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APPENDIX C - STRUCTURAL DESIGN CRITERIA

C.1 CLASSIFICATION OF STRUCTURES

C.1.1 General

Certain station structures must remain functional and/or protect vital equipment and systems, both during and following the most severe natural phenomenon which is postulated to occur at the site. In order to establish the loadings and loading combinations for which each individual structure is to be designed, buildings and their structural systems are separated into the following two seismic classes with respect to a seismic design requirements.

Seismic Class I - Seismic Class I structures and equipment are those whose failure could increase the severity of the design basis accident, and cause release of radioactivity in excess of 10 CFR 50.67 limits, or those essential for safe shutdown and removal of decay heat following a LOCA.

Seismic Class II - Seismic Class II structures and equipment are those whose failure would not result in the release of significant radioactivity and would not prevent reactor shutdown. The failure of seismic Class II structures may interrupt power generation.

A structure designated seismic Class II shall not degrade the integrity of any structure designated seismic Class I. Although a structure, as a whole, may be seismic Class I, less essential portions may be considered seismic Class II if they are not associated with loss of function, and their failure does not render the seismic Class I portions inoperable.

Seismic Class II structures are structurally separated from seismic Class I structures by means of expansion joints to provide for unequal deflections associated with independent movements of the structures. The arrangement is such that in the unlikely event that a Class II structure should collapse, it would not impair the safety function of the Class I structure.

The criteria for the relative movements under maximum earthquake loadings require that the clearance provided exceeds the combined movements. The relative movements under these loadings are accommodated by expansion joints at adjoining structures and by built-in flexibility for piping systems. A dynamic analysis has shown that the cumulative maximum displacements of adjoining concrete structures will be about one-half of the clearance provided.

In the case of structures defined as partially Class I and partially Class II rigidly interconnected, the Class I portion is checked to assure it can carry any loads that may be transmitted from the connected Class II structure.

The following list itemizes the structures, equipment, and process systems which fall under the two seismic classes defined above.

C.1.2 Seismic Class I Structures and Systems

Class I Structures

- Drywell, vents, torus, and penetrations
- Reactor building
- Spent fuel pool
- Reactor vessel support pedestal
- Main control room complex (including cable spreading room, emergency switchgear rooms, and battery rooms)
- Radwaste building
- Diesel generator building
- Pump structure (portion containing critical service water pumps)
- Emergency heat sink facility, including cooling tower
- Stack
- Structures required to protect seismic Class I equipment
- Post-LOCA CADS liquid N_2 tank building
- Recombiner building

Class I Equipment and Systems

Nuclear steam supply systems:
- Reactor vessel and internals, including:
  - CRD housing
  - CRD guide tube
  - CRD
  - CRD cap screw
  - Control rod
  - CRD thermal sleeve and key
  - In-core housing
  - Feedwater sparger
  - Jet pump adapter
  - Shroud
  - Top guide
  - Core support
  - Core support and top guide aligner
  - Core plate stud
  - Jet pump riser brace
  - Jet pump assembly
  - Jet pump instrument penetration seal
  - Differential pressure and liquid control line
  - Core spray line and clamp

Head cooling spray nozzle (for Unit 2 only)
Dry tube
Power range monitor installation hardware
Power range detector
Orificed fuel support
Fuel channel
Fuel assembly
Reactor vessel supports and stabilizers
Control rod drive system (equipment required for scram operation)
Control rod drive housing supports
Recirculating piping, including valves and pumps
Main steam piping out to second isolation valve
Nuclear boiler system safety valves
Nuclear boiler system relief valves
Piping connections from the reactor vessel, up to and including the first isolation valve external to the drywell
Core standby cooling systems (CSCS)
Standby liquid control system (except for the test tank and test connections)
High pressure service water system
Emergency service water system
Standby gas treatment system
Fuel storage facilities, to include spent fuel and new fuel storage racks
Reactor building crane
Circulating water pump structure crane (designed to Seismic Class I requirements)
Standby power systems:
Station batteries (except balance-of-plant battery and 24 volt neutron monitoring batteries)
Standby diesel generators
Emergency buses and other electrical gear for onsite power supply to engineered safeguards and nuclear safety systems
Instrumentation and controls:
Reactor level instrumentation
Reactor manual control system
Control rod instrumentation (portions)
Post-LOCA CADS

C.1.3 Seismic Class II Structures and Systems

Class II Structures

Turbine building
Shop and warehouse
Administration building

Water treatment building
Pump structure, except for portion affecting critical service water systems
Intake screen structure
Cooling towers and cooling tower pump structures for circulating water
Off-gas filter station
Auxiliary boiler house
Guardhouse
Outdoor electrical switchgear structures
Sewage treatment plant
Radwaste onsite storage facility
Recirc ASD Structure

Class II Equipment and Systems

Turbine-generator system and transformers
Condensers
Turbine building crane
Feedwater heaters and pumps
Condensate storage tanks and pumps
Refueling water storage tank
Station auxiliary power buses
Offsite AC power system
Radwaste systems
Reactor water cleanup system
Condensate filter-demineralizer system
Compressed air system
Reactor building cooling water system
Turbine building cooling water system
Instrument N_2 system
All other piping and equipment not listed under seismic Class I
24 volt neutron monitoring batteries
Feedwater zinc injection system
Hydrogen Water Chemistry System

C.2 STRUCTURAL DESIGN BASIS

Structures are designed for dead loads, live loads, seismic loads, and wind loads in accordance with applicable codes and as described in the following paragraphs. The loading conditions, and combinations thereof, are determined by the function of the structure and its importance in meeting the station safety and power generation objectives.

C.2.1 Dead and Live Loads

The structures in the power plant complex are designed for the dead loads and live loads to which the structures will be subjected. Roofs of all the structures are designed for a snow load of 30 psf.

C.2.2 Seismic Loads

The design of seismic Class I structures is based on a dynamic analysis using the spectrum response curves developed for the site. The design of seismic Class I equipment is based on a dynamic analysis using either acceleration spectrum response curves or acceleration time histories developed at points of attachment, the method of analysis being dependent on the nature of the equipment.

The list of Class I (seismic design) structures, equipment, and systems is presented in paragraph C.1.2. All structures listed in this table as Class I structures were seismically analyzed by the response spectra method, except the portion of the pump structure containing critical service water pumps was seismically analyzed by the time-history method.

The structures are analyzed for the following magnitudes of ground acceleration:

a. Design earthquake considers a maximum horizontal ground acceleration of 0.05g. Under this condition, stresses due to the earthquake combined with stresses due to other operational loadings are limited to the working stress levels of the materials used in the structures except as noted in paragraph C.2.6.3. The customary increase in normal allowable working stress due to earthquake is not used.

b. Maximum credible earthquake (MCE) considers a horizontal ground acceleration of 0.12g. Under this condition, stresses due to the earthquake combined with stresses due to other operational loadings are allowed to approach the yield strength of the materials and are limited to 90 percent of yield stress (fy) for the steel and 85 percent of ultimate compressive stress (f'c) for the concrete. In addition, all items required for safe shutdown will not lose their function.

Proof of design adequacy is accomplished by showing the criteria stated for steel and concrete for the MCE condition are not exceeded and thus the structures comply with the definition of seismic Class I in paragraph C.1.1. Structural deformations and deflections calculated are well within the linear-elastic range and cause only low stresses.

c. Vertical ground accelerations associated with the design earthquake and MCE are 67 percent of the corresponding horizontal acceleration spectrums; namely, 0.033g for the design earthquake and 0.08g for the MCE.

Table C.2.1 shows the damping factors which are used for excitations associated with the design earthquake and the MCE.

Vertical seismic stresses are not severe because they represent only a fractional increase in the dead load which the structure carries. Since the frequencies of the modes associated with vertical motion are normally large, it is sufficient to design the vertical elements for the maximum vertical ground acceleration without a detailed dynamic analysis of the structure.

The reactor building is nearly symmetrical about both perpendicular axes. The lack of symmetry is not sufficient to significantly alter stresses and may be safely ignored. However, to account for so-called "accidental" torsion, after evaluating the worst cases an arbitrary conservative allowance of 20 percent was made on all forces.

Parametric studies were carried out to determine the relative influence of the numerous variables involved which verified the adequacy of the assumption.

The vertical seismic response can be divided into two categories. The first category is the general building motion involving primarily the column or wall elements and the second category is the local response of various beam and slab elements oriented parallel to the ground.

In general, for a building founded on a rigid foundation the building response will be small compared to the dead load since the building frequencies will be higher than the primary frequencies of the earthquake spectrum.

The beams, slab, equipment, and systems may respond differently than the overall building since their frequencies may correspond to the primary frequencies of the earthquake spectrum.

All Class I equipment and structural elements including columns, walls, beams, and slabs are analyzed and designed to resist the vertical seismic forces together with any other loads as defined in the design criteria. Beams and floors are analyzed to determine their maximum response and frequency. The equipment and systems are designed to resist any amplified beam and floor accelerations.

The seismic Class II radwaste on-site storage facility structure is designed for seismic loadings corresponding to the maximum ground acceleration of 0.05g selected for the Operating Basis Earthquake (OBE). A model analysis using a lumped mass model of the facility was performed using the criteria and methodology described in USNRC Regulatory Guide 1.143. American Concrete Institute standard ACI 318-77, "Building Code Requirements for Reinforced Concrete" was used in the design of the concrete structures. For steel structure design, American Institute of Steel Construction "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," November 1978 was used. The one-third allowable stress increase was included for steel structures for load combinations involving earthquakes or wind loads. The building foundation is discussed in UFSAR section 2.7.6.4.

Analysis of other seismic Class II structures is based on the design criteria established for the structures in Zone I of the seismic zones as defined by the Uniform Building Code, 1967 Edition.

Class II structures, such as the turbine building, which adjoin Class I structures are arranged and designed in such a way that the possible failure of the Class II building will not endanger the function of any Class I building or system.

Additionally, in the case of the 1997 re-analysis of the Recirculation system piping and the Residual Heat Removal and Reactor Water Clean-up piping inside primary containment for Peach Bottom NCR 97-02267, the seismic analysis was based on NRC Regulatory Guide 1.60 (Design Response Spectra for Seismic Design of Nuclear Power Plants) with modal combination and spatial components in accordance with Reg. Guide 1.92 (Combining Modal Responses and Spatial Components in Seismic Response Analysis). These regulatory guides were used because they are required by Regulatory Guide 1.84 when using ANSI Code Case N-411-1

(Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping Section III, Division 1).

C.2.3 Wind Loads

The methods used in determining the wind pressures for the radwaste on-site storage facility are in accordance with ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and other Structures". The storage facility structures are designed to withstand a maximum windspeed of 90 miles per hour. The wind is assumed to occur 30 feet above ground and has a 100-year mean recurrance interval.

The wind loads used in the design of other portions of this plant are derived from Paper 3269, entitled "Wind Forces on Structures," published by the American Society of Civil Engineers, Transactions, Volume 126, Part II, 1961, as applied to the Peach Bottom site. The total wind pressures, listed in Table C.2.2, include positive and negative pressures and gust factors.

C.2.4 Tornado Loads

Tornado winds traversing the site could damage the reactor building superstructure, turbine building, condensate storage tanks, stack, and incoming power lines. However, the ability to shut down the reactor, the integrity of primary containment, and the capability of essential heat removal systems would not be impaired.

Components which directly affect the ultimate safe shutdown of the plant are located either in reinforced concrete structures or underground for tornado protection. These components include the following:

Reactor primary system

CRD hydraulic equipment, excluding feed pumps

Standby gas treatment system

Standby liquid control system

Primary containment and isolation valves

HPCIS

RCICS

RHRS

Emergency service water system

High pressure service water system

Station batteries

Standby diesel-generators and associated switchgear

Controls and instrumentation for above systems

Main control room complex

Intake structure (portions essential to systems listed above)

Where failure could affect the operation and function of the primary containment, the reactor primary system, or other safeguards equipment, the following tornado effects are considered in the design of these structures:

1. External wind forces resulting from a tornado having a horizontal peripheral tangential velocity of 300 mph maximum, which includes the tangential and translational components.

2. Differential pressure of 3 psi between inside and outside of fully enclosed areas. Blowout panels are included where necessary in the design of the structure to limit pressure differentials.

3. Missiles equivalent to a 4 in thick x 12 in wide x 12 ft long wood plank traveling end-on at 300 mph; or a 4,000-lb passenger auto flying through the air at 50 mph, at not more than 25 ft above ground, with a contact area of 20 sq ft.

4. A torsional moment resulting from applying the wind specified in item 1 acting on one-half the length of a building.

Walls of all open compartments were designed to withstand the differential pressure which occurs during the tornado depressurization. Blowout panels are provided to relieve excess positive pressure in all essential parts of the structure.

Building structures housing safeguards equipment are designed to withstand a tornado-induced depressurization rate of 1 psi/sec for 3 sec. To accomplish this objective, all compartments that are essentially leaktight are checked to verify that they are capable of withstanding a differential pressure of 3 psi.

Seismic Class I equipment and/or structures either protected by a tornado resistant structure, or whose loss of function during a tornado would not violate the safety requirements of the plant, are not designed against tornado effects.

The structural steel frame of the reactor building upper superstructure is designed to withstand the force of a 300-mph wind without exceeding the yield stress. The reactor building siding and roof decking, however, is designed for the normal wind loading. When this design wind loading is appreciably exceeded, portions of the siding and decking are expected to be lost. Connectors for the siding are designed to fail at stress levels associated with tornado loading to assure that the siding will blow away. However, to ensure an adequate load carrying capacity of the structural members, the individual members were designed to take the full load of the tornado if the siding directly affecting that member remained intact. However, the reinforced concrete structure of the reactor building protects the equipment necessary for the safe shutdown of the reactor, the primary containment, and the essential heat removal equipment from the effects of a tornado. Tornado effects on the spent fuel pool are discussed in General Electric Topical Report APED-5696. On the sides, the fuel pool is protected against low trajectory missiles by thick concrete walls between the turbine and the pool.

C.2.5 Special Loadings

The structures housing critical equipment required for safe shutdown of the plant are designed for special loadings.

C.2.5.1 Turbine Missiles

The turbine missile probability will be maintained to less than 1×10^{-5} per year, and the probability of damaging a critical target will be maintained less than 1×10^{-7}. This is consistent with Sections 3.5.1.3 and 2.2.3 of the Standard Review Plan. Section 11.2.4 includes the basis for determining probabilities and the inspection program that has been instituted to maintain the probability of turbine missile generation within acceptable limits.

Missiles from the RCIC turbine were also investigated to assure that they would not damage any critical piping in the vicinity of the turbine. The possibility of this type of missile is very remote.

C.2.5.2 Tornado-Generated Missiles

Tornado-generated missiles are discussed in paragraph C.2.4. The concrete shield plug above the drywell is capable of resisting missiles generated in a tornado. The large equipment openings of the diesel generator building have missile-proof doors. The personnel access doors are shielded from such missiles by baffle walls. This concept is used throughout the project to protect large openings against effects of tornado winds, depressurization, and tornado-generated missiles.

C.2.5.3 Temperature Loads

For each seismic Class I structure, temperature loads considered to be significant were included in the design. For example, the biological shield was designed for the normal operating loads listed in Table C.4.5; the reactor pressure vessel pedestal was designed for the loading conditions listed in Table C.4.4; the primary containment shell was designed for the accident conditions listed in Tables M.3.5 and M.3.6; the fuel pool walls were originally designed for normal allowable stresses. A check under loss-of-fuel pool coolant (i.e., boiling water) indicated that stresses would be still below normal allowable limits. Refer to Section C.2.6.3 for reevaluation of the spent fuel pool.

Higher temperatures than LOCA condition were not considered for other than process equipment normally encountering higher temperatures; however, the stress levels are sufficiently low to be able to tolerate a short duration increase in temperature to 305°F and still be within the allowable limits.

Transient stresses do not significantly affect concrete stresses. However, transients were considered at the point of embedment of the drywell shell. The design basis temperature was initially 281°F during a LOCA. The current bounding drywell temperature, however, occurs during a break of a steam line. A spectrum of steam line break sizes have been evaluated to ensure a bounding drywell temperature profile is established. The most limiting drywell temperature from this analysis is 340°F. Although the drywell environment may see temperatures as high as 340°F for 20 minutes, the most limiting temperature for the drywell shell has been analyzed to be within the design temperature of 281°F (Reference 24).

C.2.5.4 Flood Loads and Flood Protection

Structures required for safe shutdown of Units 2 and 3 in the event of the probable maximum flood (PMF), (causing an estimated wave runup to Elevation 136.9 ft (C.D.) assuming no accident occurs concurrently, are:

Reactor building
Main control room complex

Diesel generator building
Pump structure (portion containing critical service water pumps)
Emergency heat sink facility, including cooling tower

Components required for safe shutdown of Units 2 and 3 are:

Reactor vessel and internals
CRDS (portion essential for scram)
Recirculation piping system
RCICS
RHRS
High pressure service water system
Emergency cooling system
Emergency service water system
Standby power systems
Instrumentation and controls:
- Reactor level instrumentation
- Reactor pressure instrumentation

For description of wave runup superimposed on the PMF refer to subsection 2.4.

For drawings of structures and components listed above see Figures 12.1.1, 12.2.1, and Drawings C-84, and M-2 through M-7. The emergency heat sink structure is shown in Figure C.2.1.

Watertight doors are provided at all structures; waterproofing is installed to Elevation 135.0 ft (C.D.) and any penetration in the exterior walls is sealed to ensure leaktightness necessary to plant safety.

The integrity of the waterproofing on the external surfaces of vertical walls below grade cannot be checked since such surfaces are inaccessible. Accessible joints are visually inspected and caulked as required on a periodic basis as part of regular plant maintenance.

Plastic waterstops are used at all construction joints to maintain the integrity of joints. Penetrations and conduits in exterior walls are sealed with approved, pre-tested seal details and material which assure leaktightness against ground or flood water. Penetration seals are installed in accordance with approved specifications and procedures and are inspected to assure proper installation.

C.2.6 Loading Combinations

The following paragraphs describe the loading combinations used for the design of the seismic Class I structures. Loads and loading combinations for Class II structures are in accordance with the Uniform Building Code and normal design practice for power plants. Loading combinations used for the design of the primary containment are discussed in Appendix M.

D = Dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads, operating pressures, and live loads expected to be present when the plant is operating. 50 psf is considered normal operating live load.

W = Design wind loading conditions.

W' = Loads due to tornado.

R = Jet force on structure due to rupture of any one pipe.

H = Force on structure due to thermal expansion of pipes under operating conditions. The effect of this loading was considered on individual members where required.

E = Design earthquake load.

E' = MCE load.

T = Temperature load.

F = Flood loading (flood level at Elevation 135 ft 0 in).

For Class I structures, code allowable stress values are modified since structures of this class must sustain much more severe loads and be more accurately proportioned than structures normally considered under building codes. However, the same codes will still furnish guidance.

The criteria for seismic Class I structures with respect to stress levels and load combinations for the postulated events are noted in the following paragraph.

C.2.6.1 Reactor Building and All Other Seismic Class I Structures

1. D+E Normal allowable code stresses (AISC for structural steel, ACI for reinforced concrete). The customary increase in normal design stresses,

when earthquake loads are considered, is not permitted.

2. D+E' — Maximum allowable stresses are as follows:
Steel - 0.9 Fy (yield strength of steel);
Concrete - 0.85 f'c (compressive strength of concrete);
Reinforcement - 0.9 fy (yield strength of reinforcement).

3. D+W — Maximum allowable working stresses may be increased one-third above normal code allowable stresses.

4. D+W' — Maximum allowable stresses are as follows:
Steel - 0.9 Fy;
Concrete - 0.85 f'c;
Reinforcement - 0.9 fy.

5. D+E+T — Normal allowable code stresses. The customary increase in normal design stresses when earthquake is considered is not permitted.

6. D+E'+T — Maximum allowable stresses are as follows:
Steel - 0.9 Fy;
Concrete - 0.85 f'c;
Reinforcement - 0.9 fy.

7. D+F — Maximum allowable stresses are as follows:
Steel - 0.9 Fy;
Concrete - 0.85 f'c;
Reinforcement - 0.9 fy.

C.2.6.2 Reactor Vessel Pedestal

1. D+T+E — Normal allowable code stresses (AISC for structural steel, ACI for reinforced concrete). The customary increase in normal design stresses, when earthquake loads are considered, is not permitted.

2. D+T+R — Maximum allowable stresses are as

follows:
Steel - 0.9 Fy;
Concrete - 0.85 f'c;
Reinforcement - 0.9 fy.

3. D+T+E' Maximum allowable stresses are as follows:
Steel - 0.9 Fy;
Concrete - 0.85 f'c;
Reinforcement - 0.9 fy.

C.2.6.3 Spent Fuel Pool

The spent fuel pool has been reevaluated structurally for additional loading due to a loaded spent fuel storage cask, the higher capacity control rod blade racks, the high density fuel racks and increased number of fuel elements. This reevaluation was performed in accordance with the applicable codes and standards identified in Section C.2.7.1.

All loading combinations required by USNRC Regulatory Guide 1.142, USNRC Standard Review Plan 3.8.4, ACI and AISC were evaluated. The number of combinations to be analyzed were reduced by eliminating combinations governed by others. Final governing equations for the spent fuel pool structure are shown below for concrete structures using strength design methods and for structural steel using plastic design methods.

Load Combinations

Reinforced Concrete

1. $U = 1.4D + 1.4F + 1.7T_0$
2. $U = 1.4D + 1.4F$
3. $U = 1.4D + 1.4F + 1.7L + 1.9E$
4. $U = D + F + L + E' + T_a$
5. $U = D + F + L + E'$
6. $U = 1.05D + 1.05F + 1.3L + 1.43E + 1.3T_0$

Structural Steel

7. $Y = 1.7D + 1.7F + 1.7L + 1.7E$
8. $Y = 1.3D + 1.3F + 1.3L + 1.3E + 1.3T_0$
9. $Y = 1.1 (D + F + L + E' + T_a)$

Where: L = Live Load
T_0 = Operating Temperature
T_a = Accident Temperature

Loading Assumptions:

The dead load includes the weight of the spent fuel racks, stored fuel, spent fuel pool, and the contributing weight of the adjacent floor slabs, roof, and walls.

The live load includes the roof snow load, the distributed live loads on the adjacent floor slabs, crane loads and a buoyant weight of a loaded spent fuel storage cask.

Hydrostatic loads consist of vertical and lateral water pressures exerted on the spent fuel pool slab and walls, respectively.

Thermal loads are based on the pool water temperatures resulting from a full core discharge under normal operating conditions, and saturation temperatures for accident conditions. In all cases, a conservative Reactor Building indoor ambient temperature of 68°F is used. A stress free temperature of 70°F is used.

C.2.6.4 Reactor Building 135' with Spent Fuel Storage Cask

The 135' elevation of the reactor building, at the base of the crane hatch, has been reevaluated structurally for a loaded spent fuel storage cask and the cask transporter, in various configurations. The concrete slab was evaluated using ultimate strength design methods, using the load combinations of section C.2.6.3, above. The structural steel was evaluated using allowable stress design methods, using the load combinations listed in Section C.2.6.1.

C.2.7 Governing Codes and Regulations

The design of all structures and facilities conforms to the applicable general codes or specifications listed below except where specifically stated otherwise; for example items 2 and 3.

Each structure was analyzed by methods appropriate for its configuration; this furnished a measure of the stresses the structure would experience under the postulated conditions. Referenced codes were used as guides to establish reasonable allowable stresses.

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

2. American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," Sixth Edition and Ninth Edition (See Note 1).

3. American Concrete Institute (ACI), "Building Code Requirements for Reinforced Concrete," (ACI 318-63) "Code

Requirements for Nuclear Safety Related Concrete Structures (ACI 349-01)” (See Note 2), and "Code Requirements for Reinforced Concrete Chimneys," ACI 307 (1969).

4. American Welding Society (AWS), "Standard Code for Arc and Gas Welding in Building Construction," (AWS-D.1.0).

5. American Petroleum Institute (API), "Specification No. 650 for Welded Steel Storage Tanks."

6. ASME Boiler and Pressure Vessel Code, "Section III, Class B (governs the design and fabrication of the drywell and suppression chamber), 1965 Edition, with applicable addenda published to April, 1967.

7. U.S. Army Corp of Engineers (Regulations with respect to dredging and construction).

8. American Society of Civil Engineers Paper No. 3269, "Wind Forces of Structures."

9. American Iron and Steel Institute (AISI), "Specification for the Design of Light Gage Cold-formed Steel Structural Members."

10. Commonwealth of Pennsylvania Department of Labor and Industry "Building Regulations for Fire and Panic."

11. Electric Power Research Institute (EPRI) "Visual Weld Acceptance Criteria," EPRI Report No. NP-5380 Volume 1: Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants (NCIG-01, Revision 2), September 1987

See Notes 1 and 2 below.

C.2.7.1 Spent Fuel Pool Reevaluation

The spent fuel pool has been evaluated structurally for additional loading due to a loaded spent fuel storage cask, the higher capacity control rod blade racks, the increased number of fuel elements and high density fuel storage racks in accordance with the following codes and standards:

1. American Concrete Institute (ACI), "Building Code Requirements for Reinforced Concrete," (ACI 318-83) and "Code Requirements for Nuclear Safety Related Structures," (ACI 349-80)

2. American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," 1978

3. U.S. Nuclear Regulatory Commission, "Standard Review Plan 3.8.4, 'Other Seismic Category I Structures,'" Revision 1, NUREG-0800, July 1981

4. U.S. Nuclear Regulatory Commission, letter from B.R. Grimes to All Power Reactor Licensees, April 14, 1978, with enclosure entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," including Supplement, dated January 18, 1979

Note 1: AISC 9th Edition may be used for evaluations that are not addressed in the 6th Edition.

Note 2: NRC Regulatory Guide 1.199 approves the use of ACI 349-01, Appendix B for concrete anchorage evaluations.

TABLE C.2.1

DAMPING FACTORS

Earthquake	Percent of Critical Damping: Design Earthquake	Percent of Critical Damping: Maximum Credible Earthquake
Reinforced concrete structures	2.0	5.0
Steel framed structures	2.0	5.0
Welded steel assemblies	1.0	2.0
Bolted and riveted assemblies	2.0	5.0
Seismic Class I piping systems *	0.5	0.5 1.0 (Unit 3 only)

* 1997 Re-analysis of the Recirculation system piping, and the Residual Heat Removal and Reactor Water Clean-up piping inside primary containment for Peach Bottom NCR 97-02267 and ASME Code Case N-411-1 as shown below:

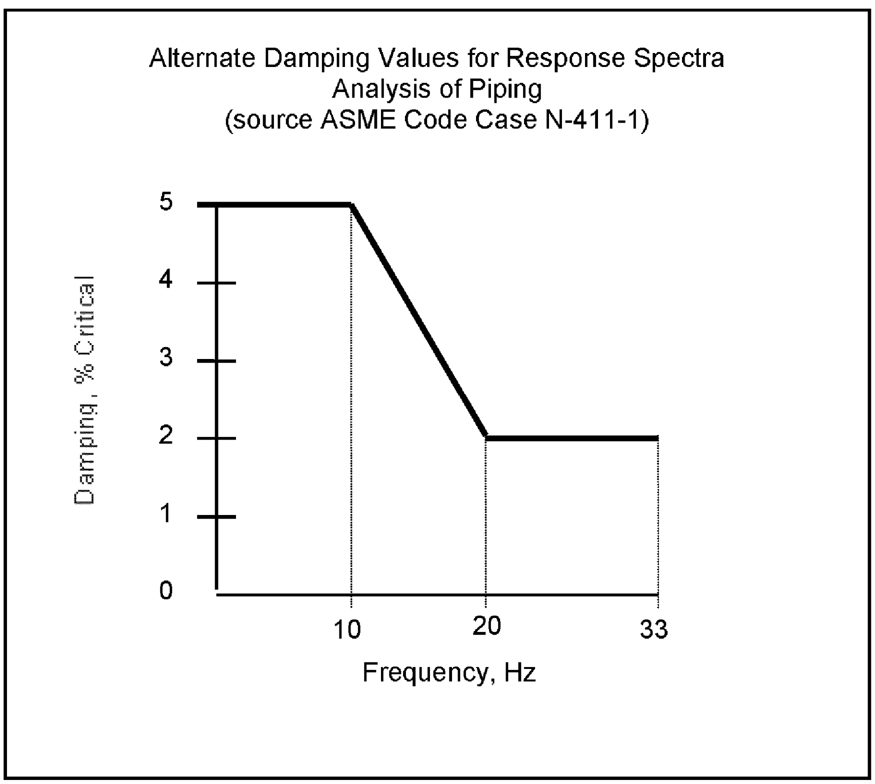

TABLE C.2.2

WIND LOADS

Height-Feet	Pressure (q) - (psf) Class I Structures - 100-Yr Recurrence	Class II Structures- 50-Yr Recurrence
0-50	25	20
50-150	35	25
150-400	45	30
Over 400	55	40

C.3 ANALYSIS OF CLASS I STRUCTURES

C.3.1 Scope

The loads, loading combinations, and allowable limits described in this appendix apply only to seismic Class I structures and components. The criteria in this appendix are intended to supplement applicable industry design codes where necessary to provide design safety margins which are appropriate to extremely reliable structures and components when account is taken of rare events associated with an MCE or postulated LOCA or a combination thereof.

Seismic Class I components are not always designed by application of the criteria using analytical techniques. Rather, the design of some components may be based upon test results, empirical evidence, or by comparison with similar items.

The seismic Class I concrete and steel structures are designed considering three inter-related primary functions for the design loading combinations described in paragraph C.2.6. 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 seismic Class I structures and components. Design code allowable stresses appropriate for the elastic design techniques were used as a guide for all stress limitations under normal design conditions. Higher stresses approaching yield for steel and ultimate for concrete were permitted under the MCE and similar conditions and as noted in paragraph C.2.6.3. Typical stresses under various conditions have been tabulated in Tables C.4.1 through C.4.5, and when these are compared with ultimate strengths, safety factors are readily apparent. The second consideration is to maintain structural deformations within such limits that seismic Class I components and/or systems will not experience a loss of function. Deformations experienced by structures under the loss of function criteria were checked and found, by elastic analysis, to be of such a small magnitude as to assure the structure would function as required. Typical deformations of the reactor building are shown in Figures C.3.8 and C.3.9. The third consideration is to limit excessive containment leakage by preventing excessive deformation and cracking where containment integrity is required.

Structural design and construction were performed in such a way as to prevent concrete cracking insofar as possible by mix design, pour limitation, and curing precautions. The stress limits in the code should result in very limited cracking on the order of a few hundredths of an inch. Such cracking would not significantly affect the leak resistance of the structure.

C.3.2 Structural Analysis

In general, the structural analysis is performed utilizing the "Working Stress Design" method as defined in "ACI Standard Building Code Requirements for Reinforced Concrete" (ACI 318-63), and in the AISC Manual of Steel Construction (Sixth Edition). "Finite element stress analysis" and other techniques are also used where applicable or necessary.

Load combinations and allowable limits on stresses are as shown in paragraph C.2.6. The maximum permissible calculated concrete compression is limited to 0.85 f'c (design compressive strength of concrete) and the maximum permissible calculated concrete shear is as given in ACI 318-63, Chapter 17, for loading involving R and E'.

Concrete structures designed for no loss of function criteria have been proportioned so as not to exceed 0.9 fy tension in the reinforcing steel and 0.85 f'c compression in the concrete. For bending, stresses have been determined on a straight line stress distribution assumption. This yields maximum allowable moments less than the ultimate strength moment as calculated by ACI-318-63 Code Section 1601. For bending every section is "under reinforced" so that the reinforcing steel reaches its allowable stress before the concrete, thus assuring ductility and reserve strength against structural collapse.

For both reinforcing steel and concrete the design criteria is: normal allowable stresses were not increased when considering operating loads with design earthquake loads. (Spent fuel pool reevaluation used ultimate strength method for the design earthquake – see C.2.6.3.) No loss of function criterion as listed in paragraph C.2.2 was used for MCE, tornado loads, flood loads, or pipe rupture jet loads when combined with normal loads.

Bond and anchorage for reinforcing steel is treated as required by ACI 318-63.

There are no loading conditions such as pressure which would cause net tension across a section resulting in biaxial and triaxial tension when combined with other loads, and thus reduce the shear strength, bond, and anchorage strength of reinforcing bars. However, there are loading conditions which produce biaxial stresses on certain members, similar to that experienced by a two-way slab. This condition is covered by ACI code allowable stresses which were used in the design except for no loss of function criteria loadings. For these criteria, reinforcing bar lap lengths and anchorage lengths that were used to develop the bars for their maximum code allowable stress are adequate to develop the higher stresses produced.

The allowable shear stresses for the no loss of function criteria are presented in Tables C.4.1, C.4.2, and C.4.4

Structural steel members designed for failure criteria have been proportioned so as not to exceed 0.9 Fy in bending and tension, 0.5 Fy in shear, and 1.5 F_a as defined in the AISC-63 code, subsection 1.5.1. Thus, the minimum factors of safety become 1.11 for bending and tension, 1.15 for shear, and from 1.11 to 1.28 for axial compression.

C.3.3 Seismic Analysis of Structures

The method used in the seismic analysis consists of the following four steps:

1. Formulation of the mathematical model of the structure or structures to be analyzed.

2. Determination of natural frequencies and mode shapes.

3. Finding the acceleration (g) levels from the response spectra curves.

4. Determination of the response of the structure to the earthquake in terms of moments, shears, and displacements.

The mathematical model of the structure consists of lumped masses and stiffness coefficients. At appropriate locations within the building, points are chosen to lump the weights of the structure. Between these locations, properties are calculated for moments of inertia, cross-sectional areas, and effective shear areas. The properties of the model are utilized in a computer program, applying unit loads at the mass points to obtain the flexibility coefficients of the building.

The natural frequencies and mode shapes of the structures are obtained by a Bechtel computer program, CE617. The program utilizes the flexibility coefficients and lumped weights of the modes. The flexibility coefficients are formulated into a matrix and inverted to form a stiffness matrix. The program then uses the technique of diagonalization by successive rotations to obtain the natural frequencies and mode shapes. Appropriate damping values of individual materials are presented in Table C.2.1.

The basic description of the earthquake is provided by spectrum response curves. Separate curves are used for the design earthquake of 0.05g horizontal acceleration and the MCE of 0.12g horizontal acceleration. These curves are presented in Figures

C.3.1 and C.3.2. Additionally, 1997 re-analysis of the Recirculation system piping, and the Residual Heat Removal and Reactor Water Clean-up piping inside primary containment for Peach Bottom NCR 97-02267 used Figure C.3.1a and C.3.2a as required by NRC Regulatory Guide 1.60 (Design Response Spectra for Seismic Design of Nuclear Power Plants). This regulatory guide was used because it is required by Regulatory Guide 1.84 when using ASME Code Case N-411-1 (Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping Section III, Division 1). The response of the structure to the earthquake is obtained by using the spectrum response technique. Appropriate acceleration levels are read from the earthquake spectrum curve corresponding to the natural frequencies of the structure. The mode shapes and lumped weights are utilized to calculate an effective weight associated with each mode.

These effective weights and the spectrum curve acceleration levels are utilized to obtain an effective force for each mode. Then, the mode shapes are used again to distribute the effective modal forces of each mode throughout the structure in order to obtain forces at each point for each mode. These forces, on a modal basis, are used as separate loading conditions to obtain the response of the structure. The individual response values per mode at different points for shear moments and displacements are combined on an absolute basis. All mode shapes of the structural system which have natural frequencies below 30 Hz are used or a minimum of four modes.

The response spectrum specified for the site design earthquake and response spectrum generated from acceleration time-history record of the July 12, 1952, Taft, California S69E Earthquake normalized for the 5 percent design earthquake are compared in Figure C.3.12 for 2 percent of critical damping except for main steam line Piping Inside Containment, since only this was used for developing floor spectrum curves. For evaluation of main steam line inside containment, the response spectrum specified for the site design earthquake and response spectrum generated from acceleration time-history records of the July 12, 1952, Taft, California S69E Earthquake normalized for the 5 percent design earthquake are compared in Figure C.3.12J (Reference 26) for 2 percent of critical damping since only this was used for developing DE floor spectrum curves.

For MCE loading for main steam line inside containment, the response spectrum specified for the site design earthquake and response spectrum generated from acceleration time-history record of the July 12, 1952, Taft, California S69E Earthquake normalized for the 12 percent maximum credible earthquake are compared for 2 and 5 percent damping in Figures C.3.12K and C.3.12L (Reference 25) since only these were used for developing MCE floor spectrum curves. The response spectrum for the 1997 Re-analysis of the Recirculation system piping and Residual Heat Removal and Reactor Water Clean-up piping inside primary containment for Peach Bottom NCR 97-02267 is compared to the site design earthquake in Figure C.3.12A.

To obtain floor spectrum curves for the MCE, the values obtained from the 2 percent damping design earthquake are multiplied by 2.4 (0.12/0.05) except for main steam line piping inside containment. Since the higher damping for the MCE is thus not considered, values employed are very conservative. For analysis of the main steam line inside containment, the MCE floor spectra curves were obtained using the structural MCE damping value specified in Table C.2.1 instead of multiplying design earthquake by 2.4 (Reference 26).

The time-history technique is used to develop spectrum curves at selected points on the structure for use in equipment analysis.

Since some of the points from the time-history spectrum fall below the site response spectrum, the ratio of the accelerations obtained by the spectrum response technique to the accelerations from the time-history analysis was used as a multiplying factor to increase the time-history spectrum for the Class I structures as appropriate.

Figure C.3.3 shows the mathematical model used for the seismic analysis of the coupled system of the reactor building, reactor vessel pedestal with sacrificial shield, and the reactor vessel. The model of the reactor vessel used in this coupled system was approximate and was used to study its effect on the reactor building. Figure C.3.3A shows the mathematical model used to generate response spectra curves for the 1997 re-analysis of the Recirculation system piping, and the Residual Heat Removal and Reactor Water Clean-up piping inside primary containment for Peach Bottom NCR 97-02267. The seismic model in Figure C.3.3A is reconstituted (Reference 25) and was used to develop the spectra for the Main Steam analysis inside containment (Reference 26). The seismic analysis of the reactor vessel and its internals is discussed in subsection C.5, "Components."

The seismic moments and shears obtained from the analysis were used for the structural design of the buildings with particular emphasis on the seismic overturning, connections of the members, and arrangement of the reinforcing in the concrete. Figures C.3.4 through C.3.11 show moments, shears, displacements, and accelerations for the reactor building.

These graphs represent the values of moments and shear used in the structural design of buildings. These values were checked from time to time to evaluate the effects of the changes associated with the design development of the project, and to assure that the design values used were always conservative.

To assure the aseismic integrity of equipment, an earthquake time-history is selected whose raw spectrum response curve is greater than or equal to the site design spectrum response curve.

This time-history is applied at the base of the building to generate, at selected elevations, additional time-histories and spectrum response curves. These time-histories and spectrum response curves are then utilized to assure the aseismic integrity of the equipment. Other seismic Class I structures were also dynamically analyzed following the same procedure.

C.4 IMPLEMENTATION OF STRUCTURAL CRITERIA

This subsection illustrates the loads and load combinations (subsection C.2) and structural static and dynamic analysis (subsection C.3) used in the structural design of seismic Class I structures.

The design analysis of the primary containment is presented in Appendix M, "Containment Report." Loads, load combinations, and methods of analysis used for the design of the primary containment are described in detail in Appendix M. That appendix also summarizes the actual stresses in the containment vessel under various loading conditions.

This subsection briefly discusses typical structural elements of the reactor building and summarizes the actual stresses in these elements in Tables C.4.1 through C.4.5.

Design procedures used for the reactor building were also used for the other seismic Class I structures, such as the diesel-generator building, the radwaste building, and the pump structure. Design procedures are identical; stresses in various elements of these structures are not illustrated.

C.4.1 Reactor Building Floor System

The selection of a particular floor system, precast concrete, poured in place concrete, composite construction, or steel and metal deck, was based on an evaluation of the economics, construction schedule and sequence, shielding requirements, and structural requirements. The reactor building has more than one type of system.

Allowable stress design methods are typically utilized to evaluate floor systems. In a few cases, the newer method of ultimate strength design has been utilized to reevaluate new loading configurations. These cases are described in Section C.2.6. Although the following example has been reevaluated using the ultimate strength design method, the paragraphs below and associated Table C.4.1 describe the original evaluation method as it demonstrates the primary method used throughout the Class I areas of the plant.

Because of its critical function and particularly heavy loading, the floor system for the ground level operating floor (Elevation 135 ft 0 in) is selected to illustrate the implementation of design criteria.

The beam selected is in the area where the railroad track enters the building. In addition to the usual railcar loading, it also supports the spent fuel cask on a special railcar. This area is also designed for a -1,000 psf live load to accommodate the transfer of heavy equipment, such as the recirculating pump motor, which may have to be transported through the railroad lock. This area was also evaluated for a live load of 125-ton cask with a 70-ton transporter.

The floor is designed to include vertical seismic loading simultaneously with the full design live load.

Wind (W), tornado (W'), and jet loads (R) do not act on this particular portion of the floor and, therefore, are omitted.

Table C.4.1 tabulates the design stresses, allowable stresses, and loading combinations as they apply to the particular beam illustrated. The design stresses are within the allowable stresses and the system is structurally adequate.

C.4.2 Reactor Building Concrete Wall

The south wall (column line 8) of the Unit 2 reactor building is selected to illustrate the implementation of the design criteria. This wall experiences several loading combinations. It is a shear wall for the seismic forces due to an earthquake (E or E') in the east-west direction. It is also a peripheral basement wall for the torus and experiences soil and hydrostatic loads (part of D) on its south face. In the superstructure of the building, this wall is designed to withstand normal wind loads (W), as well as tornado loads (W').

The combination of these loads is critical for the design of the south wall. The governing design condition was D+E.

The wall design was also investigated for the effects of tornado missiles and the thickness of the wall was determined to be adequate.

Design stresses and allowable stresses are tabulated in Table C.4.2. The design stresses are within the allowable limits.

C.4.3 Reactor Building Superstructure

The basic frame of the superstructure consists of stepped crane columns and a 12-ft deep truss.

The roof, consisting of purlins framing from truss to truss and metal deck, forms a rigid diaphragm. The bottom chords of the trusses are tied together with horizontal bracing.

All the peripheral columns, stepped crane columns on the east and west (column lines B and J), and wind columns on the north and south (column lines 8 and 18) are braced, and support girts in turn support the metal siding. The effect of metal siding as a diaphragm was neglected. Column bracing is designed for wind (W) and earthquake (E) loading. Columns are designed for dead and live loads (D), wind loads (W), and earthquake loads (E). The superstructure is also designed for tornado loads (W') on the assumption that all or part of the metal siding is blown away due to the tornado, and the basic superstructure frame is subjected to full tornado winds of 300 mph. Under this condition, the frame will withstand the loading without failure. The stresses may exceed normal allowable stresses, but will not exceed 90 percent of the yield stress of structural steel. The trusses are designed on the same basis. The effect of suction on the metal deck was taken into account.

In the design of the reactor building's superstructure, as well as its floors and walls, concentrated loads are structurally accommodated by the addition of special restraining systems as for jet loads and the support members were designed to carry the reactions. Member proportions were established to provide adequate protection wherever there was a probability of missile impingement.

Table C.4.3 summarizes the loading combinations, method of analysis, resulting design stresses, and allowable stresses. The design stresses are well within the limits of the allowable stresses.

C.4.4 Reactor Pedestal

The reactor pedestal was investigated for various loads: dead and live load (D), earthquake (E or E'), temperature (T) associated with an accident condition for a thermal gradient of 70°F, and jet forces (R) associated with a pipe rupture. Jet forces on pipe restraints attached to the sacrificial shield and pedestal were also investigated. The overall design was based on very conservative assumptions to allow for the complex interactions of the various loads.

Incorporated into the design of the reactor shield is the capability to withstand, without failure, the internal pressure and coincident jet impingement loads resulting from failures of high-pressure lines in the shield space region (from the outside diameter of the reactor shield to the outside diameter of the reactor vessel). Failure of the reactor vessel (including nozzles) is not considered credible; however, the consequences of safe-end failures are given full consideration. Safe-ends, even though attached by the reactor vessel manufacturer, are not considered to be an integral part of the reactor vessel but are regarded as a transition piece between the reactor vessel and the primary piping. Although steps have been taken, in light of recent experience, to effectively preclude safe-end failures, the design criteria developed for the reactor shield considers a full spectrum of breaks up to a double-ended recirculation line break at the nozzle to safe-end weld.

The design of the reactor pedestal included additional base anchorage above that required by calculations to assure no adverse affects from secondary stresses.

The loading combinations, resulting stresses, and allowable stresses are tabulated in Table C.4.4. The design stresses are within the allowable stresses resulting in a high safety factor. Stresses are shown at the base Elevation 119 ft 11 in and at an intermediate level, Elevation 130 ft 0 in. Base stresses are the maximum. Stresses decrease from this point to those shown for Elevation 130 ft. Above Elevation 130 ft stresses do not exceed those at that elevation. Basic reinforcement is uniform throughout the pedestal and is the basis for the reported stresses.

For Unit 2, primarily for construction considerations, permanent steel plates were used in lieu of wood forms. In the design of the anchorage of the base of the pedestal, the steel liner plates were anchored to provide the added anchorage at the base and studs were provided to secure the plate to the concrete. Based on economic considerations developed from the Unit 2 experience, it was decided to remove one or both of the liner plates for Unit 3 and to replace the anchorage so deleted by an additional row of dowels near the pedestal wall centerline.

With the added dowels, the Unit 3 reactor pressure vessel pedestal meets the same design requirements as the Unit 2 pedestal. As constructed, an inside liner similar to that for Unit 2 was used for Unit 3, and the outside of the Unit 3 pedestal was formed. With regard to the anchorage requirements, no credit was taken for the inner liner plate. The removal of the outside liner did not decrease the pedestal's capability to resist all postulated loads.

Temperature effects were accounted for in the design of the reactor pedestal using the ACI 505-54 method of analysis for the steady-state condition. Since the thickness of concrete is large, the time required to form a higher gradient than that used is beyond the expected time of exposure and therefore not considered critical.

For justification of allowable stresses see paragraph C.3.2.

The ring girder is designed to transfer the vertical and horizontal loads of the reactor pressure vessel skirt flange to the top of the reactor pressure vessel support pedestal.

The horizontal shears on the reactor pressure vessel 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 designed 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 reactor pressure vessel skirt flange and top flange of the ring girder. The total frictional force and the coefficient of sliding friction is independent of the areas in contact, so long as the total pressure remains the same. The friction-type connection of the reactor pressure vessel 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, suffer no overall loss of frictional shear resistance. 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 due to a coefficient of friction of 0.15 and a proof load of 405 kips per bolt is 3,650 kips or 2.8 times the design basis earthquake shear load of 1,300 kips or 1.4 times the MCE shear load of 2,600 kips. However, if the coefficient of friction is assumed zero, the bolts acting as bearing-type connections could resist a total horizontal shear of 4.15 (at AISC Code stresses) times the design basis earthquake shear load of 1,300 kips or 2.6 (at 90 percent yield stresses) times the MCE shear load of 2,600 kips. Therefore the high strength bolt connections of the reactor pressure vessel skirt flange to the top flange of the ring girder, with or without friction, are more than adequate for the respective design load.

The vertical loads on the reactor pressure vessel skirt flange are transferred to the top of the reactor pressure vessel support pedestal by the ring girder as a bearing plate. The ring girder is designed according to AISC Code.

For stresses between the pedestal and the spherically shaped base see Table C.4.4 at Elevation 119 ft 11 in.

Moderate friction (0.2) on the shear ring connecting the spherically shaped concrete base to the steel drywell will prevent translation. However, bearing on the external concrete structure will also prevent translation and, therefore, no reliance on the friction factor is necessary. Table C.4.4 reflects this.

C.4.5 Drywell Shielding Concrete

The drywell shielding concrete structure, due to its irregular shape and loading combinations, was designed by the finite element method and checked by other methods. Large openings were accounted for in the analysis by introducing lower stiffness elements at regions where they occur in the finite element analysis. In the design check, a shell approach was used and the worst resulting stresses were incorporated in the design. The openings were designed as frames to carry stresses around discontinuities. The most conservative results were used in the final design.

Table C.4.5 illustrates the design stresses under various loading combinations consisting of dead and live loads (D), temperature (T) with a thermal gradient of 70°F, and earthquake (E or E'). The design gradient of 70° is based on a heat balance analysis and a drywell bulk average temperature of 145°F. From Table C.4.5 stresses are quite low, providing sufficient margin to accommodate higher gradients. In any case, the time it takes to heat up this large mass of concrete is very long and, therefore, the accident transients do not significantly affect the concrete stresses. The design values tabulated are based on the finite element method used for the structural design. The structure was assumed as axisymmetrical with allowance made for local discontinuities at penetrations. In the finite element program, an uncracked section of concrete was assumed, yielding very conservative results.

The drywell shielding concrete is not subjected to tornado loads (W') or wind loads (W). The indirect application of jet forces (R) was investigated as a special case. Reactions from jet forces were taken into consideration in the design of the biological shield concrete. Load combinations listed in Table C.4.5 are representative of typical sections of structure, and since reactions at piping and equipment anchor points constitute localized conditions, they were not listed. The concrete is capable of withstanding the jet forces, as a localized load, should the drywell yield locally without rupture and close the 2-in air gap between the drywell and shield. The 2-in air space around the drywell is open at all penetrations through the biological shield and drained by pipes at the bottom. Since the 2-in gap is ventilated at several places to the atmosphere, it cannot become pressurized due to temperature inside the drywell.

The effects due to thermal expansion of pipes under operating condition (H) are insignificant inside the primary containment when compared to pressure loadings and, therefore, are not included. No hot pipes are rigidly attached to the drywell shell. Expansion bellows have been provided at critical hot penetrations.

H represents either the operating or accident condition, whichever is greater. Normal allowable stresses were not increased when using this loading.

The design stresses under all loading combinations are within the allowable limits.

C.4.6 The Sacrificial Shield

The sacrificial shield was designed without considering the concrete for any structural purpose, except the lower 10 ft of the wall. The forces considered were: seismic forces, pipe loading, pipe restraints, platform loads, and jet load reaction. The 27-in thick cylindrical structure consists of 12 steel columns equally spaced and continually tied by a 1/4-in thick steel plate on the inside and outside of the columns. For seismic design the sacrificial shield was modeled as a beam on a spring support at the top and fixed at the bottom. A space truss model was also used to check individual sections subject to combination of stresses with the aid of a computer. After the integrity of the overall structure was ascertained, local stresses, connections, and discontinuities were investigated. Proper account was made of nonaxisymmetric loads.

The design allowable stresses are based on the AISC Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings, without any increase in the normal allowable stresses when design earthquake loads are included. For the MCE condition in combination with jet loadings, the stresses are allowed to reach 90 percent of yield stresses.

For a description of the sacrificial shield see subsection 4.6. Reactor vessel penetrations which penetrate the sacrificial shield are closed with removable shield plugs which fit around the penetration pipe. The removable shield plugs allow access for in-service inspections.

C.4.7 Main Steam Pipe Chase

The main steam pipe chase is designed to sustain a static pressure of 10 psig based on the vent area available and steam release from a single pipe failure. A panel designed to blow out at less than 1.25 psig is provided to eliminate the possibility of a higher pressure buildup.

The design and construction is according to ACI 318-63 and uses the same criteria as for Class I structures under a no-loss-of-function criterion.

The design allowable stresses (safety factors) are based on the ACI 318-63 working stress method. Since the blowout panel constitutes a portion of the boundary of the secondary containment, it must remain intact for those situations for which secondary containment integrity is required. The panel was analyzed for its ability to withstand without failure the effects of the MCE (0.12g) and the design was found to be adequate.

Analyses were made of the resultant stresses in the piping within the steam tunnel for the MCE to assure that those lines with a potential for energy release to the tunnel sufficient to cause panel failure will meet the stress requirements of seismic Class I piping as defined in Appendix A.

C.4.8 Spent Fuel Pool

The fuel pool, together with the dryer-separator storage pool, forms a channel-shaped beam supported in the middle by the biological concrete shield structure and at the outer ends by the building walls.

In order to minimize the possibility of pool leakage, the pools are lined with stainless steel.

A finite element analysis was performed to determine the maximum allowable fuel rack loads which can be imposed on the pool slab. The analysis included the effects of the water in the pool, including the fluctuation of pressure due to seismic acceleration and sloshing. Thermal effects due to normal operating and accident conditions were also included.

The available section strengths for reinforced concrete elements are calculated by the strength design method in accordance with ACI 318 and ACI 349. Axial force/moment and axial force/shear interaction diagrams are generated for the entire spent fuel pool structure. These interaction diagrams were then used to manually check each critical section. The axial force/shear interaction diagram for the spent fuel pool floor includes the transverse shear strength of the steel beams. The available section strengths for structural steel members for axial loads plus bending are determined by plastic design methods in accordance with AISC.

The section strengths required to carry the fuel pool loading are based on results from the finite element analyses. Required section strengths in terms of shear forces and bending moments are determined for each element in the spent fuel pool structure and for each of the governing load combinations.

C.4.9 Concrete Block Walls

The typical block wall design was based on a static coefficient of acceleration of 0.2g. When the plant design was completed, walls were checked dynamically based on their location in the structure to ensure the block walls could withstand the worst load combinations associated with the MCE and verify that the preliminary design was conservative. A technical report in response to NRC IE Bulletin 80-11 was prepared and issued on April

17, 1981, which describes in greater detail the re-evaluation criteria for the block walls. The conclusion of this re-evaluation is noted therein and necessary modifications have been completed to assure that the block walls do not prejudice the integrity of any Class I equipment.

C.4.10 Strength Tests and Crack Control for Concrete

The entire Unit 2 reactor building and radwaste building concrete, as well as most of the remainder of the concrete in place at the time of the Bechtel Corporation report, "Concrete Strength Survey Report," dated June, 1970, were surveyed by the Swiss hammer method. Reports are included in the referenced report. Lewis H. Tuthill, past president of ACI, reviewed the number of cores taken and stated that non-destructive testing was better than drilling the structure full of holes with possible impairment of strength and agreed the Swiss hammer readings gave uniform results of the concrete in place.

Twenty-two thousand (22,000) cu yd of concrete was placed and 117 representative cores taken or approximately 1 core for every 188 cu yd of concrete placed. The project specification states one set of six cylinders for every 150 cu yd of concrete placed. This gives a ratio of approximately (1:7) 1 core to 7 cylinders, and when the Swiss hammer tests are included with the cores, the ratio of cores to cylinders is greater than one, and the structure was determined to be adequate for its intended function.

ACI 318 paragraph 504(a) states one test will be taken for every 150 cu yd. Two specimens shall be made for each test or 20,000/150 = 133 tests or 266 specimens all to be broken at 28 days. However, ACI 301 paragraph 1704(c) states that at least 3 cores be taken in areas of concrete that were considered deficient. Only five areas were deficient. Therefore, at least 15 cores are required to satisfy ACI 301. With the Swiss hammer tests, in addition to the cores, the structure was determined to be adequate for its intended function.

Concrete cores were taken from all Class I structure concrete in the affected areas, and Swiss hammer correlation calibration was made with standard cylinders from concrete being used on the project. A grid pattern was then established in each area to be tested (some 2 ft to 3 ft vertically by approximately 10 ft horizontally on centers in the biological shield area) with the calibrated Swiss hammer.

The Swiss hammer tests essentially measured the surface hardness of the concrete tested. Core tests furnished the depth and condition of the concrete in the structure. The correlation calibration was established to verify areas not cored. The Swiss hammer is also useful to check uniformity; to locate areas of unsatisfactory concrete in walls, beams, floors, and mass structures; and to serve as a substitute for test cylinders cured at the site for evaluating the compressive strength at early ages.

The Swiss hammer tests and core test were conducted in accordance with the requirements of ACI 301 Chapter 17, "Evaluation of Concrete Strength," paragraph 1704. The Swiss hammer used on affected areas was calibrated and correlated for each concrete sample tested (ACI 301, paragraph 1704(a)). Cores were taken and tested to ASTM C-42 (ACI 301, paragraph 1704(b)). More than three cores were taken in each affected area (ACI 301, paragraph 1704(c)). Approximately 10 impact hammer tests were taken within a cored area. In accordance with ACI 301, paragraph 1704(d), an evaluation of stresses of affected areas was made and compared with core test results. The strength of all cores exceeded that required for the members with a safety factor equal to or greater than specified in the ACI Code.

Ultrasonic non-destructive testing was not employed in the affected areas as the cores gave actual strengths of concrete in place and Swiss hammer readings verified areas between cores.

Microfissures in lower than anticipated strength concrete considered satisfactory for compressive strength does not decrease the shear and bond resistance below a safe level.

This is demonstrated by the shear mode of break in the cores. Thus the compressive strength of the core actually indicates the shear strength of the concrete.

Since bond transfer is mechanically done through reinforcing bar deformations, the compressive and shear strength of the concrete represents a measure of the bond also. Therefore, shear and bond strengths were lowered as the compressive strength was lowered but not below a safe level.

Recent research and testing in connection with microfissures in concrete has shown that microfissures exist before loading and stressing and are caused by normal settlement of aggregate, bleeding of the mixing water and shrinkage stresses induced by the drying process. Spreading of the cracks is retarded by interaction of surrounding particles, continuity of surrounding mass of concrete, roughness of aggregate, restraint of surrounding matrix, reinforcing steel, and pores and voids.

It has been established that for loads below about 30 percent of f'_c the increase in bond cracking is negligible. Above 30 percent and up to about 70 percent of f'_c the amount of bond cracking increases causing the initial deviation from linearity of the stress-strain curve. Further cracking or increase in sizes is restricted until about 85 percent or 90 percent of ultimate is reached for the reasons outlined above.

Slow crack growth and propagation is associated with creep and shrinkage. Stress concentrations around the end of a microfissure invariably cause creep in the matrix until an equilibrium position is reached where stress concentrations are insufficient to propagate the microfissure. Therefore, under sustained loads below ultimate, the microfissures stabilize. Besides, creep and shrinkage reduce the load carried by the mortar, further preventing the formation or propagation of a significant number of microfissures for loads below the ultimate.

Under biaxial or triaxial compressive stresses the strength is improved since normal stresses are restraining the propagation or formation of microfissures. Normal and shear stresses are interrelated by the classical Coulomb-Mohr theory, confirmed by tests.

As discussed above, the mode of failure for the concrete cores was in shear, not the hour-glass mode representative of compressive failures, thereby permitting the use of code factors for allowable stresses in structural concrete.

Actual stresses in the structures are well within those allowable for the core strengths tested.

Examination of microfissures indicated that there was no migration of gases; thus, it would not collect at rebar and have any more effect on the bond than on other characteristics. Therefore, bond is adequate.

References used are ACI journals of August 1964; January 1969; July 1969; and April 1971; and "Causes, Mechanism and Control of Cracking in Concrete," ACI publication SP-20.

TABLE C.4.1

REACTOR BUILDING FLOOR SYSTEM

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
**This corner was designed for two alternate loadings: steel Uniformly distrib. Live load of 1.0 ksf. Live load of RR car and 85T fuel cask Higher stresses due to either of the above govern the design	Working stress design method for steel and concrete	D + E	F_c = 1.35 F_v = 0.060 F_t = 24.0 F_b = 24.0	f_c = 0.072 f_v = 0.015 f_t = 4.5 f_b = 17.6	Floor at elev. 135 ft 0 in SE corner beam 36 WF 194 (compact) concrete slab 2 ft 3 in
Materials conform as follows: steel concrete f'_c = 3,000 psi at 28 days concrete maximum strength per ACI 318-63 Reinforcing ASTM Designation: * A615 Grade 40 per ACI 318-63 Structural Steel ASTM Designation: A 36-67 per AISC Manual and Specification, 1963.		D + E'	F_c = 2.55 F_v = 0.093 F_t = 32. F_b = 32.	f_c = 0.076 f_v = 0.016 f_t = 4.9 f_b = 18.4	Floor at elev. 135 ft 0 in SE corner beam 36 WF 194 (compact) slab 2 ft 3 in

Note that this table provides stresses from the original analysis method to serve as an example of the typical load cases assessed and the results obtained.

*Reinforcing bars #8 and larger are A615 Grade 60. Other bars are A615 Grade 40 throughout the Class I structures.

** Evaluation of this corner for other live load conditions is covered in next page.

TABLE C.4.1

REACTOR BUILDING FLOOR SYSTEM

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Allowable Moment (k-ft)	Max. Moment (k-ft) or Max. Stress (ksi)	Location
This corner was further evaluated for the following loading conditions: Live load of cask transporter (70T) with a TN-68 cask (125T).	Working stress design method for steel and concrete.	D**	$F_c = 1.35$ } $F_t = 24.0$	* $M_c = 525.4$	* $M = 489.1$	Floor at Elev. 135'-0", SE corner steel beam 30 WF99 with 2'-3" conc. slab
			$F_v = 0.06$		$f_v = 0.056$	
			$F_b = 24.0$		$f_b = 14.0$	
	Same as above	D+E	$F_e = 1.35$ } $F_t = 24.0$	$M_c = 525.4$	$M = 515.0$	Same as above
			$F_v = 0.06$		$f_v = 0.059$	
			$F_b = 24.0$		$f_b = 14.7$	
	Same as above	D+E′	$F_e = 2.55$ } $F_t = 32.0$	$M_c = 992.4$	$M = 551.7$	Same as above
			$F_v = 0.093$		$f_v = 0.063$	
			$F_b = 32.0$		$f_b = 15.8$	

Note that this table provides stresses from the original analysis method to serve as an example of the typical load cases assessed and the results obtained.

* M_c = Maximum allowable section bending moment capacity.

M = Maximum section bending moment

** D = Includes both dead and live loads.

TABLE C.4.2

REACTOR BUILDING CONCRETE WALLS

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Design load (D) includes all dead and equipment loads shear and soil pressure	Working stress method	D + E	F_t = 24.0	f_t = 18.2	External wall on 8 acting as re-taining and
			F_c = 1.8	f_c = 1.030	
					wall below elev.
Materials conform as follows:			F_v = 0.070	f_v = 0.053*	135 ft 0 in
			F_v = 0.188	f_v = 0.070**	
Concrete F_c' = 4,000 psi at 28 days maximum strength per ACI 318-63		D + E'	F_t = 54.	f_t = 18.2	External wall on 8 acting as re-taining and
shear			F_c = 3.40	f_c = 1.13	
Reinforcing ASTM Designation: A615 Grade 60 per ACI 318-63			F_v = 0.253	f_v = 0.120	wall below elev. 135 ft 0 in
Maximum allowable stresses for D + E'					
Concrete F_c = .85 f_c'					
Reinforcing F_t = 0.9 fy					

* Retaining wall shear

** Seismic shear

TABLE C.4.3

REACTOR BUILDING STEEL SUPERSTRUCTURE

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Material: Structural Steel ASTM Designation: A-36-67 per AISC Manual & Specification, 1963	"STRESS" computer program using "stiffness" method for D & W'	D + E	F_a = -14.7	f_a = -3.7	Corner columns
			F_b = 24.0	f_b = 1.0	
	Dynamic Analysis for Earthquake	D + W'	F_a = -14.7	f_a = -3.8	Corner columns
			F_b = 32.0	f_b = 1.8	
	Working Stress Design Method	D + E	F_a = 22.0	f_a = 2.7	End column bracing
		D + W'	F_a = 32.4	f_a = 12.0	End column bracing
		D + W'	F_a = 22.0	f_a = 3.7	Center column
			F_b = 32.4	f_b = 17.2	
		D + E	F_a = -17.99	f_a = -15.2	Truss top Chord
		D + W'	F_a = -27.0	f_a = -8.6	Truss bottom Chord
		D + E	F_a = 22.0	f_a = 13.2	Truss Diag.
		D + W'	F_a = -18.7	f_a = -8.8	Truss Diag.

TABLE C.4.4

REACTOR CONCRETE PEDESTAL

Load Combination	Maximum Allowable Stress (psi)			Units 2 and 3 No Liners or Dowels Maximum Calculated Stresses (psi) @ El 119 ft 11 in (Base)	@ El 130 ft 0 in
D+T (Vert.) + E	Rebar	Tension	$.4F_y$ = 24,000	6,730	6,730
		Compr.	$.45 f'_c$ 1,800	1,140	1,114
	Concr.	Shear**	$1.1 \sqrt{f'_c}$ 70	41	41
D+T (Circ) + E	Rebar	Tension	24,000	7,440	7,440
		Compr.	1,800	317	317
	Concr.	Shear**	70	70	70
D+T (Vert.) + E'	Rebar	Tension	$.9F_y$ = 54,000	7,345	7,345
		Compr.	$.85 f'_c$ = 3,400	1,161	1,114
	Concr.	Shear	$4 \phi \sqrt{f'_c}$ = 215	76	76
D+T (Vert.) + R	Rebar	Tension	54,000	38,430	38,430
		Compr.	3,400	1,140	1,140
	Concr.	Shear	$3.5 \phi \sqrt{f'_c}$ = 188	58	58
		Torsion*	$10.2 \phi \sqrt{f'_c}$ = 547*	326	326
D+T (Vert.) + E+R	Rebar	Tension	54,000	40,400	40,400
		Compr.	3,400	1,585	1,555
	Concr.	Shear	215	96	96
		Torsion*	547*	330	330
D+T (Vert.) + E'+R	Rebar	Tension	54,000	43,790	43,770
		Compr.	3,400	1,602	1,555
	Concr.	Shear	215	131	131
		Torsion*	547*	333	333
Unit 2 (two Liners)*** stresses not shown same as above.					
D + T + E' + R	Rebar			41,740	41,740
	Bolts			17,065	
Unit 3 (Dowels)***					
D + T + E' + R	Rebar			39,000	43,770
	Dowels			39,000	-

TABLE C.4.4 (Continued)

NOTES:

Description/Criteria

The reactor vessel pedestal is a 25 ft 6 in high cylinder with 3 ft thick concrete walls and inside diameter of 20 ft 3 in. The pedestal projects from a spherical shaped base formed by the inside of the drywell. The shears and moments in the pedestal are transferred to the drywell through a welded steel shear ring and the bearing between the drywell and the concrete.

Temperature (T) = 70°F temperature gradient Design Earthquake (E) 0.05g, 2 percent damping.

Maximum Credible Earthquake (E') 0.12g, 5 percent damping Jet Force (R) = 1,000 K @ el 130 ft 0 in.

Materials: Concrete F'_C = 4,000 psi at 28 days maximum strength per ACI 318-63, Reinforcing ASTM

Designation: A615 Grade 60 per ACI 318-63.

Maximum allowable stresses for D + T + E' and D + T + R loads are: Concrete $F_c = 0.85 f'_C$; Reinforcing $F_t = 0.90\ F_y$. The customary increase in normal design stresses for other loading combinations when earthquake loads are considered is not used.

Methods of Analysis

Working Stress Design Method.

For seismic loads response spectra are used.

Circumferential and vertical temperature analysis is in accordance with ACI 505-54.

*Torsional shear stress analysis based on "Tentative Recommendations for Design of Reinforced Concrete Members to Resist Torsion" by ACI Committee 438 in ACI Journal, January, 1969. Formula provides for interaction of flexural (υ) and torsional (τ) shears. Due to the openings in shell only 20 percent of full ring torsional constant was used in calculations of torsional shear stress.

Only 70 psi in shear are taken by concrete; rest by steel.

Tensile stresses in reinforcement are the sum of the stresses due to flexure and torsion.

**Allowable stress is for unreinforced concrete. Radial ties are provided throughout the pedestal which would permit the use of higher allowable stresses $\left(5\sqrt{f'_C} = 316\right)$.

***Taking base anchorage into account.

TABLE C.4.5

DRYWELL SHIELDING CONCRETE

Description/Criteria	Method of Analysis	Load Combination*	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Drywell shield acts as a structural wall carrying floors. Design load (D) consists of all dead loads, equipment loads, and floor live load.	Finite Element Stress Analysis	D + E	F_t = 24.0	f_t = 8.2 Circumferential reinforcing tension	El 145 ft
		D + E	F_c = 1.80	f_c = 0.45 Vertical concrete compression	
		D + E	F_v = 0.07	f_v = 0.0645 Concrete shear at El 180 ft	
	Temperature stresses are for uncracked section.	D + T + E'	F_t = 54.0	f_t = 23.5 Circumferential reinforcing tension	El 145 ft
Seismic loads (E and E') are according to the response spectra for the reactor building.		D + T + E'	F_c = 3.40	f_c = 1.57 Vertical concrete compression	
		D + T + E'	F_v = 0.253	f_v = 0.177 Concrete shear at El 180 ft	
Operational thermal load of 45°F (averaged) thermal gradient across the wall is considered.	"STRESS" computer program using "stiffness" method	D + T + E'+ R	F_t = 54.0	f_t = 38.9 Circumferential reinforcing tension	El 141 ft at main steam penetrati ons
			F_c = 3.40	f_c = 1.5 Vertical concrete compression	
			F_v = 0.253	f_v = 0.115 Concrete shear	

Materials conform as follows:
Concrete f'_c = 4,000 psi at 28 days maximum strength per ACI 318-63. Reinforcing ASTM Designation: A615 Grade 60 per ACI 318-63. Structural steel ASTM Designation A-36 per Specification and Manual of the AISC 6th Ed.

Maximum allowable stresses for D + T + E' load combination are:
Concrete $F_c = 0.85f'_c$
Reinforcing F_t = 0.9fy

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*Floor design load (D) includes dead and live loads plus 50-psi live loads (appropriate for operating conditions).

C.5 COMPONENTS

C.5.1 Intent and Scope

C.5.1.1 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 an MCE or primary pressure boundary coolant pipe rupture, or a combination of events.

C.5.1.2 Components Designed Primarily by Empirical Methods

There are many important seismic Class I components or equipment which are not normally designed or sized directly by stress analysis techniques. Simple stress anlayses 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. 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. 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 MCE or design earthquake, 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.

C.5.1.3 Components Qualified for SQUG Methodology

A specifically approved empirical method of equipment qualification is the use of seismic experience data utilized in accordance with the Seismic Qualification Utilities Group (SQUG) methodology to verify the seismic adequacy of existing, new, modified and replacement items on a case-by-case basis. Such evaluations are performed in a controlled and systematic manner to ensure that the item of equipment is properly represented in the earthquake experience or generic testing classes and that applicable caveats are met. In particular, each new or replacement item must be evaluated for any design changes that could reduce the seismic capacity of the equipment from that reflected in the experience data base, and all such evaluations must be documented in accordance with established procedures. SQUG methodology is applied in accordance with the SQUG Generic Implementation Procedure (Reference 16) and implementation of the SQUG methodology is controlled and documented in accordance with approved procedures. The use of the SQUG methodology is limited to the scope of equipment covered by equipment classes described in the SQUG Generic Implementation Procedure (GIP). The methodology is not used to verify the seismic adequacy of equipment not included within the scope of the equipment classes described in the GIP, and may not be used when the NRC commitment has been made to qualify specific equipment to IEEE 344-75 (invoked by Reg. Guide 1.100).

C.5.2 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 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.

The design loading conditions which were used for components of this plant (except the reactor vessel and reactor vessel internals) are presented in Tables C.5.7 and C.5.8.

C.5.2.1 Loading Conditions

The loading conditions may be divided into four categories: normal, upset, emergency, and faulted conditions. No categorization for loading conditions was made for the safety/relief valves, safety valves, main steam line isolation valves, recirculation system valves and pumps, or other components in the reactor coolant pressure boundary since the applicable codes did not require such categorization. The categories listed above are generically described as follows.

C.5.2.1.1 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.

*C.5.2.1.2 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 design earthquake. For recirculation systems, the design earthquake, not the MCE, is the upset condition.

C.5.2.1.3 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 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.

C.5.2.1.4 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. The faulted condition includes the effects of the MCE.

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* See Note A

*C.5.2.2 Allowable Limits

In addition to the generic definition of loading conditions in the preceding paragraphs, the meaning of these terms is expanded in the 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. The quantitative event classifications are as follows:

Loading Conditions 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-yr event encounter probability

These probabilities have been assigned to establish the appropriate structural design limits for the loading conditions in paragraph C.5.2.1. A summary of these limits is shown in the tables listed below:

Deformation Limit	Table C.5.1
Primary Stress Limit	Table C.5.2
Buckling Stability Limit	Table C.5.3
Fatigue Limit	Table C.5.4

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.

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* See Note A

The other SM 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
$10^{-5} > P_{40} > 10^{-6}$	1.125
$1.0 > P_{40} > 10^{-1}$	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.5.5.

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 permitted by the applicable code and by these criteria, the SF_{min} may be progressively lowered to a minimum acceptable level on the basis that there is a lesser need for design margin for loading conditions which have a diminishing probability of occurrence.

*NOTE A:

In Table C.5.5 of the Peach Bottom 2 and 3 FSAR, the 40-year event encounter probability for the Maximum Credible Earthquake (MCE) plus normal loads is given to be 10^{-3}. Using the currently accepted probability categories given in Table 1, MCE plus peak operating loads is a faulted condition. In the replacement recirculation piping analysis and Main Steam piping analysis inside containment, MCE plus peak operating loads was analyzed as a faulted condition.

In the original analysis, MCE plus peak operating loads was analyzed as a faulted condition and MCE plus normal operating loads was analyzed as an emergency condition.

The following tables do not apply to recirculation and Main Steam piping inside containment:

Table C.5.2
Table C.5.4
Table C.5.5
The Main Steam piping inside containment meet Table C.5.7 requirement.

(NOTE A CONTINUED)

Table 1

<u>Current Probability Categories* and Acceptance Criteria for Components other than Containment Structures</u>

Normal Conditions	$P_1 \sim 1.0$ ----> $P_{40} = 40$
Upset Conditions	$1 > P_1 \geq 10^{-2}$
	$40 > P_{40} \geq 0.4$
Emergency Conditions	$10^{-2} > P_1 \geq 10^{-4}$
	$0.4 > P_{40} \geq 4 \times 10^{-3}$
Faulted Conditions	$10^{-4} > P_1 \geq 10^{-6}$
	$4 \times 10^{-3} > P_{40} \geq 4 \times 10^{-5}$

P_1 = 1-year event encounter probability

P_{40} = 40-year event encounter probability

Safe Shutdown Earthquake (SSE) is characterized as having an encounter probability of $<10^{-4}$ per reactor year. This is equal to an encounter probability of $<4 \times 10^{-3}$ in 40 reactor years.

*For BWR 4, 5, & 6 reactors utilizing Mark I, II, and III containments.

C.5.3 Method of Analysis and Implementation of Criteria

C.5.3.1 Reactor Vessel

The reactor pressure vessels are designed, fabricated, inspected, and tested in accordance with Section III of the 1965 Edition of the ASME Boiler and Pressure Vessel Code as described in Appendix K. The ASME code does not require categorization of loading conditions.

Stress analysis requirements and load combinations for the reactor vessel have been evaluated for the cyclic conditions expected throughout the 60-year life, with the conclusion that ASME code limits are satisfied. The monitoring locations for the cycles used in the evaluation of these vessels is presented in Table 4.2.4. The results of the original stress analysis are presented in Table C.5.6. References 4, 5, 6 and 35 document the re-evaluation of reactor vessel fatigue in accordance with the CUF of the locations presented in Table 4.2.4 prior to implementation of Extended Power Uprate. Reconciliation of reactor vessel stress and fatigue evaluation performed for EPU is presented in References 19, 20, and 22, with the conclusion that ASME code limits for the affected components continue to be satisfied.

The vessel design report contains the results of the detailed design stress analyses performed for the reactor 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. Results of the most critical of those original analyses are included in Table C.5.6. The conclusion is that the limits in the criteria have been met. Results of reactor vessel stress reconciliation for EPU are presented in Reference 19.

Closure stresses and usage factors have been re-evaluated in M-1-A-411 based on a reduced number of tensioning and detensioning passes for RPV assembly and disassembly.

C.5.3.1.1 Vessel Fatigue Analysis

An analysis of the reactor 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.

References 19 and 20 document the fatigue analysis of the affected reactor pressure vessel components for EPU operating conditions, including re-analysis of the feedwater nozzle. The analysis considers effects of environmentally assisted fatigue due to reactor coolant environment for a 60-year plant life (Reference 18). For the feedwater nozzle, cumulative usage factor is based on both system and rapid cycling effects on the inner radius. The cumulative usage factors for the critical components are below the code allowable limit.

C.5.3.1.2 Vessel Seismic Analysis

A seismic analysis was performed for a coupled system consisting of reactor building, drywell, reactor vessel, and internals. The analysis is discussed in paragraph C.5.3.2.

C.5.3.2 Reactor Vessel Internals

Although not mandatory, the design of the reactor vessel internals is in accordance with the intent of Section III of the ASME Boiler and Pressure Vessel Code. Most of the material used for fabrication is solution heat-treated, unstabilized type 304 austenitic stainless conforming to ASTM specifications. Allowable stresses for the internals materials under normal operating conditions are taken directly from Section III. The methods of analysis used as a guide were 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, and results of the most critical of those original analyses are included in Table C.5.6. The conclusion is that the limits in the criteria have been met. Analysis of affected reactor vessel internals for EPU conditions is documented in References 21 and 22, with the conclusion that ASME Section Ill code limits for allowable stress and fatigue usage are satisfied.

C.5.3.2.1 Internals Deformation Analysis

Control Rod System

If there were excessive deformation of the CRDS, made up of the CRD, CRD housing, control rod, control rod guide tube and fuel channels, and the core structural elements which support them (top guide, core support, 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 MCE. Analyses of the internal components which have the highest calculated stresses are included in a subsequent paragraph. The highest calculated stresses occur where the MCE and loads resulting from the design basis accident 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 MCE and line break accident. Reconciliation analysis of the control rod system performed for EPU continues to support the conclusion that rod insertion would not be negatively affected. (Reference 21)

Core Support

The core support sustains the pressure drop across the fuel. The 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 CRD housings and thereby increase core bypass leakage. This upward deflection would have to be 1/2 in to begin to lift the 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 in. 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. Reconciliation analysis performed for EPU continues to support this conclusion for the core support (Reference 19).

C.5.3.2.2 Internals Fatigue Analysis

Fatigue analysis was performed using as a guide the ASME Boiler and Pressure Vessel Code, Section III. The method of analysis used to determine the cumulative fatigue usage is described in General Electric Topical Report 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 (leg type shroud support) was almost identical to the Peach Bottom plant. Therefore, the calculated fatigue usage is expected to be a reasonable approximation for this plant.

Loading Combinations and Transients Considered

1. Normal startup and shutdown
2. Design and MCE's
3. Ten minute blowdown from a stuck relief valve
4. HPCI operation
5. LPCI operation (design basis accident)

6. Improper start of a recirculation loop

Conclusion

The cumulative fatigue usage factor for Peach Bottom is evaluated to be less than 1 (Ref. 5). Based on the reconciliation analysis performed for EPU, the cumulative usage factor for the affected internals is less than the ASME Code limit of 1.0 (Reference 21).

Remarks

The location of maximum fatigue usage is at the bottom side of the baffle plate at the point where the baffle plate attaches to the shroud in the vicinity of the minimum ligament.

C.5.3.2.3 Internals Seismic Analysis

The seismic loads on the reactor vessel and internals are based on a dynamic analysis of the coupled model consisting of reactor building, reactor vessel, and internals. The natural frequencies and mode shapes for the system were determined. The relative displacement, acceleration, and load response of the reactor vessel 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 combined results for the critical components from the original analysis are presented in Table C.5.6. Seismic analysis of the internals is not affected by EPU implementation. Increased weight of the replacement steam dryer is reconciled as documented in PEAM-EPU-130 (Reference 23). For additional details see Appendix K, Exhibit V.

C.5.3.3 Piping

C.5.3.3.1 Piping Flexibility Analysis

The piping has been analyzed for the effects of dead loads, external loads, and thermal loads. In addition, piping attached to the torus has also been analyzed for the effects of Safety Relief Valve (SRV) discharge loads and loss of coolant accident (LOCA) loads which consists of pool swell, chugging, and condensation oscillation. Stresses calculated are combined bending and torsional stresses in accordance with ANSI B31.1, "Power Piping," and intensification factors were applied in accordance with ANSI B31.1. (See Section A.1.1) Several pressure/temperature cycles were evaluated, and the cycle representing the worst for thermal expansion stresses was selected for the design case. All critical points were evaluated to the stress limits of the design code and, in addition, events with a very low probability of occurrence were analyzed and stresses at all critical points compared with the limits defined in this loading criteria. Fatigue analysis for pipe weld joints using techniques in ANSI B31.7 was not performed. However, for some tees and points of stress concentration, other than girth welds, analyses were performed for fatigue damage using methods based on ANSI B31.7. Additionally, the effects of Mark I cyclic mechanical loads on torus attached piping was addressed in Report MPR-751, "Augmented Class 2/3 Fatigue Evaluation Method and Results for Typical Torus Attached and SRV Piping Systems" dated November 1982. Results showed that fatigue usage factors are low, typically below 0.3 and all below 0.5 compared to a code limit of 1.0 for a plant lifetime.

The recirculation and RHR piping systems were designed to ASME-III Class 1 (see Section A.1.1), and Torus Attached Piping was evaluated in accordance with the codes listed in Section A.1.1.

In the course of design progress it has been determined that the weld reinforcement limit criteria given in Appendix A are not applicable to the systems requiring in-service inspection in accordance with ASME Boiler and Pressure Vessel Code, Section XI. In order to meet the requirements of ultrasonic examination for in-service inspection of welds within the primary coolant pressure boundary (Group I), the weld reinforcement heights are equal to or less than required in ANSI B31.1. In addition, the main steam lines from downstream of the outer isolation valves to the main turbine stop valves and the feedwater lines from the pump discharge to the reactor coolant pressure boundary meet the requirements of ANSI B31.1. The justification for the use of the weld reinforcement limit criteria for Group II is that there is no essential difference in the requirements specified for piping in this group between ANSI B31.7

Class II and the project design requirements as stated in Appendix A. The load combination, allowable stresses, and identification of points of highest stress are summarized in Table C.5.7.

C.5.3.3.2 Piping Seismic Analysis

Piping, 2 inch and smaller, is analyzed by one of three methods: the span chart method, the simplified static method, or the computer method. The simplified static method and the computer method are described in Appendix A, paragraph A.3.1.4. Piping, 2-1/2 in and larger, is dynamically analyzed by the response spectrum method described in paragraph A.3.1.4 of Appendix A. 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 acceleration response spectra for the applicable floor elevations. For the design earthquake, 0.5 percent damping is used. For the maximum credible earthquake, 0.5 percent damping is used for Unit 2 and 1.0 percent damping is used for Unit 3. The increased flexibility of the curved segments of the piping systems was also considered. Except for the 1997 re-analysis of the Recirculation system piping, and the Residual Heat Removal and Reactor Water Clean-up piping inside primary containment for Peach Bottom NCR 97-02267, the results of earthquakes acting in the X and Y (vertical) directions simultaneously, and Z and Y directions simultaneously 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 thus obtained was combined with the results produced by other loading conditions to compute the resultant stresses.

Some Torus attached piping were seismically analyzed based on the response spectra at the piping center of mass although other pipe portions are located and supported from higher elevations. An analysis of piping stress calculation S/11187/D-68, which represents the worst case, or enveloping condition of all configurations per ECR 01-00077, was performed by Bechtel Corporation. The analysis was performed using NRC approved code case N-411 with an envelope response spectra of all applicable elevations. The re-analysis results supported acceptability of the original center of mass methodology. Based on results of the analysis, no re-analysis of other calculations are required using actual floor elevations. Therefore, those stress calculations which utilized center of mass spectra are concluded to be acceptable and will continue to utilize this approach in future re-analyses. Those calculations are for piping stress analyses of systems 10 (RHR), 12 (RWCU), 13 (RCIC), 14 (CS), and 23 (HPCI) and their numbers are S/11187/D-050, S/11187/D-051, S/11187/D-053, S/11187/D-059, S/11187/D-060, S/11187-015/D-064, S/11187/D-067, S/11187/D-068, S/11187/D-070, S/11187/D-072, S/11187/D-073, S/11187/D-077, S/11187/D-080, S/11187/D-096, 10-39, 10-41, 23-04, and 23-9.

The 1997 re-analysis of the Recirculation system piping, and the Residual Heat Removal and Reactor Water Clean-up piping inside the primary containment for Peach Bottom NCR 97-02267 combined the peak collinear contributions due to the three spatial components of seismic excitation by the square root-sum of the squares (SRSS) method as required by the application of ASME Code Case N-411-1. In this method, separate analyses are conducted corresponding to the three spatial components (two horizontal, one vertical) of seismic excitation resulting in an analysis of a three (3) dimensional earthquake.

As indicated in paragraph C.1.2 and Appendix A, each main steam line up to and including the main steam line isolation valve external to the primary containment is seismic Class I. The main steam line anchor is in the seismic Class I portion of the main steam line and, therefore, does not separate the seismic Class I part of the main steam line from the seismic Class II part. Additional analysis of the main steam lines from outer main steam isolation valves up to but not including the turbine stop valves indicates that because they are restrained from the steam tunnel to the turbine, including restraint for fast valve closure, the lines will meet the stress requirements of seismic Class I piping as defined in Appendix A. Additional restraints are provided for the seismic Class II portions of the main steam lines to protect the adjacent seismic Class I piping in the pipe tunnel. The design methods and design stress criteria are similar to those provided for Monticello Unit 1, AEC Docket No. 50-263.

The main steam line anchor at the penetration adapter was designed to resist dead load, thermal loads, and design earthquake loads within normal AISC code allowable stresses. The anchor was also designed to withstand dead load, thermal loads, MCE, and the design accident loads within allowable stresses as discussed in paragraph C.3.2.

All seismic restraints and snubbers were located after a dynamic analysis determined their necessity. After the supports and restraints were installed, they were field checked to ensure compliance with the assumptions in the analysis.

C.5.3.3.3 Piping Mark I Load Analysis

Torus attached piping greater than 4-inch NPS has been dynamically analyzed using displacement time-history analysis by the modal superposition method for safety relief valve and pool swell loads.

Displacement time-histories were developed and applied at the torus nozzle for each degree of freedom (three displacements and three rotations). Mass point spacing was selected based upon a maximum significant frequency of 50 Hz. In addition, for piping close to the torus nozzle, additional mass points were selected to obtain significant responses. The damping factors used are shown in Table C.5.9.

The harmonic analysis method was used to analyze the piping systems which were subjected to condensation oscillation and chugging loads. Fifty (50) individual analyses were run for chugging for frequencies from 1 to 50 Hz. The results of each of the 50 responses were summed absolutely. Thirty individual analyses were performed for condensation oscillation for frequencies from 1 to 30 Hz. The four largest results were summed absolutely and added to the remaining twenty-six which were combined by square-root-sum-of-the-squares.

For torus attached piping 4-inch NPS and less, static analysis was performed using a dynamic load factor of 2.0. Six displacements (three translations and three rotations) were prescribed at each torus nozzle. The analysis of branch piping was similarly analyzed except the rotational displacements were neglected as they have a translation displacement equivalent of less than 1/16 inch.

The evaluation of Mark I hydrodynamic loads is in accordance with the provisions of Section A.1.1.

C.5.3.4 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. Usually a design is developed based on extensive test and experience. The equipment was analyzed to determine equipment adequacy for earthquake loading. The equivalent static coefficients for equipment were obtained from applicable floor response spectra corresponding to the support elevations of the equipment. 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. The criteria, method of analysis, and summary of critical stresses for various equipment are included in Table C.5.8.

For existing, new, modified or replacement equipment installed at Peach Bottom, the Seismic Qualification Utilities Group (SQUG) methodology may be used in lieu of the methodology described above to verify the seismic adequacy of the equipment on a case-by-case basis. SQUG methodology is applied in accordance with the SQUG Generic Implementation Procedure (Reference 16) and implementation of the SQUG methodology shall be controlled and documented in accordance with approved procedures.

C.5.3.5 Cable Trays

Cable trays, battery racks, instrument racks, and control consoles which are by definition seismic Class I (paragraph C.1.1), considering the safety functions required, are supported or restrained to withstand, without loss of safety functions, the effects of the MCE (horizontal ground acceleration of 0.12g).

The design of the cable tray support systems is the product of extensive investigation of hanger systems. Design adequacy is verified by dynamic analysis. Battery racks are designed for static coefficients (0.24g), and the adequacy of these coefficients confirmed by dynamic analysis. Instrument racks and control consoles are dynamically analyzed and restrained for natural frequencies equal to or greater than 20 Hz.

C.5.3.6 Reactor Coolant System Supports

Recirculation piping stresses were calculated in accordance with the ASME B&PV Code, Section III, Article NB-3600, 1980 Edition, up to and including Winter 1981 addenda. The load combinations and allowables are shown in Table C.5.7. The following transients are considered in the stress analyses of the recirculation piping:

Transient	Category	Cycles
Startup/Shutdown	Normal	216
Turbine roll and increase to power	Normal	216
Loss of feedwater heater	Upset	10
Partial feedwater heater bypass	Upset	70
Scrams	Upset	180
Loss of feedwater pumps, isolation valves closed	Upset	10
Reactor overpressure with delayed scram	Emergency	1
Single SRV blowdown	Upset	8

Automatic Blowdown	Emergency	1
Hydrotest	Test	226
OBE - design	Normal/upset	50

All component supports for the recirculation piping, main steam piping (to the first anchor outside the drywell), and the remainder of the reactor coolant systems were designed to the codes in effect at the time the purchase order was placed. The design, materials, and fabrication of parts were in accordance with Power Piping Code, ANSI B31.1, and the Standard MSS-SP-58, as applicable. The suspension systems were designed in accordance with the criteria presented below. This table applies to both variable and constant support hangers and to seismic restraints. Design conditions, load combinations, and calculated stress for the recirculation system pipe whip restraints are presented in Table C.5.6.

The design of new component supports and parts for recirculation pump snubbers and new hanger clamps, in conjunction with recirculation pipe replacement, was in accordance with ASME B&PV Code, Section III, Subsection NF, 1980 Edition, up to and including Winter 1981 addenda. The materials and fabrication of parts for recirculation pump snubbers and new hanger clamps were in accordance with ASME B&PV Code, Section III, Subsection NF, 1980 Edition, up to and including Winter 1980 addenda for Peach Bottom 2, and Section III, Subsection NF, 1980 Edition, up to and including Winter 1981 addenda for Peach Bottom 3.

Ambient Conditions

Temperature	70°F (prior to initial startup) 135°F normal/150°F maximum (during operation and shutdown)
Relative Humidity	40% (during operation) 95% (during shutdown)
Radiation	100 Rads/hr (3.5×10^7 R/40 yr)

Load Combinations	Primary Membrane Stress Limits
Weight + Thermal Expansion, Design Earthquake	S(1)

Weight + Thermal $0.9\ S_y^{(3)}$

DESIGN OF PEACH BOTTOM 2 AND 3 RECIRCULATION PUMP SNUBBERS

New piping supports installed in conjunction with the recirculation pipe replacement program are designed in accordance with Subsection NF of the ASME B&PV Code, Section III. Supports are either designed by load rating, per Subsection NF-3260, or to the stress limits for linear supports, per Subsection NF-3231. To avoid buckling in the component supports, Appendix F of the ASME B&PV Code requires that the allowable loads be limited to two-thirds of the critical buckling loads. The critical buckling loads that are more severe than normal, upset, and emergency loads, are determined by the supplier, using the methods discussed in Appendix F of the ASME B&PV Code. In general, the load combinations used for the design of component supports correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports, since no fatigue evaluation is necessary to meet the ASME B&PV Code requirements.

Stresses in the snubber component supports under normal, upset, emergency, and faulted loads are calculated. These calculated stresses are then compared against the allowable stresses of the material, as given in the ASME B&PV Code, Section III, to make sure that they are below the allowable limits.

C.5.3.7 References

1. NEDC-32183P, "Power Rerate Safety Analysis Report for Peach Bottom 2 & 3", May 1993

2. GE NE-123-E239-1292, REV 1, "Peach Bottom Atomic Power Plant Units 2 and 3 105% Power Rerate Evaluation of Main Steam and Recirculation Piping," Class II, Dec 1993.

3. NEDC-32230P, "Peach Bottom Power Rerate Project Engineering Report" Class III, March 1994.

4. GE-NE-523-61-0493, "Fatigue Evaluation of the Peach Bottom II and III Reactor Vessels," May 1993.

5. GE Letter, GENE B13-01805-73, Oct. 18, 1996.

6. GE Letter, WFW 9607, Nov. 13, 1996.

7. Deleted

8. 23A4065, Peach Bottom Unit 2, GE Design Specification for Recirculation System Piping Section XI Replacement (PECO Document Number M-1-U-499)

9. 23A4608, Peach Bottom Unit 3, GE Design Specification for Recirculation System Piping Section XI Replacement (PECO Document Number M-1-U-502)

10. 23A4086, Peach Bottom Unit 2, GE Certified Design Report for Recirculation System Piping and Equipment Loads, Loop A (PECO Document Number M-1-U-504)

11. 23A4645, Peach Bottom Unit 2, GE Certified Design Report for Recirculation System Piping and Equipment Loads, Loop B (PECO Document Number M-1-U-505)

12. 23A8026, Peach Bottom Unit 3, GE Certified Design Report for Recirculation System Piping and Equipment Loads, Loop A (PECO Document Number M-1-U-506)

13. 23A8027, Peach Bottom Unit 3, GE Certified Design Report for Recirculation System Piping and Equipment Loads, Loop B (PECO Document Number M-1-U-507)

14. NEDC-33064P Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 & 3 Thermal Power Optimization.

15. NEDC-32938P Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization.

16. Seismic Qualification Utilities Group (SQUG) Generic Implementation Procedure (GIP) Revision 3A.

17. G-080-VC-313, "Project Task Report, Peach Bottom Atomic Power Station Unit 2 and 3, SIL 636 Evaluation," GE-NE-0000-011-4483, Rev. 0, Class III, March 2003.

18. "Project Task Report, Peach Bottom Atomic Power Station Unit 2 and 3, Extended Power Uprate task T0400: Containment System Response," GE-0000-0130-9920-R1, Rev. 1, August 2012.

19. G-080-VC-411, T0302, RPV- Stress and Fatigue Evaluation, PBAPS Units 2 and 3.

20. PEAM-EPU-128, T0302, RPV - Environmentally Assisted Fatigue Analysis (SIA), PBAPS Units 2 and 3

21. PEAM-EPU-9, T0303, RPV Internals Mechanical Evaluation for EPU, PBAPS Units 2 and 3

22. G-080-VC-423; NEDC-33566P, Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate, Revision 0, September 2012

23. PEAM-EPU-130, RSD Disposition on GEH Task Evaluations, PBAPS Units 2 and 3

24. PEAM-EPU-29, T0400 Containment System Response, PBAPS Units 2 and 3

25. Calculation PS-1042, "Development of Primary Structure N-S and E-W Seismic Models"

26. Calculation PS-1043, "Development of Primary Structure N-S and E-W Taft In-Structure Spectra for MSL Piping Analysis"

27. Calculation 1-31, "Piping Stress Analysis for Main Steam Line A Inside Containment and HPCI Line"

28. Calculation 1-32, "Piping Stress Analysis for Main Steam Line B Inside Containment and HPCI Line"

29. Calculation 1-33, Piping Stress Analysis for Unit 2 Main Steam Line C Inside Containment and HPCI Line"

30. Calculation 1-34, "Piping Stress Analysis for Main Steam Line D Inside Containment and HPCI Line"

31. Calculation 1-35, "Piping Stress Analysis for Unit 3 Main Steam Line C Inside Containment and HPCI Line"

32. Calculation 1-36, "Piping Stress Analysis for Unit 3 Main Steam Line A Inside Containment and HPCI Line"

33. Calculation 1-37, "Piping Stress Analysis for Unit 3 Main Steam Line B Inside Containment and HPCI Line"

34. Calculation 1-38, "Piping Stress Analysis for Unit 3 Main Steam Line D Inside Containment and HPCI Line"

35. Calculation PM-1164, Peach Bottom Environmentally Assisted Fatigue Screening.

TABLE C.5.1

DEFORMATION LIMIT

Either One of (Not Both)	General Limit
a. $\left[\dfrac{\text{Permissible Deformation},\ DP}{\text{Analyzed deformation causing loss of function},\ DL}\right]$	$\leq \dfrac{0.9}{SF_{min}}$
b. $\left[\dfrac{\text{Permissible Deformation},\ DP}{\text{Experimental deformation causing loss of function},\ DE}\right]$	$\leq \dfrac{1.0}{SF_{min}}$

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*

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*"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 procedure produces conservative and safe designs. Examples where deformation limits apply are: CRD alignment and clearances for proper insertion, core support deformation causing fuel disarrangement, or excess leakage of any component.

TABLE C.5.2

PRIMARY STRESS LIMIT

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

where:

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

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* See NOTES, Table C.5.3.

TABLE C.5.3

BUCKLING STABILITY LIMIT

Any One of (No More than One Required) — General Limit

a. $$\left[\frac{\text{Permissible Load, LP}}{\text{Code normal event permissible load, PN}}\right] \leq \frac{2.25}{SF_{min}}$$

b. $$\left[\frac{\text{Permissible Load, LP}}{\text{Stability Analysis Load, SL}}\right] \leq \frac{0.674}{SF_{min}}$$

c. $$\left[\frac{\text{Permissible Load, LP}}{\text{Ultimate Buckling Collapse Load from Test, SE}}\right] \leq \frac{1.0}{SF_{min}}$$

where:

NOTES

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

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.

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 ASME Section III or its equivalent. The “lower bound limit load” is here defined as that produced from the analysis of an ideally plastic (non-strain

TABLE C.5.3 (Continued)

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 uniaxial case.

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

EP = Elastic plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress 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,

TABLE C.5.3 (Continued)

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.5.4

FATIGUE LIMIT

Summation of mean fatigue[1] damage usage including emergency or fault 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

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(1) Fatigue failure is defined here as a 25 percent area reduction for a load carrying member which is required to function, whichever is more limiting. In the fatigue evaluation, the methods of linear elastic stress analysis may be used when the 3S range limit of ASME Section III has been met. If 3S is not met, account will be taken of (a) increases in local strain concentration, (b) strain ratcheting, and (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.5.5

MINIMUM SAFETY FACTORS

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

KEY:

N = Normal loads

U = Upset loads (result in maximum system pressure)

A_D = Design earthquake

A_m = MCE

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.

NOTE: The minimum safety factor decreases as the event probability diminishes, and if the event is too improbable (incredible: $P_{40} < 10^{-6}$), then the safety factor is appropriate or required.

TABLE C.5.6

REACTOR VESSEL AND REACTOR VESSEL INTERNALS

Criteria	Loading	Primary Stress Type	Allowable Stress	Calculated 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 SA 302 - Gr. B	Normal and upset condition load 1. Design earthquake 2. Design pressure	Membrane and bending	40,000 psi	29,700 psi
For normal and upset condition Stress limit = 1.5x26,700 = 40,000 psi	Emergency condition load 1. MCE 2. Design pressure	Membrane and bending	60,000 psi	32,200 psi
For emergency condition Stress limit = 1.5x40,000 = 60,000 psi	Faulted condition loads 1. MCE 2. Jet reaction forces 3. Design pressure	Membrane and bending	80,000 psi	35,100 psi
For faulted condition Stress limit = 2.0x40,000 = 80,000 psi				
Vessel Support Skirt				
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sect. III, defines stress limit for SA 302, Gr. B	Normal and upset condition loads 1. Dead weight 2. Design earthquake	General membrane	26,700 psi	6,600 psi
For normal and upset condition S_m = 26,700 psi	Emergency condition loads 1. Dead weight 2. MCE	General membrane	40,000 psi	8,100 psi
For emergency condition S_{limit} = 1.5 S_m = 1.5x26,700 = 40,000 psi For faulted condition S_{limit} = 2.0 S_m = 2.0x26,700 = 53,000 psi	Faulted condition loads 1. Dead weight 2. MCE 3. Jet reaction forces	General membrane	53,400 psi	11,900 psi

* These results represent original plant design. Analysis of power rerate conditions is documented in References 1 and 2. Analysis for Thermal Power Optimization is presented in Reference 14 and 15

TABLE C.5.6 (Continued)

Criteria	Loading	Primary Stress Type	Allowable Stress	Calculated Stress
Shroud Leg Support				
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sect. III, defines allowable primary membrane stress SB-168 material 1. Tensile Loads For normal and upset condition S_m = 23,300 psi	Normal and upset condition loads 1. Design earthquake 2. Pressure drop across shroud (normal) 3. Subtract deadweight	Tensile	23,300 psi	12,600 psi
For emergency condition S_{limit} = 1.5 S_m = 1.5x23,300 = 35,000 psi	Emergency condition loads 1. MCE 2. Pressure drop across shroud (normal) 3. Subtract dead weight	Tensile	35,000 psi	22,900 psi
For faulted condition S_{limit} = 2.0 S_m = 2.0x23,300 = 46,600 psi	Faulted condition loads 1. MCE 2. Pressure drop across shroud during faulted condition 3. Subtract dead weight	Tensile	46,600 psi	28,200 psi
2. Compressive Loads For normal and upset conditions S_A = 0.4 S_y = 0.4x35,000 = 14,000 psi	Normal and upset condition loads 1. Design earthquake 2. Zero pressure drop a across shroud 3. Dead weight	Compressive	14,000 psi	12,500 psi
For emergency condition S_{limit} = 0.6 S_y = 0.6x 35,000 = 21,000 psi	Emergency condition loads 1. MCE 2. Subtract operating pressure drop across shroud 3. Dead weight	Compressive	21,000 psi	17,000 psi
For faulted condition S_{limit} = 0.8 S_y = 0.8x 35,000 = 28,000 psi	Faulted condition loads 1. MCE 2. Zero pressure drop across shroud 3. Dead weight	Compressive	28,000 psi	26,800 psi

TABLE C.5.6 (Continued)

Criteria	Loading	Primary Stress Type	Allowable Stress	Calculated 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. Design earthquake 2. Weight of structure 3. Weight of temporary control curtain	General membrane plus bending	24,000 psi	21,600 psi
For normal and upset condition Stress Intensity $S_A = 1.5\ S_m = 1.5x16,000$ psi = 24,000 psi For emergency condition $S_{limit} = 1.5\ S_A = 1.5x24,000$ psi = 36,000 psi	Emergency condition loads 1. MCE 2. Weight of structure 3. Weight of temporary control curtains	General membrane plus bending	36,000 psi	30,700 psi
For faulted condition $S_{limit} = 2S_A = 2x24,000$ = 48,000 psi	Faulted condition loads (same as emergency condition)	General memberane plus bending	48,000 psi	30,700 psi
Top Guide Beam End Connections				
Primary Stress Limit - ASME Boiler and Pressure Vessel Code, Sec. III, defines material stress limit for type 304 stainless steel.	Normal and upset condition loads 1. Design earthquake 2. Weight of structure 3. Weight of temporary control curtains	Pure shear	9,600 psi	9,100 psi
For normal and upset condition Stress Intensity $S_A = 0.6\ S_m$ = 0.6x16,000 psi = 9,600 psi. For emergency condition $S_{limit} = 1.5\ S_A = 1.5x9,600$ psi = 14,000 psi	Emergency condition loads 1. MCE 2. Weight of structure 3. Weight of structure control curtains	Pure shear	14,400 psi	14,400 psi
For faulted condition $S_{limit} = 2S_A = 2x9,600$ psi = 19,200 psi	Faulted condition loads (same as emergency condition)	Pure shear	19,200 psi	14,400 psi

TABLE C.5.6 (Continued)

Criteria	Loading	Primary Stress Type	Allowable Stress	Calculated 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. Design earthquake	General membrane plus bending	24,000 psi	15,250 psi
For allowable stresses see top guide, longest beam, above.	Emergency condition loads 1. Normal operation pressure drop 2. MCE	General membrane plus bending	36,000 psi	22,500 psi
	Faulted condition loads 1. Pressure drop after recirculation line rupture 2. MCE	General membrane plus bending	48,000 psi	26,500 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. Design earthquake	Pure shear	9,600 psi	0
For allowable shear stresses, see top guide beam end connections above.	Emergency condition load 1. MCE	Pure shear	14,400 psi	12,000 psi
	Faulted condition load 1. MCE	Pure shear	19,200 psi	12,000 psi

TABLE C.5.6 (Continued)

Criteria	Loading	Primary Stress Type	Moment Limit Accounting for Pressure Loads	Maximum Moment
Fuel Channels				
Primary Stress Limit - Allowable stress S_m for Zircaloy determined according to methods recommended by ASME Boiler and Pressure Vessel Code, Sect. III. Allowable moment determined by calculating limit moment using Table C.3.2, equation (b), then applying SF_{min} for applicable loading conditions.	Normal and upset condition load 1. Design earthquake 2. Normal pressure load	Membrane and bending	28,230 in-lb	6,927 in-lb
	Emergency condition load 1. MCE 2. Normal pressure load	Membrane and bending	42,350 in-lb	16,625 in-lb
(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 = σ yield.	Faulted condition load 1. MCE 2. Loss of cooling accident pressure	Membrane and bending	56,500 in-lb	16,625 in-lb

TABLE C.5.6 (Continued)

Criteria	Loading	Location	Allowable Stress	Calculated Stress
RPV Stabilizer				
Primary Stress Limit AISC specification for the construction, fabrication, and erection of structural steel for buildings	Upset condition 1. Spring preload 2. Design earthquake	Rod	136,000 psi	f_t = 62,400 psi*
		Bracket	22,000 psi	f_b = 15,500 psi
			14,000 psi	f_v = 4,600 psi
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads	Emergency condition 1. Spring preload 2. MCE	Bracket	33,000 psi	f_b = 20,000 psi
			21,000 psi	f_v = 6,000 psi
For emergency conditions 1.5 x AISC allowable stresses For faulted conditions Material yield strength	Faulted Condition 1. Spring preload 2. MCE 3. Jet reaction load	Bracket	36,000 psi	f_b = 21,500 psi
			21,500 psi	f_v = 6,400 psi
*The ratio maximum stress/stress limit is highest for upset loading conditions				
RPV Support (Ring Girder)				
Primary Stress Limit AISC specification for the design fabrication and erection of structural steel for buildings	Normal and upset condition 1. Dead loads 2. Design earthquake 3. Loads due to scram	Top Flange	27,000 psi	f_b = 15,000 psi
		Bottom Flange	27,000 psi	f_b = 14,100 psi
		Vessel to girder bolts	60,000 psi	f_t = 35,200 psi
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads			22,500 psi	f_v = 4,450 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. MCE 3. Jet reaction Load	Top Flange	45,000 psi	f_b = 39,500 psi
		Bottom Flange	45,000 psi	f_b = 37,500 psi
		Vessel to girder bolts	125,000 psi	f_t =115,000 psi
			75,000 psi	f_v = 11,800 psi

TABLE C.5.6 (Continued)

Criteria	Loading	Location	Allowable Stress	Calculated Stress
CRD Housing Support				
Primary Stress Limit AISC specification for the design, fabrication and erection of structural steel for buildings	Faulted condition loads	Beams (Top cord	33,000 psi	f_a = 15,300 psi
	1. Dead weight		33,000 psi	f_b = 12,700 psi
	2. Impact force from failure of a CRD housing	Beams (bottom cord)	33,000 psi	f_a = 11,800 psi
			33,000 psi	f_b = 7,600 psi
For normal and upset condition		Grid structure*	41,500 psi	f_a = 40,000 psi
			27,500 psi	f_b = 11,100 psi

F_a = 0.60 F_y (tension)

F_b = 0.60 F_y (bending)

F_v = 0.40 F_y (shear

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

*(Dead weights and earthquake loads are very small as compared to jet force.)

Recirculating Pipe and Peach Bottom Unit 3

Pump Restraints

Information under this heading deleted.

TABLE C.5.6 (Continued)

Criteria	Loading	Location	Allowable Stress	Calculated Stress

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TABLE C.5.6 (Continued)

Criteria	Loading	Location	Allowable Stress	Calculated 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, Sect. III, for Type 304 stainless steel tubing For normal and upset conditions $S_A = 1.5\ S_m = 1.5x15{,}800 = 23{,}700$ psi For faulted condition $S_{limit} = 2.0\ S_A = 2.0x23{,}700 = 47{,}400$ psi	Faulted condition loads 1. Dead weight 2. Pressure drop across guide tube due to failure of recirculation line 3. MCE	The maximum bending stress under faulted loading conditions occurs at the center of the guide tube.	47,400 psi	7,535 psi
In-core 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 at 575°F For emergency condition $(N+A_m)$ $S_{limit} = 1.5\ S_m = 1.5x15{,}800 = 23{,}700$ psi	Emergency condition loads 1. Design pressure 2. MCE	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration.	23,700 psi	15,290 psi

TABLE C.5.6 (Continued)

Criteria	Loading	Location	Allowable Stress	Calculated Stress
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. MCE	At column to base welds	11,000 psi	9,620 psi[1]
		At base hold down lug (casting)	20,000 psi	16,600 psi[2]
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[3]			
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.

[3] 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.

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.

TABLE C.5.6 (Continued)

Results of Analysis

All criteria were met.

Analysis showed 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.5.7

LOADING COMBINATIONS AND CRITERIA FOR PEACH BOTTOM 2 AND 3
Main Steam Piping

Operating Condition / Service Level	**Design Basis Load Case Combinations**	**Code Equation**	**Stress Allowable**
Sustained	Peak Pressure + $Deadweight_{steam}$ EQN.	11	1.0 Sh (at design temperature)
Sustained	Peak Pressure + $Deadweight_{water}$ EQN.	11	1.0 Sh (at design temperature)
Occasional (Upset)	Peak Pressure + $Deadweight_{steam}$ + Design Earthquake (DE)	EQN. 12 Level B	1.2 Sh (at design temperature)
Occasional (Upset)	Peak Pressure + $Deadweight_{steam}$ + TSV Transient	EQN. 12 Level B	1.2 Sh (at design temperature)
Occasional (Upset)	Peak Pressure + $Deadweight_{steam}$ + RV Transient	EQN. 12 Level B	1.2 Sh (at design temperature)
Occasional (Emergency)	Peak Pressure + $Deadweight_{steam}$ + SRSS [Design Basis Earthquake (DE) and TSV Transient]	EQN. 12 Level C	1.8 Sh (at design temperature)
Occasional (Emergency)	Peak Pressure + $Deadweight_{steam}$ + SRSS [Design Basis Earthquake (DE) and RV Transient]	EQN. 12 Level C	1.8 Sh (at design temperature)
Occasional (Faulted)	Peak Pressure + $Deadweight_{steam}$ + SRSS [Maximum Credible Earthquake (MCE) and TSV Transient and RV Transient]	EQN. 12 Level D	2.4 Sh (at design temperature)
Thermal Expansion	Thermal Stress Range + DE Seismic Anchor Movements (SAM)	EQN. 13	S_A (S_A = f (1.25 Sc + 0.25 Sh))
Sustained + Thermal Expansion	Peak Pressure + $Deadweight_{steam}$ + Thermal Stress Range + SAM	EQN. 14	(Sh + S_A)

For the occasional load the pressure associated with the event is used to obtain stresses.

The material allowable stress Sh and Sc are based on a factor of safety of 3.5 based on Evaluation 01137961-04 (EC EVAL 388022).

The stresses for the Main Steam line inside containment for lines A, B, C, and D are documented in Reference 31 through 34 and all stresses are within the allowables per applicable codes.

TABLE C.5.7 (Continued)

STEAM TO HPCI TURBINE

Criteria	Loading	Allowable Stress	Calculated Stress
Secondary Stress The sum of the thermal expansion stress and earthquake equipment displacement stress should conform to allowable stress as given in B31.1.	1. Thermal 2. Earthquake equipment displ. stress	22,500 psi	6,617 psi
Primary Stress The sum of the longitudinal stresses due to internal pressure, dead weight, and inertia effects of a design earthquake should be less than 1.2 times the hot allowable stress.	1. Internal pressure 2. Dead weight 3. Design earthquake	18,000 psi	12,028 psi
Primary Stress The sum of the longitudinal stresses due to internal pressure, dead weight, and inertia effects of an MCE should be less than the hot yield stress.	1. Internal pressure 2. Dead weight 3. MCE	26,000 psi	17,561 psi

NOTE: 1. The calculated stress is the sum of the various maximum stresses, which do not necessarily all occur at the same point in the piping system. Therefore, comparing this calculated stress with the allowable stress is a conservative procedure.

2. The 105% rerate condition raised the thermal stress by 2% and the pressure stress by 4%. Overall stress is still well within the allowable stress.

3. Maximum thermal expansion stress for Unit 3 HPCI piping inside the torus compartment from Modification P00634 is 17,315 psi.

TABLE C.5.7 (Continued)

CORE SPRAY PUMP SUCTION

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TABLE C.5.7 (Continued)

ASME CODE CLASS 1 PEACH BOTTOM 2 RECIRCULATION PIPING - HIGHEST STRESS LOCATION

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
ASME B&PV Code Section III, NB-3600						
Design condition: Eq. 9 ≤ 1.5 S_m	Primary	24760 psi	25875 psi	0.96	1. Pressure 2. Weight 3. DE	Loop A Lug discharge (096)
Service levels A & B (normal & upset) condition: Eq. 12 ≤ 3.0 S_m	Secondary	36010 psi	51750 psi	0.70	3. Thermal expansion	Loop A Riser Tee (062)
Service levels A & B (normal & upset) condition: Eq. 13 ≤ 3.0 S_m	Primary plus secondary (except thermal expansion)	45171 psi	51750 psi	0.87	1. Pressure 2. Weight 3. DE	Loop A Disch. Riser Reducer (062)
Service levels A & B (normal and upset) condition:					1. Pressure 2. Weight 3. Thermal expansion 4. DE	Loop A Header extruded Outlet (085)
Cumulative usage factor		0.109	1.0	0.109		Loop A Disch.Riser Reducer (062) Loop B Riser Tee (085)
Service level B (upset) condition: Eq. 9 ≤ 1.8 S_m & .15 S_y	Primary	25219 psi	29388 psi	0.86	1. Pressure 2. Weight 3. DE	Loop A Lug discharge (096)
Service level C (emergency) condition:	None					

TABLE C.5.7 (Continued)

ASME CODE CLASS 1 PEACH BOTTOM 2 RECIRCULATION PIPING - HIGHEST STRESS LOCATION

Acceptance Criteria	Limiting Stress Type	Calculated Stress(1) or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
Service level D (faulted) condition:						
Eq. 9 < 3.0 S_m & 2.0 S_y	Primary	26970 psi	39184 psi	0.69	1. Pressure 2. Weight 3. MCE	Loop A Lug Discharge (096)

TABLE C.5.7 (Continued)

ASME CODE CLASS 1 PEACH BOTTOM 3 RECIRCULATION PIPING - HIGHEST STRESS LOCATION

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
ASME B&PV Code Section III, NB-3600						
Design condition:						
Eq. 9 ≤ 1.5 S_m	Primary	23962 psi	25875 psi	0.93	1. Pressure 2. Weight 3. DE	Loop A Lug discharge (181)
Service levels A & B (normal & upset) condition:						
Eq. 12 ≤ 3.0 S_m	Secondary	24073 psi	51750 psi	0.47	3. Thermal expansion	Loop A RWCU/Recirc. Tee (500)
Service levels A & B (normal & upset) condition: Eq. 13 ≤ 3.0 S_m	Primary plus secondary (except thermal expansion)	41779 psi	51750 psi	0.79	1. Pressure 2. Weight 3. DE	Loop A Recirc. Pump Suction Elbow (47)
Service levels A & B (normal and upset) condition:					1. Pressure 2. Weight 3. Thermal expansion 4. DE	Loop A RHR Tee (602)
Cumulative usage factor		0.009	1.0	0.009		Loop A RHR/Recirc. Tee (500)
Service level B (upset) condition:					1. Pressure 2. Weight 3. DE	Loop A Lug discharge (181)
Eq. 9 ≤ 1.8 S_m & .15 S_y	Primary	24400 psi	29388 psi	0.83		
Service level C (emergency) condition:	None					

TABLE C.5.7 (Continued)

ASME CODE CLASS 1 PEACH BOTTOM 3 RECIRCULATION PIPING - HIGHEST STRESS LOCATION

Acceptance Criteria	Limiting Stress Type	Calculated Stress[1] or Usage Factor	Allowable Limits	Ratio Actual/ Allowable	Loading	Identification of Locations of Highest Stress Points - NODG Point Numbers
Service level D (faulted) condition:						
Eq. 9 < 3.0 S_m & 2.0 S_y	Primary	29261 psi	39184 psi	0.75	1. Pressure 2. Weight 3. MCE	Loop A Suction Elbow (045F)

TABLE C.5.7 (Continued)

LOADING COMBINATIONS AND CRITERIA FOR PEACH BOTTOM 2 AND 3
RECIRCULATION ASME B&PV CODE CLASS 1
PIPING AND COMPONENTS

The loading combinations given below were considered and the calculated stresses are reported for the governing load.

Event[1]	ASEM B&PV Code Service Limit
P_D + W + DE	Design
N	A
N + OBE + SOT	B
None	C
P_P + W + MCE	D

[1] Key to load definitions:

P_D = Design Pressure

W = Weight
N = Normal load consisting of pressure, dead weight, and thermal loads.
DE = Design earthquake
SOT = Systems operating transients

MCE = Maximum credible earthquake
P_P = Peak system operating pressure

TABLE C.5.8

EQUIPMENT

MAIN STEAM ISOLATION VALVES

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
1. Body Minimum Wall Thickness Loads: Design pressure & temperature Primary Membrane Stress Limit: S = 7,000 lb/in^2 per ASA B16.5	Minimum wall thickness in the cylindrical portions of the valve shall be calculated using the following formula: $t = 1.5\left[\frac{Pd}{2S - 1.2P} + C\right]$ where: S = allowable stress of 7,000 psi P = primary service pressure, 655 psi d = inside diameter of valve at section being considered, in C = corrosion allowance of 0.12 in	t = 1.875 in	Body wall thickness t = 1.83 at 23" diam
2. Cover Minimum Thickness Loads: Design pressure & 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, in d = diameter or short span, in C = attachment factor S = allowable stress, psi W = total, bolt load, lb h_G = gasket moment arm, in C_1 = corrosion allowance, in	t = 5.469 in	Valve cover thickness and stress t = 4.888 in S_{allow} = 17,800 lb/in^2

TABLE C.5.8 (Continued)

MAIN STEAM ISOLATION VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
3. Cover Flange Bolt Area: Loads: Design Pressure & temperature Gasket load Stem operational load Seismic load - maximum credible earthquake Bolting Stress Limit: Allowable working stress per ASME Nuclear Pump & Valve Coe, Class I	Total, bolting loads and stresses shall be 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 shall be included in the total load carried by bolts. The horizontal and vertical seismic forces shall be applied at the mass center of the valve operator assuming that the valve body is rigid and anchored.	S = 30,900 lb/in^2 at 575°F	Flange Bolt Area & Stress A_b = 55.29 in^2 S_b = 15,400 lb/in^2
4. Body Flange Thickness & Stress Loads: Design pressure & temperature Gasket load Stem operational load Seismic load - maximum credible earthquake 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 shall be 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 shall be included in the total load carried by the flange. The horizontal and vertical seismic forces shall be applied at the mass center of the valve operator assuming that the valve body is rigid and anchored.	S = 26,700 lb/in^2 S = 26,700 lb/in^2 S = 26,700 lb/in^2	Body Flange Thickness and Stress for t = 4 in S_H = 25,520 lb/in^2 S_R = 14,120 lb/in^2 S_T = 4,900 lb/in^2
5. Valve Disc Thickness Loads: Design pressure & temperature lb/in^2	$S_r = S_t = \frac{3\ (3 + \nu)\ PR^2}{8\ t^2}$ where: S_r = radial stress, psi S_t = tangential stress	S = 17,800 lb/in^2	Valve Disc Thickness and Stress for t = 3.563 in S_r = S_t = 16,830

TABLE C.5.8 (Continued)

MAIN STEAM ISOLATION VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
Primary bending stress limit: Allowable working stress per ASME Section VIII	γ = Poisson's ratio P = design pressure, psi R = radius of disc, in t = thickness of disc, in		
6. Valve Operator Supports Loads: Design pressure & temperature Stem operational load Equipment dead weight Seismic load - maximum credible earthquake Support Rod Stress Limit: Allowable working stress per ASME Section VIII.	The valve assembly shall be 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 shall be determined for the operator support components.	S = 18,000 lb/in^2	Operator Support Stress & Deflection Combined bending & tensile stress S = 5,400 lb/in^2 Deflection at Operator S = 0.032 in

TABLE C.5.8 (Continued)

MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
1. Inlet Nozzle Wall Thickness			
Loads:	$t = \frac{PR}{SE - 0.6P} + C$		
1.1 x Design Press. at 600°F	where:		
Primary Membrane Stress Limit: Allowable stress intensity as defined by ASME Standard Code for Pumps & Valves for Nuclear Power	t = min. required thickness, in S = allowable stress, lb/in^2 P = 1.1 x design press., lb/in^2 R = internal radius, in E = joint efficiency C = corrosion allowable, in	t = 0.784 in	t = 0.183 in
2. Valve Disc Thickness			
Loads:	$S_S = \frac{W}{A} = \frac{PA_1}{A}$		
1.1 x Design Press. at 600°F	where:		
Diagonal Shear Stress Limit: 0.6 x allowable stress intensity as defined by ASME Standard Code for Pumps & Valves for Nuclear Power.	W = shear load, lb A = shear area, in^2 P = 1.1 x design press., lb/in^2 A_1 = disc area, in^2 and: A = πS (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 = 20,190 lb/in^2	S = 13,617 lb/in^2

TABLE C.5.8 (Continued)

MAIN STEAM SAFETY VALVES

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
3. Inlet Flange Bolt Area Loads: Design pressure & temperature Gasket load Operational load Maximum credible earthquake 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 shall be calculated in accordance with procedures of Para. 1-704.5.1 Flanged Joints, of B31.7 Nuclear Piping Code.	S_b = 27,700 lb/in^2	S_b = 17,297 lb/in^2
4. Inlet Flange Thickness Loads: Design pressure & temperature Gasket load Operational load Seismic load - maximum credible earthquake Flange Stress Limits: S_H , S_R , S_T 1.5 S_m per ASME Nuclear Pump & Valve Code	Flange thickness and stresses shall be calculated in accordance with procedures of Para. 1-704.5.1 Flanged Joints, of B31.7 Nuclear Piping Code.	S_H = 27,300 lb/in^2 S_R = 27,300 lb/in^2 S_T = 27,300 lb/in^2	S_H = 21,339 lb/in^2 S_R = 10,798 lb/in^2 S_T = 4,581 lb/in^2

TABLE C.5.8 (Continued)

MAIN STEAM SAFETY VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
5. Valve Spring - Torsional		Set Point	Set Point
Stress	$S_{max} = \frac{8PD}{\pi d^3}\left[\frac{4C-1}{4C-4} = \frac{0.615}{C}\right]$	S = 82,500 lb/in^2	S = 65,693 lb/in^2
Loads:		Torsional Stress Limit:	
W_1 = Set point load, lbs W_2 = Spring load at maximum lift, lb	where:	S = 112,500 lb/in^2	S = 112,500 lb/in^2
0.67 x torsional elastic limit when subjected to a load of W_1	S_{max} = torsional stress, lb/in^2 P = W_1 or W_2 = spring load, lb D = mean diameter of coil, in d = diameter of wire, in C = $\frac{D}{d}$ = correction factor		
0.90 x torsional elastic limit when subjected to a load of W_2.			
6. Yoke Rod Area	$A = \frac{F}{2s_m}$		
Loads:			
Spring load at maximum lift	where:		
Primary Stress Limit:	A = required area per rod, in^2 F = total spring load, lb S_m = allowable stress, lb/in^2	A = 2.67 in^2	A = 0.852 in^2
Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps & Valves for Nuclear Power			

TABLE C.5.8 (Continued)

MAIN STEAM SAFETY VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
7. Yoke Bending & Shear Stresses	$S_b = \frac{M}{Z}$, $S_s = \frac{V}{A}$	S_b = 18,200 lb/in^2 S_s = 10,900 lb/in^2	S_b = 17,932 lb/in^2 S_s = 10,900 lb/in^2
Loads: Spring load at maximum lift	where: S_b = bending stress, lb/in^2 S_s = shear stress, lb/in^2 M = bending moment, in-lb Z = section modulus, in^3 V = vertical shear, lb A = shear area, in^2		
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.			
8. Body Minimum Wall Thickness	$t = 1.5\left[\frac{Pd}{2S - 1.2P} + C\right]$		
Loads: Primary Service pressure		t = 0.562 in	Body Bowl t = 0.3312 in
Primary Stress Limit: Allowable stress, 7,000 lb/in^2, in accordance with ASA B16.5.	where: P = primary service pressure, 150 lb/in^2 t = required thickness, in S = allowable stress, 7,000 lb/in^2 d = inside diameter of valve at section being considered, in	t = 1.224 in t = 0.562 in	Inlet Nozzle t = 0.231 in Outlet Nozzle t = 0.2823 in

TABLE C.5.8 (Continued)

MAIN STEAM SAFETY VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
9. Inlet Nozzle Combined Stress Loads: Spring load at maximum lift Operational load Seismic load - maximum credible earthquake 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, lb/in^2 F_1= maximum spring load, lb F_2= vertical component of reqction 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 seismic force, in-lb	S = 27,300 lb/in^2	S = 5,997 lb/in^2
10. Spindle Diameter Loads: 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, lb/in^2 I = moment of inertia, in^4 L = length of spindle in compres-sion, in	Actual Load F = 110,383 lb	Load Limit ($0.2F_c$) F = 30,210 lb

TABLE C.5.8 (Continued)

MAIN STEAM SAFETY VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
11. Spring Washer Shear Area Loads: 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, lb/in^2 F = spring load, lb A = shear area, in^2	S_s = 15,960 lb/in^2	S_s = 2,312 lb/in^2

TABLE C.5.8 (Continued)

MAIN STEAM RELIEF VALVES

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
1. Body Minimum Wall Thickness	$t = 1.5\left[\frac{Pd}{2S - 1.2P}\right] + C$		Main Body:
Loads:			
Design pressure & temperature		t = 1.47 in	t = 0.625 in
Primary Membrane Stress Limit:	where:		Bonnet:
Allowable working stress as defined by USAS B16.5 (7,000 psi at primary service pressure).	t = minimum required thickness, in S = allowable stress, 7,000 lb/in^2 P = primary service pressure, 655 d = inside diameter of valve at section being considered, in C = corrosion allowance, 0.12 in	t = 0.312 in	t = 0.287 in
2. Bonnet Cap & Pilot Base Minimum Thickness	$t = d\left[\frac{CP}{S_m} + \frac{1.78\ Wh_G}{S_m d^3}\right]^{1/2} + C_1$		
Loads:			Bonnet Cap:
Design pressure & temperature		t = 1.0 in	t = 0.612 in
Gasket Load Primary Stress Limit:	where:		Pilot Base:
Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power.	t = minimum required thickness, in d = diameter or short span, in C = attachment factor, ASME Section VIII P = design pressure, lb/in_2 S_m = allowable stress, lb/in^2 W = total bolt load, lb h_G = gasket moment arm, in C_1 = corrosion allowance, 0.12 in	t = 2.219 in	t = 2.117 in

TABLE C.5.8 (Continued)

MAIN STEAM RELIEF VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Design Dimension	Calculated Stress or Minimum Dimension Required
3. Flange Bolt Area - Inlet Flange, Outlet Flange, Body to Bonnet, Bonnet to Base	Total bolting loads and stresses shall be 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	A_b = 2.854 in^2
Loads:			Bonnet to Cap:
Design pressure & temperature Gasket load Operational load Maximum credible earthquake		A_b = 1.452 in^2	A_b = 0.995 in^2
			Inlet Flange:
		A_b = 13.9 in^2	A_b = 6.25 in^2
Bolting Stress Limit:			Outlet Flange:
Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps and Valves for Nuclear Power.		A_b = 12.2 in^2	A_b = 5.5 in^2
4. Flange Thickness - Inlet, Outlet, Bonnet Flanges	Flange thickness and stresses shall be calculated in accordance with procedures of Paragraph 1-704.5.1 Flanged, Joints of B31.7 Nuclear Piping Code.		Body to Base:
Loads		S_H = 26,250 lb/in^2 S_R = 26,250 lb/in^2 S_T = 26,250 lb/in^2	S_H = 24,412 lb/in^2 S_R = 17,837 lb/in^2 S_T = 7,554 lb/in^2
Design pressure & temperature Gasket load Operational load			Cap to Bonnet:
Maximum credible earthquake Flanged Stress Limits, S_H, S_R, S_T		S_H = 26,250 lb/in^2 S_R = 26,250 lb/in^2 S_T = 26,250 lb/in^2	S_H = 15,598 lb/in^2 S_R = 3,325 lb/in^2 S_T = 3,380 lb/in^2
			Inlet Flange:
1.5 S_m per ASME Nuclear Pumps and Valve Code.		S_H = 26,250 lb/in^2 S_R = 26,250 lb/in^2 S_T = 26,250 lb/in^2	S_H = 15,200 lb/in^2 S_R = 5,200 lb/in^2 S_T = 8,600 lb/in^2
			Outlet Flange:
		S_H = 26,250 lb/in^2 S_R = 26,250 lb/in^2 S_T = 26,250 lb/in^2	S_H = 12,437 lb/in^2 S_R = 12,213 lb/in^2 S_T = 3,088 lb/in^2

TABLE C.5.8 (Continued)

MAIN STEAM RELIEF VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Design Dimension	Calculated Stress or Minimum Dimension Required
5. Valve Disc. Thickness & Stress Loads: Design pressure & temperature Primary Stress Limit: Allowable stress intensity, S_m, as defined by ASME Standard Code for Pumps & Valves for Nuclear Power.	$S_r = S_t = \frac{3\ (3 + \nu)\ PR^2}{8\ t^2}$ where: S_r = radial stress, lb/in^2 S_t = tangential stress, lb/in^2 ν = Poisson's ratio P = design pressure lb/in^2 R = radius of disc, in t = thickness of disc, in	S_m = 15,800 lb/in^2	Disc Stress: $S_t = S_r$ = 10,620 lb/in^2
6. Inlet Nozzle Diameter Thickness & Stress Loads: Design pressure & temperature Operational load Maximum credible earthquake 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, lb/in^2 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	S = 26,250 lb/in^2	Inlet Nozzle Stress: S = 19,289 lb/in^2

TABLE C.5.8 (Continued)

RECIRCULATION PUMPS

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
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	$t = \frac{PR}{SE - 0.6P} + C$ 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	2.75 in	2.68 in

TABLE C.5.8 (Continued)

RECIRCULATION PUMPS (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
2. Casing Cover Minimum Thickness Loads: Normal and upset condition Design pressure & temperature	$S_r = \frac{3w}{4t^2}\left[a^2 - 2b^2 + \frac{b^4(m-1) - 4b^4(m+1)\ln\frac{a}{b} + a^2b^2(m+1)}{a^2(m-1) + b^2(m+1)}\right]$ $+ \frac{3w}{2\pi t^2}\left[1 - \frac{2mb^2 - 2b^2(m+1)\ln\frac{a}{b}}{a^2(m-1) + b^2(m+1)}\right]$	= 15,075 psi	S_r = 6.40 psi
Primary Bending Stress Limit: 1.5 S_m per ASME code for Pumps and Valves for Nuclear Power Class I	$S_t = -\frac{3w(m^2-1)}{4mt^2}\left[\frac{a^4 - b^4 - 4a^2b^2\ln\frac{a}{b}}{a^2(m-1) + b^2(m+1)}\right] +$ $\frac{3w}{2\pi mt^2}\left[1 + \frac{ma^2(m-1) - mb^2(m+1) - 2(m^2-1)a^2\ln\frac{a}{b}}{a^2(m-1) + b^2(m+1)}\right]$ 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	= 15,075 psi	S_t = 5,243 psi
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 Sect. III, Class C	Bolting loads, areas, and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Appendix II.	20,000 psi 20,000 psi	Cover Flange Bolts 17,850 psi Seal Flange Bolts 17,750 psi

TABLE C.5.8 (Continued)

RECIRCULATION PUMPS (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required	
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 Sect. III, Class C	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" - ASME Section VIII, Appendix II.	9.25 in	Flange Thickness & Stress 8.9 in	
5. Pump Nozzle Membrane and Bending Stress Loads: Normal and upset condition Design pressure & temperature Piping reactions during normal operation Combined Stress Limit: 1.5 S_m per ASME code for Pumps and Valves for Nuclear Power Class I.	$S_L = \frac{\pi}{4} \frac{D^2P}{A} + \frac{M}{Z} + \frac{F}{A}$ $S_C = \frac{PD}{2t}$ $S_S = \frac{TR_O}{J}$ $S = \frac{S_L + S_C}{2} \pm \left[\left(\frac{S_L - S_C}{2}\right)^2 + S_S^2\right]^{1/2}$	28,650 psi	Max. Pump Nozzle Stresses Suction S = 16,800 psi	 Discharge 16,800 psi

TABLE C.5.8 (Continued)

RECIRCULATION PUMPS (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
	where: S_L = longitudinal stress, lb/in^2 S_C = circumferential stress, lb/in^2 S_S = torsional stress, lb/in^2 D = nozzle internal diameter, in P = design pressure, lb/in^2 A = nozzle cross section metal area, in M = maximum bending moment, in-lb F = maximum longitudinal force, lb t = nozzle wall thickness, in J = polar moment of inertia, in^4 R_O = nozzle outside radius, in T = torsional moment		
6. Mounting Bracket Combined Stress Loads: Flooded weight Maximum credible earthquake Combined Stress Limit: Yield Stress	Bracket vertical loads shall be determined by summing the equipment and fluid weights and vertical seismic forces. Bracket horizontal loads specific seismic force at mass center shall be determined by applying the of pum-motor assembly (flooded). Horizontal and vertical loads shall be applied simultaneously to determine tensile, shear and bending stresses in the brackets. Tensile, shear, and bending stress shall be combined to determine maximum combined stresses.	17,300 psi	Maximum Combined Stress Bracket #1: 16,845 psi #2&#3: 11,458 psi

TABLE C.5.8 (Continued)

RECIRCULATION PUMPS (Continued)

Criteria	Method of Analysis	Allowable Stress or Actual Dimension	Calculated Stress or Minimum Dimension Required
7. Stresses Due to Seismic Stress:	The flooded pump-motor assembly shall		Motor Bolt Tensile
Loads	be analyzled as a free body supported by constant support hangers from the	11,200 psi	8,828 psi
Loads:	pump brackets. Horizontal and vertical seismic forces shall be applied at mass		Pump Cover Bolt Tensile Stress:
Operating pressure and temperature	center of assembly and equilibrium reactions shall be determined for the		
Maximum credible earthquake	motor and pump brackets. Load, shear, and moment diagrams shall be construc-	32,000 psi	19,148 psi
Combined Stress Limit:	ted using live loads, dead loads, and calculated snubber reactions. Combined bending, tension, and shear stresses		Motor Support Barrel Combined Stress:
Yield stress	shall be determined for each major component of the assembly including motor, motor support barrel, boling, and pump casing. The maximum combined tensile stress in the cover boling shall be calculated using tensile stresses determined from loading diagram plus tensile stress from operating pressure.	22,400 psi	1,678 psi

TABLE C.5.8 (Continued)

RECIRCULATION VALVES

PB II

Criteria	Method of Analysis	Allowable Stress orDesign Dimension	Calculated Stress or Minimum Dimension Required
1. Body Minimum Wall In Pipe Run 28" Suction Valve 28" Discharge Valve Loads: Design Pressure Design Temperature Codes and Standards 1) USAS B31.1 1967 2) Manufacturers Standards Society MSS-SP.66	$t = \frac{1.5\ Pd}{2S-2P\ (1-y)} + 0.1$ where: t = minimum wall thickness, in P = design pressure, psig d = minimum diameter of flow passage, but not less than 90% of inside diameter at welding end, in S = allowable working stress, psi y = plastic stress distribution factor, 0.4	28" (Suction Valve) t = 1.940 in 28" (Discharge Valve) t = 1.940 in	28" (Suction Valve) t = 1.667 in 28" (Discharge Valve) t = 1.938 in
2. Body-to-Bonnet Bolt Area Loads 2" Equal Bypass Valve 4" Discharge Bypass Valve 22" Equalizer Valve 28" Suction Valve 28" Discharge Valve Loads: Design Pressure and Temp. Gasket Load Stem Operational Load Seismic Load	Total bolting loads and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections" except that the stem operational load and seismic loads carried by bolts. The horizontal and vertical seismic forces shall be applied at the equivalent mass center of the valve operator assuming that the valve body is rigid and anchored.	Flange Bolt Area and Stress S_{allow} = 29,000 lb/in^2 A_b (in^2) 28" suc 64.51 28" dis 64.51	A_m (in^2) S (psi) 28" suc 33.46 14,838 28" dis 39.13 17,634

TABLE C.5.8 (Continued)

RECIRCULATION VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Design Dimension	Calculated Stress or Minimum Dimension Required
Bolting Stress Limit Allowable working stress per ASME Boiler & Pressure Vessel Code Sec. VIII App. II 1968 Edition			
3. Flange Stress 2" Equal. Bypass Valve 4" Discharge Bypass Valve 22" Equalizer Valve 28" Suction Valve 28" Discharge Valve	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" except that the stem thrust load and seismic loads shall be included in the total load carried by the flange. The horizontal and vertical seismic forces shall be applied at the equivalent mass center of the valve operator assuming that the valve body is rigid.	S_H : 20,139 lb/in^2 (Hub Stress) S_R : 13,426 lb/in^2 (Radial Stress) S_T : 13,426 lb/in^2 (Tangential Stress)	Body Flange S_H S_R S_T 28"suc 14,474 9,111 6,113 28"dis 15,966 10,265 6,772
Loads: Design pressure & temperature Gasket Load Stem Operational Load Seismic Load			Bonnet Flange S_H S_R S_T 28"suc 13,160 10,969 6,826 28"dic 14,211 12,308 7,930
Codes--ASME Boiler & Pressure Vessel Code Section VIII Appendix II, 1968			
Flange Stress Limits; S , S , S : S per ASME Boiler & Pressure Vessel Code Sec. VIII App. II, 1968 Edition			

TABLE C.5.8 (Continued)

RECIRCULATION VALVES (Continued)

Criteria	Method of Analysis	Allowable Stress or Design Dimension	Calculated Stress or Minimum Dimension Required	
4. Yoke Bolts	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 equivalent mass center of the operator assembly equivalent.	S_b (allowable) = 20,000 lb/in^2	S_b, (bolt stress, lb/in^2)	S_b
28" Suction Valve 28" Discharge Valve	Using these forces, stresses and deflections are determined for the operator support components.		28" (Suction) 28" (Discharge)	1,322 6,326
Loads: Seismic load				

TABLE C.5.8 (Continued)

RECIRCULATION VALVES

PB III

Criteria	Method of Analysis	Allowable Stress or Design Dimension	Calculated Stress or Minimum Dimension Required
1. Body Minimum Wall In Pipe Run 2" Equal. Bypass Valve 4" Discharge Bypass Valve 22" Equalizer Valve 28" Suction Valve 28" Discharge Valve Loads: Design Pressure Design Temperature Codes and Standards 1) USAS B31.1 1967 2) Manufacturers Standards Society MSS-SP.66	$t = \frac{1.5\ Pd}{2S-2P\ (1-y)} + 0.1$ where: t = minimum wall thickness, in P = design pressure, psig d = minimum diameter of flow passage, but not less than 90% of inside diameter at welding end, in S = Allowable working stress, psi y = plastic stress distribution factor, 0.4	2" (Equal. Bypass Valve) t = 0.400 in 4" (Disch. Bypass Valve) t = 0.700 in 22" (Equal. Valve) t = 1.540 in 28" (Suction Valve) t = 1.940 in 28" (Discharge Valve) t = 1.940 in	2" (Equal. Bypass Valve) t = 0.253 in 4" (Disch. Bypass Valve) t = 0.405 in 22" (Equal. Valve) t = 1.520 in 28" (Suction Valve) t = 1.677 in 28" (Discharge Valve) t = 1.938 in
2. Body-to-Bonnet Bolt Area Loads 2" Equal Bypass Valve 4" Discharge Bypass Valve Loads: Design pressure & temperature Gasket Load Stem Operational Load Seismic Load	Total bolting loads and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections" except that the stem operational load and seismic loads shall be included in the total load carried by bolts. The horizontal and vertical seismic forces shall be applied at the mass center of the valve operator assuming that the valve body is rigid and anchored.	Flange Bolt Area and Stress S_{allow} = 29,000 psi A_b (in^2) 2" 1.81 4" 9.29	A_b (in^2) S_b (psi) 2" 1.68 26,883 4" 3.40 10,613

TABLE C.5.8 (Continued)

RECIRCULATION VALVES (Continued)

PB III

Criteria	Method of Analysis	Allowable Stress or Design Dimension	Calculated Stress or Minimum Dimension Required
Bolting Stress Limit Allowable working stress per ASME Boiler & Pressure Vessel Code Sec. VIII App. II 1968 Edition			
3. Flange Stress 2" Equal. Bypass Valve 4" Discharge Bypass Valve Loads: Design pressure & temperature Gasket load Stem Operational load Seismic Loads Codes--ASME Boiler & Pressure Vessel Code Section VIII App. II ASME Boiler & Pressure Vessel Code, Section VIII, App. II, 1968 Edition.	Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" except that the stem thrust load and seismic loads shall be included in the total load carried by the flange. The horizontal and vertical seismic forces shall be applied at the mass center of the valve operator assuming that the valve body is rigid.	S_H: 20,139 lb/in^2 (Hub Stress) S_R: 13,426 lb/in^2 (Radial Stress) S_T: 13,426 lb/in$_2$ (Tangential Stress)	Body Flange S_H(psi) S_R(psi) S_T(psi) 2" 10,175 5,103 9,352 4" 13,408 6,303 11,935 Bonnet Flange S_H S_R S_T 2" NA NA NA 4" NA NA NA

TABLE C.5.8 (Continued)

RECIRCULATION VALVES (Continued)

PB III

Criteria	Method of Analysis	Allowable Stress or DesignDimension	Calculated Stress or Minimum Dimension Required	
			22" Valve	28" Valve
4. A) Body & Bonnet Flange Stress B) Body Neck Wall Stress 22" Equalizer Valves 28" Suction Valves 28" Discharge Valves	Primary, secondary, and peak stresses were analyzed using finite element computer analyses. The model was verified by strain gage tests. The seismic and stem thrust loads are to be converted to an equivalent pressure and proportional stress added directly.	Primary Membrane plus bending = 23,700 psi	20,660 psi	22,770 psi
		Local Membrane = 23,700 psi	16,168 psi	17,860 psi
		Secondary Membrane plus Bending Range = 47,400 psi	98,230 psi*	48,750 psi*
Loads: Design pressure & temperature Seismic Load Stem Thrust Codes--ASME Boiler & Pressure Vessel Code Sec. III			*See ASME Code Case No. 1441.	

TABLE C.5.8 (Continued)

RECIRCULATION VALVES (Continued)

PB III

Criteria	Method of Analysis	Allowable Stress or Design Dimension	Calculated Stress or Minimum Dimension Required	
			22" Valve	28" Suc.& Disch.
5. Body-to-Bonnet Bolting 22" Equalizer Valve 28" Suction & Discharge Valve Loads: Design Pressure Design Temperature Seismic Load Stem Thrust Codes--ASME Boiler & Pressure Vessel Code Sec. III 1968 Edition	Primary, secondary, and peak stresses were analyzed using finite element computer analyses. The model was verified by strain gage tests. The Seismic and stem thrust loads are to be converted to an equivalent pressure and proportional stress added directly.	Under operating conditions 67,000 psi Maximum conditions 100,500 psi	33,900 psi 85,000 psi	42,900 psi 59,000 psi
6. Valve Operator Support Bolting 2" Equal. Bypass Valve 4" Discharge Bypass Valve 22" Equalizer Valve 28" Suction Valve 28" Discharge Valve Loads: Equipment dead weight Seismic load - Codes--ASME Boiler & Pressure Vessel Code Section VIII 1968 Edition.	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. Using these forces, stresses and deflections are determined for the yoke leg to bonnet bolts.	S_b allowable = 20,000 psi	S_b, (bolt stress, psi) 2" (Equal. Bypass) 4" (Discharge Bypass) 22" (Equal.) 28" (Suction) 28" (Discharge)	S_b 3,532 10,622 2,602 2,906 3,840

TABLE C.5.9

DAMPING FACTORS FOR MARK I LOADS

	Percent of Critical Damping — Piping Diameter	
	NPS ≤ 12 inch	NPS > 12 inch
Safety relief valve discharge loading	1	2
Pool swell, condensation oscillation, chugging	2	3