate to foundation) due to the combination of the OBE (E) acting on the turbine while running plus the total piping loads [weight, thermal & OBE (E)] does not exceed the allowable tensile stress nor the allowable shear stress for the bolting materials as specified in the ASME Boiler & Pressure Vessel Code, Section VIII.

Method of Analysis

where "F" is the resultant force in lbs, and "M" is the resultant moment in ft-lbs

Typical acceptable area on the force-moment diagram is indicated below:



4. Vertical forces on the anchor bolts, subjecting them to tension, are the sum of the following:

a) Weight of the turbine assembly times the vertical component of acceleration,
b) The vertical pipe force reactions,
c) The pipe moment reactions tending to tip the turbine and subject the bolting to tension

Horizontal forces on the anchor bolts, subjecting them to shear and tension, are the sum of the following:

a) Weight of the turbine assembly times the horizontal component of acceleration acting on the center of gravity,
b) The horizontal pipe force reactions,
c) The effect of pipe moment reactions causing horizontal loading at the anchor bolts.

NOTE: Friction shall not be considered to be restrictive

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
HPCI Turbine (Cont'd)

Statement of Criteria	Method of Analysis
5. The stresses in the turbine anchor bolts (turbine to baseplate and baseplate to foundation) due to the combination of SSE (E') acting on the turbine in standby plus the total piping loads [weight, thermal and SSE (E')] does not exceed 0.9 times the yield stress in tension, nor twice the allowable shear stress for the bolting materials as specified in the ASME Boiler and Pressure Vessel Code, Section VIII.	5. Same as analysis under (4), above.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RCIC Turbine

OBE (E) Coefficients	0.75 g horizontal	0.07 g vertical
SSE (E') Coefficients	1.50 g horizontal	0.14 g vertical

The RCIC turbine is a horizontally mounted solid wheel turbine. The turbine drives its pump through a flexible coupling. The turbine to baseplate mounting is accomplished by bolting and doweling at the exhaust end, and by a sliding pedestal (axial movement only) at the inlet end. The RCIC turbine is mounted on a common baseplate with its pump.

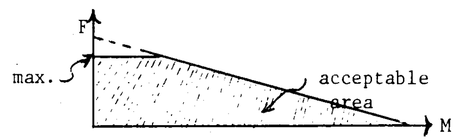
Statement of Criteria	Method of Analysis	Results
1. Closure bolting is designed to contain the internal design pressure of the turbine casing without exceeding the allowable working stress of the bolting material. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	1. Bolting loads and stresses are calculated in accordance with the "Rules for Bolted Flange Connections", ASME Boiler and Pressure Vessel Code, Section VIII, Appendix II.	1. Maximum Allowable Stress 20,000 psi
2. The minimum wall thickness of the turbine casing limits stress to the allowable working stress when subjected to design pressure plus corrosion allowance. Allowable stresses are in accordance with ASME B&PV Code, Section VIII.	2. Stresses in the various pressure containing portions of the turbine casing are calculated according to the rules of Part UG, Section VIII, of the ASME Boiler & Pressure Vessel Code.	2. Maximum Allowable Stress 17,500 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RCIC Turbine (Cont'd)

Statement of Criteria	Method of Analysis
3. The forces and moments imposed by the attached piping on the turbine inlet and exhaust connections satisfy the following conditions:	3. The total resultant of the forces and the total resultant of the moments on both the inlet and the exhaust connections of the turbine satisfy the following conditions:
The resultant force and moment from the combination of dead weight and thermal expansion are less than that stipulated by the equipment vendor.	For the combination of dead weight and maximum thermal expansion, Inlet-----F = 870 lbs. M = 2,620 ft. lbs. Exhaust---F = 2,000 lbs. M = 6,000 ft. lbs.
The resultant force and moment from the combination of dead weight, thermal expansion and OBE or SSE is less than that demonstrated acceptable by detailed seismic analysis of the equipment.	For the combination of dead weight, maximum thermal expansion, and OBE (E), Inlet-----F = 1,200 lbs M = 3,000 Exhaust---F = 18,000 lbs, but not to exceed 8,370 lbs. M = 6,000 ft. lbs.
	For the combination of dead weight, maximum thermal expansion, and SSE (E'), Inlet-----F = 1,800 lbs. M = 4,500 ft. lbs. Exhaust---F = 27,000 lbs, but not to exceed 12,555 lbs. M = 9,000 ft. lbs.
	where "F" is the resultant force in lbs and "M" is the resultant moment in ft. lbs.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RCIC Turbine (Cont'd)

Statement of Criteria	Method of Analysis
	Typical acceptable area on the force-moment diagram is indicated below:



Statement of Criteria	Method of Analysis
4. The stresses in the turbine anchor bolts (turbine to baseplate) due to the combination of the OBE (E) acting on the turbine while running plus the total piping loads [weight, thermal and SSE (E)] shall not exceed the allowable tensile stress nor the allowable shear stress for the bolting materials as specified in the ASME Boiler and Pressure Vessel Code, Section VIII.	4. Vertical forces on the anchor bolts are the sum of the following: a) Weight of the turbine assembly times the vertical component of acceleration, b) The vertical pipe force reactions, c) The pipe moment reactions tending to tip the turbine and subject the bolting to tension. Horizontal forces on the anchor bolts, subjecting them to shear and tension, are the sum of the following: a) Weight of the turbine assembly times the horizontal component of acceleration acting on the center of gravity, b) The horizontal pipe force reactions, c) The effect of pipe moment reactions causing horizontal loading at the anchor bolts. NOTE: Friction shall not be considered to be restrictive

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
RCIC Turbine (Cont'd)

Statement of Criteria	Method of Analysis
5. The stresses in the turbine anchor bolts (turbine to baseplate) due to the combination of SSE (E') acting on the turbine in standby plus the total piping loads [weight, thermal and SSE (E')] does not exceed 0.9 times the yield stress in tension nor twice the allowable shear stress for the bolting materials as specified in the ASME Boiler and Pressure Vessel Code, Section VIII.	5. Same as analysis under (4), above.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Isolation Valves

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
1. Body Minimum Wall Thickness Load: Design pressure and temperature Primary membrane stress limit: S = 7,000 psi per ASA B16.5	Minimum wall thickness in the cylindrical portions of the valve is calculated using the following formula: $t = 1.5\left[\frac{Pd}{2S-1.2P} + C\right]$ where: S = allowable stress of 7000 psi P = primary service pressure, 655 psi d = inside diameter of valve at section being considered, inches C = corrosion allowance of 0.12 inches	t = 1.74 in.
2. Cover Minimum Thickness Loads: Design pressure and temperature Design bolting load Gasket load Primary stress limit: Allowable working stress per ASME Section VIII.	$t = d\left[\frac{CP}{S} + \frac{1.78\ Wh_G}{Sd^3}\right]^{1/2} + C_1$ where: t = minimum thickness, inches d = diameter or short span, inches C = attachment factor S = allowable stress, psi W = total, bolt load, pounds ${}^{h}C$ = gasket moment arm, inches C_1 = corrosion allowance, inches	t = 4.50 in

TABLE C-3-7 (Cont'd)
Main Steam Isolation Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
3. Cover Flange Bolt Area: Loads: Design pressure and temperature Gasket load Stem operational load Seismic load - E' Bolting stress limit: Allowable working stress per ASME Nuclear Pump & Valve Code, Class I.	Total, bolting loads and stresses are calculated in accordance with "Rules for Bolted Flange Connections" - ASME Boiler & Pressure Vessel Code, Section VIII, Appendix II, except that the stem operational load and seismic loads are included in the total load carried by bolts. The horizontal and vertical seismic forces are applied at the mass center of the valve operator assuming that the valve body is rigid and anchored	S = 33,100 psi @ 575°F
4. Body Flange Thickness & Stress Loads: Design pressure & temperature Gasket load Stem operational load Seismic load - E' Flange Stress Limits: S_H, S_R, S_T 1.5 S_m per ASME Nuclear Pump & Valve Code, Class I	Flange thickness and stress are calculated in accordance with "Rules for Bolted Flange Connections" - ASME Boiler & Pressure Vessel Code, Section VIII, Appendix II, except that the stem operational load and seismic loads are included in the total load carried by the flange. The horizontal and vertical seismic forces are applied at the mass center of the valve operator assuming that the valve body is rigid and anchored.	S = 35,000 psi S = 35,000 psi S = 35,000 psi
5. Valve Disc Thickness Load: Design pressure & temperature Primary bending stress limit: Allowable working stress per ASME Section VIII.	$\mathrm{MaxSr} = \frac{3W}{2\pi t^2}\left[\frac{2a^2(m+1)\log\frac{a}{\sqrt{b}} + a^2(m-1) - b^2(M-1)}{a^2(m+1) + b^2(m-1)}\right]$ where: W = total applied load, lb. t = thickness of disc, in. a = outer radius of disc b = inner radius (fixed) of disc m = 3.85, reciprocal of Poisson's ratio	S = 23,300 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Isolation Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
6. Valve Operator Supports Loads: Design pressure & temperature Stem operational load Equipment dead weight Seismic load - E' Support Rod Stress Limit: Allowable working stress per ASME Section VIII.	The valve assembly is analyzed assuming that the valve body is an anchored, rigid mass and that the specified vertical and horizontal seismic forces are applied at the mass center of the operator assembly, simultaneously with operating pressure plus dead weight plus operational loads. Using these loads, stresses and defelections are determined for the operator support components.	S = 20,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Safety Valves

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
1. Inlet Nozzle Wall Thickness Load: 1.1 x Design Press. @ 600°F Primary Membrane Stress Limit: Allowable stress intensity as defined by ASME Standard Code for Pumps & Valves for Nuclear Power	$t = \frac{PR}{SE - 0.6P} + C$ where: t = min. required thickness, inches S = allowable stress, psi P = 1.1 x design press., psi R = internal radius, inches E = joint efficiency C = corrosion allowable, inches	t = 0.143 in.
2. Valve Disc Thickness Load: 1.1 x Design press. @ 600°F Diagonal Shear Stress Limit: 0.6 x allowable stress intensity as defined by ASME Standard Code for Pumps & Valves for Nuclear Power.	$S_s = \frac{W}{A} = \frac{PA_1}{A}$ where: W = shear load, lb. A = shear area, in^2 P = 1.1 x design press, psi A_1 = disc area, in^2 and: A = $\pi S (R + R^1)$ S = slope of frustrum of shear cone, in. R = radius at base of cone, in. R^1 = radius at top of cone, in.	S_s = 12,780 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Safety Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
3. Inlet Flange Bolt Area Loads: Design pressure & temperature Gasket load Operational load SSE (E') Bolting Stress Limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps & Valves for Nuclear Power	Total bolting loads and stresses are calculated in accordance with procedures of Para. 1-704.5.1 Flanged Joints, of B31.7 Nuclear Piping Code.	S_b = 27,700 psi
4. Inlet Flange Thickness Loads: Design pressure & temperature Gasket load Operational load Seismic load - E' Flange Stress Limits, S_H, S_R, S_T 1.5 S_m per ASME Nuclear Pump & Valve Code	Flange thickness and stresses are calculated in accordance with procedures of Para. 1-704.5.1 Flanged Joints, of B31.7 Nuclear Piping Code.	S_H = 27,300 psi S_R = 27,300 psi S_T = 27,300 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Safety Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
5. Valve Spring-Torsional Stress Loads: W_1 = Set point load, lbs W_2 = Spring load at maximum lift, lbs. Torsional Stress Limit: 0.67 x torsional elastic limit when subjected to a load of W_1. 0.90 x torsional elastic limit when subjected to a load of W_2.	$Smax = \frac{8PD}{Hd^3}\left[\frac{4C-1}{4C-4} + \frac{0.615}{C}\right]$ where: S_{max} = torsional stress, psi P = W_1 or W_2 = spring load, lbs D = mean diameter of coil, in. d = diameter of wire, in. C = $\frac{D}{D}$ = correction factor	Set Point S = 62,500 psi Max. Lift S = 93,750 psi
6. Yoke Rod Area Load: Spring load at maximum lift Primary Stress Limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps & Valves for Nuclear Power	$A = \frac{F}{2S_m}$ where: A = required area per rod, in^2 F = total spring load, lbs S_m = allowable stress, psi	A = 0.575 in^2

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Safety Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
7. Yoke Bending & Shear Stresses Load: Spring load at maximum lift Bending & Shear Stress Limits: Bending - allowable stress intensity, S_m, per ASME Nuclear Pump & Valve Code Shear - 0.6 x allowable stress intensity, 0.6 S_m, per ASME Nuclear Pump & Valve Code.	$S_b = \frac{M}{Z},\ S_s = \frac{V}{A}$ where: S_b = bending stress, psi S_s = shear stress, psi M = bending moment, in. lbs. Z = section modulus, in^3 V = vertical shear, lb. A = shear area, in^2	S_b = 18,200 psi S_s = 10,900 psi
8. Body Minimum Wall Thickness Load: Primary service pressure Primary Stress Limit: Allowable stress, 7,000 psi, in accordance with ASA B16.5.	$t = 1.5\left[\frac{Pd}{2S\ \ 1.2P}\right] + C$ where: t = required thickness, in. S = allowable stress, 7,000 psi P = primary service pressure, 150 psi d = inside diameter of valve at section being considered, in.	Body Bowl t = 0.311 in. Inlet Nozzle t = 0.218 in. Outlet Nozzle t = 0.250 in.
9. Inlet Nozzle Combined Stress Loads: Spring load at maximum lift Operational load Seismic load - E' Combined Stress Limit: 1.5 x allowable stress intensity, 1.5 S_m, per ASME Code for Pumps & Valves for Nuclear Power.	$S = \frac{F_1 + F_2}{A} + \frac{M_1 + M_2}{Z}$ where: S = combined bending & tensile stress, F_1 = maximum spring load, lb. F_2 = vertical component of reaction thrust, lb. A = cross section area of nozzle, in^2 M_1 = moment resulting from horizontal component of reaction, lb. in. M_2 = moment resulting from horizontal	S = 27,300 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Safety Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
10. Spindle Diameter Load: Spring load at maximum lift Spindle Column Load Limit: 0.2 x critical buckling load	$F_c = \frac{\pi^2 EI}{L^2}$ where: F_c = critical buckling load, lb. E = modulus of elasticity, psi I = moment of inertia, in^4 L = length of spindle in compression, in.	Load Limit ($0.2F_c$) F = 85,900 lb.
11. Spring Washer Shear Area Load: Spring load at maximum lift Shear Stress Limit: 0.6 x allowable stress intensity, 0.6 S_m, per ASME Nuclear Pump & Valve Code.	$S_s = \frac{F}{A}$ where: S_s = shear stress, psi F = spring load, lb. A = shear area, in^2	S_s = 15,960 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Relief Valves

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
1. Body Minimum Wall Thickness Load: Design Pressure & Temperature Primary Membrane Stress Limit: Allowable working stress as defined by USAS B16.5 (7,000 psi @ primary service pressure).	$t = 1.5\left[\frac{Pd}{2S-1.2P}\right] + C$ where: t = minimum required thickness, in. S = allowable stress, 7,000 psi P = primary service pressure, 655 d = inside diameter of valve at section being considered, in. C = corrosion allowance, 0.12 in.	Main Body t = 0.625 in. Bonnet t = 0.287 in.
2. Bonnet Cap & Pilot Base Minimum Thickness Loads: Design pressure & temperature Gasket load Primary Stress Limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps & Valves for Nuclear Power.	$t = d\left[\frac{CP}{S_m} + \frac{1.78\ Wh_G}{S_m d^3}\right]^{1/2} + C_1$ where: t = minimum required thickness in. d = diameter or short span, in. C = attachment factor, ASME Section VIII P = design pressure, psi S_m = allowable stress, psi W = total bolt load, lb. h_G = gasket moment arm, in. C_1 = corrosion allowance, 0.12 in.	Bonnet Cap t = 1.0 in. Pilot Base t = 2.219 in.

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Relief Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
3. Flange Bolt Area - Inlet Flange, Outlet Flange, Body to Bonnet, Bonnet to Base	Total bolting loads and stresses are calculated in accordance with procedures of Para. 1-704.5.1 Flanged Joints, of B31.7 Nuclear Piping Code.	Body to Base A_b = 10.26 in^2
Loads: Design pressure & temperature Gasket load Operational load SSE (E')		Bonnet to Cap A_b = 1.452 in^2 Inlet Flange A_b = 13.9 in^2
Bolting Stress Limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps & Valves for Nuclear Power.		Outlet Flange A_b = 12.2 in^2
4. Flange Thickness - Inlet, Outlet, Bonnet Flanges Loads: Design pressure & temperature Gasket load Operational load SSE (E')	Flange thickness and stresses are calculated in accordance with procedures of Para. 1-704.5.1 Flanged Joints, of B31.7 Nuclear Piping Code.	Body to Base S_H = 26,250 psi S_R = 26,250 psi S_T = 26,250 psi Cap to Bonnet S_H = 26,250 psi S_R = 26,250 psi S_T = 26,250 psi
Flange Stress Limits, S_H, S_R, S_T 1.5 S_m per ASME Nuclear Pumps and Valve Code.		Inlet Flange S_H = 26,250 psi S_R = 26,250 psi S_T = 26,250 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Main Steam Relief Valves (Cont')

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD.
		Outlet Flange S_H = 26,250 psi S_R = 26,250 psi S_T = 26,250 psi
5. Valve Disc Thickness & Stress Load: Design pressure & temperature Primary Stress Limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps & Valve for Nuclear Power.	$S_r = S_t = \frac{3\,(3 + \upsilon)\;PR^2}{8\;t^2}$ where: S_r = radial stress, psi S_t = tangential stress, psi ν = Poisson's ratio P = design pressure, psi R = radius of disc, in. T = thickness of disc, in.	Disc Stress S_m = 15,800 psi
6. Inlet Nozzle Diameter Thickness & Stress Loads: Design pressure & temperature Operational load SSE (E') Primary Stress Limit: 1.5 x allowable stress intensity, 1.5 S_m, as defined by ASME Standard Code for Pumps & Valves for Nuclear Power.	$S = \frac{F_1 + F_2}{A} + \frac{M_1 + M_2}{Z}$ where: S = combined bending and tensile stress, psi F_1 = vertical load due to design pressure, lb. F_2 = vertical component of reaction thrust, lb. A = cross section area of nozzle, in^2 M_1 = moment resulting from horizontal reaction, in. lb. M_2 = moment resulting from horizontal seismic force at mass center of valve, in. lb.	Inlet Nozzle Stress S = 26,250 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Recirculation Valves

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS
1. Body Minimum Wall Thickness Load: Design pressure and temperature Primary membrane stress limit: S = 7,000 psi per ASA B16.5	Minimum wall thickness in the cylindrical portions of the valve are calculated using the following formula: $t = 1.5\left[\frac{Pd}{2S-1.2P}\right] + C$ where: S = allowable stress of 7000 psi P = primary service pressure, 655 psi d = inside diameter of valve at section being considered, inches C = corrosion allowance of 0.12 inches	Body Wall Thickness t = 1.875 in.
2. Cover Minimum Thickness Loads: Design pressure and temperature Design bolting load Gasket load Primary stress limit: Allowable working stress per ASME Section VIII.	$t = d\left[\frac{CP}{S} + \frac{1.78\ Wh_G}{Sd^3}\right]^{1/2} + C_1$ where: t = minimum thickness, inches d = diameter of short span, inches C = attachment factor S = allowable stress, psi W = total, bolt load, pounds h_C = gasket moment arm, inches C_1 = corrosion allowance, inches	Valve Cover Thickness and Stress t = 5.469 in. S_{allow} = 17,800 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Recirculation Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS
3. Cover Flange Bolt Area Loads: Design pressure and temperature Gasket load Stem operational load Seismic load - DBE Bolting stress limit: Allowable working stress per ASME Nuclear Valve and Pump Code, Class I.	Total, bolting loads and stresses are calculated in accordance with "Rules for Bolted Flange Connections" - ASME Boiler & Pressure Vessel Code, Section VIII, Appendix II, except that the stem operational load and seismic loads are included in the total load carried by bolts. The horizontal and vertical seismic forces are applied at the mass center of the valve operator assuming that the valve body is rigid and anchored.	S = 30,900 psi @ 575°F
4. Body Flange Thickness & Stress Loads: Design pressure & temperature Gasket load Stem operational load Seismic load - DBE Flange Stress Limits: S_H, S_R, S_T 1.5 S_m per ASME Nuclear Pump & Valve Code, Class I	Flange thickness and stress are calculated in accordance with "Rules for Bolted Flange Connections" - ASME Boiler & Pressure Vessel Code, Section VIII, Appendix II, except that the stem operational load and seismic loads are included in the total load carried by the flange. The horizontal and vertical seismic forces are applied at the mass center of the valve operator assuming that the valve body is rigid and anchored.	S = 26,700 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Recirculation Valves (Cont'd)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS
5. Valve Disc Thickness Load: Design pressure & temperature Primary heading stress limit: Allowable working stress per ASME Section VIII.	$S_r = S_t = \frac{3\,(3 + \upsilon)\;PR^2}{8\;t^2}$ where: S_r = radial stress, psi S_t = tangential stress ν = Poisson's ratio P = design pressure, psi R = radius of disc, inches t = thickness of disc, inches	S = 17,800 psi
6. Valve Operator Supports Loads: Design pressure & temperature Stem operational load Equipment dead weight Seismic load - SSE Support Rod Stress Limit: Allowable working stress per ASME Section VIII.	The valve assembly is analyzed assuming that the valve body is an anchored, rigid mass and that the specified vertical and horizontal seismic forces are applied at the mass center of the operator assembly, simultaneously with operating pressure plus dead weight plus operational loads. Using these loads, stresses and deflections are determined for the operator support components.	S = 18,000 psi

TABLE C-3-7 (Cont'd)
LOADING CRITERIA
Standby Liquid Control Tank

OBE (E) Coefficients	0.33 g horizontal	0.07 g vertical
SSE (E') Coefficients	0.66 g horizontal	0.14 g vertical

The standby liquid control tank is a vertical tank under atmospheric pressure. All vertical and horizontal shell joints are double welded, full penetration butt joints. The baseplate supports the tank on its foundation. The baseplate is fastened to the foundation by means of anchor bolts.

Statement of Criteria

1. The minimum thickness of the shell plates is computed from the stress on the vertical joints using a joint efficiency of 0.85. The corrosion allowance is added to the minimum thickness to obtain design thickness. In no case is the nominal thickness of shell plates less than 3/16".

   The allowable stress is in accordance with ASME Boiler and Pressure Vessel Code, Section VIII.

2. The dead weight of the tank, the seismic acceleration [SSE (E')] on its center of gravity and pipe reaction on the outlet nozzle from maximum thermal expansion and SSE (E') does not produce excessive shell stress.

Method of Analysis

1. The following formula is used in calculating the minimum thickness of the shell plate

$$t = \frac{2.6(D)\ (H-1)\ (G)}{fo\ E}$$

where:
- t = minimum thickness - inches
- D = nominal inside diameter of tank-ft
- H = height from bottom of tank to overflow level-ft
- G = specific gravity of fluid
- fo= Allowable design stess-psi
- E = Joint efficiency dimensionless

2. The dead weight of shell, liquid and attachments, the seismic bending moment, and the pipe reaction forces and moments does not produce tensile stresses in excess of allowable or compressive (buckling) stresses in excess of 1/3 of yield strength.

Results

1. Minimum Thickness 0.0154"

2. Allowable Stress

Tensile	17,500 psi
Compressive	10,000 psi

3.3 Method of Analysis and Implementation of Criteria

3.3.1 Reactor Pressure Vessel

The Reactor Pressure Vessel (RPV) has been designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, its interpretations, and applicable requirements for Class A vessels as defined therein, 1965 Edition, using addenda up to and including the Winter 1966 Addenda.

Stress analysis requirements and load combinations for the Reactor Pressure Vessel have been evaluated for the cyclic conditions expected throughout the 40 year life, with the conclusion that ASME Code limits are satisfied.

The vessel design report contains the results of the detailed design stress analyses performed for the Reactor Pressure Vessel to meet the Code requirements. Selected components, considered to possibly have higher than Code design primary stresses as a result of rare events or a combination of rare events, have been analyzed in accordance with the requirements of the loading criteria in this appendix. Descriptions of the most critical of those analyses are included in Table C-3-7, LOADING CRITERIA. The analyses show that the limits in the criteria have been met.

3.3.1.1 Reactor Pressure Vessel Fatigue Analysis

An analysis of the Reactor Pressure Vessel shows that all components are adequate for cyclic operation by the rules of Section III of the ASME Code. The critical components of the vessel are evaluated on a fatigue basis, calculating cumulative usage factors (ratios of required cycles to allowed cycles-to-failure) for all operating cycle conditions. The cumulative usage factors for the critical components of the vessel are below the Code allowable of 1.0.

3.3.1.2 Reactor Pressure Vessel Seismic Analysis

A seismic analysis was performed for a coupled system consisting of the Reactor Building, the Drywell, the Reactor Pressure Vessel, and the Reactor Pressure Vessel internals. The analysis is discussed below in USAR Section C-3.3.2.3.

3.3.1.3 Reactor Pressure Vessel Support Structure

The Reactor Pressure Vessel (RPV) is supported on the bottom by a steel cylindrical skirt welded to the vessel. The skirt sets on top of a ring shaped girder, which in turn is bolted to the top of the Reactor Pedestal. The RPV is also supported, near the top, by lateral stabilizers which transfer the loads through the Sacrificial Shield Wall and the Drywell shell to the Drywell Biological Shield Wall. Some of the more critical components of the RPV support structure include the RPV stabilizer, the RPV stabilizer brackets and their connection to the RPV shell, the RPV vessel support skirt, and the RPV ring girder. The results of the analysis for the Reactor Pressure Vessel support structure are documented in the calculations of record and show that the acceptance criteria described above are met.

3.3.1.4 Miscellaneous Associated Components and Supports

There are several miscellaneous components and supports associated with the Reactor Pressure Vessel which are evaluated for various loading conditions. These include the Control Rod Drive (CRD) housing support, the Reactor Recirculation (RR) pipe and pump restraints, the hydraulic control unit piping, and the spent fuel storage racks. The results of the analysis for these components show that the acceptance criteria described above are met.

3.3.2 Reactor Pressure Vessel Internals

Although not mandatory at the time, the design of the Reactor Pressure Vessel (RPV) internals is in accordance with the intent of Section III of the ASME B&PV Code. The Material used for fabrication of most of the materials is solution heat treated, unstabilized Type 304 austenitic stainless steel conforming to ASTM specifications. Allowable stresses for the internals materials under normal operating conditions are taken directly from Section III. Methods of analysis, use as a guide, the design procedures of Section III. For rare events or a combination of rare events, the internals have been analyzed in accordance with the requirements of the loading criteria in this appendix. Details of the most critical analyses for RPV internal components are included in Table C-3-7, LOADING CRITERIA. The analyses show that the limits in the criteria have been met.

3.3.2.1 Internals Deformation Analysis

Control Rod System

If there were excessive deformation of the control rod system, made up of the control rod drive, control rod drive housing, control rod, control rod guide tube and fuel channels and the core structural elements which support them (top guide, core support and shroud and shroud support) it could possibly impede control rod insertion. The maximum loading condition that would tend to deform these long, slender components is the Safe Shutdown Earthquake. Descriptions of the analyses of the internal components, which have the highest calculated stresses, are included in the following paragraphs. The highest calculated stresses occur where the Safe Shutdown Earthquake and loads resulting from the Design Basis Accident (DBA) line break are considered to occur simultaneously. Even in these cases, the general stress levels are relatively low. No significant deformation is associated with these calculated stresses; therefore, rod insertion would not be impeded after an assumed simultaneous maximum possible earthquake and line break accident.

Core Support

The core support sustains the pressure drop across the fuel. This pressure drop is the only load which causes significant deflection of the core support. Excessive core support deflection could lift the control rod guide tubes off their seats on the control rod drive housings and thereby increase core bypass leakage. This upward deflection would have to be 1/2-inch to begin to lift guide tubes. The maximum deflections under normal operation conditions and pipe rupture differential pressures for the core support are calculated to be very small as compared to 1/2-inch. The guide tubes will, therefore, not be lifted off, although even if they were, this would not be of concern because bypass leakage at this time is not important.

3.3.2.2 Internals Fatigue Analysis

Fatigue analysis is performed using ASME Section III as a guide. The method of analysis used to determine the cumulative fatigue usage is described in APED - 5460, "Design and Performance of GE-BWR Jet Pumps", September, 1968. The most significant fatigue loading occurs in the jet pump - shroud - shroud support area of the internals. The analysis was performed for a plant where the configuration (gusset type shroud support) was almost identical to the Cooper Station. Therefore, the calculated fatigue usage is expected to be a reasonable approximation for this station.

Loading Combinations and Transients Considered

1. Normal startup and shutdown

2. Operating Basis and Safe Shutdown Earthquakes

3. Ten minute blowdown from a stuck relief valve

4. HPCI operation

5. LPCI operation (DBA)

6. Improper start of a Reactor Recirculation loop

The allowable cumulative fatigue usage factor is 1.0. The analysis shows that the actual usage factor is less than the allowable.

3.3.2.3 Internals Seismic Analysis

The seismic loads on the Reactor Pressure Vessel (RPV) and internals are based on a dynamic analysis of the coupled model consisting of the Reactor Building, the RPV and internals as described in the succeeding paragraphs. The natural frequencies and mode shapes for the system were determined. The relative displacement, acceleration and load response of the RPV and internals were then determined using the time history method of analysis. The dynamic response was determined for each mode of interest and added algebraically for each instant of time. Resulting response time-histories were then examined, and the maximum value of displacements, accelerations, shears and moments were used for design calculations. These results were combined with the results of other loads for the various loading conditions. The loading conditions for the critical components are also presented in Table C-3-7, LOADING CRITERIA. The dynamic model of the RPV and internals is briefly described below.[19]

The presence of fluid and other structural components, e.g., fuel within the RPV, introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix.

The seismic model of the RPV and internals analyzed has one horizontal translation coordinate for each node point considered in the analysis. Due to the approximate coincidence of the mass center and elastic axis of the building and symmetry of the RPV and internals, one horizontal coordinate was excluded, i.e., motion in the two horizontal directions do not interact and separate analysis can be performed. The remaining translational coordinate for the various node points was the vertical coordinate. This coordinate (vertical) was excluded because the frequency content of earthquakes is such that the vertical frequencies of the RPV and internals are well above those of earthquakes. Dynamic loads due to vertical motion were added to, or subtracted from, the static loads of the components, whichever is the more conservative. The two rotational coordinates about each node point were excluded because the moment contribution of rotary inertia is negligible. The remaining rotational coordinate has been omitted since the seismic response of the RPV is negligible.

Seismic analyses were performed by coupling the lumped mass model of the RPV and internals with the building model to determine the system natural frequencies and mode shapes. The load response of the RPV and internals was then determined by the time history method. The maximum ground acceleration time histories for the design earthquakes are described in USAR

Section II-5.2.3. The seismic response was determined by uncoupling the equations of motion with a coordinate transformation, dividing the ground acceleration time history into small time increments and then finding the modal response by way of duhamels integral. Applying the coordinate transformation by the modal responses, the time history displacements acceleration and loads were determined. Finally maximum responses were isolated from the time history response results and used to verify seismic design adequacy of the Reactor Pressure Vessel and internals.

3.3.3 Piping

3.3.3.1 Piping Flexibility Analysis

The piping has been analyzed for the effects of dead loads, external loads, and thermal loads. Combined bending and torsional stresses were calculated in accordance with USAS B31.1.0-1967, Power Piping, including intensification factors. Several pressure temperature cycles were evaluated and the cycle representing the worst for thermal expansion stresses was selected for the design case. Critical points were evaluated to the stress limits of USAS B31.1.0-1967 for the standard load combination events. In addition, for events with very low probability of occurrence, stresses at critical points were compared with the limits defined in this appendix. Fatigue analyses were also performed for Class IN piping using methods and allowable limits of ANSI B31.7-1969. The loading criteria and allowable stresses are summarized in Table C-3-7, LOADING CRITERIA.

Analysis of the plant piping systems demonstrate compliance with the above criteria. Note- the load combinations involving P_{max} (peak pressure) were evaluated on a generic basis and were shown not to control in comparison to other load combinations. Also, enveloping load combinations are used in some of the calculations of record to simplify the analysis.

Certain piping systems, such as those associated with the Piping Replacement Project and the Mark I Program, are designed and analyzed to a different set of criteria and/or are subjected to additional dynamic loads. These other piping analyses are described in more detail in USAR Section App.C-3.3.3.5.

3.3.3.2 Piping Seismic Analysis

Seismic Class IS piping systems 2½" and greater in diameter were dynamically analyzed using the "response spectrum method" of analysis. For each of the piping systems, a mathematical model consisting of lumped masses at discrete joints connected together by weightless elastic elements was constructed. Valves were also considered as lumped masses in the pipe, and valve operators as lumped masses acting through the operator center of gravity. Where practical, a support is located on the pipe at or near each valve. Stiffness matrix and mass matrix were generated and natural periods of vibration and corresponding mode shapes were determined. Input to the dynamic analyses were the 0.5% damped acceleration response spectra for the applicable floor elevation. The increased flexibility of the curved segments of the piping systems was also considered. The results for earthquakes acting in the X and Y (vertical) directions simultaneously (combined by absolute summation), and Z and Y directions simultaneously (combined by absolute summation) were computed separately. The maximum responses of each mode are calculated and combined by the root mean square method to give the maximum quantities resulting from all modes (the response of closely spaced modes was combined by absolute summation). The response thus obtained was combined with the results produced by other loading conditions to compute the resultant stresses. For Seismic Class IS piping systems less than 2½" in diameter, piping and supports were field routed using span and load chart tables.[66]

The dynamic seismic stress analyses includes both bending and torsional effects of the eccentric masses on the piping. The torsional effect of eccentric masses (e.g., valve operators, etc.) having a significant effect on the results of the analysis have been included in the mathematical model. However, if the pipe stress due to the torsional effect is expected to be less than 500 psi (based upon hand calculations and experience), the offset moment due to the eccentric mass may be neglected.[20]

In addition to the piping tabulated in Table C-3-7, other Class IS piping systems were analyzed for stress due to normal and upset loadings and were shown not to exceed the limits of B31.1.0.[22] These piping systems were also analyzed for the following load combinations that have a very low probability of occurrence:

1. The sum of the primary longitudinal stresses due to pressure, dead weight and inertia effects of maximum possible earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40-year plant life is 10E-3 and the safety factor (SF) = 1.5.

2. The sum of the primary longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum probable earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40-year plant life is 10E-2 and SF = 1.8.

3. The sum of the primary longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum possible earthquake must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40-year plant life is 0.25 x 10E-3 and SF = 1.36.

3.3.3.3 Pipe Rupture Loading[23]

For pipe rupture loading (R), the maximum capacity of the process pipe to deliver load was utilized. This concept defines the maximum capability of piping to transmit load, as the load caused by an ultimate bending moment (M_U), and occurs when all the piping material is at the actual yield strength of the material. Any additional pipe loading would cause increase in strain without increase in stress. The actual yield strength of the material was derived from mill test reports for the process piping in question.

In addition to the moment (M_U), rupture loading (R) also considers a force (P) applied to the nozzle. This force was assumed to be the process pipe pressure multiplied by the cross-sectional area of the pipe.

When rupture loading is applied to a containment penetration, the assembly deflects and contact is made between the flued head fitting and the limit stops at the flange. The flange, which is provided at this location, serves as a stiffening ring which transmits the ensuing rupture loading to the embedded sleeve, which in turn transmits the load to the Drywell Biological Shield Wall.

3.3.3.4 Drywell Penetration Limit Stops[23]

When required by analysis of the Drywell penetration nozzles and/or when required to provide positive bellows seal protection, limit stops are utilized in the vicinity of the flued head fittings. There are three different types of Drywell penetrations:

Type 1 Penetrations have expansion joint bellows seals to allow thermal expansion (refer to USAR Figure V-2-3) Type 1 limit stops are shown typically on Burns and Roe Drawing SKM200. Note that the axial limit stops provide assurance that the bellows seals will not fail due to excessive torsional rotation, or due to axial collapse of the joint at the time of pipe rupture.

Type 2 Penetrations have no provisions for thermal expansion (refer to USAR Figure V-2-4). Limit stops are not required for Type 2 penetrations.

Type 3 Penetrations have thermal sleeves for minor thermal expansion (refer to USAR Figure V-2-5). Type 3 limit stops are shown typically on Burns and Roe Drawing SK101670R. No axial or torsional stops are required, and only certain Type 3 penetrations require lateral stops.

The loading combinations which control the design of the limit stops are dead load plus pipe rupture.

As shown on Burns and Roe Drawings SKM200 and SK101670R, the limit stops consist of sleeves, embedded in the Drywell Biological Shield Wall and extend to the flued head fitting outside the shield wall, coaxially with the containment penetration. In the vicinity of the flued head fitting, a flange is attached to the sleeve extension. A gap is maintained between the sleeve and the penetration assembly, so that no contact is made due to thermal differential movements.

3.3.3.5 Other Piping Analyses

3.3.3.5.1 Main Steam Piping

An analysis was performed[24] to verify that the Main Steam piping within the Reactor Coolant Pressure Boundary, the relief valve discharge piping, and the associated piping suspension systems are designed to withstand the dynamic effects produced by a turbine stop valve closure and by Safety Relief Valve (SRV) discharge[44].

Two of the four Main Steam lines, with their associated SRV discharge piping, were analyzed for their dynamic response to a turbine stop valve closure and SRV operation using the SAP IV Computer Program. The maximum stresses from either of these two dynamic analyses were combined with the seismic inertia stresses by SRSS and included in the standard piping load combinations as discussed in USAR Section C-3.3.1. These resultant stresses were then compared to the corresponding allowable stresses. The results of these analyses are documented in the calculations of record, and conclude that the pipe stresses are within the appropriate allowable stresses. Also, the analysis shows that excessive deflections, which may cause interference problems, will not occur. The effects on the SRV discharge lines during an SRV discharge transient is evaluated in the CNS Plant Unique Analysis Report (see USAR Section App.C-3.3.3.5.3). The turbine stop valve closure transient was not considered a critical load case for the SRV discharge lines in that analysis.

Selected portions of the MSIV leakage pathway to the condenser, as shown in Drawing CNS-MS-43, were analyzed for seismic ruggedness as part of the licensing of the LOCA dose calculations. Walkdowns and associated analyses evaluated the seismic capacity of the MSIV leakage pathway piping. The Main Steam System (including the bypass piping), the primary leakage pathway, and the alternate leakage pathway were selected for detailed computer analysis using the methods of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, 2001 Edition, Appendix N. Since the consideration of a Safe Shutdown Earthquake (SSE) was not in the original design basis for the piping systems under review, the capacity criteria given below was established for use in piping system limited analytical reviews and detailed analyses:

$$P + .75i[(M_A/Z)] < 1.0\ S$$

$$i(M_c/Z) < S_A + \{S-(P + .75i[M_A/Z])\}$$

$$P + .75i[(M_A/Z) + (M_{BI}/Z)] < 2.4\ S$$

$$i[(M_c/Z) + (M_{Esam}/Z)] < 2.5\ S_A$$

$$i[(2 \times M_{Esam})/Z)] < 2.5\ S_A$$

Where:

P = Pressure Loadings

M_A = Applied Moments due to Deadweight Loadings

M_{BI} = Applied Moments due to SSE Seismic Intertial Loadings

M_{Bsam} = ½ the range of Applied SSE Moments due to Seismic Anchor Motion Loadings

M_c = Range of Applied Moments due to Thermal Expansion and Thermal Anchor Motions

Z = Piping Section Modulus

S = Allowable Primary Stress limit per the B31.1 Code

S_A = Allowable Expansion Stress range per B31.1 Code

i = Stress intensification factor as defined in the B31.1 Code

For detailed dynamic analysis of piping in both the Turbine Building and the Reactor Building, the horizontal seismic demand is the 5% damped specific floor response spectra which have been calculated following the guidance in NUREG-0800 Sections 3.7.1 and 3.7.2, using a Regulatory Guide 1.60 spectral shape as the input ground response spectrum.

For static analysis, the methodology incorporated into Appendix N of ASME Section III was used. For localized evaluation of piping systems containing multiple changes of directions between lateral supports which provide the same direction of restraint, the piping can be projected into single beams in each of two orthogonal horizontal directions and the vertical direction. This approach called the Equivalent Static Load Method is used in conjunction with the methodology of NEDC-31858P and Regulatory Guide 1.61. For the CNS MSIV Leakage Pathway, the equivalent static seismic ruggedness evaluations of piping systems utilized a factor of 1.0 applied to the peak acceleration of the amplified floor response spectra.

3.3.3.5.2 SRV Discharge and Torus Attached Piping

As part of the Mark I Containment Program, dynamic analyses were performed to evaluate piping which would experience hydrodynamic loads associated with a postulated Loss of Coolant Accident (LOCA) event or a Safety Relief Valve (SRV) discharge. The design bases for the SRV Discharge and the torus attached piping are described in detail in the CNS Plant Unique Analysis Report. A summary level discussion is provided below. In addition, a summary of the Mark I containment program and the modifications performed is provided in USAR Section C-2.5.7.1.

Evaluations were performed for the SRV discharge piping (extending from the Main Steam connection through the SRVs to the T-quencher assemblies in the Suppression Chamber) and all piping systems which have an attachment point on the Suppression Chamber (torus) shell. Most of the torus attached piping systems have a segment both external to, and inside the Suppression Chamber. The piping external to the Suppression Chamber was modeled from the Suppression Chamber shell connection point to a point along the pipe beyond which the load effects from the Suppression Chamber excitation are no longer significant. Components included in these evaluations are the piping and piping supports, branch lines along the piping system, and all associated equipment. The piping inside the Suppression Chamber is modeled and analyzed separately, and generally consists of short partially submerged suction and discharge piping.

The torus attached piping systems were originally modeled and analyzed using the SUPERPIPE computer program developed by EDS Nuclear. The piping systems attached to the Suppression Chamber shell were analyzed for the hydrodynamically induced Suppression Chamber vibrations using the response spectra method. The response spectra representing the Suppression Chamber vibrations were developed with consideration of the dynamic coupling between the piping and the Suppression Chamber shell. Reanalysis for selected piping systems was performed using the ADLPIPE, ANSYS, and PISTAR computer programs.

The Mark I hydrodynamic stress results were combined with stresses due to the originally defined loads (dead weight, thermal and seismic) in accordance with the CNS Plant Unique Analysis Report. The resultant stresses were then compared to the allowable stresses for ASME Code Class 2 or Class 3 (depending on the system) piping systems per the applicable subsections of ASME Section III, S77. The piping components satisfy the applicable stress requirements of the Code. Fatigue evaluations due to thermal loading were performed to meet the applicable requirements of the Code. An augmented fatigue evaluation method for ASME Code Class 2/3 piping subjected to Mark I hydrodynamic loads was developed in MPR Report-751, titled, "Augmented Class 2/3 Fatigue Evaluation Method and Results for Typical Torus Attached on SRV Piping System," dated November 1982. This report was reviewed by the NRC and it was concluded that all CNS torus attached piping systems have a fatigue usage of less than 0.5 during the plant life.[61]

The original Mark I program methodology used for combining dynamic loads allowed up to two dynamic loads to be combined with a Square-Root-of-the-Sum-of-Squares (SRSS) methodology. The SRSS combination included an additional conservative increase factor of 1.1. The original methodology also required that if there were more than two independent dynamic loads, the additional loads had to be summed absolutely to the (1.1) SRSS combination. An alternate methodology for combining dynamic loads for torus attached piping (for pipe stress combinations, pipe support load combinations, Suppression Chamber penetration piping reactions, and equipment loads) which removes the 1.1 factor from the SRSS dynamic load combination and allows all dynamic loads to be combined via SRSS is used on subsequent analyses. This alternate methodology is acceptable since it meets the same criteria (i.e., meeting an 84% non-exceedance probability with more than a 90% confidence level) that was used to justify the original 1.1 SRSS method as described in Appendix D of the CNS Plant Unique Analysis Report. This combination method was submitted to the NRC by the BWR Owners Group and was reviewed and found acceptable by the NRC.[35,36]

3.3.3.5.3 Replacement Piping

Portions of the Reactor Recirculation, Core Spray, Reactor Water Cleanup, and Residual Heat Removal systems were replaced in 1985 with piping made of new material and, in some cases, configuration changes were made. Flexibility and seismic analyses of these systems have been performed which demonstrate compliance with the requirements of 1983 Edition of the ASME Boiler and Pressure Vessel Code. The load combinations and allowable stresses are provided in Impell Corp. Report No. 01-0840-1268[34].

3.3.3.5.4 Intake Structure Piping

In addition to the standard piping loads, piping systems attached to the Intake Structure are also subjected to a dynamic transient associated with a barge impact with the Intake Structure. A response spectrum for this transient has been developed for the operating floor of the Intake Structure. Piping attached to this structure is analyzed using this response spectrum. This load is considered similar in probability to an SSE. Therefore for piping subjected to this transient, the maximum stresses due to the inertia effects of an SSE or the barge impact load is substituted for SSE in the applicable load combinations (see USAR Section C-3.3.3.1). The allowable stress is the same as for load combinations involving SSE.

3.3.4 Equipment

The extent of stress analyses performed on equipment is dependent upon the type of equipment and the type of fabrication. Fabricated shapes are generally made from plate or rolled shapes with uniform thickness and shapes with regular geometric configurations. Cast shapes are generally made with non-uniform material thickness in complicated shapes that are not regular geometric configurations. Manufacturers have traditionally designed cast shapes conservatively since they do not lend themselves to rational analysis. Typically a design is developed based on extensive test and experience.

For the seismic design of the equipment and components, the analytical results from both the horizontal and vertical earthquake loadings are considered to act concurrently and in the most disadvantageous directions. The earthquake loads in the horizontal direction were obtained from the appropriate amplified floor response spectra plotted for the supporting structure. In lieu of determining the natural frequency of the equipment, the peak value of the applicable floor response spectrum was used in calculating the earthquake induced loads. Alternately, the natural frequency of the equipment was determined and corresponding input acceleration was obtained from the appropriate floor response spectra. Where equipment is represented by multi-degree of freedom systems, dynamic analyses which accounts for the contribution of all the significant modes of vibration are used.[27]

In the vertical direction, the buildings are considered to be very stiff as compared to the horizontal direction. The equipment and components (pumps, motors, tanks, valves, etc.) are also considered to be very stiff in the vertical direction. These factors tend to keep the vertical earthquake loads quite low. Since this equipment is designed to withstand vertical loads equal to its weight (lg) the effect of the vertical earthquake loads is, therefore, not as significant in the overall design of the equipment. The original vertical seismic acceleration component was specified as one-half of the horizontal ground acceleration. This design requirement was increased to two-thirds of the horizontal ground acceleration, which is the current Licensing/Design Basis.[28]

The original criteria, method of analysis and summary of critical stresses for various equipment are included in Table C-3-7, LOADING CRITERIA.

Dynamic testing is accepted as an alternate procedure to prove seismic adequacy of Class I equipment in cases where analytical procedures can not be used to evaluate equipment adequacy.[26] Use of the earthquake experience-based method, as described in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) was accepted for seismic qualification of certain electrical and mechanical equipment in the scope of the GIP.[68]

For new and replacement electrical equipment, either IEEE 344-1971 or IEEE 344-1975 "*IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations*" is acceptable for seismic qualification of Class 1E electrical equipment. CNS is designed to the IEEE 344-1971 standard, however the IEEE 344-1975 standard provides additional guidance and is commonly used in industry.

Depending on the design objective, the following two types of tests can be used in the dynamic testing of equipment.[26]

1) <u>Free Vibration Test</u>

This test is done on equipment whose response is dominated by the fundamental mode. The critical damping ratio and fundamental frequency can be determined from this test. In this test, an initial displacement or initial velocity is imparted to the equipment. The initial displacement is introduced by forcibly displacing the equipment and then suddenly releasing the force. The initial velocity is obtained by impinging the equipment with an impulse. Accelerometers or strain gauges are mounted on the equipment. Several readings are made first to assure that the equipment is vibrated in its primary mode. The critical damping ratio is then calculated by the standard formula for determining damping from logarithmic decrement.

2) <u>Forced Vibration Test</u>

The equipment is mounted on a shake table to which a prescribed acceleration is applied. By this test, both the critical damping ratio and the equipment's functional capability can be determined. The test can be performed also by means of eccentric shakers. In order to obtain the critical damping ratio of the equipment, a sinusoidal acceleration is applied by the shake table or eccentric shaker. The forced response curve (maximum amplitude vs. forcing frequency curve) of the equipment is obtained first. The critical damping ratio is then calculated by using the half-power method of fitting a theoretical forced response curve through the data points at and near resonance.

In order to prove the equipment's functional capability, a prescribed acceleration, which can be sinusoidal, sine beat, random or shock type is applied at the shake table. The vibratory motion used is such that vibratory loads are equal to or more than the seismic loads represented by the applicable floor spectra. All loads normally acting on the equipment are also simulated. When seismic loading on mechanical equipment are considered analytically, the resultant stresses are added to those from normal and accident conditions in appropriate loading combinations to assure that the equipment will function as specified.

When dynamic testing is used to verify the seismic adequacy of Class I mechanical equipment, the equipment is operated during and after the test to ensure operability.

When testing is used to supplement analysis, the equipment is not in the operating mode but, the dynamic loads computed are added to those from normal and accident conditions in appropriate loading combinations to assure that the equipment will function as specified[26].

The components satisfy the applicable requirements of the ASME and ANSI Codes.

Revision 3 of the Generic Implementation Procedure (GIP-3)[69], as modified and supplemented by the U.S. Nuclear Regulatory Commission Supplemental Safety Evaluation Report (SSER) No. 2[70] and SSER No. 3[71], may be used as an alternative to other authorized methods for the seismic design and verification of modified, new and replacement equipment classified as Class I.

Only those portions of GIP-3 listed below, which apply to the seismic design and verification of mechanical and electrical equipment, electrical relays, tanks and heat exchangers, and cable and conduit raceway systems shall be used. The other portions of the GIP are not applicable since they contain administrative, licensing, and documentation information which is applicable to the USI A-46 program.

Part I, Section 2.3.4, Future Modifications and New and Replacement Equipment.

Part II, Sections 2.1.2 and 2.4, Seismic Capability Engineers.

Part II, Section 4, Screening Verification and Walkdown.
The following differences shall be used when applying this section of GIP-3 for modified, new, and replacement equipment.

Use of Method A for determining the seismic demand is directly applicable to equipment mounted in those buildings and at those elevations which are at or below the locations where safe shutdown equipment was mounted and evaluated as a part of the USI A-46 program[72] and accepted by the NRC in its review of that program.[68] Method A may be used in other buildings and elevations, but with appropriate justification consistent with the NRC's acceptance of Method A for USI A-46 at CNS[68] and guidance provided to SQUG member utilities.[73]

For new anchorage installations and where previous bolt sizes and patterns are not used, the allowable anchorage capacities should be based on factors of safety specified in this Appendix (or, if not specified herein, as recommended by the anchorage manufacturer), instead of the GIP-3 factor of safety of 3.0.

Documentation of the results of the Screening Verification and Walkdown in Section 4.6 may be limited to the use of walkdown checklists. It is not necessary to complete Screening Verification Data Sheets (SVDSs).

Part II, Section 6.4, Comparison of Relay Seismic Capacity to Seismic Demand. It is not necessary to identify "essential relays" and perform functionality screening as defined in Section 6 of GIP-3. Only the seismic capacity compared to seismic demand evaluation in the GIP will be applied to relays designated as Class I.

Part II, Section 6.5, Relay Walkdown.

Part II, Section 7, Tanks and Heat Exchangers Review.
This section of GIP-3 may be used in its entirety for replacement of existing tanks and heat exchangers, as well as for the design and construction of new tanks and heat exchangers, except for new flat-bottom vertical tanks. For new flat-bottom vertical tanks, the following attributes, in addition to appropriate GIP-3 criteria shall be used:

The cast-in-place anchor bolts and associated hardware (chairs, transfer plates, etc.) in Subsection 7.3.3 shall be designed and installed in accordance with embedment depth, edge distance, anticipated concrete cracking, and corrosion allowance specified in GIP-3. The maximum strain in the anchor bolts shall not exceed that corresponding to the yield strength of the bolt material.

In Step 16 of Subsection 7.3.3.3, change the equation to:

$$\sigma_c = 0.6 \text{ [min. } (\sigma_{pe} , \sigma_{pd})] \text{ [psi]}$$

In Subsection 7.3.7, the tank foundation (ring-type or otherwise) shall be designed to resist uplifting and the overturning moment.

Part II, Section 8, Cable and Conduit Raceway Review.
The additional evaluations and alternate methods for resolving raceway outliers in Sections 8.4.1 through 8.4.8 may be used. However, the generic methods for resolving outliers in Part II, Section 5 shall not be used.

Part II, Section 10, References.

Appendix B, Summary of Equipment Class Descriptions and Caveats.

Appendix C, Generic Equipment Characteristics for Anchorage Evaluations.

Appendix D, Seismic Interaction.

Appendix G, Screening Evaluation Work Sheets (SEWS).

Notes:

(1) The GIP method should be used only for equipment located in mild environments and when concurrent vibratory loads, e.g., hydrodynamic loads from DBA, in combination with SSE loads are less than the GIP Reference Spectrum.

(2) The GIP method should not be used on equipment for which seismic qualifications have been imposed or committed to in connection with the resolution of other specific issues (e.g., Regulatory Guide 1.97, Three Mile Island (TMI) Action Item II.F.2, IPEEE) unless justified on a case-by-case basis.

Specific details of the criteria and method of analysis for various plant equipment is provided below:

Core Spray, RHR, HPCI, and RCIC Pumps and HPCI and RCIC Turbines

The Core Spray, RHR, HPCI and RCIC pumps and the HPCI and RCIC turbines are evaluated for the expected loading conditions. The structural design criteria and method of analysis for these pumps and turbines are described in Table C-2-7. The Plant Unique Analysis Report describes additional analyses performed on this equipment. The results of the analysis for these pumps and turbines show that the applicable acceptance criteria have been met.

Reactor Recirculation Pumps

The Reactor Recirculation pumps are evaluated for the expected loading conditions associated with both normal operating and seismic conditions. The structural design criteria and method of analysis for the key components of these pumps are described in Table C-2-7. The results of the analysis for the Reactor Recirculation pumps show that the applicable acceptance criteria have been met.

Standby Liquid Control Tank and Pumps

The Standby Liquid Control tank and pumps are evaluated for the expected loading conditions. The structural design criteria and method of analysis for the key components of these pumps are described in Table C-2-7. The results of the analysis for the Standby Liquid Control Pumps show that the applicable acceptance criteria have been met.

Main Steam Isolation Valves, Safety/Relief Valves, and Reactor Recirculation Valves

The Main Steam Isolation Valves, Safety/Relief Valves, and the Reactor Recirculation Valves are evaluated for the expected loading conditions associated with both normal operating and seismic conditions. The results of the analysis for these valves show that the applicable acceptance criteria have been met.

4.0 REFERENCES FOR APPENDIX C

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9. Deleted.

10. Deleted.

11. Deleted.

12. Q/A 12.17; Amendment 9.

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14. Q/A 12.18; Amendment 9.

15. Q/A 12.19; Amendment 9.

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18. Q/A 12.50; Amendment 13.

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20. Q/A 12.59; Amendment 13.

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35. "Mark I Containment Program Cumulative Distribution Functions for Typical Dynamic Responses of a Mark I Torus and Attached Piping Systems," NEDE-24632, Revision 0, December 1980.

36. "Safety Evaluation by the Office of Nuclear Reactor Regulation for Acceptability of the SRSS Method for Combining Dynamic Responses in Mark I Piping Systems," forwarded by NRC letter D. Vassallo (NRC)/H. Pfefferlen (GE) dated March 10, 1983.

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