3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

The following information identifies the transients used in the design and fatigue analysis of ASME Code Class 1 components, reactor internals and component supports. Cyclic data for the design of ASME Code Class 2 and 3 components, as applicable, are discussed in Subsection 3.9.3. All transients are classified with respect to the component operating condition categories identified as normal, upset, emergency, faulted, and testing as defined in the ASME Boiler and Pressure Vessel Code, Section III, Division I. The transients specified below represent conservative estimates for design purposes only and do not purport to be accurate representations of actual transients, or necessarily reflect actual operating. procedures; nevertheless, all envisaged actual transients are accounted for, and the number and severity of the design transients, exceeds those which may be anticipated during the life of the plant.

→(EC-8458, R307)
Pressure and temperature fluctuations resulting from the normal, upset, emergency and faulted transients are computed by means of computer simulations of the Reactor Coolant System, pressurizer, and steam generators. Design transients are detailed in the design specifications. The component designer then uses the specification transient curves as the basis for design and fatigue analysis.
←(EC-8458, R307)

In support of the design of each Code Class 1 component, a fatigue analysis of the combined effects of mechanical and thermal loads is performed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code. The purpose of the analysis is to demonstrate that fatigue failure will not occur when the components are subjected to typical dynamic events which may occur in the power plant.

→(EC-8458, R307)
The fatigue analysis is based upon a series of dynamic events depicted in the respective component specifications. Associated with each dynamic event is a mechanical, thermal-hydraulic transient presentation along with an assumed number of occurrences for the event. The presentation is generally simple and straightforward, since it is meant to envelop the actual plant response. The intent is to present material for purposes of design. A best-estimate representation of the expected plant dynamic response may be simplified and exaggerated to cover a wider range of evolutions with one design transient. (A design transient could be a best estimate transient or based on one.)
←(EC-8458, R307)

Similarly, the characterization of a given dynamic event with a specific name is unimportant. Any plant dynamic occurrence with consequences which fall within the envelope associated with one of these dynamic events is by definition represented by that dynamic event. The fundamental concept is to ensure that the consequences of the normal and upset conditions which are expected to occur in the power plant are enveloped by one or more of the dynamic event portrayals in the component specifications. The number of occurrences selected for each dynamic event is considered to be conservative, so that in the aggregate, a 40 year useful life will be provided by this design process.

Design load combinations for ASME Code Class 1, 2, and 3 components are given in Subsection 3.9.3. Design loading combinations for reactor internals structures are presented in Subsection 3.9.5.2.

The principal design bases of the Reactor Coolant System (RCS) and reactor internals structures are given in Section 5.2 and Subsection 3.9.5, respectively.

Table 3.9-1 summarizes the transients used in the stress analysis of Code Class 1 NSSS components except NSSS valves (see Subsection 3.9.3.1.1.2 for NSSS valves). Additional specific component transients for the reactor coolant pumps, steam generators, reactor coolant piping, and the pressurizer are provided in Subsections 5.4.1, 5.4.2, 5.4.3, and 5.4.10 respectively. The basis for the transients is indicated, and the number of occurrences specified is to provide a system/component design that will not be limited by expected cyclic operation over the life of the plant. The number of occurrences is generally based on a once/day, once/week, once/ month, etc. type of evaluation. It is expected that the frequency of cyclic transients will be greater than design at the beginning of plant life and significantly less than design after the first year of operation with cumulative occurrences less than design values. System integrity is further assured by using conservative methods of predicting the range of pressure and temperature for the transients. The list of transients is intended to include startup and shutdown operations, inservice hydrostatic tests, emergency and recovery operations, switching operations, and seismic events. An explanatory discussion of each transient is also given. The applicable operating condition category as designated by the ASME Code Section III is also indicated in each case.

The transients listed include allowance for less severe transients, such as rod withdrawal incident or boron dilution incident. The number of transients listed are believed to be far in excess of any number and severity that can be anticipated to occur during the life of the facility.

Pressure and thermal stress variations associated with the above design transients are considered in the design of component supports as described in Subsection 3.9.3.4.

In addition to the design transients listed above and included in the fatigue analysis, the loadings produced by the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) are also applied in the design of components and support structures of the RCS. The OBE and SSE are classified as upset and faulted condition events, respectively. For the number of cycles pertaining to the OBE, refer to Subsection 3.7-3-2.

3.9.1.1.1 Non-NSSS Safety Class 1 Piping

→(EC-935, R302)

Class 1 stress analyses were performed in accordance with the ASME Boiler and Pressure Vessel Code, 1974 Edition and all addenda up to and including Summer 1975 addenda. A stress report is made in accordance with paragraphs NB 3200 and NB 3600 of ASME Section III. The safety class 1 piping analyzed is defined by line numbers as listed in Table 3.9-2 and shown on the figures indicated.

←(EC-935, R302)

All the ASME Code Class I piping listed in Table 3.9-2 is analyzed with the cyclic duty, pressure, temperature and flow transients for the normal, upset, emergency, faulted and test conditions specified in Table 3.9-3.

Design load combinations and stress limits for the above piping are given in Subsection 3.9.3.

All Code Class I piping is classified as seismic Category I and is analyzed as such. The OBE loading is considered to occur five times over the plant life with 40 cycles for each event.

One SSE event is assumed to occur for Code Class I piping for the life of the plant.

3.9.1.1.2 Non-NSSS Supplied Safety Class 1 Valves

The ASME Code Class 1 valves are designed in accordance with Article NB-3000 of ASME Section TTI. The manufacturers supplying Class 1 valves, the types of valves, and ASME Code edition and addenda are as follows:

Manufacturer	Types of Valves	ASME Code Edition/Addenda
Yarway Corp	Stainless steel, 600 lb ANSI and higher, 2 in. and smaller	1971/Winter 1973
Velan Engineering Co	Stainless steel, 600 lb ANSI and higher, 2 in. and smaller	1971/Winter 1973
→(EC-13981, R304) Anchor/Darling Co / Flowserve →(EC-13981, R304)	Stainless steel, 2-1/2 in. and larger	1971/Winter 1972
Lunkenheimer Co	Stainless steel, 2-1/2 in. and larger	1971/Winter 1972
ACF Industries (W-K-M Valve Div)	Control valves	1971/Summer 1973
Crosby Valve and Gage Co	Safety and Relief	1974/Sumner 1974
Valcor (Masoneilan)	Solenoid	1977/Summer 1977
Target Rock Corp	Solenoid	1980/Summer 1980
→(LBDCR 15-021, R309) Dresser ←(LBDCR 15-021, R309)	Safety	1974/Summer 1975

The Class 1 valves of nominal pipe size greater than four inches are designed for the applicable cyclic loading conditions shown in Table 3.9-3. The system transients for valves are supplied to the manufacturers by Non-NSSS suppliers. They in turn perform cyclic and transient analyses, in accordance with the ASME Code. Design of valves four inches and smaller is discussed in Sub-Section 3.9.3.1.2. A stress report is submitted by each manufacturer to show that the requirements of Subarticle NB-3500 are satisfied.

3.9.1.2 Computer Programs Used in Analyses

3.9.1.2.1 Non-NSSS Systems and Components

The computer programs employed in these analyses are strictly controlled as to their use and modification. Any significant change in any of the program requires reverification of the program. The QA methodology used is consistent with 10CFR50, Appendix B.

3.9.1.2.1.1 PIPESTRESS2010

a) Description

PIPESTRESS2010 is a computer program for linear elastic analysis of three - dimensional piping systems including multiple branches and closed loops. Pipes are modeled using the load-deflection relationships based on the displacement method.

b) Application

PIPESTRESS2010 has the following capabilities:

1) Stress calculation in conformance with either

   (a) American National Standards Institute B31.1 Piping Code, 1967 Edition, or

   (b) ASME Boiler and Pressure Vessel Code, Section III 1974 Edition for safety class 1, up to and including 1975 summer addenda, or

   (c) ASME Boiler and Pressure Vessel Code, 1971 Edition Section III for safety classes 2 and 3, up to and including 1972 winter addenda.

2) Static analysis for loading conditions due to pressure, applied loads, thermal expansion, dead weight, support movement, differential settlement, and seismic acceleration.

3) Frequency analysis of lumped mass model to compute natural frequencies and mode shapes.

4) Response analysis uses floor response spectra data to compute inertial forces at mass points for each vibrational mode. These forces are applied to the structure and solved as a static loading. The resulting forces and moments for individual modes are then combined.

5) Thermal transient analysis using the finite difference approximation to find thermal gradients in the pipe walls, due to step or ramp temperature changes The program determines the times during each transient when the various stress terms will be maximized, by an iterative technique.

6) Combination cases to combine components of forces, moments and deflections from independent loading conditions, using a choice of methods:

   (a) algebraic addition,

   (b) addition to absolute values,

   (c) square root of sum of squares, or

   (d) addition in the direction of a specified loading case.

7) Fatigue analysis as prescribed in ASME Section III, for code class 1 piping. The forces and stresses due to cyclic load sets are found, in order to determine the cumulative usage factor, which must not exceed 1.0 for an acceptable design.

c) Verification

PIPESTRESS2010 solutions to ASME sample problems have been compared with the solutions to the same sample problems generated by similar, independently written programs in the public domain, namely, ANSYS(1), PIPESD(2) and ADLPIPE(3). The comparison shows the PIPESTRESS2010 results to be substantially identical to results generated by the above programs and by hand calculations. The results were summarized in the Washington Public Power Supply System Nuclear Units No. 3 and 5 PSAR (Docket Nos. STN 50-508 and 509).

3.9.1.2.1.2 EQLOADFACT2423

a) Description

The purpose of this program is to develop response spectra using the time history response of the structure to specified earthquake excitations, to calculate dynamic load factors at relief valves, and to calculate the dynamic response of a piping system to combinations of time dependent forces or to combinations of time dependent accelerations. Solutions are limited to small elastic deformations resulting from a transient force or acceleration input.

b) Application

This program is used to find the dynamic response of a lumped parameter piping system to a variety of transient forces or accelerations and to furnish dynamic solutions when modal - response techniques are inadequate. Specifically, it may be used to find dynamic load factors for relief valves and for response spectra generation for main steam and feedwater lines.

c) Verification

EQLOADFACT2423 is a time - history dynamic analysis program which has the same pedigree as the Ebasco program code PLAST2267. While PLAST2267 performs a non-linear dynamic analysis, the EQLOADFACT2423 code is limited to elastic analysis solutions.
. The details of the analytic description of PLAST2267 is in Appendix 3.6B. The validation cases described in Appendix 3.6B for PLAST2267 for the elastic solution are applicable to EQLOADFACT2423.

3.9.1.2.1.3 PRTHRUST

a) Description

PRTHRUST is a modification of RELAP 3 which computes and plots force time history curves of the reaction fluid thrust loads resulting from a circumferential or longitudinal pipe break for subcooled liquid, flashing liquid and steam systems. This modification to RELAP 3 consisted of adding thrust equations to establish the fluid reaction forces acting on a ruptured pipe.

b) Application

This program is used to determine pipe rupture thrust loads resulting from breaks in Main Steam or Feedwater piping only (see Subsection 3.6.2.2.2.a).

c) Verification

RELAP 3 is an NRC approved loss of coolant accident program (see "Interim Acceptance Criteria for Emergency Core Cooling System for Light Water Power Reactor", Interim Policy Statement, Atomic Energy Commission, Federal Register, June 20, 1971, Page 12217) and as such is a verified program. Only the force equations added to the program required verification.

Predicted thrusts (forces) from PRTHRUST and a completely independent technique have been compared. PRTHRUST comparison with Moody's work (Moody, F J, "Fluid Reaction and Impingement Loads", Specialty Conference on structural design of Nuclear Power Plant Facilities", Chicago, 1973) shows that PRTHRUST predicts about the same or higher reaction thrusts. Since PRTHRUST predicts similar reaction thrusts as Moody's solution, and Moody's solution compares well with experimental data, it has been concluded that PRTHRUST accurately predicts reaction thrust values.

3.9.1.2.1.4 RAP

a) Description

RAP (Restraint Actuation Program) models segments of a piping system as a single degree of freedom system to provide an estimate of maximum loads and deflections of pipe whip restraints due to the impact of high energy piping resulting from postulated breaks.

b) Application

This program is used to determine maximum loads and deflections in pipe whip restraints due to breaks in Main Steam or Feedwater piping only (see Subsection 3.6.2.2.2.b).

c) Verification

RAP has been verified by comparing computer results and hand calculations for a given sample problem. This comparison yielded identical results.

3.9.1.2.1.5 PIPERUP

a) Description

The PIPERUP computer program performs dynamic, nonlinear, elastic-plastic analysis of three dimensional piping systems subjected to concentrated static or time history forcing functions. These forces result from fluid jet thrust at the location of postulated longitudinal or circumferential rupture of high energy piping. The program calculates support reactions, internal forces, moments and system deflections as a function of time. In addition, strains in each section of pipe which have exceeded the yield criteria are also identified.

b) Application

This program is used to determine the pipe whip potential resulting from postulated beaks in Main Steam or Feedwater piping only (see Subsection 3.6.2.2.2.c).

c) Verification

The PIPERUP program has been checked for its element matrix, hinge mechanism, support impact, integration scheme and overall logics. Verification of these items was done by either test results, a closed form solution or an independent computer program. The results are all as anticipated.

3.9.1.2.1.6 CALPLOTF2421

a) Description

The CALPLOTF2421 computer code converts the transient flow conditions calculated in a piping system by the RELAP code into transient flow forces. The fluid transient may be initiated by a pipe break or any other condition modeled by RELAP.

b) Application

The CALPLOTF2421 computer code uses a solution of the momentum equation to calculate forces on bends in a piping system. The derivation of the equations used is contained in Appendix 3.6C.

c) Verification

The CALPLOTF2421 computer code has been verified by comparisons with experimental measurements. Experimental comparisons using the RELAP and CALPLOTF computer codes are contained in Appendix 3.6C. Comparisons of the CALPLOTF computer code calculations with the transient fluid properties calculated using a method of characteristics approach are also contained in Appendix 3.6C.

3.9.1.2.1.7 PIPESHK2389

a) Description

The PIPESHK2389 computer code determines the forces developed by a shock wave traveling in the air of a piping system downstream of a fast opening relief valve discharging saturated or superheated steam.

b) Application

This computer code assumes one-dimensional, quasi-steady, adiabatic, frictionless flow in the discharge piping system downstream of the opening safety relief valve. The shock wave is assumed to travel through the air in the discharge piping. Using the equations of conservation of mass, momentum and energy, the states of the air in front of and in back of the moving shock wave are related by the Rankine - Hugonoit equations. The state conditions of the watersteam mixture downstream of the safety relief valve are determined by applying the equations of conservation of mass and energy across the relief valve. The forces on the piping system are calculatd using the same methodology described in the CALPLOTF2421 computer code (Subsection 3.9.1.2.1.6).

c) Verification

The actual methodology employed in this computer code were adapted from Reference 31. The computer code's thermodynamic and fluid flow calculations have been verified by hand calculation. The verification of the force calculations is contained in Subsection 3.9.1.2.1.6.

3.9.1.2.1.8 BPCYLNOZ (CYLNOZ)

a) Description

BPCYLNOZ is a computer program for analysis of the localized effect at the junction of a run-pipe and integrally welded pipe attachment such as a trunnion or rectangular lug. The computation of these local stresses is performed in accordance with the method outlined in Welding Research Council (WRC) Bulletin No. 107, August 1965 for circular trunnions and for rectangular lugs not subjected to Torque loadings. For lugs subjected to Torque loadings the method outlined in WRC Bulletin No. 198 is employed.

b) Application

BPCYLNOZ is capable of computing the stresses resulting from various loading conditions (such as Weight, Thermal, Seismic, etc.) and combining them in accordance with the rules and limits of ASME BPVC, Section III, Subsections NC and ND. The stresses are combined according to load type and service level as stipulated in the appropriate sections of the code. The program also accepts as input the general run-pipe stresses computed by piping analysis programs such as ADLPIPE or NUPIPE and adds them to the local stresses so that the total stress at the attachment point is evaluated and compared to the appropriate allowable stresses. The program is also capable of evaluating the effect of a reinforcing pad at the point of attachment. The program will also compute and evaluate the stresses induced in the attachment due to the same combinations of loadings.

c) Verification

BPCYLNOZ has been extensively checked by hand calculations performed using WRC Bulletins No. 107 and 198. Further hand results also verify the combination and evaluation of stresses according to the requirements of the applicable code sections.

3.9.1.2.1.9 WERCO

a) Description

WERCO is a computer program for the analysis of local stresses in shells due to external loadings. The program is based on Welding Research Council Bulletin No. 107/August 1965, updated per the March 1979 Revision of the Bulletin.

b) Application

WERCO program has the capability of calculating the stresses induced in pipe wall by thrust and moment loadings applied to trunnions or similar integral attachments. Stresses are considered in the pipe shell at the attachment-to-shell juncture in both the circumferential and longitudinal directions. Any load combinations, i.e., weight, thermal, seismic etc., can be applied simultaneously, and local stresses resulting from the combinations are then added to the general run-pipe stresses computed by piping analysis programs, such as EBASCO 2010 program, and others, and compared to the appropriate allowable stresses.

c) Verification

WERCO program was evaluated against CYLNOZ program, and against BP Technical Services, Inc. program for integral welded attachment analysis. According to the evaluation there are no differences between the three programs. For the same input data, the resultant output obtained by the three programs is the same.

3.9.1.2.1.10 Superpipe

a) Description

SUPERPIPE is a comprehensive computer program for the structural analysis and design checking of piping systems, with particular emphasis on nuclear power piping. SUPERPIPE is an Impell Corporation proprietary program.

b) Application

In addition to a wide array of static analysis options, SUPERPIPE performs dynamic response spectra analyses, force time history and acceleration time history analyses and executes on several large mainframe computers. In addition to the traditional in-phase single-level analysis method using enveloped spectra, SUPERPIPE uses a multiple level response spectral technique to analyze piping subjected to dynamic excitations which vary significantly between different anchor/support locations.

A comprehensive series of design checking options are available for Class 1 and Class 2/3 stress checking, break locations, and support load summaries.

c) Verification

SUPERPIPE was benchmarked by comparison with results published by the NRC in NUREG/CR-1677 for seven sample problems. Verification addressed the response spectrum method of dynamic analysis. The program has been thoroughly tested and verified for a comprehensive set of sample problems, including extensive comparison with ASME benchmark problems.

3.9.1.2.1.11 ME 101 PIPING ANALYSIS

a) <u>ME101</u>

<u>Description:</u>

ME101 is a finite element computer program for performing linear elastic response of piping systems. ME101 is a Bechtel Corporation proprietary program.

<u>Application:</u>

The program includes all traditional piping stress options, such as static thermal, weight, uniformly distributed loads, external loads, effective weight, SAM (Seismic Anchor Movement) analysis, and dynamic and static seismic analysis with enveloped spectrum methodology. ME101 also includes ISM (Independent Support Motion) or MRS (Multiple Response Spectrum); closely-spaced modes and modal coupling using CQC and double-sum methodologies; ISM time history analysis in the form of arbitrary support displacements or accelerations; force or pressure transient time history; ZPA (Zero Period Acceleration) analysis using static or missing mass correction; Bechtel's non-linear energy absorber analysis; direct integration time history method considering non-linear kinematic hardening support; and harmonic and steady state vibration analysis. The program can analyze and evaluate piping systems in accordance with the latest NRC requirements of 1.61, 1.92, 1.48, and ASME code cases N-411 and N-420. The program provides great flexibility in load combination, support and hanger guidance, and stress check and stress summary based on the ANSI B31.1 Power Piping Code and the ASME Section III Nuclear Class 2 and 3 code criteria (including the latest -1983).

b) ME101sp

Description:

ME101sp performs the seismic analysis of small diameter piping systems (2-inch and under) using the approved modified response spectrum method. ME101sp is Bechtel Corporation proprietary program.

Application:

This program generates a set of tables of seismic spans, support reactions, and stresses for various pipe sizes. It also performs response spectrum curve merging along with calculation of the seismic span. However, this program can be used independently for the sole purpose of merging spectrum curves, and stores the combined spectrum data for ME101 analysis. The spectrum input for ME101sp can be in the form of free, fixed, ME101, SUPERPIPE data, and other data tape formats. A neutral plot file of the raw or combined spectrum curves can be generated for plotting on RMS, Tektronix, CALCOMP, SEIKO, or any other neutral-file-compatible plotter.

c) ME101dt

Description:

ME101dt is a quasi-two dimensional transient analysis program. It is a Bechtel Corporation proprietary program.

Application:

ME101dt computes radial thermal gradients through pipe walls, as well as axial discontinuity temperature differences for input to Nuclear Class 1 code compliance analysis. Pipe wall sections and fluid layers at equal axial spacing are considered. Each fluid layer exchanges heat with the discretized wall sections as it flows by them. For each time step, the one dimensional diffusion equation is numerically solved for each wall section and the fluid layers that come in contact with it.

d) ME101cl

Description:

ME101c1 is a code compliance program that determines stress intensity and cumulative usage factor for Nuclear Class 1 power piping components per equations nine through fourteen of subsection NB-3650, ASME Boiler and pressure vessel code, Section III. It is a Bechtel Corporation proprietary program.

Application:

This program provides a summary of stress intensity, stress intensity range, and usage factor that can be incorporated in a Nuclear Class 1 piping stress report.

e) ME101ls

Description:

ME101ls computes local stresses in cylindrical shells caused by external moments and forces acting on rigid attachment of circular or rectangular shape. This is a Bechtel Corporation proprietary program.

Application:

ME101ls calculates piping stress intensities caused by internal pressures and moments in accordance with the pressure and moment stress calculations specified in equations nine and ten of ASME Section III NB-3650. This module can be used to perform stress checks and design qualification of pipe at welded attachment locations.

Validation on ME101, ME101sp, ME101dt, ME101cl, ME101ls:

Software Validation:

The ME101/ME101sp/ME101dt/ME101cl/ME101ls, software has been validated by total of the fifty-two validations problems (ME101ver1 through ME101ver52) and the eleven NUREG/CR 1677's benchmark problem (NRC1 through NRC11). The problems selected are based on their numerical sensitivities, system characteristics, unique features, available solutions, program capacities, built-in criteria, defaulted values, etc. The results have been compared against hand calculations, commercially available computer programs, and standard benchmark problems from ASME and the Nuclear Regulatory Commission.

System Validation:

During system installation and system validation of ME101/ME101sp/ME101dt/ME101cl/ME101ls, a set of problems is rerun on the new system. The results are then electronically compared to those results obtained from 9 validation problems. The test plan includes system validation to verify that the software was installed correctly and is working properly.

3.9.1.2.1.12 ME150 - Frame Analysis Program for Pipe Supports (FAPPS)

Description:

FAPPS is an interactive frame analysis program specifically developed for the analysis and design of pipe support frames, associated welds, baseplates, embedments, and local effects such as punching shear, web crippling, and local flange bending. This is a Bechtel Corporation proprietary program.

Application:

The program performs code check for AISC, ASME Section III, Subsection NF and AIJ codes for normal, upset, emergency, and faulted load conditions. FAPPS allows use of various load sets and allows algebraic, absolute, and/or SRSS combination of results for each vector within a load set, as well as each load set that is to be combined in one load set. The program addresses recent NRC and industry issues; provides margin factors for structural members, welds, baseplates with anchors and embedments. Use of this program minimizes the need for manual calculations. The program uses a screen management system that greatly eases the input process.

3.9.1.2.1.13 ME-152 - Standard Frame Analysis Program for Pipe Supports (SMAPPS)

Description:

SMAPPS combines the benefits of a structural frame analysis program with the simplicity of pre-engineered standards. This is a Bechtel Corporation proprietary program.

Application:

SMAPPS analyzes and designs commonly used standard frames for pipe supports, including associated welds and baseplates with anchors for AISC and ASME Section III, Subsection NF requirements, as well as project deflection/stiffness requirements. SMAPPS provides margin factors for frame, welds, and baseplates with anchors that minimize the need for reevaluation of pipe support due to load changes and as-built reconciliation.

3.9.1.2.1.14 ME153 - Miscellaneous Application of Pipe Supports (MAPPS)

Description:

MAPPS is an interactive program that performs various pipe support related analyses. This is a Bechtel Corporation proprietary program.

Application:

Uniform Weld Analysis - This program presents a method of determining weld size for the connecting structural member based on the approach described in "Welded Sructures" and "Solution of Design of Weldments" by O.W. Blodgett. It accepts five different types of shapes and analyzes from two to sixteen weld configurations. The program considers proper allowables, computes weld properties and minimum weld requirements, and adjusts the effective throat due to skewed angle. It also computes weld stresses and margin factors.

Non-Uniform Weld Analysis - The program presents a method of welding and analyzing any two-dimensional, non-uniform weld pattern with variable effective throat. It permits the user to model the weld as a series of joints and elements, depending upon actual "weld" information, where it can be determined that a constant effective throat over the entire weld configuration is not achieved. The program gives the user the choice of either algebraic or absolute combinations of stress components, and computes the combined stress at every joint of the weld.

Clip Angle Analysis - The program is developed to analyze stresses in the clip angle used in pin end connections. The program also analyzes welds between the clip angles and the framed member, and the clip angles and the supporting members. The program analyzes the clip angles for three forces and axial moment. It also permits variation of the angle between the framed member and supporting structure.

Non-Standard Clamp Analysis - The program analyzes and/or designs six different types of non-standard clamps. It computes the stresses in clamp-plate, stanchion and associated welds, clamp bolts, and required pretension in the bolts.

Beta Angle Check - The program determines and verifies beta angles used in programs like STRUDL and FAPPS.

Anchor Plate Analysis - the program analyzes the plate used in anchoring the process pipe and associated welds. The program can also be used to analyze intermediate plates used between two structural members.

Local Effect Evaluation - the programs analyzes W-beam to W-beam connection for local effects due to local flange bending and web crippling. It also performs the punching shear and web crippling analysis for tube and pipe members.

Bolt Spacing - The bolt spacing module analyzes the allowable pullout based on the concrete capacity and spacing for randomly located concrete anchors. Various checks are performed to ensure that a minimum spacing and a minimum edge distance are met. A plotting option is also available.

3.9.1.2.1.15 ME035 - Non-linear Baseplate Analysis

Description:

ME035 is intended for analyzing and designing baseplates of pipe supports. This is a Bechtel Corporation proprietary program.

Application:

ME035 is combination of pre-processor, SAP, and post-processor. It is capable of analyzing flexible baseplates on a geometrically non-linear foundation. The pre-processor performs geometry calculations to generate the finite element model and data sets for SAP. The SAP performs analysis execution. Post-processing reformats the results into report tables.

ME035 has a library of standard attachments and accepts any non-standard attachments, as well as multiple attachments. It accommodates welded and/or bolted conditions of the baseplate. The post-processor provides a very comprehensive output.

Validation on ME035, ME150, ME152, ME153

In the ME035/ME150/ME152/ME153 validation, the results of the above programs have been compared with hand calculations and commercially available computer programs. The values of validation output files executed on Waterford's computer were compared with the validation output files of current release of each program under strictly QA procedures of Bechtel Corporation.
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3.9.1.2.1.16 HSTA - Hydraulic System Transient Analysis

Description:

HSTA is a program that performs hydraulic system transient analysis. This is a Bechtel Corporation proprietary program.

Application:

In order to analyze one dimensional transient flow problems, the mass and momentum conservation equations are used to obtain pressure and velocity as functions of time and distance. The numerical solution of the finite difference equations is based on the method of characteristics (MOC); which is well known for its ability to analyze fast transients. The pressure and velocity at each node are then used internally by the code to calculate forcing functions.

The program HSTA models a piping system by dividing it up into equi-distant nodal points. Several nodal points having the same pipe diameter can be combined as "links". The links are then separated by boundary conditions at their junctions. However, the calculations are performed for all the nodes comprising a link. For the purpose of calculating pipe run forces, straight pipe segments between two consecutive bends are specified in the HSTA model.

3.9.1.2.1.17 GT STRUDL, PD STRUDL, E/PD-STRUDL, and MC STRUDL

Description:

STRUDL is a structural analysis design software program. The original STRUDL computer software program was developed at Massachusetts Institute of Technology (MIT) from 1965 to 1972 as as subsystem of the Integrated Civil Engineering System (ICES). Since 1972 numerous enhancements and efficiency modifications have taken place.

STRUDL is an information processing system capable of supplying the user with accurate and complete technical data for structural design. This software provides a highly user-oriented, comprehensive, state-of-the-art, and integrated general prupose structural analysis and design information processing system as a practical structural engineering design tool.

GT STRUDL is the proprietary version of STRUDL which is maintained and enhanced by Georgia Institute of Technology, Computer Aided Structural Engineering (CASE) Center. PD STRUDL is a proprietary version of STRUDL which is maintained and enhanced by Phi-DELTA, Inc. E/PD STRUDL is the proprietary version of STRUDL which is maintained and enhanced by Ebasco. MC STRUDL is the proprietary version of STRUDL which is maintained and enhanced by McDonald Douglas. GT STRUDL, PD STRUDL, E/PD-STRUDL, and MC STRUDL are based on the STRUDL material and methodologies placed in the public domain by MIT.
➜

Application:

The three types of analysis procedures applied to structures performed by STRUDL are stiffness analysis, nonlinear analysis, and dynamic analysis. Each of these procedures requires that the geometry, topology, boundary conditions, element (member and finite element) properties, material properties, and loads to be specified in some manner prior to the analysis. Analytic procedures apply to any combiniation of framed structures and continuum mechanics problems of arbitrary configuration and composition. STRUDL contains facilities for design which are broadly applicable to a wide variety of steel and reinforced concrete structures.

Verification:

GT STRUDL, PD STRUDL, E/PD-STRUDL, and MC STRUDL are vendor supplied software purchased as off the shelf computer programs. The quality assurance program for each of the STRUDL programs are based on the guidelines outlined by the nuclear industry and meets the necessary requirements for the use in safety related analysis calculations. Initial Validation/Verification of the software is performed by the developer in accordance with their QA Program. Validation of the program is again verified upon installation of the software. The results are compared to the results generated and provided by the vendor. Periodic verification of the program shall be done annually as a minimum, and prior to the use of any new software revision. The purpose of the revalidation is to assure that the program is properly operating on the installed system as expected. The validation process is intended to duplicate the software verification performed by the vendor by running the vendor supplied problems with the same acceptance criteria.

3.9.1.2.1.18 STAAD III/ISDS

Description:

STAAD III/ISDS is a full scale structural analysis and design software program. This program is supplied by Research Engineers, Inc. It is an information processing system capable of supplying the user with accurate and complete technical data for structural design. This software provides a highly user-oriented, comprehensive, state-of-the-art, and integrated general purpose structural analysis and design information processing system as a practical structural engineering design tool.

Application:

STAAD III/ISDS contains facilities for desgin which are broadly applicable to a wide variety of steel, reinforced concrete, and timber structures. Program capabilities include 2D and 3D static dynamic P-delta analysis with frame, shell, and plate finite elements. The program extends to the use of steel, concrete, and timber design materials and supports AIDC, LRFD, ACI, and AASHTO design codes.

Verification:

STAAD III/ISDS is a vendor supplied software purchased as an off the shelf computer program. The quality assurance program for STAAD III/ISDS is based on the guildelines outlined by the nuclear industry and meets the necessary requirements for the use in safety related analysis calculations. Initial Validation/Verification of the STAAD-III/ISDS software is performed by Resarch Engineers Inc. is in accordance with their QA Program. Validation of the program is again verified upon installation of the software. The results are compared to the results generated and provided by Research Engineers. Periodic verification of the program shall be done annually as a minimum, and prior to the use of any new software revision. The purpose of the revalidation is to assure that the program is properly operating on the installed system as expected. The validiation process is intended to duplicate the software verification performed by the vendor by running the vendor supplied problems with the same acceptance criteria.

→(DRN 02-1586, R12-A)

3.9.1.2.1.19 HYTRAN

Description

HYTRAN is a PC-based program for fluid transient analysis that computes the time-dependant forces, the pressures, and flow velocities in the legs of a liquid filled piping system. Starting from an initial steady state, the solution is advanced step by step in time until the user selected end time is reached. Transients may be initiated by pump start-up or trip, valve opening or closure, or by variations of pressures or flow at an exterior node in the network.

The input consists of an ASCII file describing the system configuration. The input data formatting is described in the user manual. Printed output of head and flow velocity at the ends of each leg and of the unbalanced forces acting on them is available at the user's option. Screen or printer plots of leg force or nodal pressure/flow time histories can also be created.

Application:

HYTRAN is a program for water hammer analysis for liquid filled piping systems. Transients may be initiated by pump start-up or trip, valve opening or closing, or by variations of pressures or flow at an exterior node in the network. The results can be used for time history analyses of piping systems.

Software Validation:

HYTRAN is a Sargent and Lundy proprietary program. Several sample problems were used to verify and validate various program options. The results of these sample problems were verified by comparison with the results provided by the HYTRAN analyses.

System Validation:

During system installation and system validation a set of problems were rerun on the new system. The results are then compared to those results obtained from the validation problems. The test plant includes system validation to verify that the software was installed correctly and working properly.

3.9.1.2.1.20 SYSFLO

Description:

SYSFLO is a transient hydraulic analysis computer program, which uses a fully implicated numerical technique to solve integrated mass, energy and momentum conservation equations throughout the piping system.

Application:

SYSFLO is a program used for water hammer analysis for liquid filed piping systems. Transients may be initiated by pump start-up or trip, valve opening and closing, or by variations of pressures or flow at an exterior node in the network. The results can be used for time history analyses of piping systems. SYSFLO is an MPR proprietary program used by MPR in calculation performed for Waterford 3.

Verification:

Several sample problems were used to verify and validate various program options. The results of these sample problems were verified by comparison with the results provided by the SYSFLO program.

←(DRN 02-1586, R12-A)

→(DRN 02-1586, R12-A)

3.9.1.2.1.21 PIPER

Description:

SYSFLO is a computer program used for modeling piping systems. The program calculates the volume of the piping segments based on diameter and length of the specified piping.

Application:

Piper is a program used as a preprocessor for SYSFLO (Section 3.9.1.2.1.20). PIPER is a MPR proprietary program used by MPR in calculations performed for Waterford 3.

Verification:

Several sample problems were used to verify and validate various program options. The results of these sample problems were verified by comparison with the results provided by the PIPER program.

←(DRN 02-1586, R12-A)

→(EC-935, R302)

3.9.1.2.1.22 DST/PIPESTRESS

a) Description

DST/PIPESTRESS is a group of interrelated computer programs for performing linear elastic analysis of three-dimensional piping systems subject to a variety of loading conditions. Chemical process piping, nuclear and conventional power generation piping systems may be investigated for compliance with piping codes and with other constraints on system response.

b) Application

Full feature nuclear piping analysis program

1) Stress calculation in conformance with

a. ASME Classes 1, 2 and 3 and ANSI/ASME B31.1

b. ASME B31.3

c. CODETI

d. RCC-M Classes 1 and 2

e. KTA Classes 1 and 2

f. EDF Piping Code for Composite Materials

g. EN 13480-3 European Piping Code

Code versions from 1967 to present

2) Analysis capabilities include

a. Heat transfer and thermal gradient stress

b. Fatigue analysis usage factor

←(EC-935, R302)

→(EC-935, R302)

c. Thermal stratification

d. Response cases with up to 99 independent support levels

e. Response spectra in Cartesian or cylindrical coordinates

f. Modal superposition by grouping, double sum CQC and other methods

g. Time history analysis by generalized response method

h. Selective “true” time history analysis

i. Determination of rigid cut-off frequency for time history analysis

j. Rigid mode correction for all dynamic analysis method

k. Up to 500 user-defined load and combination cases

l. Structures with 3,000 elements or more

m. Element dimensions in Cartesian, cylindrical, spherical or sloping coordinates

n. Translators from other analysis programs

c) Verification

DST, the PIPESTRESS vendor, is responsible for the design, testing and release of their computer code, and maintains their own in-house QA program in their office in Geneva, Switzerland.

The purpose of this program is to comply with quality assurance standards for software used in the nuclear industry which have been developed in various countries. In particular, it is based on the applicable requirements of

1. Title 10, U.S. Code of Federal Regulations, Part 50, Appendix B

2. ANSI/ASME NQA-1-1983, Quality Assurance Program Requirements for Nuclear Facilities

3. U.S. NRC Regulatory Guide 1.28

4. Title 10, U.S. Code of Federal Regulations, Part 21, Reporting Defects and Noncompliance

As part of the QA program, DST maintains a library of formal test problems referred to as the Verification Test Set (VTS). DST is constantly adding verification problems to the VTS. Each major release of PIPESTRESS is tested by executing the VTS. The VTS includes benchmark problems from

1. NUREG/CR-1677 – Vol. 1

2. NUREG/CR-1677-51267 Vol. 2

←(EC-935, R302)

→(EC-935, R302)

3. NUREG/CR6414-52487

4. “Guide de Validation des Progiciels de Calcul de Structure”

←(EC-935, R302)
→(EC-8435, R307)

3.9.1.2.1.23 NPLATE

Description:

As part of pipe rerouting and pipe support structural work performed for the steam generator replacement, the NPLATE computer program was used to model/qualify baseplates. NPLATE is a non-linear finite element program for analysis of rectangular baseplate assemblies. NPLATE is an AREVA proprietary program used in qualification of baseplates at Waterford 3.

Application:

The NPLATE computer program was used to qualify baseplate assemblies that required evaluation in support of the replacement of the steam generators.

Verification:

Several sample problems were used to verify and validate various program options. The results of these sample problems were verified by comparison with the results provided by the NPLATE program.

←(EC-8435, R307)
→(LBDCR 14-015, R309)

3.9.1.2.1.24 AUTOPIPE

Description:

AutoPIPE is a stand-alone computer aided engineering (CAE) program for calculation of piping stresses, flange analysis, pipe support design, and equipment nozzle loading analysis under static and dynamic loading conditions. In addition to 18 piping codes, AutoPIPE incorporates ASME, British Standard, API, NEMA, ANSI, ASCE, AISC, UBC, and WRC guidelines and design limits to provide a comprehensive analysis of the entire system.

Application:

AutoPIPE provides unique capabilities for process, power, oil and gas, nuclear, underground, offshore floating, production, storage, and offloading (FPSO) platform and subsea pipeline areas with 25 international piping codes. Advanced AutoPIPE capabilities include built-in wave loading, buried pipeline analysis, jacketed piping, dynamic loadings, orthotropic fiberglass reinforced plastic (FRP/GRP), and HDPE plastic piping analysis. It also includes thermal stratification or bowing, thermal transient, pipe/structure interaction, fluid transient with closure time and relief valve utilities, advanced load sequencing, non-linear support gaps and friction and jacketed piping. Local stress calculation to WRC 107, WRC 297, PD 5500, KHK, API 650 is available using AutoPIPE Nozzle.

Verification:

AutoPIPE quality assurance program has passed numerous nuclear and Nuclear Procurement Issues Committee (NUPIC) audits to 10 CFR 50 App. B, ISO9001, CSA N286.7-99, ASME NQA-1, and ANSI N45.2 standards. AutoPIPE Nuclear V8i provides design of critical safety pipework to ASME Class 1, 2, or 3.

←(LBDCR 14-015, R309)

3.9.1.2.2 NSSS Systems and Components

→(EC-8435, R307)

Control of all computer programs used in design at Combustion Engineering is maintained under the provisions of QADP5.2, Section 4, of the C-E Quality Assurance of Design Manual (QADM), Reference 32. In Reference 33, the NRC granted approval to the referencing of that letter (Reference 33) is satisfaction of the requirements of Criterion VII, Appendix B, 10CFR50.

←(EC-8435, R307)

3.9.1.2.2.1 Reactor Coolant System

The following subsections provide a summary of the applicable computer programs used in the structural analyses for ASME code class 1 systems, components, and supports. The summaries include individual descriptions and applicability data. The computer programs employed in these analyses have been verified in conformance with design control methods, consistent with 1OCFR50, Appendix B.

→(EC-8458, R307)

3.9.1.2.2.1.1 DELETED

3.9.1.2.2.1.2 DELETED

←(EC-8458 R307)

→(EC-1020, R307)

3.9.1.2.2.1.3 Bottom Head Penetration Reinforcement Program

←(EC-1020, R307)

This program calculates reinforcement available and reinforcement required for penetration in hemispherical heads. The technique described in paragraph NB-3332 of the ASME Boiler and Pressure Code, Section III is used.

→(EC-1020, R307)

This program is used to perform preliminary sizing and reinforcement calculations for the bottom hemispherical head in the reactor vessel.

←(EC-1020, R307)

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.4 Flange Fatigue Program BCH10102

This program computes the redundant reactions, forces, moments, stresses and fatigue usage factors in a reactor vessel head, head flange, closure studs, vessel flange, and upper vessel wall for pressure and thermal loadings. Classical shell equations are used in the interaction analysis.

➔(EC-1020, R307)

This program is used to perform the fatigue analysis of the reactor vessel flange assembly.

🡐(EC-1020, R307)

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.5 Nozzle Fatigue Program BCH10105

This program computes the redundant reactions, forces, moments and fatigue usage factors for nozzles in cylindrical shells.

This program is used to perform the fatigue analysis of reactor vessel nozzles and steam generator feedwater nozzle.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.6 Edge Coefficients BCH10026

This program calculates the coefficients for edge deformations of conical cylinder and tapered cylinders when subjected to axisymetric unit shears and moments applied at the edges.

This program is used to perform the fatigue analysis of reactor vessel wall transition.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.7 Generalized 4 x 4 BCH10124

This program computes the redundant reactions, forces, moments, stresses and fatigue usage factors for the reactor vessel wall at the transition from a thick to thinner section and at the bottom bead juncture.

This program is used to perform fatigue analysis of reactor vessel bottom head juncture.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results-with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.8 Load Transfer Program BCHEP007

This program transfers input loads to the nozzle cross-section being evaluated. For cross-sections outside the limit of reinforcement, the loads are applied and stresses calculated in one degree increments around the perimeter of the nozzle. This determines the worst load combination. For the nozzle-to-vessel juncture, the stresses are calculated in 90 degree increments, i.e., longitudinal and circumferential planes with respect to the reactor vessel.

This program is used to perform structural and faulted analysis of the reactor vessel.

In accordance with Standard Review Plan 3.9.1 Section II.2.cl the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

→(EC-1020, R307)

←(EC-1020, R307)

3.9.1.2.2.1.10 ANSYS

This is a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis. This program is in the public domain. Additional descriptive information on this program is provided in Subsection 3.9.1.2.2.2.3.

This program is used for numerous applications for all components in the areas of structural, fatigue, thermal and eigenvalue analysis.

3.9.1.2.2.1.11 Mare Island Computer Program, MEC-21/MECOL

This program is used for piping flexibility checks and to do a vibration analysis on the pressurizer heaters. This program is in the public domain and further verification is not required.

This program is used in numerous piping applications.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.12 Reinforcement Analysis of Skewed Penetrations and Non-Radial Nozzles, BC101047

This program is designed to compute the limits of compensation for penetrant openings that are non-radial or skewed to a spherical head. The program is to be used as an aid in satisfying the requirements of Section III, ASME Boiler and Pressure Vessel Code.

This program is used in the preliminary sizing and reinforcement calculations for hemispherical beads in the pressurizer.

In accordance with Standard Review Plan 3.9.1 Section 11.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.13 Primary Plus Secondary and Peak Stresses for the Pressurizer Manway, BC10324

The program is designed to compute and tabulate the primary plus secondary stresses and the peak stresses in the manway assembly. The program is to be used as an aid to satisfying the requirements of Section III, ASME Boiler and Pressure Vessel Code.

This program is used in the fatigue analysis of the pressurizer manway.

In accordance with Standard Review Plan 3.9.1 Section 11.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially) identical.

3.9.1.2.2.1.14 The Structural Analysis for Partial Penetration Nozzles, Heater Tube Plug Welds, and the Water Level Boundary of the Pressurizer Shell, BCH10301

This program (computes various analytical parameters, primary plus secondary stresses and stress intensities, peak stresses and stress intensities and the cyclic fatigue analysis with usage factors at cuts of interest. This program is utilized to satisfy the requirements of Section III, Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code.

This program is used in the fatigue analysis of partial penetration nozzles in the pressurizer and piping.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.15 Nozzle Primary Plus Secondary Stress Range Check, Peak Stress Calculation and Maximum Usage Factor Location, BCII 10192

This program is designed to combine the required stress components to develop the primary, plus secondary stress intensities and determine their maximum ranges. The program also calculates peak stresses and stress intensities and develops a usage factor guide.

This program is used in the fatigue analysis of nozzles in the pressurizer and piping.

In accordance with Standard Review Plan 3.9.1, Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical. As requested by the NRC, a tabular summary of the verification results is provided in Appendix 3.9.D.

3.9.1.2.2.1.16 A Three Variable Summation Program for Computing Thermal Stresses BC10126

The program is a three variable computer program which evaluates for each time (transient) and location along nozzle, the summation of the constant thermal stress term (K) and the product of H (thermal force) and M (thermal moment) by their respective coefficients, C_1 and C_2.

This program is used in the fatigue analysis of nozzles in the pressurizer and piping.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.17 Seal, Shell II Code, WIN 10470

This program computes stresses and deformations of axisymmetric shells for pressure and thermal loads.

This program is used in the fatigue analysis of various nozzles in the pressurizer, piping and steam generator.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.18 ICES STRUDL II, WIN 15658

General purpose, finite element program for framed structures and continuous mechanics problems. Additional descriptive information on this code is provided in Subsection 3.9.1.2.2.2.1.

This program is used in the eigenvalue analysis of piping and component internals.
In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.19 Primary Structure Interaction, BC10223

This program calculates redundant loads, stresses, and fatigue usage factors in the primary head, tubesheet, secondary shell, and stay cylinder for pressure and thermal loadings.

This program is used in the fatigue analysis of the steam generator primary structure.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.20 Tube-to-Tubesheet Weld, BC10362

This program performs a three body interaction analysis of the tube-to-tubesheet weld juncture. The program calculates primary, secondary, and peak stresses and computes range of stress and fatigue usage factors.

This program is used in the fatigue analysis of steam generator tube-to-tubesheet weld.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.21 Support Skirt Loading, BC10286

This program calculates the stresses in the conical support skirt of the steam generator for external loads.

This program is used in the structural analysis of steam generator support skirt.

In accordance with Standard Review Plant 3.9.1, Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical. As requested by the NRC, a tabular summary of the verification results is provided in Appendix 3.9D.

3.9.1.2.2.1.22 Principal Stress Program, BC10210

This program sums stresses for three load conditions and computes principal stress intensity, stress intensity range, and fatigue usage factor.

This program is used in the fatigue analysis of steam generator components.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.23 OUTRND Program, BCH 10319

This program calculates the bending stresses in an out-of-round cylinder subjected to internal pressure. The application of this program is limited to evaluation of secondary shell out-of-round deviation exceeding the ASME Code allowables.

This program is used for fabrication deviations on steam generator shells.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.24 Nozzle Load Resolution, BC10211

A special purpose program, used to calculate stresses in nozzles produced by piping loads in combination with internal pressure.

This program is used in the fatigue analysis of steam generator nozzles.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.25 Analysis of Axisymmetric Solids, BCH10311

A finite element program used to determine stresses and deformations of axisymmetric structures.

This program is used in the fatigue analysis of the steam generator secondary shell.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.26 Effective Bending Area of a Bolted Flange, ZIPPER

This program is used to determine the neutral axis in bending for the bolted flange of the steam generator support skirt.

This program is used in the structural analysis of the steam generator support skirt.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.27 CHAT 12100

A general purpose finite difference heat transfer program is used for steady state and transient thermal analysis.
This program is used in numerous thermal relaxation analysis for all components.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.28 CEFLASH-4A

This program is used to calculate transient conditions resulting from a flow line rupture in a water/steam flow system. The program is used to calculate steam generator internal loadings following a postulated main steam line break.

This program is used in a steam line break accident structural analysis.

Refer to Items 2 and 3 in the Bibliography of Section 6.3.

3.9.1.2.2.1.29 CRIB

This program is one dimensional, two phase thermal hydraulic code, utilizing a momentum integral model of the secondary flow. This program is used to establish the recirculation ratio and fluid mass inventories as a function of power level. The program is in the public domain and further verification is not required.

This program is used for determining steam generator performance.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.30 PWR Code BCH10107

This program is used for preliminary sizing of the steam generator heat transfer area. The required number of tubes and the average tube length is calculated to satisfy the specification performance and pressure drop requirements.

This program is used for determining steam generator performance.

In accordance with Standard Review Plan 3.9.1 Section II.2.c, the validity of this computer program was established by running test problems through the program and comparing the results with hand calculations. The results so obtained were substantially identical.

3.9.1.2.2.1.31 HEAT05

The HEAT05 computer program applies the finite element analysis techniques to the transient and steady state heat conduction analysis of axisymmetric solids with temperature and heat flux boundary conditions. The finite element idealization of the structure may be represented by ring elements of either triangular or quadrilateral cross-section which are interconnected along modal circles.

The solution for the temperature distribution within the structure is determined by the standard Rayleigh-Rinz procedure in which the generalized coordinates are selected as the temperatures at the nodal points of the finite element idealization. The form of the assumed temperature field within an element depends on the specific element type.

This program is used to analyze thick walled pump components subjected to internal pressure and thermal loads, and to external piping loads.

This program is based on a finite element heat transfer program developed by Professor Wilson of the University of California in Berkeley. After compilation, it was verified against both Professor Wilson's program and the theoretical solutions Wilson used to verify his program.

3.9.1.2.2.1.32 SOLIDS II

The computer program, SOLIDS II, applies the finite element analysis to axisymmetric solids subjected to either axisymmetric or non-axisymmetric distributed, concentrated, temperature loading. The finite element idealization represents the continuous structure by a system of ring elements which are interconnected at circumferential points or nodal circles.

Equilibrium equations are developed at each nodal circle. A solution of this net of equations for the unknown nodal circle displacements constitutes a solution for the system, since stresses within each element can be calculated from the appropriate nodal circle displacements.

The formulation of the element stiffness matrices assumes a linear displacement field to assure continuity between adjacent elements. For non-axisymmetric loads, which are symmetric about a plane containing the axis of revolution, the formulation expands the nodal circle displacements, the temperature distribution, and the nodal circle forces in Fourier series.

This program has been used to analyze thick walled pump components subjected to internal pressure and thermal loads, and to external piping loads.

3.9.1.2.2.1.33 BJS-BJT

The BJS-BJT analysis program is a generalized computer program developed to perform complex thermal gradient and stress analysis problems. The program logic is divided into two sections, one of which performs the thermal analysis, and one which calculates stresses due to thermal distributed and concentrated loadings. Both portions of the program use a finite element method of analysis.

The thermal analysis portion determines the steady state or transient temperature distribution throughout the structure being analyzed. These temperature distributions are presented as specific temperatures specified for each nodal point in the finite element model as a function of time for each thermal condition analyzed.

The stress analysis portion of the computer program is a general purpose three dimensional linear, elastic analysis program based on the finite element displacement method.

Loads imposed on the structure to be analyzed may consist of temperature, concentrated mechanical loads, or distributed loads such as gravity and pressure. Load cases may also consist of a linear combination of node and element type loads.

This program is used to analyze thick walled pump components subjected to internal pressure and thermal loads, and to external piping loads.

This program was verified against theoretical closed form solutions (hand calculations).

➔(EC-8458, R307)

3.9.1.2.2.1.34 WESTEMS

WESTEMS is a post-processor which is able to develop transfer function representation of transient stress states based on finite element results derived from ANSYS input. WESTEMS is capable of doing heat transfer and structural analyses covering stress, fracture and fatigue evaluations.

WESTEMS was used for the fatigue analysis 8x6 inch cone handhole components and 4x3 inch cone inspection port components of the replacement steam generators which are secondary side components.

The application of this program has been approved by the NRC (Reference 43) and has been applied in a manner consistent with its design and use.

3.9.1.2.2.1.35 TRANFLOW

TRANFLOW is a code that solves the mass, energy, and momentum conservation equations for transient thermal/hydraulic phenomena using a fully implicit backward differencing technique.

The code was used for Waterford 3 replacement steam generator analyses for determining the detailed distribution of fluid temperatures and heat transfer coefficients for SG secondary side stress analyses and the steamline break accident, determining the pressure drops across SG secondary side internal components.

🡐(EC-8458, R307)

→(EC-8458, R307)
TRANFLOW provides similar analytical approaches to CEFLASH-4B. TRANFLOW has been evaluated and benchmarked against CEFLASH-4B for its use and acceptability.
←(EC-8458, R307)

3.9.1.2.2.2 Reactor Internals, Fuel and CEDMs

The following computer programs are used in the static and dynamic analyses of reactor internals, fuel, and control element drive mechanisms (CEDMs).

3.9.1.2.2.2.1 ICES/STRUDL-II

a) Description:

The ICES/STRUDL-II computer program provides the ability to specify characteristics of problems (framed structures and three-dimensional solid structures), perform analyses (static and dynamic) and reduce and combine results.

Analytic procedures in the pertinent portions of ICES/STRUDL-II apply to framed structures. Framed structures are two or three dimensional structures composed of slender, linear members which can be represented by properties along a centroidal axis. Such a structure is modeled with joints (including support joints) and members connecting the joints. A variety of force conditions on members or joints can be specified. The member stiffness matrix is computed from beam theory. The total stiffness matrix of the modeled structure is obtained by appropriately combining the individual member stiffness.

The stiffness analysis method of solution treats the joint displacements as unknowns. The solution procedure provides results for joints and members. Joint results include displacements and reactions and joint loads as calculated from member end forces. Member results are member end forces and distortions. The assumptions governing the beam element representation of the structure are as follows: linear, elastic, homogeneous, and isotropic behavior, small deformations, plane sections remain plane, and no coupling of axial, torque, and bending. Further description is provided in Reference 5.

b) Application
→(EC-2800, R307)
The ICES/STRUDL-II program is used in the analysis of reactor internals. The program is used to obtain stiffness properties of lower support structure and upper guide structure grid beams due to transverse loads. The results of the analyses are incorporated into overall reactor vessel internals' models, which calculate the dynamic response due to seismic and loss-of-coolant accident (LOCA) conditions. These latter results, at a given time, are fed back into the grid beam models to yield dynamic stresses.
←(EC-2800, R307)

c) Verification

ICES/STRUDL-II is in the public domain and further verification is not required.

The design control measures established in the OADM (Reference 32) ensure the applicability and validity of this program, and meet Criterion II.2.a of Standard Review Plan 3.9.1.

3.9.1.2.2.2.2 MRI/STARDYNE

a) Description

The MRI/STARDYNE program uses the finite-element method for the static and dynamic analysis of two and three dimensional solid structures subjected to any arbitrary static or dynamic loading or base acceleration. In addition, initial displacements and velocities may be considered. The physical structure to be analyzed is modeled with finite elements that are interconnected by nodes. Each element is constrained to deform in accordance with an assumed displacement field that is required to satisfy continuity across element interfaces. The displacement shapes are evaluated at nodal points. The equations relating the nodal point displacements and their associated forces are called the element stiffness relations and are a function of the element geometry and its mechanical properties. The stiffness relations for an element are developed on the basis of the theorem of minimum potential energy. Masses and external forces are assigned to the nodes. The general solution procedure of the program is to formulate the total assembly stiffness matrix K and apply it to either of following equations:

$$[K] \bullet \{\delta\} = \{P\} \qquad (1)$$

$$\omega 2[m]\{q\} - [K]\{q\} = 0 \qquad (2)$$

where: $\{\delta\}$ = the nodal displacement vector

$\{p\}$ = the applied nodal forces

$[m]$ = the mass matrix

ω = the natural frequencies

$\{q\}$ = the normal modes

Equation (1) applies during a static analysis which yields the nodal displacements and finite elements internal forces. Equation (2) applies during an eigenvalue/eigenvector analysis, which yields the natural frequencies and normal modes of the structural system. Using the natural frequencies and normal modes together with related mass and stiffness characteristics of the structure, appropriate equations of motion may be evaluated to determine structural response to a prescribed dynamic load.

The finite element used to date in NSSS analyses is the elastic beam member. The assumptions governing its use are as follows: small deformation, linear-elastic behavior, plane sections remain plane, no coupling of axial, torque and bending, geometric and elastic properties constant along length of element.

Further description is provided in Reference 6.

b) Application

→(EC-2800, R307)

The MRI/STARDYNE program is used in the analysis of reactor internals. The program is used to obtain the response of the internals to prescribed seismic excitation. The structural components are modeled with beam elements. The geometric and elastic properties of these elements are calculated such that they are dynamically equivalent to the original structures. The response analysis is then conducted using both modal response spectra and modal time history techniques. Both method are compatible with the program.

←(EC-2800, R307)

c) Verification

MRI/STARDYNE is in the public domain and further verification is not required.

The design control measures established in the QADM (Reference 32) ensure the applicability and validity of this program, and meet Criterion II.2.a of Standard Review Plan 3.9.1.

3.9.1.2.2.2.3 ANSYS

a) Description

ANSYS is a general purpose finite element program with structural and heat transfer capabilities. It is described in Reference 1.

b) Application

➔(EC-2800, R307; EC-8458, R307)

For the evaluation of the CEDMS for power uprate 3716 MWt and Replacement Steam Generator, the eigenvalue and dynamic analysis capabilities of ANSYS are used to determine the CEDM natural frequencies and mode shapes and response loads due to seismic, BLPB, pump pulsation and random turbulence.

🡐(EC-2800, R307; EC-8458, R307)

➔(EC-1020, R307)

ANSYS is used for the stress analysis of the CEDM pressure housings. The finite element capabilities of the program are used to determine the primary, primary-plus-secondary stresses and peak stresses due to design normal, upset and faulted conditions and to evaluate the resulting stress intensities.

🡐(EC-1020, R307)

c) Verification

➔(EC-2800, R307)

ANSYS is in the public domain and further verification is not required. The design control measures established in the QADM (Reference 32) and Inter-Business Unit Edition Policies & Procedures Manual, Sections NSNP 3.6.1-3.6.6 (Reference 44) ensure the applicability and validity of this program, and meet Criterion II.2.a of Standard Review Plan 3.9.1.

🡐(EC-2800, R307)

3.9.1.2.2.2.4 ASHSD

a) Description

The ASHSD program uses a finite-element technique for the dynamic analysis of complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The three-dimensional axisymmetric continuum is represented as an axisymmetric thin shell, a solid of revolution, or a combination of both. The axisymmetric shell is discretized as a series of frustums of cone and the solid of revolution as triangular or quadrilateral "toroids" connected at their nodal circles.

Hamilton's variational principle is used to derive the equations of motion for these discrete structures. This leads to a mass matrix, stiffness matrix, and load vectors which are all consistent with the assumed displacement field. To minimize computer storage and execution time, the non-diagonal "consistent" mass matrix is diagonalized by adding off-diagonal terms to the appropriate diagonal terms. These equations of motion are solved numerically in the time domain by a direct step-by-step integration procedure.

The assumptions governing the axisymmetric thin shell finite element representation of the structure are those consistent with linear orthotropic thin elastic shell theory. Further description is provided in Reference 7.

b) Application

ASHSD is used to obtain the dynamic response of the core support barrel due to a LOCA. An axisymmetric thin shell model of the structure is developed. The spatial Fourier series components of the time varying LOCA loads are applied to the modeled structure.

The program yields the dynamic shell and beam mode response of the structural system.

c) Verification

ASHSD has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results. The details of these comparisons may be found in References 7 and 8.

3.9.1.2.2.2.5 CESHOCK

a) Description

The computer program CESHOCK solves for the response of structures which can be represented by lumped-mass and spring systems and are subjected to a variety of arbitrary type loadings. This is done by numerically solving the differential equations of motion for an n^{th} degree of freedom system using the Runge-Kutta-Gill technique. The equations of motion can represent an axially responding system or a laterally responding system; i.e., an axial motion, or a coupled lateral, and rotational motion. The program is designed to handle a large number of options for describing load environments and includes such transient conditions as time-dependent forces and moments, initial displacements and rotations, and initial velocities. Options are also available for describing steady-state loads, preloads, accelerations, gaps, non-linear elements, hydrodynamic mass, friction, and hysterisis.

The output from the program consists of minimum and maximum values of translational and angular accelerations, forces, shears, and moments for the problem time range. In addition, the above quantities are presented for all printout times requested. Plots can also be obtained for displacements, velocities and accelerations as desired. Further description is provided in Reference 9.

b) Application

The CESHOCK program is used to obtain the transient response of the reactor vessel internals and fuel assemblies due to LOCA and seismic loads.

Lateral and vertical lumped-mass and spring models of the internals are formulated. Various types of springs; linear, compression only, tension only, or non-linear springs are used to represent the structural components. Thus, judicious use of load-deflection characteristics enables effects of components impacting to be predicted.

Transient loading appropriate to the horizontal and vertical directions is applied at mass points and a dynamic response (displacements and internals forces) is obtained.

c) Verification

CESHOCK has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted experimental test or analytical results. The details of these comparisons may be found in References 8 and 9. A summary of the verification is provided in Appendix 3.9D.

3.9.1.2.2.2.6 SAMMSOR/DYNASOR

a) Description

SAMMSOR-DYNASOR provides the ability to perform non-linear dynamic analyses of shell structures represented by axisymmetric finite elements and subjected to arbitrarily varying load configurations.

The program employs the matrix displacement method of structural analysis, utilizing a curved shell element. Geometrically nonlinear dynamic analyses can be conducted using this program.

Stiffness and mass matrices for shells of revolution are generated utilizing the SAMMSOR part of this program. This program accepts a description of the structure in terms of the coordinates and slopes of the nodes, and the properties of the elements joining the nodes. Utilizing the element properties, the structural stiffness and mass matrices are generated for as many as 20 harmonics and stored on magnetic tape. The DYNASOR portion of the program utilizes the output tape generated by SAMMSOR as input data for the respective analyses.

The equations of motion of the shell are solved in DYNASOR using Houbolt's numerical procedure with the non-linear terms being moved to the right-hand side of the equilibrium equations and treated as generalized pseudo-loads. The displacements and stress resultants can be determined for both symmetrical and asymmetrical loading conditions. Asymmetrical dynamic buckling can be investigated using this program. Solutions can be obtained for highly nonlinear problems utilizing as many as five circumferential Fourier harmonics. Further description is provided in References 10 and 11.

b) Application

This program is used to analyze the dynamic buckling characteristics of the core support barrel during a LOCA hot-leg break. The program's non-linear characteristics provide this capability.

A finite element model of the core support barrel (CSB) is formulated which is consistent with the computer program. Taking into account the initial deviation of the structure and the shell mode which is most likely to give the minimum critical pressure, the time-dependent pressure load is applied to the barrel. The maximum displacement occurring in the barrel is obtained.

c) Verification

SAMMSOR/DYNASOR has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations, accepted experimental test or analytical results; and results obtained with a similar independently written program in the public domain.

3.9.1.2.2.2.7 SAAS

a) Description

The program performs finite element static analyses of axisymmetric solids. The continuous body to be analyzed is replaced by a system of ring elements with triangular or quadrilateral cross sections. The elements are interconnected at their apexes, then referred to as nodes. The displacement method of finite element analysis is used to derive the element stiffness matrix. This method proceeds by selecting a displacement expansion over the element, consistent with elemental boundary conditions, and assuming displacements in the interior of the element depend only on nodal quantities. The elemental stiffness matrices are computed and combined to yield the total stiffness matrix of the modeled structure. The principle of minimum potential energy is then applied to yield displacements and elemental forces (stresses). Since these elements are of relative arbitrary shape, the procedure can be applied to bodies of complex geometry. The program performs static analyses due to both boundary forces and thermal loads by converting these effects into equivalent nodal quantities.

Assumptions governing the use of the aforementioned finite elements are those consistent with linear elasticity theory of solid structures. Further description is provided in Reference 12.

b) Application

The program is employed to determine the stiffness properties of flanged regions as related to axisymmetric loads. Specifically, the CSB upper and lower flanges with connecting cylinders and the upper guide structure flange with a connecting cylinder, were analyzed. Displacements due to a known external load were determined. The resulting stiffnesses were incorporated into an overall model of the reactor vessel internals, and were then employed in determining the dynamic response during vertical seismic and LOCA excitation. The results of the dynamic analyses were fed back into the flange analyses to determine their maximum stresses and deformations.

c) Verification

SAAS is in the public domain and further verification is not required.

The design control measures established in the QADM (Reference 32) ensure the applicability and validity of this program, and meet Criterion II.2.a of Standard Review Plan 3.9.1.

3.9.1.2.2.2.8 NAOS

a) Description

The NAOS program applies the finite element analysis to axisymmetric solids subjected to arbitrary non-axisymmetric loadings by expanding the various kinematic and forcing functions into Fourier series.

The continuous body to be analyzed is replaced by a system of ring elements with triangular or quadrilateral cross sections and/or thin conical shell elements. The elements are interconnected at their apexes and ends (for the case of shell elements) and then referred to as nodes. The displacement method of finite element analysis is used to derive the element stiffness matrices. This method proceeds by selecting a displacement expansion over the elements, consistent with elemental boundary conditions and, assuming displacements in the interior of the element depend only on nodal quantities. The elemental stiffness matrices are computed and then combined to yield the total stiffness matrix of the modeled structure. The principle of minimum potential energy is then applied to obtain displacements and element forces (stresses). The program performs static analyses due to both boundary forces and thermal loads, by converting their effects into equivalent nodal quantities. Since the elements employed are of relatively arbitrary shape, the procedure can be applied to bodies of complex geometry.

Assumptions governing the aforementioned analyses are those consistent with linear elasticity theory of solids and thin shell structures. Further description is provided in Reference 13.

b) Application

The program is employed to determine the stiffness properties of flanged regions due to lateral loads. Specifically, the CSB upper and lower flanges with connecting cylinders and the upper guide structure plate flange with connecting cylinders were analyzed for lateral shear and bending moment loads. Displacements due to known magnitudes of these loads were determined. The resulting stiffnesses were incorporated into overall models of the reactor vessel internals and then employed in determining the dynamic response during horizontal seismic and LOCA excitation.

c) Verification

NAOS has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results. The details of these comparisons may be found in Reference 13.

3.9.1.2.2.2.9 WATERHAMMER and CEFLASH - 4

a) Description

These programs determine the space-time variation of reactor vessel pressures in the subcooled regime by representing the system with a connected grid. WATERHAMMER technique is based on wave propagation analysis in an elastic medium, wherein the local instantaneous pressure is determined from a superposition of component wave intensities.

CEFLASH technique involves a simultaneous solution of the equations of mass, energy, and momentum along with a representation of the equation of state. The mathematical details of both of these programs are provided in References 14, 15, and 16.

b) Application

These programs are used to determine the blowdown loads resulting from the double-ended breaks, in either hot or cold leg piping, that cause pressure differences to develop across reactor internal components and also induce flow rates causing drag forces on those components. Details of the application of these programs to the dynamic analysis of blowdown loads is provided in Subsection
3.9.2.5 and Reference 8.

c) Verification

References 14 and 15 for the WATERHAMMER and CEFLASH-4 programs are part of the public domain. References 8 and 16 provide additional verification of the use of these programs to blowdown loads calculations.

➔(DRN 03-2056, R14)

3.9.1.2.2.2.10 CEFLASH-4B

The CEFLASH-4B computer code (Reference 36 predicts the transient distributions of pressure, flow rate and density throughout the primary reactor coolant system during the subcooled and saturated portion of the blowdown period of a Loss-of-Coolant-Accident (LOCA). CEFLASH-4B is used primarily to predict the blowdown hydraulic loads on the walls and internal structures of the reactor pressure vessel, including the core, during the early period of subcooled decompression during a LOCA. The equations for conservation of mass, energy and momentum along with a representation of the equation of state are solved simultaneously in a node and flow path network representation of the primary system. CEFLASH-4B provides transient pressures, flow rates and densities throughout the primary system following a postulated pipe break in the reactor coolant system. The CEFLASH-4B computer code is a modified version of the CEFLASH-4A code (References 37 to 39) and has been approved by the NRC for evaluating the subcooled decompression and early saturation response portions of the blowdown of a PWR following a postulated LOCA (References 40 and 41). The capability of CEFLASH-4B to predict experimental blowdown data is presented in Reference 36.

🡐(DRN 03-2056, R14)
➔(EC-12610, R307)

3.9.1.2.2.3 Reactor Coolant Gas Vent System (RCGVS)

3.9.1.2.2.3.1 ANSYS

a) Description

🡐(EC-12610, R307)

→(EC-12610, R307)
ANSYS is a general purpose finite element program with structural and heat transfer capabilities. It is described in Reference 1.

b) Application

ANSYS was used to calculate the static and dynamic loadings for both the class 1 and class 2 portions of the RCGVS piping. The finite element capabilities of the program are used to determine these pipe loadings due to normal, upset and faulted conditions.

c) Verification

ANSYS is in the public domain and further verification is not required.

The design control measures established in the Inter-Business Unit Edition Policies & Procedures Manual, Sections NSNP 3.6.1-3.6.6 (Reference 42), ensure the applicability and validity of this program, and meet Criterion II.2.a of Standard Review Plan 3.9.1.
←(EC-12610, R307)

3.9.1.3 Experimental Stress Analyses

Neither, the NSSS nor Non-NSSS are utilizing experimental methods of stress analysis on the Waterford 3 project.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition

3.9.1.4.1 Seismic Category I Items

3.9.1.4.1.1 Code Items

The system or subsystem analyses performed to establish the loadings included in the faulted conditions are based upon elastic methods. Accordingly, elastic methods are used in the design calculations performed to demonstrate the suitability of the components, including supports to withstand the faulted conditions.

Inelastic analysis of components and their supports and pipe restraints is permitted only during postulated pipe rupture conditions and is limited to elements within the reactor coolant cold leg or hot leg, and Main Steam and Feedwater Systems. When inelastic analysis is permitted, the system or subsystem analysis used to develop the system loads properly accounts for the inelastic action. Calculated inelastic strains, produced by the faulted condition, are limited to 50 percent of the ultimate strain (strain at ultimate stress of the material).

For ASME Code Class 1 components, excluding supports, the calculated primary stresses produced by the faulted conditions shall not exceed the primary stress limits of paragraphs NB3221 and NB3230 of ASME Section III using an S_m value equal to the lesser of 2.4 times the tabulated S_m value and V.70 times the tensile strength of the material with both values taken at the appropriate temperature. For supports, including attachment welds to the item assembly, the primary stress limits of NB3221 shall be satisfied by using an S_m value equal to the greater of 1.5 times the tabulated S_m value and 1.2 times the tabulated yield strength, but not exceeding 0 7 times the tensile strength, with values taken at the appropriate temperature.

As an alternate to the above paragraph for main reactor coolant loop piping, the stress limit for faulted conditions is satisfied if the requirement of Equation (9) of NB3652 is met using a limit of three times the tabulated S_m value and, in addition, the pressure resulting from the faulted condition loading combinations does not exceed twice the design pressure.

The stress limits for ASME Class 2 and 3 components and supports are provided in Subsection 3.9.3.1.1.4.

Table 3.9-21 provides the summary of the loadings, stress ranges, S_m m, n, and K_e values for each location where the primary plus secondary stress intensity range is either just below or just above a value of 3 S_m

3.9.1.4.1.2 Non-Code Items

Those components not covered by the ASME Code and related to plant safety include: (1) reactor vessel internals, (2) fuel, (3) control element drive mechanisms (CEDMS) and (4) control element assemblies (CEAs). Each of these components is designed and fabricated in accordance with specific procedures and criteria to insure their operability as it relates to safety. The fundamental criterion to be met following a postulated faulted condition is to ensure relationships and assumptions employed in the analysis.

The method of analysis used in the evaluation of stress in the reactor internals and CEDMs during the faulted condition is elastic component analysis. This is compatible with the elastic system analysis employed to determine the dynamic loads acting on the internals and CEDMS, as described in Subsections 3.7.3, 3.9.2.5 and 3.9.4.

3.9.1.4.1.3 Fuel Assemblies

The evaluation of stresses in fuel assemblies during a combined design basis earthquake and loss-of-coolant accident is described in Reference 17. The report includes full details of the analytical methods, material strength requirements, theoretical relationships and assumptions employed in the analysis.

3.9.1.4.1.4 CEAs

The stresses, produced in the CEAs during the various faulted condition events, are evaluated by elastic analysis methods. These stresses arise from the accelerations and resultant deflections and reactor coolant pressure variations predicted to occur during such events. For withdrawn CEAs, the predicted accelerations are calculated from the motion of the upper guide structure. For inserted CEAS, the deflection is assured to be equal to the deflection of the fuel assembly in which the CEAs are inserted.

→(DRN 03-2056, R14)

3.9.1.5 Analysis Methods Under Pipe Break Loadings

The major components of the Reactor Coolant System (RCS) are designed to withstand the forces associated with the design basis pipe breaks discussed in Subsection 3.6.2, in combination with the forces associated with the safe shutdown earthquake and normal operating conditions. The forces associated with the postulated pipe breaks include pipe thrust forces at the break location, jet impingement loads, resultant subcompartment differential pressurization forces and hydraulic forces acting on the reactor internals.

3.9.1.5.1 Original Dynamic Analysis of Postulated Main Coolant Loop Pipe Breaks

The original dynamic structural analysis of main coolant pipe breaks (MCLBs) was performed using a lumped parameter model as discussed in CENPD-168[(18)], including details of the reactor vessel and supports, major connected piping and components, and the reactor internals. The pipe break thrust force, asymmetric subcompartment pressurization forces and asymmetric reactor internal hydraulic forces were applied as simultaneous time history forcing functions.

The pipe break thrust forces were determined by the methods discussed in CENPD-168. The time and spacially dependent hydraulic loads acting on the reactor internals were determined by the methods discussed in Subsection 3.9.2.5. The subcompartment pressurization forces are discussed in Section

←(DRN 03-2056, R14)

➔(DRN 03-2056, R14; EC-19087, R305; EC-29816, R306)
6.2. A dynamic analysis of the reactor vessel supports under MCLB conditions is presented in Appendix 5.4A. Other aspects of asymmetric loads due to MCLBs are discussed in Appendix 3.9E. The resultant component and support reactions were combined with the appropriate normal operating and seismic reactions.

The maximum reactor vessel displacements were imposed upon a flexibility analysis model of the RCS in order to account for the load on other component supports due to RV motion.

➔(EC-8458, R307)
3.9.1.5.2 Dynamic Analysis of Postulated Branch Line Pipe Breaks for Extended Power Uprate to 3716 MWt and Replacement Steam Generators

Using LBB methodology described in 3.6.3, consideration of mechanical (dynamic) effects of MCLBs and surge line breaks were eliminated. Branch line pipe breaks (BLPBs) in the remaining largest tributary piping systems interfacing the RCS (as listed in Section 3.6.2) replaced the MCLBs. The RCS was analyzed for the effects of BLPBs for extended power uprate to 3716 MWt and Replacement Steam Generators.
🡐(EC-8458, R307)
The RCS model developed for the remaining BLPB analyses is a full RCS lumped parameter ANSYS model, which includes details of the RV, RV internals and core, steam generators and internals, reactor coolant pumps, main coolant loop piping, and gapped and non-gapped RCS supports. Pipe break thrust force, jet impingement loads on RCS components, asymmetric subcompartment pressurization forces and asymmetric reactor internal hydraulic forces are applied as simultaneous time history forcing functions. The time and spacially dependent hydraulic loads acting on the reactor internals are determined by the methods discussed in Subsection 3.9.2.5. The subcompartment pressurization forces are discussed in Section 6.2. The subcompartment pressurization forces that are dynamically applied to the NSSS components credit LBB methodology.
🡐(EC-19087, R305; ; EC-29816, R306)

The dynamic responses of the RCS to BLPBs are determined. The resultant RCS component and support reactions are combined with the appropriate normal operating and seismic reactions for evaluation. The RCS response motions and response spectra are provided as input to structural analyses and evaluations of piping and components that interface with the RCS.
🡐(DRN 03-2056, R14)

3.9.2 DYNAMIC SYSTEM ANALYSIS AND TESTING

3.9.2.1 Preoperational Vibration, Thermal Expansion and Dynamic Testing on Piping

Piping vibration, thermal expansion and dynamic effect testing will be conducted during preoperational and startup testing. The purpose of these tests is to confirm, by observation or measurement, as appropriate, that the piping systems, restraints, components and supports are capable of withstanding the flow-induced dynamic loadings under steady state and anticipated transient operating conditions. In addition, thermal motions will be monitored to verify movements predicted by analysis and ensure that adequate clearances exist to allow the required normal thermal movement of systems, components and supports.

This testing program is designated to fulfill the requirements of Regulatory Guide 1.68, Revision 2. The following piping is included in the Test Program:

- ASME Code Class 1, 2, and 3 Systems
- Other high energy systems within seismic Category I structures
- High energy portions of non-safety systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable level
- Seismic Category I portion of moderate energy piping systems located both inside and outside containment.

Certain lines however which fall in the categories above will be exempted from testing for the following reasons:

- Line is rarely used, or when used, is not related to plant shutdown
- Line is both isolated from source of vibration and has a low momentum flow
- Line is continuously supported (e.g., buried lines)
- Line cannot be tested under the operational conditions for which it is designed during preoperational or startup testing (e.g., containment spraying headers).

Test boundaries of each system, subject to test, will be marked up on isometrics as well as points which are to be observed and corresponding allowable vibratory and thermal motion.

3.9.2.1.1 Vibration Testing

The vibration tests are performed during those system operating modes where significant vibratory response is anticipated, based on operating experience with similar system in nuclear power plants. Prior to the implementation of the test program, a test procedure will be written which will contain a description of the tests, a complete listing of the systems to be tested and of the various modes of operations under which they are to be tested and the acceptance criteria for each test. For example, Table 3.9-19 gives a summary listing of possible testing modes for selected systems.
They are divided into two categories:

Steady State - Repetitive vibrations, such as when pumps are operating, which occur for relatively long periods of time during the normal plant operation;
Transient - Vibrations which occur during relatively short periods of time. Examples are single and multiple pump start, rapid valve opening or closing and safety relief valve operation.

To simplify the testing efforts, four (4) levels of test (based on their sophistication) are identified:

3.9.2.1.1.1 Level 1 - Visual Observation Test

The purpose of this test is to visually determine the acceptability of the vibration for the piping subject to test.

Testing at Level 1 is judged sufficient to determine the acceptability of steady state and transient vibration for many cases, based on industrial experience with similar systems and the fact that piping systems are very flexible. This flexibility results in high allowable peak-to-peak displacements which might be easily observed visually. Locations having allowable peak-to-peak displacements in excess of 20 mils will be clearly observable visually and require no specific definition of their location. All locations with allowable peak-to-peak displacements less than 20 mils will be marked up on the isometrics as well as the respective distances from which these vibrations must be imperceivable to be acceptable. The distances, marked up on the isometrics, will be derived by determination of a visually observable maximum amplitude which would result in a dynamic stress less than or equal to 50 percent of the alternating stress amplitude at 10 cycles as shown in the ASME Code. In addition to marked points, special attention will be paid to observing:

a) Elbow spans and spans adjacent to elbows;
b) Spans with lumped masses such as valves and flanges;
c) Vents, drains, and instrumentation lines.

Simple charts, which quickly and conservatively determine allowable peak-to-peak displacement for any piping span configuration, will be provided for this purpose. Should the Level 1 test procedure lead to inconclusive results then a Level 2 test is to be performed.

3.9.2.1.1.2 Level 2 - Hand Held Amplitude Test

The purpose of this test is to determine the vibratory displacement of those piping segments for which Level 1 visual observations are questionable.

This Test Procedure is applicable for both steady state and transient conditions. A Level 2 Test, utilizing and hand-held vibration indicator to measure peak-to-peak displacement, will be performed at prescribed locations. The locations will be chosen on the basis of dividing the piping systems into a series of representative spans. A span is defined as any part of a piping system between two consecutive restraints which function in the same direction, or a cantilever. Instruction on how to break down each piping system into different span configurations will be provided as part of the test procedure. The measurement locations and acceptable criteria for the different span configurations will be given in the Test Procedure.

Stress amplitudes due to vibration will be considered acceptable if they do not exceed 50% of S_a at 10^6 cycles as shown in Figures I-9 of the ASME B&PV Code, Section III 1971 edition up to and including the Winter 1972 addenda.

For low cycle ($<10^6$ cycles) transient vibrations, the acceptance criteria is predicated on the following:

a) If observed displacements are such that the maximum dynamic amplitude stress does not exceed 50% of S^a at 10_6 cycles as shown in Figures I-9 of the ASME B&PV Code a Section III 1971 edition up to and including the Winter 1972 addenda, then the vibration is acceptable.

b) If observed displacements are larger than a) above, then:

1) A cumulative usage factor U_V is computed from

$$Uv = \sum i \frac{N^1}{N^i_{AL}}$$

where:

N_i = number of type i transients times the effective number of cycles for each type i transient

N^i_{AL} = allowable number of cycles for type i transient corresponding to the alternating stress, S_i, where

S_i = the maximum alternating stress produced by the type i transient.

The vibration is acceptable if $U_V < 0.1$.

Instrumentation Requirements for Level 2 test are given in Table 3.9-20.

If the test results do not meet the Level 2 acceptance criteria, then a Level 3, Hand Held Amplitude/Frequency Test will be performed for cases of steady state vibration and Level 4, Instrumentation-Stress Test for cases of transient vibrations.

3.9.2.1.1.3 Level 3 - Hand Held Amplitude/Frequency Test

The purpose of this test is to determine the vibratory and respective peak-to-peak displacements of piping segments for which the results of the Level 2 testing are inconclusive. Portable instruments are used for inconclusive results then a Level 2 test is to be performed.

3.9.2.1.1.2 Level 2 - Hand Held Amplitude Test

The purpose of this test is to determine the vibratory displacement of those piping segments for which Level 1 visual observations are questionable.

This Test Procedure is applicable for both steady state and transient conditions. A Level 2 Test, utilizing and hand-held vibration indicator to measure peak-to-peak displacement, will be performed at prescribed locations. The locations will be chosen on the basis of dividing the piping systems into a series of representative spans. A span is defined as any part of a piping system between two consecutive restraints which function in the same direction, or a cantilever. Instruction on how to break down each piping system into different span configurations will be provided as part of the test procedure. The measurement locations and acceptable criteria for the different span configurations will be given in the Test Procedure.

Stress amplitudes due to vibration will be considered acceptable if they do not exceed 50% of S_a at 10^6 cycles as shown in Figures I-9 of the ASME B&PV Code, Section III 1971 edition up to and including the Winter 1972 addenda.

For low cycle ($<10^6$ cycles) transient vibrations, the acceptance criteria is predicated on the following:

a) If observed displacements are such that the maximum dynamic amplitude stress does not exceed 50% of S^a at 10_6 cycles as shown in Figures I-9 of the ASME B&PV Code a Section III 1971 edition up to and including the Winter 1972 addenda, then the vibration is acceptable.

b) If observed displacements are larger than a) above, then:

1) A cumulative usage factor U_V is computed from

$$Uv = \sum i \frac{N^1}{N^i_{AL}}$$

where:

N_i = number of type i transients times the effective number of cycles for each type i transient

N^i_{AL} = allowable number of cycles for type i transient corresponding to the alternating stress, S_i, where

S_i = the maximum alternating stress produced by the type i transient.

The vibration is acceptable if $U_V < 0.1$.

Instrumentation Requirements for Level 2 test are given in Table 3.9-20.

If the test results do not meet the Level 2 acceptance criteria, then a Level 3, Hand Held Amplitude/Frequency Test will be performed for cases of steady state vibration and Level 4, Instrumentation-Stress Test for cases of transient vibrations.

3.9.2.1.1.3 Level 3 - Hand Held Amplitude/Frequency Test

The purpose of this test is to determine the vibratory and respective peak-to-peak displacements of piping segments for which the results of the Level 2 testing are inconclusive. Portable instruments are used for this test. Acceptance criteria incorporated in the same charts used for Level 2 tests and/or computer analysis used to determine dynamic stresses, based on the measurement results, will enable a final construction regarding the acceptability of steady state vibration.

3.9.2.1.1.4 Level 4 - Instrumentation - Stress

This test will be performed for those transient events for which the results of Level 2 testing are questionable. A time history analysis of the piping system response to the transients will be performed utilizing the computer program PLAST. The location of maximum stress points, maximum displacement points and maximum restrained loads will be calculated.

The results will give all necessary information to establish acceptance criteria and to select proper testing sensors. Fluid parameters will be measured if required. A data acquisition system will be used to record information during testing.

3.9.2.1.1.5 Corrective Action

In the unlikely event that the piping vibration exceed the acceptance criteria for Level 3 or 4 tests, then corrective actions will be initiated. Possible corrective action includes: (1) identification and reduction or elimination of the offending force, (2) detuning of resonant piping spans by appropriate modifications to the restraint system, (3) addition of bracing to stiffen the system, and (4) changes in operating procedures to eliminate troublesome operating conditions.

Following corrective action, additional testing shall be performed to determine if the vibrations have been sufficiently reduced to satisfy the acceptance criteria and the piping stress analysis shall be revised to include the corrective measures.

The methodology described above is summarized in the General Flow Chart in Figure 3.9-20.

3.9.2.1.2 Thermal Expansion Testing

Thermal expansion testing will be performed to verify that the measured movements at particular locations are approximately equal to those predicted by analysis and to ensure that the piping is not restrained due to interferences with other components.

Prior to the implementation of the testing program a test procedure identifies systems to be tested and expected movements at those chosen points.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

All seismic Category I safety-related mechanical equipment is qualified by analysis and/or testing to ensure structural adequacy. In addition, for equipment considered active, operability is concluded by analysis or verified by testing. The methods and procedures used and the results of tests and analyses that confirm implementation of the design criteria for safety-related mechanical equipment are provided in the following Subsections:

3.9.3.1 Loading Combinations, Design Transients and Stress Limits (for ASME Code Class 1, 2 and 3 Components)

3.9.3.2.1 Operability Assurance of Active Pumps

3.9.3.2.2 Operability Assurance of Active Valves

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment (associated with mechanical equipment) Safety-related heat exchangers, tanks and fans are seismically qualified by analysis.

3.9.2.3.4 Response Analysis

3.9.2.3.4.1 Deterministic Response

The natural frequencies and mode shapes of the core support barrel, which form the basis for all forced response analyses, were obtained through the use of the axisymmetric shell finite-element computer program, ASHSD[7] This computer program is capable of obtaining natural frequencies and mode shapes of complex axisymmetric shells; e.g., arbitrary meridional shape, varying thickness, branches, multi-materials, orthotropic material properties, etc. An inverse iteration technique is used in the program to obtain solutions of the characteristic equation, which is based on a diagonalized form of consistent mass and stiffness matrices developed using the finite element method. Four degrees of freedom - radial displacement, circumferential displacement, vertical displacement and meridional rotation - are taken into account in the analysis giving rise to coupled mode shapes and frequencies.

A finite-element model of the core support barrel system was developed, as shown in Figure 3.9-1. Evaluation of the reduction of these frequencies for the system immersed in the coolant was made by means of the "virtual mass" method. The normal mode method was used to obtain the structural response of the core support barrel to the deterministic forcing functions. Generalized masses based on mode shapes and the mass matrix from the shell finite element computer program were calculated for each core support barrel mode of vibration. Modal force participation factors, based on the mode shapes and the predicted periodic forcing functions, were calculated for each mode and forcing function. The generalized coordinate response for each mode was then obtained through solution of the corresponding set of independent second order, single-degree-of-freedom equations. Utilizing displacement and stress mode shapes from the shell finite-element computer program, the structural response of the core support barrel for each mode was obtained by means of the appropriate coordinate transformation. Response to any specific forcing function was obtained through summation of the component modes for that forcing function.

3.9.2.3.4.2 Random Response

The random response analysis considered the response of the core support barrel system to the turbulent downcomer flow during steady-state operation. The random forcing function was assumed to be a wideband stationary random process with a pressure spectral density equal to the peak value associated with the turbulence. The rms vibration level of the core support barrel system in terms of a beam mode was obtained based upon a damped, single-degree-of-freedom analysis assuming the rms random pressure fluctuations to be spatially invariant. The maximum rms response calculated was considerably less than the design operating clearances available at the snubbers.

3.9.2.3.5 Transient Forcing Conditions

The transients that occur during loop startup or shutdown represent gradual transitions from one steady-state mode of operation to another taking place over many seconds. It was recognized, therefore, that no dynamic magnification of structural response would occur and no dynamic transient response calculations were required.

3.9.2.4 <u>Preoperational Flow-Induced Vibration Testing of Reactor Internals</u>

The Maine Yankee and Fort Calhoun precritical vibration monitoring programs (PVMPs) together constitute a valid prototype design for Waterford 3. Waterford 3 has been designated as a non-prototype seismic Category I design.

In accordance with NRC Regulatory Guide 1.20, Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing, Revision 1, prototype prediction, measurement, and inspection programs were developed and performed for the Maine Yankee and Fort Calhoun reactor internals. Theoretical prediction analyses were performed for Maine Yankee(22) and Fort Calhoun(23) to estimate the amplitude, time, and spatial dependency of the steady-state and transient hydraulic and structural responses to be encountered during precritical testing. The precritical vibration monitoring programs for Maine Yankee and Fort Calhoun were completed successfully(24)(25) Comparisons of the measured and predicted responses for Maine Yankee and Fort Calhoun demonstrate that the theoretical prediction methods used provided accurate estimates of the steady-state response of the core support barrel system, when reasonable best estimate values for the magnitude of the inlet pressure fluctuations are used. It was concluded from these programs that flow induced vibrations of the Maine Yankee and Fort Calhoun reactor internals are well within design allowables and are acceptable for all normal, steady-state, and transient flow modes of reactor coolant pump operation.

Presented in Table 3.9-4 is a summary of the significant hydraulic and structural design parameters for each of the three reactor designs. The effects of these structural and hydraulic parameters on the flow-induced vibratory response of the reactor internals are presented in Subsection 3.9.2.6, where it is shown that the nominal differences have no significant effects on the stress levels. In general, the analysis for Waterford 3 demonstrates that:

a) The predicted structural response of the Waterford 3 internals are well within design allowables and are acceptable for all normal, steady-state, and transient flow modes of reactor coolant pump operation.

b) The prototype precritical vibration monitoring programs for Maine Yankee and Fort Calhoun adequately account for the specific design features of Waterford 3 which are shared by the valid prototype reactor designs.

The applicant is proceeding to implement a PVMP for Waterford 3 which is consistent with the recommendations of NRC Regulatory Guide 1.20, Revision 1, as it relates to non-prototype seismic Category I units. The reactor vessel internals will be subjected, during the preoperational and functional testing program, to all significant flow modes of normal reactor operation for a sufficient period of time to determine whether the reactor vessel internals exhibit any unexpected vibration problems. Prior to and during the PVMP, the reactor vessel internals will be examined to detect any evidence of unanticipated or excessive vibrations. The internals will be removed from the vessel for those visual and nondestructive inspections. The following points will be investigated:

a) all major load bearing elements,

b) lateral, vertical and torsional restraints within the vessel,

c) locking and bolting devices,

d) all other locations examined on the prototype designs and

e) the reactor vessel interior for loose parts and/or foreign material.

The results of the full examination program will be separately reported in a summary report submitted to the NRC to confirm that the observed vibrational characteristics are similar to those of the prototype design.

3.9.2.5 <u>Dynamic System Analysis of the Reactor Internals Under Faulted Conditions</u>

Dynamic analyses were performed to determine blowdown loads and structural responses of the reactor internals and fuel to postulated LOCA (Loss-of-Coolant Accident) loadings and to verify the adequacy of their design. A brief description of these methods is provided below.

The LOCA maximum stress intensities in the reactor internals were determined using conservative combinations of lateral and vertical LOCA time-independent loadings. The maximum LOCA stresses and the maximum stresses resulting from the SSE were then combined using the root-sum-square method to obtain the total stress intensities. The analytical results indicate that the maximum deformations do not exceed the allowable safe shutdown limits and that the maximum component stresses do not exceed the ASME code allowables for faulted conditions. Subsections 3.9.3 to 3.9.5 provide the design bases which were used in the structural evaluation of the reactor internals.

3.9.2.5.1 Dynamic Analysis Forcing Functions

→(DRN 03-2056, R14; EC-19087, R305)

These subsections provide a description of the methods used to obtain the forcing functions that were employed for the dynamic analysis of the reactor vessel internals during a LOCA. The original analyses considered forcing functions (blowdown loads) resulting from double-ended MCLBs in either the hot or cold leg piping at the RV nozzles. For the power uprate to 3716 MWt, only blowdown loads from BLPBs were considered. The forcing functions consist of transient pressure differences that develop across the reactor vessel internal components and induced flow rates which cause drag forces on the components.

←(EC-19087, R305)

3.9.2.5.1.1 Pressure Loads

The original analysis determined the blowdown loads using the WATERHAMMER and CEFLASH-4 codes. For the power uprate to 3716 MWt, the CEFLASH-4B code was used. These codes are described below:

The determination of the space-time variation of reactor vessel pressures, in the subcooled regime, can be accomplished by either of two analytical techniques, both of which represent the system by a connected grid. The first technique is based on a wave propagation analysis in an ' elastic medium, wherein the local instantaneous pressure is determined from a superposition of component wave intensities. This technique was formulated into the WATERHAMMER code[14] . The second technique involves a simultaneous solution of the conservation equations of mass, energy and momentum along with a representation of the equation of state. This technique has been formulated into the CEFLASH-4 code, among others. Descriptions of the mathematical details are given in References 15 and 16.

←(DRN 03-2056, R14)

Because of the greater generality of the CEFLASH-4 technique, it can yield information in both the subcooled and saturated regimes. However, the WATERHAMMER code permits a greater amount of spatial detail than does CEFLASH-4. Although both the WATERHAMMER and CEFLASH-4 techniques can be shown to have generally good agreement with experimentally determined absolute pressure values, the calculated pressure difference across a given vessel component can be different depending on which technique is used to calculate it. Because of this fact, analyses of the subcooled pressure loads have been determined with both the WATERHAMMER and CEFLASH-4 programs. This investigation has indicated that the subcooled pressure differences resulting from WATERHAMMER generally are more adverse across the critical components than those obtained from CEFLASH-4. Hence, in order to emphasize conservatism, the WATERHAMMER results are employed during the subcooled blowdown.

The saturated portion of the blowdown can only be analyzed with the CEFLASH-4 code. However, this is not a significant limitation, as the saturated hydraulic loads are smaller than the subcooled hydraulic loads.

Neither the WATERHAMMER nor the CEFLASH-4 codes account for changes in local volumes that are separated by a boundary.

It is reasonable to expect such boundaries (e.g., core support barrel) to displace in response to the applied loads and thereby cause the actual subcooled loads to be less than those predicted from a fixed volume analysis.

→(DRN 03-2056, R14)
The WATERHAMMER and CEFLASH-4 codes have been replaced by the CEFLASH-4B code. CEFLASH-4B (Reference 36) predicts the transient distributions of pressure, flow rate and density throughout the primary reactor coolant system during the subcooled and saturated portion of the blowdown period of a LOCA. See Section 3.9.1.2.2.2.10 for more information concerning CEFLASH-4B. Consistent with WATERHAMMER and CEFLASH-4, CEFLASH-4B does not account for displacement of local boundaries during the blowdown, resulting in conservative loads on reactor internals components.
←(DRN 03-2056, R14)

3.9.2.5.1.2 Drag Loads
→(DRN 03-2056, R14)
Following the development of large pressure differentials throughout the post break system, the local fluid velocity will accelerate in various regions and will cause higher than normal drag forces on certain components. Particular attention was paid to the drag loads that occur across the core and the CEA shrouds for double-ended (DE) hot leg breaks. These loads were higher than for the cold leg break because the hot leg flow area is approximately twice that of the cold leg. Furthermore, a DE break in a hot leg will cause the core and CEA shroud (transverse) flow rates to accelerate in the directions in which they are normally flowing. Hence, the transient drag force will add to the steady-state values. The core and transverse shroud flow rates reverse for a DE cold leg break. Hence, these transient drag forces are opposite to the steady-state values and must overcome them before producing net loads in the reverse direction. The same methodology was used to evaluate drag loads due to the BLPBs considered for the power uprate to 3716 MWt.
←(DRN 03-2056, R14)

3.9.2.5.1.3 Core Loads

The total instantaneous load across the core may be given as a summation of the pressure forces acting in the direction of the pressure gradient and the drag forces acting parallel to the flow. The loads are obtained using a control volume approach utilizing an integrated fluid momentum equation. The drag forces are represented by the fluid shear term in this equation and consist of both frictional and form drag.

3.9.2.5.1.4 CEA Shroud Loads

During normal steady-state operation, the coolant flows axially from the core up into the core outlet region, where the coolant flows around the outside of the control element assembly shroud tubes. Within this shroud tube region, the coolant flow direction changes so that it exits radially via the reactor vessel outlet nozzles. The transverse flow of coolant across the CEA shroud tubes gives rise to loads which tend to deflect these shroud tubes.

The magnitude of this transverse flow, and the resulting deflections, are greatest following a double-ended break in a hot leg.

The transverse forces on the CEA shroud tubes were determined by first conducting a model flow test and then scaling up the model loads to the appropriate values for a pressurized water reactor.

3.9.2.5.1.5 Blowdown Loads Results

Typical results for the blowdown loads analysis for this plant are given on Figures 3.9-2 and 3.9-3.

→(DRN 03-2056, R14)

Figure 3.9-2 shows the transient pressure difference across the wall of the core support barrel, at the elevation of the vessel nozzles, following a double-ended cold leg break. This pressure difference is shown for the radial direction towards the broken cold leg. The core support barrel inside - outside pressure difference is maximized in the direction of the broken leg. The transient delta pressure across the core support barrel (CSB) is significantly reduced for BLPBs considered for the power uprate to 3716 MWt evaluation. For example, the largest delta pressure across the CSB for the safety injection line break is 181 psi.

Following a double-ended hot leg break, the pressure distribution in the vessel annulus is uniform with respect to circumferential location. For such a break, Figure 3.9-3 shows the transient pressure difference across the core support barrel wall at the elevation of the vessel nozzles. Again, the delta pressure across the CSB for the BLPBs on the outlet line are significantly lower than those for the double-ended hot leg break. The largest delta pressure for the outlet breaks considered for the power uprate to 3716 MWt evaluation was 109 psi.

←(DRN 03-2056, R14)

3.9.2.5.2 Structural Response Analyses

The dynamic LOCA analyses of the reactor internals and core determined the shell, beam and rigid body motions of the internals, using established computerized structural response techniques. The analyses consisted basically of three parts. In the first part, the time-dependent shell response of the core support barrel to the transient loading was calculated using the finite-element computer code, ASHSD(7). The second part of the analysis evaluated the buckling potential of the core support barrel for hot leg break conditions using the finite-element computer code, SAMMSOR-DYNASOR(10)(11). In the third part, the dynamic time history responses of the reactor internals and core to vertical and horizontal loads resulting from hot and cold leg breaks were determined with the CESHOCK code (Subsection 3.9.1.2.2.2.5).

→(DRN 03-2056, R14)

The ASHSD and SAMMSOR-DYNASOR analyses were not repeated for the BLPBs considered for the power uprate to 3716 MWt evaluation, because the shell forces on the CSB due to the MCLBs clearly bound those due to the BLPBs. However, the CESHOCK analyses were repeated for the BLPB evaluations, and RV internals and core response was significantly reduced.

←(DRN 03-2056, R14)

3.9.2.5.2.1 Shell Response of the Core Support Barrel

A cold leg break causes a pressure transient on the core support barrel that varies circumferentially as well as longitudinally. The ASHSD finite element computer code was used to analyze the shell response of the CSB to the pressure transient from a cold leg break.

The CSB was modeled as a series of shell elements joined at their nodal point circles as shown on Figure 3.9-1. The length of the elements in each model was selected to be a fraction of the shell attenuation length. Since rapid changes in the stress pattern occur in the regions of structural discontinuity, the nodal points were more closely spaced in such regions.

A damped equation of motion was formulated for each degree of freedom of the system. Four degrees of freedom, radial displacement, circumferential displacement, vertical displacement, and meridional rotation were taken into account in the analysis. The differential equations of motion were solved numerically using a step-by-step integration procedure.

The circumferential variation of the pressure time-history was considered by representing the pressure as a Fourier expansion. The pressure at each node in the model was determined by linear interpolation. Thus a complete spatial time load distribution compatible with the ASHSD computer program was obtained. Each load harmonic was considered separately by ASHSD. The results for each harmonic were then added to obtain the total results.

The ASHSD code computed the nodal point displacement, resultant shell forces and shell stresses as a function of time.

3.9.2.5.2.2 Dynamic Stability Analysis of CSB

A hot leg break causes a net external radial pressure on the core support barrel. A stability analysis of the CSB was performed using the finite element computer code, SAMMSOR-DYNASOR. The effects of an initially imperfect shape based on the out-of-roundness tolerances were included in the analysis.

The CSB was modeled as a series of shell elements, as shown on Figure 3.9-4. Stiffness and 0mass matrices for shells of revolution were generated utilizing the SAMMSOR part of the code. The equations of motion of the shell were solved in DYNASOR using the Houbolt numerical procedure.

An initial imperfection was applied to the core support barrel by means of a pseudoload for each circumferential harmonic considered. The actual pressure transient loading generated by the outlet break is uniform circumferentially but varies longitudinally. The response was obtained for each of the imperfection harmonics.

Appendix F, Section III of the ASME Boiler and Pressure Vessel Code requires that permissible dynamic external pressure loads be limited to 75 percent of the dynamic instability pressure loads, or alternately, the dynamic instability loads must be greater than 1.33 times the actual loads. Consequently this analysis was repeated with the imperfection applied in the critical harmonic and the pressure loading was increased beyond 1.33 times the actual loads in order to demonstrate the stability of the core support barrel.

3.9.2.5.2.3 Dynamic System Analysis of the Reactor Internals

Dynamic analyses were performed to determine the structural response of the reactor internals to postulated asymmetric LOCA loading (including reactor vessel motion effects) and to verify the adequacy of their structural design. The postulated pipe breaks result in horizontal and vertical forcing functions which cause the internals to respond in both beam and shell modes.

Detailed structural mathematical models of the reactor internals were developed based on the geometrical design. These models were constructed in terms of lumped masses connected by beam or bar elements, and included nonlinear effects such as impacting and friction. The models were developed for input to the CESHOCK code which solves the differential equations of motion for lumped parameter models by a direct step-by-step numerical integration procedure. The model definitions employed the procedures established in Combustion Engineering Topical Report CENPD-178-P Rev. 1, (Reference 34) and included hydrodynamic coupling effects and a detailed representation of the core support barrel to upper guide structure to reactor vessel interfaces. Separate models were formulated for the horizontal (Figure 3.9-7) and vertical (Figure 3.9-6) directions to more efficiently account for structural and response differences in those directions.

The models for the horizontal directions were developed in terms of lumped masses connected by beam elements. The stiffness values for the beam elements were evaluated using beam characteristic equations. The lumped mass weights were based upon the mass distribution of the internals structures. Local masses such as plates and snubber blocks were included at appropriate nodes. The effect of the surrounding water on the dynamics of the internals for horizontal motion was accounted for by hydrodynamically coupling the components separated by a narrow annulus - the vessel, core barrel, and core shroud. The clearance between the core support barrel and the reactor vessel snubbers as well as the clearance between the core shroud, guide lugs and the fuel alignment plate was simulated by nonlinear springs which account for the loads generated when impacting occurs. A representation of the core was included in the internals models and provides appropriate inertial and impact feedback effects on the internals response.

The vertical model stiffness values were calculated using bar characteristic equations. Nonlinear couplings were included between components to account for structural interactions such as those between the fuel and core support plate, and between the core support barrel and upper guide structure upper flanges. Preloads, which are caused by the combined action of applied external forces, dead weights, and holddowns were also included. Friction elements were used to simulate the coupling between the fuel rods and spacer grids.

A reduced model of the reactor vessel internals (Figure 3.9-8) was developed for incorporation into the reactor coolant system model. The detailed nonlinear horizontal and vertical internals (plus core) models were condensed and combined into a three-dimensional model compatible with the reactor coolant system model and the computer programs through which the latter model was analyzed.

The purpose of this reduced internals model was to account for the effects of the internals LOCA loads on the reactor vessel support motion and the structural loading interaction between the internals and the vessel. The reduced internals model was developed so as to produce reactor vessel support motions and loadings equivalent to those produced by the detailed internals models.

The dynamic responses of the reactor internals to the postulated pipe break were determined with the CESHOCK code utilizing the detailed models. Horizontal and vertical analyses were performed for both hot and cold leg breaks to determine the lateral and axial responses of the internals to the simultaneous internal fluid forces and vessel motion excitation.

The vertical excitation of the internals was calculated using a control volume method of analysis. In this method, the reactor internals were subsectioned and enclosed within volumes of solid plus fluid. The momentum equation was then applied to each volume, and a resultant force was calculated and applied to the structural nodes within the volume. This method takes into consideration pressure, fluid friction, momentum changes, and gravitational forces acting on each volume. The resulting load time histories in a form consistent with CESHOCK code input requirements.

In order to achieve an initial (prior to the pipe break) equilibrium, the initial static deflections and gaps were calculated. The resulting initial conditions and load time histories were input to the CESHOCK code and the dynamic response of the model was calculated.

The horizontal input excitation resulting from a cold leg break were the core support barrel force time history and the vessel motion time history determined from the reactor coolant system analysis. The core support barrel forces were obtained by representing the asymmetric pressure distribution time history as a Fourier expansion. The two terms (sinθ) and cosθ) which excite the beam mode of vibration were then integrated over the core support barrel and transformed into nodal force time histories.

The horizontal input excitation resulting from a hot leg break were the CEA shroud crossflow load time histories and the vessel motion time history determined from the reactor coolant system analysis. The forces applied to the shroud mass points were determined directly from the blowdown pressure time history and included the drag force an forces due to the pressure differential on the shrouds.

→(DRN 03-2056, R14)
←(DRN 03-2056, R14)

The results from these horizontal and vertical analyses consisted of the time dependent member forces, and nodal displacements, velocities, and accelerations. The maximum member forces were combined with the results from the seismic analysis of the reactor internals and core as described in Subsection 3.7.3.14 to verify the structural adequacy of the internals.

3.9.2.6 Correlation of Reactor Internals Vibration Tests With the Analytical Results

3.9.2.6.1 Introduction

Since Waterford 3 has been classified as a non-prototype Category 1 design, an analysis and full inspection program will be performed for the plant in lieu of a measurement program. The results of the analysis are presented in this subsection. The results of the full inspection will be separately reported in a summary report.

The analysis procedures utilized are presented in detail in References 19 through 21. Only the pertinent results are presented in the following subsections.

3.9.2.6.2 Comparison of Structural and Hydraulic Parameters

Evacuation views of the Waterford 3, Maine Yankee, and Fort Calhoun reactor internals are shown on Figures 3.9-9 through 3.9-11, respectively. Table 3.9-4 is a summary of the significant hydraulic and structural design parameters for each reactor design. In general, the designs are similar, but some variations do exist. For example, the Waterford 3 design is simpler than either prototype in that it does not include a thermal shield.

The most significant hydraulic region is the downcomer annulus, where the coolant flow is undeveloped and highly turbulent. These reactor designs vary in three aspects with regard to the evaluation of hydraulic pressure fluctuations in the downcomer. These are the number of coolant loops, the presence or absence of a thermal shield, and the magnitude of the coolant velocity. A brief discussion of each is presented below.

The number of loops and their azimuthal relationship affect the spatial distribution of the fluctuating pressure field within the downcomer. That there is a nonuniform distribution of the has been shown in model tests and the Maine Yankee PVMP.[28][29][30] Using the principle of superposition and having determined the pressure field in the annulus for the case of one functioning pump, it is a simple matter to develop the fields corresponding to any azimuthal pattern of operating loops with varying phase relationships. The validity of superimposing pump effects was initially investigated during the hydraulic forcing function development. Subsequently, with actual PVMP measurements of the fluctuating pressure at various locations in the downcomer annulus for various operating conditions, the applicability of the principle was checked. Specifically, data obtained from single pump operation were combined to predict multiple pump pressures at various transducer locations. These values were compared with the actual measurements. The results indicated an average variation from perfect correlation of less than 25 percent.[24][25]

The majority of the predicted values exceeded the measured values indicating conservatism in the estimates. From these results, it was concluded that the superposition principle is an acceptable procedure for developing hydraulic forcing functions in the downcomer annulus.

The random component of the hydraulic loading on the CSB could be affected by the presence of a thermal shield in the annulus. Waterford 3 has no thermal shield. However, the random hydraulic load was developed from PVMP pressure data with the shield present. This results in a conservative estimate of the loading.

A usual assumption in the prediction of hydraulic fluctuations, whether of a periodic or random character, is that the magnitude is dependent on the fluid density multiplied by the square of a characteristic velocity. Data obtained in the Maine Yankee and Fort Calhoun PVMPs indicate this assumption to be valid Comparison of the Fort Calhoun hot post-core estimate and measurement(24) indicates the validity of the assumption over a sizeable range of the postulated variables (e.g., a change in density compounded with a change in velocity). The agreement for the other prototype is nearly as good, despite Maine Yankee being a three-loop system. This would indicate that the effects of flow velocity in Waterford 3 would be to reduce the hydraulic load as they have a slightly lower downcomer coolant velocity (see Table 3.9-4).

3.9.2.6.3 Deterministic Structural Response Results

Predictions of the periodic forcing functions were based on the steady state hot core coolant conditions. All predictions were made using the best estimate of the inlet duct pressure pulsations available as derived from the Maine Yankee and Fort Calhoun PVMPS. The fundamental forcing frequencies were the pump rotational speed (20 Hz) and the blade passing speed (100 Hz). A higher harmonic of each of the fundamental frequencies was included (40 and 200 Hz, respectively).

From the results of the response analysis (described in Subsection 3.9.2.3.4.1) for Waterford 3, the maximum stress intensity is experienced in the CSB upper flange region and is below the allowable stress criteria, viz:

$$\left(\sigma_{\max} = 640 \text{ psi}\right) \langle\langle \left(\sigma_{allowable} = 26000\text{psi}\right)$$

The analytical results provide a high degree of assurance that the structural integrity of the reactor internals will be maintained during all normal operating steady-state and transient conditions of reactor coolant pump operation.

3.9.2.6.4 Random Structural Response Results

The random response analysis considers the response of the CSB system to the turbulent component of the flow during steady-state operation. The random forcing function is assumed to be a wide-band stationary random process representing the random pressure fluctuations that result from the flow turbulence. The power spectral density of the pressure fluctuations was estimated from a representative analytical expression modified by the results of flow model testing. The power spectral density used was for full design flow conditions (all pumps operating). The response of the CSB system in the beam mode at the snubber elevation was considerably less than the nominal design gap at the CSB-RV snubbers.

3.9.3 ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

ASME Code Class 1 fluid system components are required to be designed in accordance with rules and methods specified in the ASME Code. The design stress limits of the ASME Code are selected to insure the pressure-retaining integrity of safety class equipment. Regulatory Guide 1.48 was issued in 1973. As an acceptable alternative, the loading combinations described in Subsections 3.9.3.1.1 and 3.9.3.1.2 were considered.

The ASME Code is recognized by industry and in the Code of Federal Regulations, as a standard whose rules and procedures provide a reliable, conservative basis for the design of nuclear safety related equipment of very high integrity. The design limits specified by the ASME Code inherently contain safety factors themselves, so that if certain maximum stresses, deformations, or fatigue usage factors are less than the allowable limits by any amount, the design is conservative.

A summary of maximum total stresses, deformations, usage factors, and identification of items within 10 percent of the Code allowable would require increased applicant and staff review effort without improving plant safety. Therefore, this specific information is not included. However, statements are provided indicating that particular pieces of safety equipment are in compliance with the appropriate ASME Code requirements.

3.9.3.1.1 Loading Combinations, Design Transients, and Stress Limits for NSSS Components and Supports

3.9.3.1.1.1 NSSS ASME Code Class 1 Supports

Design transients for Code Class 1 supports are discussed in Subsections 3.9.3.4 and 3.9.1.1. Loading combinations and stress limits are discussed in Subsection 5.4.14.

3.9.3.1.1.2 NSSS ASME Code Class 1 Components

Design transients for ASME Code Class 1 components are discussed in Subsection 3.9.1 and Section 5.4. Loading combinations for ASME Code Class I Components (other than valves) are shown in Table 3.9-5. Stress limits for ASME Code Class 1 components (other than valves) are shown in Table 3.9-6.

Class 1 line valves (RC 100E, 100F, 236, 237, CH 515, SI 618, 628, 638 and 648) are designed and manufactured to the draft ASME Code for pumps and valves for Nuclear Power - 1968 including the March 1970 Addenda. In addition valves with extended operators were analyzed for the combined loadings of maximum seismic loads (3.0g in any direction), maximum operator thrust load, dead weight, and design pressure load. This information as well as the stress limits used for the design of these valves is defined in greater detail by Subsection 3.9.3.2.2.1. Design transient loads were not analyzed based on the design criteria of the draft ASME Code for pumps and valves, paragraph 410, which excludes valves 4" IPS and under. All Class 1 line valves fall within this exclusion.

→(LBDCR 15-021, R309)

Class 1 safety valves (RC 200, 201) are designed and manufactured to ASME Section III, 1974 Edition including the Summer 1975 Addenda. Valves were analyzed for the combined loadings of maximum seismic loads, operating pressure, reaction force, pipe loads and dead weight. The stress intensity due to primary membrane plus primary bending was limited to LI.5Sm. Safety valves were also designed to be capable of operation during and after a seismic acceleration of 3.0g in any direction applied at the pipe connections. Design transients are negligible.

←(LBDCR 15-021, R309)

3.9.3.1.1.3 Reactor Internals Structures

Design transients for reactor internal structures are discussed in Subsection 3.9.1.1. Loading combinations and stress limits are presented in Subsection 3.9.5.

3.9.3.1.1.4 NSSS ASME Code Class 2 and 3 Components and Supports

The loading combinations applicable to ASME Class 2 and 3 vessels, PUMPS, and supports, are the design operating condition loading plus earthquake loadings associated with the OBE and SSE as defined for Class 1 components in Table 3.9-5 Pipe break loads are not specified for Code Class 2 and 3 components (see Subsection 3.9.3.1.1.4.2 and Tables 3.9-14 and 3.9-15).

3.9.3.1.1.4.1 NSSS Tanks and Heat Exchangers

Pressure vessels supplied for the auxiliary systems are listed in Table 3.9-14.

Vessel assemblies, including supports, support attachment welds, and anchor bolts are capable of withstanding specified horizontal and vertical seismic accelerations applied simultaneously at the vessel center of gravity, acting in the direction that yields the highest stress.

All ASME Code Class 2 and 3 pressure vessel pressure retaining parts are designed such that under the loading conditions defined in Table 3.9-5, the stresses do not exceed the stress limits specified in the ASME Boiler and Pressure Vessel Code, Section III.

Vessel components not subject to fluid pressure, such as supports, attachment welds and anchor bolts, are designed to the stress criteria of the AISC Manual of Steel Construction, Seventh Edition. The most severe loading condition is used, except that the increase in allowance stresses due to seismic or wind loads, per paragraph 1.5.6, part 5 of the manual, is not permitted

3.9.3.1.1.4.2 NSSS Valves Class 2 and 3

Class 2 and 3 line valves are designed and manufactured to articles 24 and 34 of the draft ASME Code for pumps and valves for Nuclear Power-1968 including the March 1970 Addenda. The maximum operating pressures are limited to those specified at the maximum operating temperature in ANSI B16.5 - 1968, Steel pipe flanges and flanged fittings. Articles 24 and 34 of the pump and valve Code did not have a design method specified for Class 2 or 3 valves. The standard did, however, require that a suitable method, which had been previously demonstrated, be used for the design of valves. ANSI B16.5 was the recognized industry standard whose rules and procedures provided a reliable, conservative basis for the design of valves of high integrity. The design limits specified by the standard inherently contain safety factors themselves; so that if certain maximum stresses, deformations, or fatigue usage factors are less than the allowable limits by any amount, the design is conservative. Valves with extended operator were analyzed for the combined loadings of maximum seismic loads (3.0g in any direction), maximum operating thrust load, design pressure load and weight. This, as well as the stress limits used for the design of these valves, are defined in greater detail by Subsection 3.9.3.2.2.1. The loading combinations of Regulatory Guide 1.48 were not in effect at the time of purchase of Class 2 and 3 valves. Pipe break loads were not specified for one or more of the following reasons:

a) Pipe breaks in lines attached to the valve renders the system inoperative. Operation of the system under these conditions is not required.

b) Pipe break loads not transmitted to the valve.

c) Valve located in moderate energy system.

d) Valve is not active.

3.9.3.1.1.4.3 NSSS Pumps

Pumps supplied for the auxiliary systems are listed in Table 3.9-15.

Pump assemblies, including supports, support attachment welds, and anchor bolts are capable of withstanding; specified horizontal and vertical seismic accelerations applied simultaneously.

For Class 2 pressure-retaining parts in the active (charging and safeguard) pumps, under the concurrent loadings of the OBE and normal operating (upset conditions), the primary membrane stress is less than 1.1S, and the primary membrane plus bending stress is less than 1.65S. Under the concurrent loadings of the normal operating conditions and the SSE (faulted conditions), the primary membrane stress is less than 1.2S, and the primary membrane plus bending stress is less than 1.8S.

For the non active Class 2 and 3 pumps, under the concurrent loadings of the OBE and normal operating (upset conditions), the primary membrane stress is less than 1.1S, and the primary membrane plus bending stress is less than 1.65S- Under the concurrent loadings of the normal operating conditions and the SSE (faulted conditions), the primary membrane stress is less than 2.0S, and the primary membrane plus bending stress is less than 2.4S.

S = Allowable value of stress per the ASME Draft Code for Pumps and Valves, for the active pumps and per ASME Code Section III for the non active pumps.

Charging and safeguard pump components that are not subjected to fluid pressure (such as supports, attachments welds and anchor bolts) are designed to the stress criteria of the AISC Manual of Steel Construction, Seventh Edition. The most severe loading condition is used, the exception being the increase in allowable stresses due to seismic or wind loads, per Paragraph 1.5.6, Part 5 of the manual.

Design thermal transients applicable to the high pressure safety injection pumps are:

a) increase in suction temperature from 40 F to 300 F in 10 sec, 40 times, and

b) decrease in suction temperature from 300 F to 40 F in 10 sec, 40 times.

Design thermal transients applicable to the low pressure safety injection pumps are:

a) increase in suction temperature from 40 F to 300 F in 10 sec, 40 times,

b) increase in suction temperature from 70 F to 300 F in one min, 500 times, and

c) decrease in suction temperature from 300 F to 70 F in several hours, 500 times.

3.9.3.1.2 Loading Combinations, Design Transients, and Stress Limits for Non-NSSS Components

Loading combinations for ASME Code Class 1 components other than valves are given in Table 3.9-5 and stress limits for these components are given in Table 3.9-6. Design transients for Code Class 1 components are discussed in Subsection 3.9.1.1.
Table 3.9-7 specifies loading combinations and stress limits for ASME Code Class 2 and 3 piping. The worst case design conditions are used for each Code Class 2 or 3 component. This condition for each component will be the same as the specified component operating condition, except that where a pump function must be assured during the emergency or faulted condition, the emergency or faulted condition for the plant is considered the normal condition for the pump or valve.

Code Class 1, 2 and 3 valves are designed in accordance with the design rules of ASME Code Section III, Subsections NB-3500, NC-3500 and ND-3500, respectively, of the Code Edition and Addenda in effect at the time of issuance of the initial purchase order. The manufacturers and applicable Code and Addenda for Code Class 1 valves are summarized in Subsection 3.9.1.1.2. The manufacturers and applicable Code and Addenda for Code Class 2 and 3 valves are given below:

Manufacturer	Type of Valve	ASME Code Edition/Addenda
Anchor-Darling Valve Company	Stainless Steel and Carbon Steel 2-1/2 inch and larger	1971/Winter 1972
Fisher Controls Company (Continental Division)	Butterfly	1971/Summer 1973
Jamesbury Corporation	Butterfly	1974/Summer 1975
Velan Engineering Company	Stainless Steel and Carbon Steel, 2 inch and smaller	1971/Winter 1973
ACF Industries, Inc. (W-K-M Valve Division)	Control	1971/Summer 1973
ACF Industries, Inc (W-K-M Valve Division)	MSIV's	1974/Summer 1975
Yarway Corporation	600-lb ANSI and higher, 2 inches and smaller	ASME Code 1974/ ---
Pacific Valve Company	600-lb ANSI and higher, 2-1/2 inches and larger	1971/Summer 1973
ITT Grinnell Company	Diaphragm	1971/Winter 1973
TRW Mission	Stainless Steel and Carbon Steel Wafer Check	1971/Winter 1973
Masoneilan International, Inc.	Control	1977/Summer 1977
Crosby Valve and Gage Co.	Miscellaneous Safety & Relief	1974/Summer 1974

Manufacturer	Type of Valve	Edition/Addenda
Crosby Valve and Gage Co.	MSRV's	1977/Winter 1978
J E Lonergan Co.	Miscellaneous Safety & Relief	1971/Winter 1973
Valcor (Masoneilan)	Solenoid	1977/Summer 1977
GPE Controls	HVAC Check	1974/Winter 1975
Control Components International	Atmospheric Dump	1974/Summer 1975
Target Rock Corp	Solenoid	1980/Summer 1980
Target Rock Corp	Pressure Regulators	1980/Summer 1980
→(DRN 02-1196, R13)		
BNL Industries, Inc.	Carbon and Stainless Steel Check	1989/None
←(DRN 02-1196, R13)		
BNL Industries, Inc.	Carbon Steel Ball	1977/Summer 1978
→ (DRN 99-1081, R11)		
Anderson, Greenwood & Co.	Safety & Relief	1986/None
← (DRN 99-1081, R11)		

The loads considered for Code Class 1 valves with nominal inlet sizes greater than four inches are: Operating, End, Thermal Transient, Dead Weight, and SSE. For valves with external actuators, the actuator thrust loads are considered as operating loads.

Stresses are calculated in accordance with the rules of the ASME Code Section III Subsection NB-3500 and verified to be within the limits of the Code. Safety Class 1 design reports as required by the Code, are submitted by the valve manufacturers as documentation of the above.

Code Class 1 valves of nominal inlet size of 4 in. and smaller are designed to the Standard Valve Design Rules of ASME Code Section III, Subsection NB. Safety Class 1 Design reports, as required by the Code, are submitted as documentation of adherence to the wall thickness requirements of the Code Design Rules.

All Code Class 2 and 3 valves with external actuators or appurtenances are seismically qualified for SSE loads. For valves located outdoors, if hurricane, tornado, or explosion loads are greater than SSE loads, the greatest of these are considered in lieu of the SSE loads. There are no Code Class 1 valves located outdoors.

Code Class 1 safety and relief valves have nominal inlet sizes 2 inches and smaller, and are designed to meet the requirements of ASME Code Section III Subsections NB-3500 and NB-7155 and Code Case 1711. These rules require that minimum wall thickness requirements be met in lieu of analyzing body stresses. Adherence to minimum wall thickness requirements are documented in Code Class I Design Reports submitted by the manufacturer.

Code Class 2 safety & relief Valves were analyzed to meet S Values in Table 1-7.1 or 1-7.2 (as applicable) from the appropriate ASME Code for all operating conditions at the minimum cross-section of the inlet neck for the following loads taken concurrently: Discharge Forces, Spring Forces, Internal Pressure, and the greatest of either SSE, Hurricane, Tornado, or Explosion Forces. There are no other applicable design basis events.

Design of Code Class 1, 2 and 3 valve bodies is in accordance with the Code, which assures that the piping system, not the valve body, is limiting and that the design of the pressure retaining parts are adequately covered by the rules of NB-3500, NC-3500, and ND-3500. Analysis of the piping as required by the ASME Code, Section NB-3600, NC-3600, and ND-3600 for system normal, Emergency, Upset, and Faulted conditions to meet the requirements of Tables 3.9-5 and 3.9-7 assures adequacy of the valve design.

Valves are supported and restrained by the supports and restraints of the piping system except for two inch and smaller motor and air operated valves. These valves are analyzed per FSAR Subsection 3.7.3.11 and supports and restraints are added to the operators to prevent adverse torsional effects. Also, supports and restraints may be added to any valve operator if required to satisfy stress allowances.

All Non-NSSS supplied vessels, i.e. tanks and heat exchangers, are ASME Code Class 3. The vessels are designed and built to Subsection ND of ASME Code Section III in effect at the time of their purchase. These vessels and the ASME Code Edition to which they are designed to are summarized below:

Description	ASME Code Edition/Addenda
Diesel Oil Storage Tanks A and B	1974/Summer 1974
Diesel Oil Storage Feed Tanks 1A and IB	1974/Summer 1974
Component Cooling Water Surge Tank	1974/Summer 1974
Chilled Water Expansion Tank	1974/Summer 1976
Component Cooling Water Heat Exchangers A and B.	1971/Winter 1973

The following loads are considered in the design of the tanks:
Hydrostatic, Nozzle and SSE. These loads are combined as required and the resulting stresses are verified to be within the limits of ASME Code Section III, Paragraph ND-3640.

Supports for the Diesel Oil Storage Feed Tanks, Component Cooling Water Surge Tank and Chilled Water Expansion Tank are designed for all loads considered above in accordance with the AISC requirements. The Diesel Oil Storage Tanks are supported on grade without legs.

The following loads are considered in the design of the heat exchangers: Internal Pressure, Nozzle, Deadweight and SSE. The loads are combined to produce the most severe resultant stresses and the stresses are verified to be within the limits of ASME Section III, Table I-7.0.
Supports for the heat exchangers are designed for all loads considered above in accordance with AISC requirements.

All Non-NSSS supplied pumps are Code Class 2 and 3 and are considered active. The design of the pumps and their supports, loading combinations, and stress limits are discussed in Subsection 3.9-3-1.2 and Appendix 3.9B.

3.9.3.2 Operability Assurance Program

Active pumps and valves are defined as components that require a mechanical motion in performing a safety function. The operability (i.e., performance of this mechanical motion) of active components during and after exposure to design bases events is assured by the following:

a) Design of each component to be capable of performing all safety functions during and following design bases events. The design specification includes applicable loading conditions. The requirement that the manufacturer demonstrate operability by analysis or test is also included in the specifications.

b) Analysis and/or test demonstrates the operability of each design under the seismic loadings. Results of the operability demonstration are detailed in Subsections 3.9.3.2.1 and 3.9.3.2.2.

c) Inspection of each component to assure compliance of critical parameters with specifications and drawings. This inspection confirms that specified materials and processes are used, that wall thicknesses meet code requirements, and that fits and finishes meet the specification requirements.

d) Testing of each component to demonstrate as-built condition. This testing confirms acceptability of structural integrity (hydrostatic tests) and leakage characteristics of each active pump and valve. The operation of each pump and valve is also demonstrated by test.

e) Start-up and periodic in-service testing to demonstrate that the active pumps and valves are in operating condition throughout the life of the plant.

f) Thorough review of listings (a) through (e) to assure that they have been successfully accomplished.

CE (NSSS supplier) and Ebasco (Non-NSSS supplier) provided active pumps and valves which are listed in Tables 3.9-8 and 3.9-9, respectively, with a brief description of the active safety function of each.

Tables 3.9-8 and 3.9-9 also indicate components that contain replacement parts which were supplied per NRC Generic Letter 89-09. This letter allows for replacement parts to be supplied from the manufacturer, who no longer has an ASME Section III Certification of Authorization. Without this, the manufacturer cannot supply components or parts with an ASME Section III "N" stamp, or Code Data Sheet. The note in Tables' 3.9-8 and 3.9-9 identifies which components contain replacements, and provides the documentation in this FSAR required by NRC Generic Letter 89-09.

3.9.3.2.1 Operability Assurance of Active Pumps

3.9.3.2.1.1 NSSS Supplied Active Pumps

The high pressure and low pressure safety injection pumps are ordered to the requirements of the Draft ASME Code for Pumps and Valves for Nuclear Power, dated November, 1968, including the March, 1970 addenda.

The Draft ASME Code for Pumps and Valves does not establish pump design criteria but rather it places reliance on the cumulative past experience of the pump designer. Ingersoll Rand, the designer of the HPSI and LPSI pumps, has been established as a centrifugal pump designer and manufacturer for more than 50 years and none of the safety injection pumps' features are of a developmental nature.

Operability of the safety injection pumps is assured both by design and by testing of the pump and motor combination as described in this subsection. Loading conditions are evaluated and conservative loading requirements based upon the most adverse combination of loads are included in the pump specifications. The maximum allowable pump nozzle loads are established by the pump manufacturer. Ingersoll Rand is required to demonstrate by analysis or testing that the pumps would not suffer loss of function when seismic loads are imposed in addition to other applicable loads.

During the safety injection pump design process, limitations are placed on stress levels that may be attained in areas where deformation might affect performance. The stresses in the foundation bolting and bearing bracket bolting are examined to ensure that the pump and motor combination could remain functional during all loading conditions, including the SSE. These stresses were found to be extremely low which precludes any significant deformation in the pump and motor assembly.

Prototype test and analytical data are obtained from the safety injection pumps' motor supplier to verify that the motors are designed with conservative margins to drive the pumps under all required operating conditions.

Routine NEMA tests are also performed on each motor to verify its electrical integrity per MG1-20-46. In addition, the pump and motor assembly designs are subjected to a thermal transient test. This test reproduces the maximum temperature change in the pumped fluid that is expected to occur during the plant faulted condition. Pump and motor performance throughout this transient test were satisfactory. The pump and motor assembly was also satisfactorily tested over the entire required performance range, including NPSH limit tests.

Site installation procedures and a preoperational testing program are relied upon to provide assurance of proper installation and onsite performance of the safety injection pumps. The quality assurance program ensures that the safety injection pumps are designed, manufactured, installed and tested in accordance with applicable component specifications, codes and regulatory requirements. Periodic testing and in-service inspection during their service life provide continued assurance that the safety injection pumps perform as required.

The overall operability assurance program for safety injection pumps described above provides adequate assurance that these pumps will function when called upon to do so. In addition, the component redundancy provided in the safety injection system design accommodates single active random component failures without a loss of required performance levels.

The boric acid pumps are ordered to the Class 2 requirements of Section III of the ASME Boiler and Pressure Vessel Code, dated 1971, including the Summer 1972 addenda. Crane-Deming analytically demonstrated operability of the boric acid makeup pump design under concurrent 1.5g horizontal and 1.0g vertical seismic loads. Stresses at all critical points are in the elastic range and below allowable limits. The impeller clearance is well within the manufacturer's limits. The calculated impeller natural frequency is 54.6 Hz which is beyond the range of significant seismic effects.

The charging pumps are ordered to the requirements of the Draft ASME Code for Pumps and Valves for Nuclear Power, dated November 1968, including the March 1970 addenda. Gaulin Corporation performed an analysis to show that the assembly can withstand the seismic loadings.

3.9.3.2.1.2 Non-NSSS Supplied Active Pumps

Design basis events considered for operability evaluations of active pumps are pipe breaks, OBE, and SSE. Because the pumps are located in moderate energy systems and the piping is restrained in a manner that will not impose pipe rupture loads on the pump nozzles, the only design basis events applicable for operability evaluations are seismic events. Operability of active pumps is therefore based on seismic qualification.

Appendix 3.9A, Seismic Considerations for Equipment Specifications, is the information provided to manufacturers which presents the criteria for seismic qualification of seismic Category 1 equipment and supports. Table 3.9-9 contains design conditions of active pumps.

The pump manufacturers provided documentation in the form of reports explaining their methods of seismic analysis and the results for each piece of equipment supplied.

Appendix 3.9B provides a description of the analyses which references the qualification reports and summarizes results for the following active pumps and their supports:

a) Containment Spray Pumps,

b) Component Cooling Water Pumps,

c) Auxiliary Component Cooling Water Pumps,

d) Chilled Water Pumps,

e) Emergency Feedwater Pumps,

f) Diesel Oil Transfer Pumps,

g) Diesel Engine Driven Jacket Water Pumps,

h) Diesel Engine Driven Lube Oil Pumps, and

i) Component Cooling Water Make-Up Pumps.

The seismic analysis consists of determination of the lowest natural frequency, confirmation of structural integrity and conclusion of operability of the pump and driver assemblies.

3.9.3.2.1.2.1 Natural Frequencies

The lowest natural frequency of the pump-driver assembly is calculated to determine rigidity of the structure. The pump, when having a natural frequency above 33 Hz is considered essentially rigid. This ensures that the seismic accelerations specified in Appendix 3.9A **(1.0g** horizontal, 0.67 vertical) envelope the floor response spectra for frequencies greater than 33 Hz-

3.9.3.2.1.2.2 Structural Integrity

Each pump-driver is analyzed to show that the assembly is structurally adequate to withstand the loads during the design basis event. The parts analyzed include:

a) Pressure retaining parts (casing, nozzles),

b) Supports (bolts (attachment bolts, anchor bolts frame, pedestal), and

c) Shaft.

The stresses for pressure retaining materials are calculated in accordance with the applicable ASME Code and confirmed to be below the allowable stress values, S, of the Tables I-7.1 or I-7.2 whichever is applicable. The applicable Code Edition and Addenda for each pump is determined when the purchase order for the pumps were placed. These are summarized in Appendix 3.9B.

3.9.3.2.1.2.3 Operability

Operability is concluded on the basis of evaluation of the critical parts for structural adequacy, analysis of load bearing parts, clearances and misalignments to ensure that relative motion between moving parts is not impeded by the deflection caused by the design basis event. The parts analyzed include:

a) Pump bearing loads,

b) Flexible coupling misalignment,

c) Impeller to casing clearance,

d) Impeller key stresses,

e) Shaft mechanical seal, and

f) Motor rotor to stator clearance.

3.9.3.2.1.2.4 Loading Criteria

The following load combinations (as described in Appendix 3.9B) were used to calculate stresses:

a) Normal Operating Loads + Nozzle Loads + OBE Loads
b) Normal Operating Loads + Nozzle Loads + SSE Loads

Normal operating loads include internal pressure, shaft torsional and impeller radial and axial loads.

The maximum seismic nozzle loads are considered in analyses of the pump supports to ensure that the pumps are capable to withstand the forces and moments transmitted from the system's piping at the interface of pipe to pump nozzle connections.

The Component Cooling Water Pumps, Component Cooling Water Make-Up Pumps, Auxiliary Component Cooling Pumps and Containment Spray Pumps are designed to sustain suction and discharge nozzle loads determined by Equation Set A. The Diesel Oil Transfer Pump and Chilled Water Pumps are designed to sustain suction and discharge nozzle loads determined by Equation Set B. These loads are conservative and greater than the loads imposed by the piping due to piping and pipe support design.

Equation Set A	Equation Set B
$F_R = 600\ A$	$F_R = 300\ A$
$M_R = 750\ SM$	$M_R = 375\ SM$

where:

F_R = Resultant Force in any Direction, lb

M_R = Resultant Moment in any Direction, ft-lb

A = Area of metal of pipe connected to pump nozzle, sq. in.

SM = Section Modulus of metal pipe connected to pump nozzle, cu. in.

For the Emergency Feedwater Pumps (motor and turbine driven pumps) the pump nozzle design loads were compared with actual piping loads and found to be acceptable. Refer to Appendix 3.9B.6 for the nozzle loads comparison.

Nozzle loads were not specified for the Diesel Engine Jacket Water and Lube Oil Pump, because the pumps do not interface with the main piping system. The pumps are mounted on and furnished with the Diesel Generator, and qualified by the Diesel Engine manufacturer. Refer to Subsection 8.3.1.1.2.13.k.

3.9.3.2.1.2.5 Factory Tests

In addition to the above, each active pump is given factory tests as necessary to determine that the work and materials are free from defects and to establish that the design and construction are satisfactory. A shop hydrostatic test is performed on all pump casings. Each pump is tested over the full operating range by the manufacturer. Performance curves showing head, efficiency and power requirements at various capacities are supplied by the manufacturer. Test runs in the field demonstrate that performance meets the requirements of the equipment specification.

3.9.3.2.2 Operability Assurance of Active Valves

3.9.3.2.2.1 NSSS Supplied Active Valves

The design requirements for active valves require that several conditions be satisfied. The pressure requirements ensure that the pressure boundary of the valve will not be violated as a result of a system pressure transient. Since code rules require that valves have a greater cross section than the matching piping, the pressure boundary integrity is assured. Design requirements for seismic conditions ensure the top works, (i.e., weakest portion of the valve yoke) will be stressed below ASME primary membrane plus bending allowables., Small deflections result since the stresses, hence strains, are kept in the elastic range. Deflections of this magnitude will not inhibit valve operability during a seismic event.

CE design specifications require that the lowest natural frequency of each valve assembly be greater than 20 Hz.

CE specifications require that the seller demonstrate by analysis or test that active valves be capable of withstanding a 3.0g acceleration in any direction.

a) Fisher Controls Company provided evidence that their valves would meet these requirements by analysis, with the exception of yoke strength, which was substantiated by actual yoke fracture tests. Control Components, Inc. provided an analysis to substantiate their design.

   The following valve features were checked:

   1) yoke legs,

   2) yoke lock nuts,

   3) combined bonnet tensile and bending stress,

   4) bonnet bolt stress,

   5) body-line connection welded shear stress, and

   6) natural frequency of actuator frame and yoke.

Pressure loaded components such as the valve bonnet and body bolts were analyzed by the equivalent pressure method in ANSI B31.7-1969 and ASME Code. The combination of normal operating loads, deadweight, and seismic loads was considered.

The allowable stress of each component checked was 1.2 Sm or Sy, whichever was greater as provided for in the 1968 edition the Draft ASME Code.

b) Target Rock Corporation provided analytical evidence that their valves would meet seismic requirements.
The following valve features were checked:

1) yoke legs,

2) combined bonnet tensile and bending stress,

3) bonnet bolt stress, and

4) natural frequency of actuator frame and yoke.

Pressure loaded components such as the valve bonnet and body bolts were analyzed by the equivalent pressure method in ANSI B31.7-1969 and ASME Code. The combination of normal operating loads, deadweight, and seismic loads was considered.

The allowable stress for each component checked was Sm.

c) Wm Powell Co. provided analytical evidence that their valves would meet seismic requirements.

The following valve features were checked:

1) yoke legs,

2) body neck maximum operator thrust,

3) combined bonnet tensile and bending stress,

4) bonnet to body bolt stress,

5) operator to bonnet bolt face stress, and

6) natural frequency of valve.

Pressure and operator thrust loaded components such as the body neck and bonnet to body bolts were analyzed by the equivalent pressure method in ANSI B31.7-1969 and ASME Section III. The combination of normal operating loads, deadweight, and seismic loads was considered.

The allowable stress for each component checked was Sm.

Valve motor operators were manufactured by Limitorque. These operators are representative of the prototype units that were successfully seismically tested in accordance with Ogden Technology Laboratories Report No. 7192-9 dated 09-26-72 and Lockheed Electronics Company Test Report No. 2120-4594 dated 07-31-68 and No. 2639A-4723 dated 09-28-73. These reports show that the operators comply with the intent of IEEE Standard 344-1971 requirements.

Valve motor actuators were manufactured by Limitorque and are identified as models SMB-00-10, SMB-00-15, and SMB-1-40. Limitorque Report B0037, Seismic Qualification Envelope, dated 1-11-80 provides test data that qualifies the entire generic family of available

Limitorque actuators, including those listed above, to IEEE-344-75. The actuators tested were mounted on a fixture capable of simulating seating torque and the fixture was mounted to the shaker table.

Relating to mounting of the actuator for seismic test, section 2.0.8.1 of Report B0037 (included in Report B0058) discusses and illustrates the fixturing of the actuator to the seismic test table. Due to the fact that there is such a wide variation in valve yoke design, it is necessary that Limitorque performs seismic tests with the actuator mounted rigidly to the test table defining the seismic excitation it can withstand that is applied directly to its mounting flange. The same tapped holes that are used to mount the actuator on the valve are used to attach the actuator to the fixture. Application of the maximum g level in each of the 0three axis would qualify the actuator for any mounting position.

Regarding monitoring of the actuator during the seismic test, the switch contacts are monitored for chatter as defined in section 4.1.@ of report B0037 and visually monitored as indicated in section 2.09, of the same report. During the seismic test the actuator is operated from a limit switch position to torque switch trip and back to limit switch position simulating closing and opening of the valve during seismic excitation.

As a minimum three accelerometers were mounted on the actuator undergoing test in each of two horizontal and one vertical axis. Three matching accelerometers were installed on the shaker table.

The tests accomplished to provide generic qualification were completed in 1979 and included the most severe actuator configurations in the generic line as well as typical configurations. Twelve different actuators were tested at three laboratories. None of the above listed actuators were included, however, SMB-1-40 had previously been tested by Aero Nav Laboratories per Limitorque Report B0003. That test was conducted in the same manner as for the generic testing and qualified the operator to IEEE-382-72.

3.9.3.2.2.2 Non-NSSS Supplied Active Valves

The design basis events considered for operability evaluation of Non-NSSS supplied active valves are pipe breaks (MSLB and LOCA whichever is applicable), OBE and SSE. Valves in moderate energy systems are not subject to pipe break loads. Valves in high energy lines have been evaluated for susceptability to pipe break loads. This evaluation is summarized in Table 3.9-16. This evaluation determined that with the exception of the Main Steam Isolation Valves (Tag Nos. 2MS-V602A and 2MS-V604B), pipe break loads are not applicable for operability evaluations for Non-NSSS supplied valves.

Because SSE loads are greater than OBE and will not occur simultaneously, SSE is the only applicable design basis event for operability evaluation of the valves. Operability is, therefore, concluded on the basis of seismic qualification of the valves. Appendix 3.9A summarizes the criteria for seismic qualification of seismic Category I equipment. Appendix 3.9C outlines the methods for seismic qualification of valves and references the qualification reports, submitted by the manufacturers as documentation. This documentation includes an explanation of the qualification method, procedures, and a summary of the results.

The Main Steam Isolation Valves have been evaluated for operability under SSE loads and pipe break loads, in addition to normal operating and actuator loads. Seismic qualification is discussed in Appendix 3.9C Flow conditions against which the valves are to close during a MSLB were specified in the design specification (see Table 3.9-17) and evaluated by the manufacturer. The design of the valve internals was evaluated and concluded to be adequate for MSLB conditions.

For valves with electrical components, i.e. solenoid or motor operators, limit switches, positioners, electro-pneumatic transducers, etc, where these components are required for a safety related function, the components are Class 1E qualified to environmental requirements in accordance with IEEE 323-1971, 1974 and IEEE 344-1971, 1975 or IEEE 382-1972. These qualifications ensure operability of the components during LOCA, MSLB and seismic events.

Although safety and relief valves are not considered active, they have been evaluated for operability during SSE and discharge reaction loads in the same manner as active valves (see above description and Appendix 3.9C).

All valves are hydrostatically shell and seat leak tested at the shop. In addition, valves with actuators are functionally tested to ensure proper operation and compliance with closure/opening time requirements where applicable (i.e. containment isolation valves). Modulating type control valves are also tested for hysteresis and linearity.

3.9.3.3 <u>Design and Installation Details for Mounting of Pressure Relief Devices</u>

→(LBDCR 15-021, R309)
Safety and relief valves for overpressure protection of ASME Code Class 1 and 2 components are designed and installed in accordance with ASME Section III, 1971 edition up to and including winter 1973 addenda with the exception of valves RC200 and 201 which were designed to the 1974/Summer 1975 Addenda. All Code Class 1 and 2 safety and relief valves are listed in Table 3.9-10 including sizes and design conditions.
←(LBDCR 15-021, R309)
The maximum combined stress at the header connection, including the allowable stress for all Code Class 1 and 2 safety and relief valves may be obtained from the pertinent stress analysis calculation given in Table 3.9-10. Analysis of the safety and relief valves includes the effects of the local stresses at the junction of the valve branch and the header and the stresses at the valve inlet.

Discharge piping effects due to valve openings for each valve listed in Table 3.9-10 are analyzed. Dynamic load factors are utilized in the calculations and the total forces and moments on the header resulting from simultaneous discharge of all safety and relief valves are considered in the header design and the support/restraint system design.

Safety and relief valves can be categorized as either open discharge or closed discharge.

a) Open discharge implies blowing into the atmosphere either directly or through a vent stack. The following information is included in the design of an open discharge system:

    1) Thrust forces include pressure and momentum effects.

    2) The minimum moments used in the stress analyses are those specified in ASME Code Case 1569.

    3) The valve thrust loads are considered on the valve inlet piping from the header.

    4) The reaction forces and moments used in the stress calculations are modified by a dynamic load factor (DLF) or by the maximum instantaneous value obtained from a dynamic time-history analysis. A dynamic load factor of 2.0 is used for static analysis if a dynamic analysis is not performed.

5) Stresses due to thermal, internal pressure, seismic effects and thrust loads are compared to those allowable for the header, local stresses and valve inlet piping. These stresses are calculated according to ASME Code, Section III and combined as shown in Table 3.9-11.

6) The analysis of multiple safety or relief valves installed on the same header do follow the guidance of Regulatory Guide 1.67. The analysis considers all valves blowing at a steady state utilizing loads based on a 60%-40% discharge split on the double ported safety valves and on the valve with the highest set pressure a DLF of 2 was used with the loads to simulate the last POP which provided a conservative case. Code Case 1569 was also used in evaluating the safety and relief valves.

b) A closed discharge is restricted from the atmosphere by a piping system from the valve to a component (e.g., a tank). The stresses developed after the initial valve thrust is calculated by either a conservative static method or a time-history computer solution. Water slug effects are included if required.

Conservative calculations described below are performed on each valve and the resulting stresses compared to the ASME Allowables. If the stresses exceed the allowables, this does not necessarily mean that the configuration is unacceptable but rather that more realistic assumptions have to be considered rather than the very conservative ones. The methods used to analyze the valves listed in Table 3.9-10 are described below.

The initial analysis of the valves conservatively assumes that they are open discharge valves supported only by the valve inlet pipe as a cantilever beam from the header. The valve discharge forces conservatively include a dynamic load factor of two. The seismic loads are based on the maximum accelerations given in the appropriate floor response spectra. No credit is taken for any support on the valve discharge piping. The following valves meet the ASME Code allowables under these very conservative assumptions:

2SI-R612B	2SI-R823A	2CC-R21
2SI-R613A/B	2SI-R824B	2CC-R22
2SI-R614A	2SI-R350B	2CC-R23
		2CC-R24
		2SI-R340B
		2CH-R184A/B

The valves that do not pass in accordance with the procedure above are reanalyzed. Using the same assumptions except that since none of the seismically restrained piping has a natural frequency with a period greater than 0.20 seconds, instead of assuming the peak of the entire earthquake floor response spectrum, the maximum accelerations for periods of 0.20 seconds or less are used. Valves meeting the ASME Code allowables under this method of analysis are as follows:

2CH-R626A/B	2SI-R1529	2SI-R1530A
2CH-Rl526A	2SI-R1531A	
2CH-Rl527A/B	2CH-R1528B	
	2CH-R629A/B	

If the valves still do not meet the allowable code limits, the supports/ restraints on the valve discharge pipe are considered. A typical closed system arrangement is shown in Figure 3.9-12. If support/restraint locations are known, enough piping and supports/restraints are modeled to adequately represent the system. A minimum of three restraints are needed of which at least one is on the valve discharge piping.

If support/restraint locations are not known, locations are selected to limit the maximum natural period of the piping system to 0.20 seconds. A computer piping analysis program is used in this analysis. Valves meeting the ASME Code allowables with this method of analysis are as follows:

2SI-R617 TK1A	2SI-R619 TK1B	2SI-R124A
2SI-R618 TK2A	2SI-R620 TK2B	2SI-R125B
2CH-R1515 A/B	2CH-R182 A/B	2SI-R126A/B

Dynamic analyses of the valves listed below were performed using the computer program PLAST 2267 to determine the internal moments and stresses caused by the Safety Valve discharge forces. These loads are then combined with the piping moments and stresses as described above and compared to the allowable stress in accordance with Equation 9 of ASME Section III Code, Paragraphs NB or NC 3652.

2SI-R339A		
2MS-R613A	2MS-R619B	2CH-R1515A/B
2MS-R614A	2MS-R620B	1SI-R2501A
2MS-R615A	2MS-R621B	1SI-R2502B
2MS-R616A	2MS-R622B	
2MS-R617A	2MS-R623B	
2MS-R618A	2MS-R624B	

Dynamic analyses of the valves listed below were performed) using the following computer programs. The computer programs RELAP5[35] and CALPLOTFII were used to calculate the forcing functions acting on the pressurizer relief valve discharge piping system following valve opening. The RELAP5 computer code was verified by test comparisons as part of the EPRI S/RV Test Program to meet the requirements of NUREG-0737 Item II.D.1. The operability of these valves were also verified as part of the EPRI S/RV Test Program and the subsequent plant specific calculations.

The results are used in a computer piping analysis program and then compared to the ASME Code allowable stress.

1RC-R2573A	1RC-R2574B

3.9.3.4 Component Supports

Loading combinations and design transients specified for the components are also taken into account in designing Code Classes 1, 2, and 3 component supports. These include, but are not limited to, the following:

a) weight of component and fluid under operating and test conditions,

b) weight of the component support,

c) loads and reactions induced by the connecting system components,

d) dynamic loads, including seismic loads,

e) restrained thermal expansion, and

f) anchor and support movement effects.

→(DRN 03-2056, R14)
The design loading combinations and transients for NSSS supplied ASME Code Class 1 reactor coolant pressure boundary supports and internal structures are discussed in Subsection 5.4.14 and Subsection 3.9.1.1, respectively. Loadings and stress limits for NSSS supplied ASME Code Classes 2 and 3 component supports are described in Subsection 3.9.3.1. Supports for NSSS supplied active components are considered in Subsection 3.9.3.1.
←(DRN 03-2056, R14)

Pump structural supports and the allowable and calculated stresses are identified in Appendix 3.9B.

The stress limits and loading combinations for pipe supports are listed in Table 3.9-18.

3.9.4 CONTROL ELEMENT DRIVE MECHANISMS

3.9.4.1 Descriptive Information of CEDM

→(DRN 01-1102, R12)
The control element drive mechanisms (CEDMS) are magnetic jack type drives used to vertically position and indicate the position of the control element assemblies (CEAS) in the core. Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its 150 in. nominal stroke in response to operation signals.
←(DRN 01-1102, R12)

→(DRN 01-1102, R12. LBDCR 15-039, R309)
The CEDM is designed to function during and after all normal plant transients. The CEA drop time for 90 percent insertion is 3.2 seconds maximum. The drop time is defined as the interval between the time power is removed from the CEDM coils and the time the CEA has reached 90 percent of its fully inserted position. The CEDM has a design life of 40 years. The CEDM is designed to operate without maintenance for a minimum of 1.5 years and without replacing components for a minimum of three years. The CEDM is designed to function normally during and after being subjected to the operating basis earthquake loads. The CEDM will allow for tripping and drive-in of the CEA during and after a safe shutdown earthquake.
←(DRN 01-1102, R12, LBDCR 15-039, R309)
The design and construction of the CEDM pressure housings fulfill the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for Class 1 vessels. The CEDM pressure housings are part of the reactor coolant pressure boundary, and they are designed to meet stress requirements consistent with those of the vessel. The pressure housings are capable of withstanding, throughout the design life, all normal operating loads, which include the steady state and transient operating conditions specified for the vessel. Mechanical excitations are also defined and included as a normal operating load. The CEDM pressure housings are service rated at 2500 psia and 650 F. The loading combinations and stress limit categories are presented in Table 3.9-12 and are consistent with those defined in the ASME Code.
→(DRN 01-1102, R12)
An extension shaft is coupled to the CEA and joins this system to the CEDM. The extension shaft and CEA coupling system are designed to the allowable stress values of Section III of the ASME Boiler and Pressure Vessel Code. These components are also designed to function during and after a safe shutdown earthquake condition. The components are designed such that the resonant frequencies are outside pump and fluid flow excitation frequencies.
←(DRN 01-1102, R12)
The design life of the CEDM is defined as 40 years of operation or 100,000 ft. of rod travel without loss of function.

The test programs performed in support of the CEDM design are described in Subsection 3.9.4.4.

3.9.4.1.1 Control Element Drive Mechanism Design Description

The CEDMs are mounted on nozzles on top of the reactor vessel closure head. The CEDMs consist of the upper and lower CEDM pressure housings, motor assembly, coil stack assembly, reed switch assemblies, and extension shaft assembly. The CEDM is shown in Figure 3.9-13. The driver power is supplied by the coil stack assembly, which is positioned around the CEDM housing. Two position indicating reed switch assemblies are supported by the upper pressure housing shroud, which encloses the upper pressure housing assembly.

The lifting operation consists of a series of magnetically operated step movements. Two sets of mechanical latches are utilized engaging a notched extension shaft. To prevent excessive latch wear, a means has been provided to unload the latches during the engaging operations. The magnetic force is obtained from large dc magnet coils mounted on the outside of the lower pressure housing.

→(DRN 01-1102, R12)

Power for the electromagnets is obtained from two separate supplies. A control programmer actuates the stepping cycle and obtains the CEA position by a forward or reverse stepping sequence. CEDM "hold" is obtained by energizing one coil at a reduced current, while all other coils are deenergized. The CEAs are tripped upon interruption of electrical power to all coils. Each CEDM is connected to the CEAs by an extenion shaft. The weight of the CEDMs and the CEAs is carried by the pressure vessel head. Installation, removal, and maintenance of the CEDM is possible with the reactor vessel head in place; however, the CEDM is inaccessible during operation of the plant.

The axial position of a CEA in the core is indicated by three independent readout systems. One counts the CEDM steps electronically, and the other two consist of magnetically actuated reed switches located at regular intervals along the CEDM. These systems are designed to indicate CEA position to within ± 2.5 in. of the true location. This accuracy requirement is based on ensuring that the axial alignment between CEAs is maintained within acceptable limits.

←(DRN 01-1102, R12)

The materials in contact with the reactor coolant used in the CEDM are listed in Subsection 4.5.1.

3.9.4.1.1.1 CEDM Pressure Housing

The CEDM pressure housing consists of the motor housing assembly and the upper housing assembly. The motor housing assembly is attached to the reactor vessel head nozzle by means of a threaded joint and is seal welded. Once the motor housing assembly is seal welded to the head nozzle, it need not be removed, since all servicing of the CEDM is performed from the top of the housing. The upper pressure housing is threaded into the top of the motor housing assembly and seal welded. The upper pressure housing encloses the CEDM extension shaft and contains a vent. The top of the upper pressure housing is closed by means of a Versa-Vent tube.

3.9.4.1.1.2 Motor Assembly

The motor assembly is an integral unit that fits into the motor housing and provides the linear motion to the CEA. The motor assembly consists of a latch guide tube, driving latches, and holding latches.

→(DRN 01-1102, R12; 02-1476, R12)

The driving latches are used to perform the major stepping of the CEA. The holding latches hold the CEA during repositioning of the driving latches and perform a load transfer function to minimize latch and extension shaft wear. Engagement of the extension shaft occurs when the appropriate set of magnetic coils are energized. This moves sliding magnets which cam a two bar linkage moving the latches inward. The driving latches move vertically a maximum of 3/4 in. The holding latches move vertically 1/16 in. to perform the load transfer.

←(DRN 02-1476, R12)

3.9.4.1.1.3 Coil Stack Assembly

The coil stack assembly for the single acting CEDM consists of five large dc magnet coils mounted on the outside of the motor housing assembly. The coils supply magnetic force to actuate mechanical latches for engaging and driving the CEA extension shaft. Power for the magnet coils is supplied from two separate supplies. A magnetic coil power programmer actuates the stepping cycle and obtains the correct CEA position by a forward or reverse stepping sequence. CEDM hold is obtained by energizing one coil at a reduced current while all other coils are deenergized. The CEAs are tripped upon interruption of electrical power to all coils. Electrical pulses from the magnetic coil power programmer provide one of the means for transmitting CEA position indication.

←(DRN 01-1102, R12)
→(EC-2800, R307)

A conduit assembly containing the lead wires for the coil stack assembly is located at the side of the upper shroud assembly.

←(EC-2800, R307)

3.9.4.1.1.4 Reed Switch Assembly

→(DRN 01-1102, R12)

Two reed switch assemblies provide separate means for transmitting CEA position indication. Reed switches and voltage divider networks are used to provide two independent output voltages proportional to the CEA position. The reed switch assemblies are positioned so as to utilize the permanent magnet in the top of the extension shaft. The permanent magnet actuates the reed switches as it passes them. The reed switch assemblies are provided with accessible electrical connectors at the top of the upper pressure housing. Three additional pairs of reed switches on each CEDM provide upper electrical limit, lower electrical limit, and dropped rod indications.

3.9.4.1.1.5 Extension Shaft Assembly

The extension shaft assemblies are used to link the CEDMs to the CEA. The extension shaft assembly is a Type 304 stainless steel rod with a permanent magnet assembly at the top for actuating reed switches in the reed switch assembly, a center section called the drive shaft, and a lower end with a coupling device for connection to the CEA.

← (DRN 01-1102, R12)

The drive shaft is a long tube made of Type 304 stainless steel. It is threaded and pinned to the extension shaft. The drive shaft has circumferential notches along the shaft to provide the means of engagement to the CEDM.

→ (DRN 01-1102, R12)

The magnetic assembly consists of a housing, magnet, and plug. The magnet is made of two cylindrical Alnico-5 magnets. This magnet assembly is used to actuate the reed switch position indicators. The magnets are contained in a housing, which is plugged at the bottom. The housing also provides a means of attaching the lifting tool for disengaging the CEA from the extension shaft.

← (DRN 01-1102, R12)

3.9.4.1.1.6 Coupling Between the Drive Shaft and the Control Element

The low end of the extension shaft has a coupling device for connection to the CEA. The device consists of an outer gripper with notched fingers and an inner tapered plunger. The inner plunger is spring loaded in the extension shafts coupled position so as to force the notched fingers of the gripper radially outward. This causes the outer diameter of the gripper to expand and the finger notches "lock" into internal horizontal grooves in the CEA hub. During the uncoupling operation, a coupling/uncoupling tool overcomes the plunger spring preload and pulls the tapered plunger up out of the gripper fingers. This operation allows the gripper fingers to collapse radially and allows the extension shaft assembly to be removed from the CEA.

No inadvertent uncouplings have occurred either during in-house testing or on similar extension shaft designs which presently are in use at six operating plants.

3.9.4.2 Applicable CEDM Design Specifications

→(EC-2800, R307)

The pressure boundary components are designed and fabricated in accordance with the requirements for Class 1 vessels per the 1998 Edition of Section III of the ASME Boiler and pressure vessel Code up to and including the 2000 addenda. The pressure boundary material complies with the requirements of Sections II and IX of the ASME Boiler and Pressure Vessel Code including the 2000 addenda and including Code Case N-4-12.

←(EC-2800, R307)

The adequacy of the design of the CEDM non-pressure boundary components has been verified by prototype accelerated life testing as discussed in Subsection 3.9.4.4.1.1. These are non-pressure boundary components.

→(EC-2800, R307)

The electrical components of the CEDMs were designed to comply with IEEE 323-1971, Standard for Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations, and IEEE 344-1975, Recommended Practice for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations. The electrical components are external to the pressure boundary and are non-pressurized.

←(EC-2800, R307)

The test program to verify the CEDM design is discussed in Subsection 3.9.4.4.

3.9.4.3 Design Loads, Stress Limits and Allowable Deformations

The CEDM stress analyses consider the following loads:

a) reactor coolant pressure and temperature,

b) reactor operating transient conditions (including turbine trip and loss of coolant flow),

c) dynamic stresses produced by seismic loadings,

d) dynamic stresses produced by mechanical excitations,

e) loads produced by the operation and tripping of the mechanism, and

→(DRN 03-2056, R14)

f) dynamic stresses due to BLPBs

Computer codes used in analysis of the above loads are listed in Subsection 3.9.1.2.

The methods used to demonstrate that the CEDMs operate properly under combined seismic and BLPB conditions are presented in Subsection 3.7.3.14.

← (DRN 03-2056, R14)

The design and fabrication of the CEDM pressure boundary components fulfills the requirements of the ASME Code, Section III, for Class 1 vessels. The pressure housings are capable of withstanding throughout the design life, all the steady state and transient operating conditions specified in Table 3.9-12.

The adequacy of the design of the CEDM pressure boundary and non-pressure boundary components has been verified by prototype accelerated life testing as discussed in Subsection 3.9.4.4.1.1.

Clearances for thermal growth and for dimensional tolerances were investigated, and tests have proven that adequate clearances are provided for proper operation of the CEDM.

The latch locations are set by a master gage, and settings are verified by testing at reactor conditions.

A weldable seal closure, per Section III of the ASME Code, is provided for the vent closure assembly in case of leakage.

The motor housing fasteners are positively captured mechanically, and all threaded connections are preloaded before capturing.

The coil stack assembly can be installed or removed simply by lowering or lifting the stack, relative to the CEDM pressure housing, for ease of coil replacement or maintenance.

3.9.4.4 CEDM Performance Assurance Program

3.9.4.4.1 CEDM Testing

3.9.4.4.1.1 Prototype Accelerated Lift Tests

This subsection describes tests performed on a typical magnetic jack control element drive mechanism. Waterford 3 will use essentially the same basic magnetic jack design with the exception of a longer upper pressure housing to accommodate the 150 in. stroke.

The following describes tests performed on a prototype CEDM of a 138 in. stroke magnetic jack design. The tests described below demonstrate that the CEDM will meet specified operating requirements for the duration of plant life with normal maintenance. Since Waterford 3 uses the same basic magnetic jack design, testing of the prototype mechanism for Waterford 3 consisted of verification of the mechanism trip characteristics.

A prototype standard CEDM was subjected to accelerated life tests accumulating 100,000 ft. of travel, equivalent to a 40 year lifetime. The first phase of the accelerated life test consisted of continuous operation of the mechanism at 40 in./min., lifting 230 lbs. for a total travel of 32,500 ft. This test was performed at simulated reactor operating conditions.

Upon completion of the test, the motor bearing surfaces were inspected and measured. A maximum bearing wear of 0.003 in. was measured. This degree of wear is considered acceptable based on the 40 yr. design life.

→(EC-2800, R307)

The second phase of the accelerated life test consisted of 200 full height gravity drops and 20,000 reversals at one extension shaft position. This test was also performed at reactor operating temperatures and pressures with 230 lbs. lift weight. The additional travel accumulated during this phase of testing was 5,400 ft. for a total travel of 38,400 ft. for the 40 in./min. mechanism operation. All drops were completed satisfactorily. Inspection of the extension shaft after this test showed no excessive wear on the shaft in the area of the reversals.

←(EC-2800, R307)

For the third phase of the lift test, the prototype magnetic jack was coupled to a CEA and extension shaft assembly (335 lbs. dry). The test was conducted under operating conditions of 600 F and 2200 psia, and the water chemistry adjusted to 1100 ppm boron. The mechanism was operated lifting the CEA and extension shaft assembly at 20 in./min. for a total of 15,625 ft. and 200 full height drops. Post test inspections again showed that motor bearing wear was negligible. No motor failures were encountered during this test phase. Total accumulated travel up to this phase of testing of the mechanism was 60,000 ft. of travel or 24 years of design life.

Two inadvertent rod drops occurred during the above test phase. These were encountered during rod drop testing, when the mechanism was primarily operated in the withdrawal mode of operation. Driving the mechanism in the insertion mode, after each series of 10 drops, corrected this condition. It was concluded that crud buildup under the pull-down magnet caused difficulty in resetting the mechanism during withdrawal, since the pull-down force on this magnet is less in the withdrawal cycle than in the insertion cycle. It was discovered that by driving the CEA and extension shaft in occasionally, instead of continuously dropping the rods, the crud under the pull-down magnet was displaced, and the malfunction eliminated.

The prototype CEDM was converted to a PLCEDM (non-tripping mechanism) by installing a non-tripping assembly. A sixth coil was added to operate the non-trip magnet and associated latches. The remainder of the mechanism was the same as the previous tripping prototype mechanism and was utilized in subsequent testing to accumulate additional travel. The mechanism was installed on a high temperature test facility and coupled to a test CEA, having a total weight of 230 lbs. dry. Its operating speed was increased to 30 in./min. The test loop conditions were set at 600 F and 2200 psig, and a fourth phase of the accelerated life test was initiated. The object of this accelerated life test was to accumulate an additional 30,000 ft. of continuous operation, utilizing the prototype mechanism as a part-length drive. After a total of 15,000 ft. of operation was accumulated, the holding latch spring in the motor assembly failed. The total travel accumulated on this particular component to that point was 75,000 ft., or an equivalent of 30 yrs. of design life. The holding latch spring was replaced and the life test continued without modification to the other parts.

The spring design was reviewed and an improved spring design incorporated into the production units with a design life of twice that of the failed spring.

During the latter half of the fourth accelerated life test, the drive shaft was distorted. The cause of this distortion was attributed to the operating rod in the extension shaft assembly having been pinned at its upper end, causing differential thermal growth between the drive shaft and operating rod. This design was changed for the production units to allow for the differential thermal expansion between the drive shaft and operating rod.

Drive shaft bowing has not occurred with the new design. This distortion of the drive shaft caused the mechanism to occasionally fail to complete a step. At no time did the CEA drop due to this distortion. Under these test conditions, an additional 18,000 ft. was completed for a total accumulated travel on this prototype mechanism as a part-length unit of 33,000 ft. Total travel on the drive shaft and major components of this motor assembly was 92,000 ft. Upon disassembly and inspection, no additional failures were found. Upon completion of the non-tripping accelerated life test, the prototype mechanism was converted back to a tripping device by removal of the non-tripping device and additional coil.

The reconverted mechanism was again operated in the high temperature test facility at normal reactor temperature and pressure. The driven weight was 230 lbs. dry, including drive shaft and CEA. Limit switches were positioned to give a total stroke of 120 in. over the upper end of the mechanism travel. The remainder of the life test consisted of operating the mechanism for 7,000 ft. Upon completion of this travel, the facility was shut down, and the bearing surfaces remeasured. The measurements were recorded, and the wear was found to be within acceptable limits.

The fast shutdown capability of the tripping magnetic jack mechanism was verified by dropping the minimum effective weight of 130 lbs. dry with the prototype mechanism. A curve showing CEA position versus time was generated utilizing the reed switch position transmitter. The facility test conditions were set at 40 psi pressure, ambient temperature, and no flow. Six drops were completed with acceptable insertion times. Time deviations between the various drops were less than 0.1 sec.

A preproduction mechanism was installed on a full-flow test facility for drop testing under reactor operating conditions including flow.

Two hundred and six full-height trips and 126 partial-height trips were completed under reactor operating conditions including flow. Drop times for 90 percent insertion at full flow reactor operating conditions met the required criteria and displacement curve.

3.9.4.4.1.2 First Production Test

A qualification test program was completed on the first production CE magnetic jack CEDM. During the course of this program, over 4,000 ft. of travel was accumulated and 30 full-height gravity drops were made without mechanism malfunction or measurable wear on operating parts. The program included the following:

a) Operation at 40 in./min. lifting 230 lbs. dry at ambient temperature and 2000 psig pressure for 800 ft.

b) Six full-height 230 lbs. dry weight gravity drops at ambient temperature.

c) Operation at simulated reactor operating conditions at 40 in./min. lifting 230 lbs. for 1,700 ft.

d) Six full-height drops at simulated reactor operating conditions with 230 lbs. of weight.

e) An operational test at ambient temperature and 2300 psig pressure, lifting 335 lbs. at 20 in./min. for 500 ft-

f) Six full-height drops of the 335 lbs. weight.

g) Operation at simulated reactor conditions for 1,700 ft- at 20 in./min. lifting 335 lbs.

→(EC-2800, R307)

h) Operation at ambient temperature and 2300 psig for 1,100 ft. and 20 full-height drops with an attached dry weight of 130 lbs.

←(EC-2800, R307)

The mechanism operated without malfunction throughout the test program and, upon final inspection, no measurable wear was found.

→(EC-2800, R307)

A qualification test program was additionally completed on the first production replacement CE-type magnetic jack CEDM motor assemblies. During the course of this program, over 4,000 ft of travel was accumulated at ambient temperature and pressure and 30 full-height gravity drops were made without mechanism malfunction or measurable wear on operating parts. The program including the following:

a) Operation at 40 in/min, lifting 234 lbs. dry at ambient temperature and pressure for 100 ft.

b) 30 full-height 234 lb dry weight gravity drops at ambient temperature and pressure.

c) Operation at 40 in/min, lifting 234 lbs dry at simulated reactor operating conditions for 300 ft.

d) Six full-height drops at simulated reactor operating conditions with 234 lbs of weight.

←(EC-2800, R307)

3.9.4.4.1.3 Production Tests

→(EC-2800, R307)

All CEDM production units are tested for a minimum of 400 ft. of total travel at combinations of pressure and temperature from ambient up to reactor operating conditions. The CEDM motors are also tested for six full-height gravity drop tests at simulated reactor operating conditions.

←(EC-2800, R307)

3.9.4.4.1.4 Field Tests

→(EC-2800, R307)

After installation of the CEDMs and prior to power operation, the CEDM motors are tested in accordance with the procedures described in Section 14.2.

←(EC-2800, R307)

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

3.9.5.1 Design Arrangements

The components of the reactor internals are divided into two major parts consisting of the core support structure and the upper guide structure assembly. The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the internals and is affixed to the bottom head of the pressure vessel. The in-core instrumentation support system is also considered as part of the reactor internals structures and assemblies. The arrangement of these components is shown in Figure 3.9-9.

3.9.5.1.1 Core Support Structure

The major structural member of the reactor internals is the core support structure. The core support structure consists of the core support barrel and the lower support structure. The material for the assembly is Type 304 stainless steel.

→ (DRN 03-2056, R14)

The core support structure is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel. Alignment is accomplished by means of four equally spaced keys in the flange, which fit into the keyways in the vessel ledge and closure head. The lower flange of the core support barrel supports, secures, and positions the lower support structure and is attached to the lower support structure by means of a welded flexural connection. The lower support structure provides support for the core by means of a core support plate supported by columns mounted on support beams that transmit the load to the core support barrel lower flange. The core support plate provides support and orientation for the lower ends of the fuel assemblies. The core shroud, which provides a flow path for the coolant is also supported and positioned by the core support plate. The lower end of the core support barrel is restricted from excessive lateral and torsional movement by six snubbers that interface with the pressure vessel wall.

← (DRN 03-2056, R14)

3.9.5.1.1.1 Core Support Barrel

The core support barrel is a right circular cylinder including a heavy external ring flange at the top end and an internal ring flange at the lower end. The core support barrel is supported from a ledge on the pressure vessel. The core support barrel, in turn, supports the lower support structure upon which the fuel assemblies rest. Press-fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head, and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

The upper section of the barrel contains two outlet nozzles that interface with internal projections on the vessel nozzles to minimize leakage of coolant from inlet to outlet.

→(DRN 03-2056, R14)
Since the weight of the core support barrel is supported at its upper end, it is possible that seismic and/or pipe break excitation could induce lateral deflection of the structure. Therefore, amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally-spaced lugs around the circumference of the barrel and act as a tongue-and-groove assembly with the mating lugs on the pressure vessel. Minimizing the clearance between the two mating pieces limits the amplitude of deflection. During assembly, as the internals are lowered into the pressure vessel, the pressure vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted. The pressure vessel lugs have bolted, captured in conel X shims, and the core support barrel lug mating surfaces are hardfaced with Stellite to minimize wear. The shims are machined during initial installation to provide minimum clearance. The snubber assembly is shown in Figure 3.9-14.
←(DRN 03-2056, R14)

3.9.5.1.1.2 Core Support Plate and Lower Support Structure

The core support plate is a Type 304 stainless steel plate into which the necessary flow distribution holes for the fuel assemblies have been machined. Fuel assembly locating pins (four for each assembly) are shrunkfit into this plate.

The fuel assemblies and core shroud are positioned on the core support plate, which forms the top support member of a welded assembly consisting of a cylinder, a bottom plate, support columns, and support beams. The core support plate is supported by an arrangement of columns welded at the base to support beams. The bottoms of the beams are welded to the bottom plate, which contains flow holes to provide proper flow distribution. The ends of the beams and the top periphery of the bottom plate are welded to a cylinder that supports the outer edge of the core support plate. The cylinder guides the main coolant flow and limits the core shroud bypass flow by means of holes located near the base of the cylinder.

3.9.5.1.1.3 Core Shroud

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The shroud consists of two Type 304 stainless steel ring sections welded to each other and to the core support plate.

→(DRN 03-2056, R14)
A small gap is provided between the core shroud outer perimeter and the core support barrel, and holes are provided in the girth rings in order to provide upward coolant flow between the core shroud and the core support barrel, thereby minimizing thermal stresses in the core shroud and eliminating stagnant pockets. The core shroud is shown in Figure 3.9-15.

3.9.5.1.2 Upper Guide Structure Assembly

The upper guide structure assembly, shown in Figure 3.9-16, consists of the upper guide structure support plate assembly, control element assembly shrouds, and a fuel assembly alignment plate. The upper guide structure assembly aligns and supports the upper end of the fuel assemblies, maintains the CEA spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during a severe accident condition, protects the control element assemblies (CEAS) from the effect of coolant cross flow in the upper plenum, and supports the in-core instrumentation plate assembly. The upper guide structure assembly is handled as one unit during installation and refueling.
←(DRN 03-2056, R14)

→(DRN 01-1102, R12; 02-1476, R12)
The upper end of the assembly is a structure consisting of a support flange welded to the top of a cylinder. A support plate is welded to the inside of the cylinder approximately in the middle. The support plate is welded to a grid array of deep beams, the ends of which are welded to the cylinder. The support flange contains four accurately machined and located alignment keyways, equally spaced at 90 degree intervals, which engage the core barrel alignment keys. This system of keys and slots provides an accurate means of aligning the core with the closure head and thereby with the CEA drive mechanisms. The support plate aligns and supports the upper end of the CEA shrouds. The shrouds extend from the fuel assembly alignment plate to an elevation above the upper guide structure support plate. The five element CEA shroud consists of a cylindrical upper section welded to a base, and a flow channel structure shaped to provide flow passage for the coolant through the alignment plate, while shrouding the CEAs from cross flow. The shrouds are bolted and lockwelded to the fuel assembly alignment plate. At the upper guide structure support plate, the shrouds are connected to the plate by spanner nuts. The spanner nuts are tightened with proper torque to assure a rigid connection and lockwelded.
←(DRN 01-1102, R12; 02-1476, R12)

→(DRN 01-1102, R12)
←(DRN 01-1102, R12)

→(DRN 03-2056, R14)
The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate engage machined posts on the fuel assembly upper end fittings to provide accurate alignment. The fuel assembly alignment plate also has four equally spaced slots on its outer edge that engage with Stellite hardfaced lugs protruding from the core shroud to limit lateral motion of the upper guide structure assembly during seismic and/or pipe break excitation. The fuel alignment plate bears the upward force of the fuel assembly holddown devices. This force is transmitted from the alignment plate through the CEA shrouds to the upper guide structure support plate. The flange of the upper guide structure support plate is designed to resist axial upward movement of the upper side structure assembly and to accommodate axial differential thermal expansion between the core barrel flange, upper guide structure and pressure vessel flange support ledge, and head flange recess.
←(DRN 03-2056, R14)

3.9.5.1.3 Flow Skirt

The Inconel flow skirt is a right circular cylinder, perforated with flow holes, and reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is supported by nine equally spaced machined sections that are welded to the bottom head of the pressure vessel.

3.9.5.1.4 In-Core Instrumentation Support System

The complete In-core Neutron Flux Monitoring System includes self-powered in-core detector assemblies, supporting structures and guide paths, and an amplifier system to process detector signals.

→(DRN 03-2056, R14)
The self-powered in-core detector assemblies and the computer system are described in Section 7.7. The instrumentation supporting structures and guide paths are described in this subsection and are shown in Figures 3.9-17 and 18.
←(DRN 03-2056, R14)

The support system begins outside the pressure vessel, penetrates the vessel boundary, and terminates at the lower end of the fuel assembly. Each instrument is guided over its full length by the external guidance conduit, the instrument plate structure guide tubes, and the thimbles that extend downward into selected fuel bundles. The in-core instrumentation guide tubes route the instruments so that the detectors are located and spaced throughout the core. The guide tubes and the in-core thimbles are attached to and supported by the instrument plate assembly shown in Figure 3.9-17.

→(DRN 03-2056, R14; EC-1020, R307)
The instrumentation plate assembly fits within the confines of the reactor vessel head and rests in the recessed section of the upper guide structure assembly. Its weight is supported by four bearing pins. The upper guide structure CEA shrouds extend through the instrumentation plate clearance holes. Above the instrumentation plate, the guide tubes bend and are gathered to form stalks that extend into the reactor vessel head instrumentation nozzles. The instrumentation plate assembly is raised or lowered during refuel to insert or withdraw all instruments and their thimbles simultaneously. The pressure boundaries for the individual instruments are at the instrumentation nozzle (Quickloc II) closure, where the external electrical connections to the in-core instruments are also made.
←(DRN 03-2056, R14)

The in-core instrument assemblies utilize a swagelok fitting for the primary pressure boundary seal between the assemblies and the (Quickloc II) closure. The in-core instrument nozzle and (Quickloc II) closure sealing arrangement is shown in Figure 3.9-18.
←(EC-1020, R307)

3.9.5.2 Design Loading Conditions

The following loading conditions are considered in the design of the reactor internals

a) normal operating temperature differences,

b) normal operating pressure differences,

c) flow impingement loads,

d) weights, reactions and superimposed loads,

e) vibration loads,

f) shock loads (including operating basis, safe shutdown earthquakes, and loads resulting from dropping control elements during a scram),

g) anticipated transient loadings,

h) handling loads (not combined with other loads above), and

i) loads resulting from postulated loss-of-coolant accidents.

3.9.5.3 Design Loading Categories

The design loading conditions are categorized in this subsection.

3.9.5.3.1 Normal Operating and Upset

This category includes the combinations of design loadings consisting of normal operating temperature and pressure differences, loads due to flow, weights, reactions, superimposed loads, vibration, shock loads including an operating basis earthquake, and transient loads.

3.9.5.3.2 Faulted

→(DRN 03-2056, R14)
This category consists of the loading combinations of Subsection 3.9.5.3.1 with the exception that the safe shutdown earthquake (in lieu of the OBE) and the loads resulting from the loss-of-coolant accident are included. SSE and LOCA loads are combined via SRSS per NUREG-0484 Rev.1.
←(DRN 03-2056, R14)

3.9.5.4 Design Bases

3.9.5.4.1 Reactor Internals

The stress limits to which the reactor internals are designed are listed in Table 3.9-13.

No emergency condition has been identified for the applicable components, therefore, no appropriate stress criteria are provided.

→(DRN 03-2056, R14)
The operating categories and stress limits are defined in Subsection NG of Section III of the ASME Boiler and Pressure Vessel Code.

The LOCA maximum stress intensities in the reactor internal components are determined utilizing the most conservative combinations of the lateral and vertical LOCA time-dependent loadings in the structural analysis. This LOCA maximum stresses and the maximum stresses resulting from the SSE are then combined via SRSS to obtain the total stress intensities.
←(DRN 03-2056, R14)

To properly perform their functions, the reactor internal structures are designed to meet the deformation limits listed below:

a) Under design loadings plus operating basis earthquake forces, deflection will be limited so that the control element assemblies can function and adequate core cooling is preserved.

→(DRN 03-2056, R14)
b) Under normal operating loadings, plus the SRSS combination of SSE forces and pipe rupture loadings resulting from a break equivalent in size to the largest line connected to the reactor coolant loop piping, deflections are limited so that the core will be held in place, adequate core cooling is preserved, and all CEAs can be inserted. Those deflections which would influence CEA movement are limited to less than 80 percent of the deflections required to prevent CEA insertion.

   The deformation limits listed in paragraph (b) for BLPBs also applied to MCLBs, except the CEA insertability was not required for MCLBs. With elimination of the MCLBs via LBB, the deformation limits associated with these breaks no longer apply.
←(DRN 03-2056, R14)

The allowable deformation limits are listed in the following tabulation. Allowable limits are established as 80 percent of the loss-of-function deflection limits.

Location	Allowable Deflection (in.)
Fuel lower end fitting, lower support structure (vertical)	1.842 (Disengagement)
Fuel upper end fitting, upper guide structure (vertical)	1.243 (Disengagement)
CEA shroud (lateral)	1.178 (CEA Insertion)

→(DRN 03-2056, R14)
In the design of critical reactor vessel internals components which are subject to fatigue, the stress analysis is performed utilizing the design fatigue curve of Figure 1-9.2 of Section III of the ASME Boiler and Pressure Vessel Code. A cumulative usage factor of less than one is used as the limiting criterion.
←(DRN 03-2056, R14)

3.9.5.4.2 In-Core Instrument Support System

The stress limits for the in-core instrument support system are in accordance with those of Subsection NG of Section III of the ASME Boiler and Pressure Vessel Code. These limits apply to the normal operating and upset conditions, and the OBE.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

The preservice and in-service testing programs for Code Class 1, 2 and 3 pumps and valves are being developed to the requirements of ASME Boiler and Pressure Vessel Code, Section XI and 10CFR50.55a (g) to the extent possible. The programs will be implemented to assess operational readiness. The initial in-service tests conducted during the first 120 month period following initial plant operation and tests conducted during successive 120 month periods throughout the service life of the facility will comply, where practical, with those requirements in editions of the Code and addenda in effect twelve months prior to the start of each 120 month period. In addition to this program, operability assurance of active pumps and valves is discussed in Subsection 3.9.3-2.

3.9.6.1 In-service Testing of Pumps

The pumps specified in the programs will be in-service tested (if applicable) according to the requirements of ASME Section XI, Subsection IWP. Applicability will be determined when as-built conditions are obtained in the appropriate systems. The hydraulic and mechanical parameters to be measured or observed are defined in Subsection IWP.

Methods of measurement and appropriate records will be developed in accordance with Subsection IWP of Section XI. The in-service testing program of applicable pumps will be discussed in detail in Chapter 16.

3.9.6.2 In-service Testing of Valves

The valves specified in the programs will be in-service tested (if applicable) according to the requirements of the ASME Code.

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TRANSIENTS USED IN STRESS ANALYSIS OF CODE CLASS 1 COMPONENTS

Normal Conditions

→(DRN 03-2056, R14; 06-555, R15; 06-1001, R15)

Occurrence	Conditions
Heatup and cooldown cycles	500* heatup and cooldown cycles during the design life of the components in the system. The rate of heating and cooling is 100 F/hr between 70 F and 543°F except for the pressurizer which has a rate of 200 F/hr between 70 F and 653 F. The heatup and cooldown rate of the system is administratively limited to assure that these limits will not be exceeded. This condition is based on a normal plant cycle of one heatup and cooldown per month rounded up to the next highest hundred. Analysis for the Pressurizer Surge Line accounts for thermal stratification and striping which result in higher localized heat up and cool down rates.
Power Changes	15,000** power change cycles over the range of 15 percent to 100 percent of full load at a rate of five percent of full load per minute either increasing or decreasing. The ± 10% step Changes in Power transient is defined as ± 100 psi and ± 10 °F (± 20 °F for the pressurizer and surge line) when at operating conditions. The Normal Cycle transient for all but the RSGs is identical to the ± 10% step Changes in Power transient. The 10^6 cycles are selected based on one million-cycles approximating an infinite number of cycles so that the limiting stress is the endurance limit. Grouped together in these transients are: pressure variations associated with fluctuations in pressurizer pressure between the set point for actuation of the backup heaters and the opening of the spray valves; temperature variations associated with the CEA control deadband. The RSG is designed for 2,000 ± 10% step Changes in Power. The 2000 cycles is based on assuming one cycle per week for 50 weeks of the year. (Two weeks of the year are assumed required for refueling.) The Normal Cycle transient for the RSGs is defined as ± 50 psi and ± 6 °F.

←(DRN 03-2056, R14; 06-555, R15; 06-1001, R15)
→(EC-8458, R307) (Power Changes)
←(EC-8458, R307)

Upset Conditions

→(EC-8458, R307)

Occurrence	Conditions
Reactor trip, turbine trip, loss of reactor coolant flow	480 cycles to include any combination. This includes reactor trips due to operator error, equipment malfunction, and a total loss of reactor coolant flow (i.e. a total loss of reactor coolant pump electrical power). This is based on one occurrence per month for the life of the plant.

→(DRN 06-1001, R15)

* The pressurizer is analyzed for 200 pressurizer heatup and 200 pressurizer cooldown cycles. The RSG is analyzed for 300 plant heatup and 300 plant cooldown cycles.

** The RSG is analyzed for 12,000 up power and 12,000 down power change cycles.

←(EC-8458, R307)

TRANSIENTS USED IN STRESS ANALYSIS OF CODE CLASS 1 COMPONENTS

Upset Conditions

Occurrence	Conditions
OBE condition	See Subsection 3.7.3.2 for the procedures used to determine the number of earthquake cycles during the seismic event.

Faulted Condition

➔(DRN 03-2056, R14)

The concurrent loadings produced by normal operation at full power, plus the design basis earthquake, plus pipe rupture are used to determine the faulted plant loading condition.

Emergency Condition

Loss of Secondary Pressure: five cycles postulated loss of secondary pressure due to a severance of a steam nozzle. This is not considered a credible event in forming the design basis of the reactor coolant system. However, it is included to demonstrate that the reactor coolant system components will not fail structurally in the unlikely event that this event does occur. The number of occurrences is an arbitrary number.

🡐(DRN 03-2056, R14)

Test Condition

Occurrence	Conditions
Primary system hydrostatic	10 primary side cycles from 15 psia to 3125 psia at a temperature between 100 F to 400 F. These cycles are based on one initial hydrostatic test plus a major repair every four years for 36 years which includes equipment failure and normal plant cycles. The secondary side of the steam generator is at atmospheric pressure during this test.
➔(EC-8458, R307)	
Primary system leak	Pressure is increased from 15 to 2250 psia at a temperature between 100 F to 400 F. Fatigue consideration of leak test are already accounted for in the plant heatup and cooldown transients (i.e. no additional pressure transients at normal RCS operating pressure need to be included in the plant component design specification).

🡐(EC-8458, R307)

TABLE 3.9-2 (1 of 2) Revision 10 (10/99)

SAFETY CLASS I NON-NSSS PIPING

Line Number	System	Figure
➜ 1SI3 - 117RL1A 1SI8 - 118RL1A 1SI12 - 119RL1A 1SI12 - 120TK1A ← 1RC12 - 37RL1A 1SI3 - 125RL1B 1SI8 - 126RL1B 1SI12 - 127RL1B 1SI12 - 128TK1B 1RC12 - 38RL1B 1SI3 - 133RL2A 1SI8 - 134RL2A 1SI12 - 135RL2A 1SI12 - 136TK2A 1RC12 - 39RL2A 1SI3 - 141RL2B 1SI8 - 142RL2B 1SI12 - 143RL2B 1SI12 - 144TK2B 1RC12 - 40 RL2B	Safety Injection Cold Leg Injection Piping	6.3-1 (for Fig. 6.3-1, Sht. 1, refer to Dwg. G167, Sht. 1)
➜ 1SI14 - 146A ← 1RC14 - 45RL2 1SI14 - 146B 1RC14 - 44RL1	Shutdown Cooling Piping	6.3-1 (for Fig. 6.3-1, Sht. 1, refer to Dwg. G167, Sht. 1) 5.1-3
➜ 1SI3 - 214 1SI3 - 215	Hot Leg Injection	6.3-1 (for Fig. 6.3-1, Sht. 1, refer to Dwg. G167, Sht. 1)
1RC2 - 43 A/B ← 1RC2 - 49RL2B 1CH2 - 58 A/B 1RC2 - 41RL1 1RC2 - 42RL2 1RC2 - 55 A/B 1RC2 - 28 1CH2 - 230	Chemical Volume & Control System; Letdown Piping, Charging and Auxiliary Spray Piping	9.3-6 (for Fig. 9.3-6, Sht. 1, refer to Dwg. G168, Sht 1) 5.1-3

TABLE 3.9-2 (2 of 2) Revision 10 (10/99)

SAFETY CLASS I NON-NSSS PIPING

Line Number	System	Figure
→		
1RC2 - 148		9.3-6 (for Fig. 9.3-6, Sht. 1, refer to Dwg. G168, Sht 1)
←		
1CH2 - 57 A/B 1CH2 - 57A 1CH2 -57B 1CH2 - 78A		5.1-3
1RC3 - 12A 1RC3 - 13B 1RC3 - 14A 1RC3 - 15B 1RC4 - 16 A/B	Reactor Coolant System; Pressurizer Spray and Auxiliary Spray Piping	5.1-3 5.1-3
1RC2 - 55 A/B		
→		
1SI2 - 206 1SI2 - 295	HPSI to SI tank	6.3-1 (for Fig. 6.3-1, Sht. 1, refer to Dwg. G167, Sht. 1)
←		
→		
1SI3 - 278 1SI3 - 279	SDCS to Relief Valve	6.3-1 (for Fig. 6.3-1, Sht. 1, refer to Dwg. G167, Sht. 1)
←		
1RC6 - 18A 1RC6 - 18B	Pressurizer Relief Piping	5.1-3
1RC2 - 46RL1A 1RC2 - 47RL1B 1RC2 - 48RL2A 1RC2 - 50 A/B	Drain Piping	5.1-3

TABLE 3.9-3 Revision 307 (07/13)

TRANSIENTS AND OPERATIVE CONDITIONS
FOR CODE CLASS 1 NON-NSSS PIPING

Plant Event	Cyclic Duty	Plant Condition*	Component Condition*
→(DRN 06-1001, R15)			
Cooldown	500**	N	N
Heatup	500**	N	N
←(DRN 06-1001, R15)			
Power Operation		N	N
Isolation Check Valve Leaks	600	N	N
Check Valve Operability Test	160	N	N
LOCA (No Hot Leg Inj)	1	F	E
Isolation Valve Leaks	40	U	U
LOCA (Hot Leg Inj)	1	F	E
Safety Valve Releases	40	U	U
Loading 5%/Min., Step Increase of 10%	17,000	N	N
→(EC-8458, R307)			
Turbine Reactor Trip Loss of Flow, Loss of Load	480	U	U
←(EC-8458, R307)			
Max. Purification, Boron Dilution	11,000	N	N
Unloading 5%/Min., 10% Step Decrease	17,000	N	N
Loss of Charging, Long and Short Term Isolation	100	U	U
Loss of Letdown	100	U	U

*Definition of the events Normal (N), Upset (U), Emergency (E), Faulted (F) and Test (T) are given in ASME Code Section III, Paragraph NB-3113.

→ (DRN 06-1001, R15; EC-8458, R307)

**The pressurizer is analyzed for 200 pressurizer heatup and 200 pressurizer cooldown cycles. The RSG is analyzed for 300 plant heatup and 300 plant cooldown cycles.

←(DRN 06-1001, R15; EC-8458, R307)

TABLE 3.9-4 (1 of 2)

COMPARISON OF STRUCTURAL AND HYDRAULIC DESIGN PARAMETERS

Parameters[(a)]			Fort Calhoun	Maine Yankee	Waterford 3
Structural					
	R_{mean},	in.	61-5/16	75-1/4	75-3/4
Upper CSB	t,	in.	2	2-1/2	3
	L,	in.	101-3/8	135-5/8	139-1/16
	R_{mean},	in.	61-1/16	74-7/8	75-1/2
Middle CSB	t,	in.	1-1/2	1-3/4	2-1/2
	L,	in	166-1/8	144-3/4	132-1/4
	R_{mean},	in.	60-11/16	75-5/8	75-1/4
Lower CSB	t,	in.	2-1/4	2-1/4	3
	L,	in.	35-5/8	38	58-1/8
Lower cylinder ID,		in.	Integral	141	141
Core cylinder OD,		in.	Integral	145	145
Support cylinder L,		in.	Integral	42	42
Structure supported			Integral	CSB flange	CSB flange
Core shroud supported			Bolted to CSB	Core support plate	Core support plate
	R_{mean},	in.	59-1/16	72-5/8	72-1/4
Cylinger	t,	in.	1-1/2	2	2
UGS	L,	in.	24	24	24
Beams,		in.	24x1-1/2	24x1-1/2	24x1-1/2
Plate	t,	in.	3-1/4	4	4
Thermal shield			Yes	Yes	No
Hydraulic					
No. of loops			2	3	2
Design min flow, 10^6 lbm/hr			71.7	122	147.8
Inlet design temperature, F			547	546	553
Inlet ID, in.			28-3/4	39	35-3/16
Outlet ID, in.			37	39-5/8	48-1/8

(a)
CSB = Core Support Barrel
UGS = Upper Guide Structure
Vel.= Design minimum velocity

TABLE 3.9-4 (2 of 2)

COMPARISON OF STRUCTURAL AND HYDRAULIC DESIGN PARAMETERS

Parameters[a]	Fort Calhoun	Maine Yankee	Waterford 3
Hydraulic (Cont'd)			
Inlet pipe velocity, ft/sec	33.7	39.0	45.0
Downcomer velocity, ft/sec	25.2	24.9	23.7
Core inlet velocity, ft/sec	12.8	12.9	16.7
Outlet pipe velocity, ft/sec	41.5	39.0	51.0
Pump rotational speed, rpm	1200.	1200.	1200.

TABLE 3.9-5

LOADING COMBINATIONS FOR ASME CODE

CLASS I COMPONENTS OTHER THAN VALVES

Condition	Design Loading Combination[a]
Design	PD
Normal[b]	PO+DW
Upset [b]	PO+DW+OBE
	$PO+DW+(OBE^2+DU^2)^{1/2}$
Faulted	$PO+DW+(SSE^2+DF^2)^{1/2}$

(a) Legend:

PD = design pressure

PO = operating pressure

DW = dead weight

OBE = operating basis earthquake

SSE = safe shutdown earthquake

DF = dynamic system loadings associated with a postulated pipe rupture (LOCA)

DU = other transient dynamic events associated with the upset plant condition. (e.g., valve opening and/or closure)

(b) As required by ASME Code Section III, Division I, other loads such as thermal transient thermal gradient, and anchor point displacement portions of the OBE require consideration in addition to the primary stress producing loads listed.

TABLE 3.9-6

STRESS LIMITS FOR ASME CODE

CLASS I COMPONENTS OTHER THAN VALVES

Condition	Stress Limits [(a)]
Normal and upset	NB 3223 and NB 3654
Emergency	NB 3224 and NB 3655
Faulted	See Paragraph 3.9.1.4

(a) As specified in ASME Section III, 1971 and applicable addenda.

TABLE 3.9-7 Revision 2 (12/88)

DESIGN LOADING COMBINATIONS AND STRESS LIMITS FOR ASME CODE CLASS 2 AND 3 PIPING

Plant Operating Condition	Design Stress Limits	Design Loading Combinations
Normal (3), (4)	1.0 S_h	PO + DW
Upset (3), (4)	1.2 S_h	PO + DW + RVC + OBE(2)(5) PO + DW + FVC + OBE(3)(5) PO + DW + RVO + OBE(2)(5) PO + DW + DBW
Faulted	2.4 S_h	PO + DW + SSE + RVC(5) PO + DW + SSE + FVC(5) PO + DW + DF PO + DW + SSE + RVO(5)

Notation

PO -	Operating Pressure
DW -	Dead Weight (includes sustained mechanical loads)
OBE -	Operating Basis Earthquake (50 percent of the SSE)
SSE -	Safe Shutdown Earthquake
RVC -	Relief Valve- closed system (transient)
RVO -	Relief Valve - open system
FVC -	Fast Valve Closure
DF -	Dynamic Events associated with the faulted condition (e.g, Pipe Rupture, Jet Impingement, Tornadoes, Explosions - effects)
S -	Allowable material stress at the maximum temperature
DBW -	Design Basis Wind

Notes:

1) As defined in ASME Section III, Paragraph NB-3113.
2) OBE will be considered in all safety-related ASME III, COde Class 2 and 3 components.
3) Thermal expansion stresses will be included when evaluating Equations (10) and (11) of NC-3652.
4) Anchor displacement effects due to earthquake will be included Equations (9) or (11) of NC-3652.
5) The Square Root Sum of Squares (SRSS) Combination for earthquake (OBE or SSE) and valve opening and closure (RVC, RVO or FVC) may be used.

TABLE 3.9-8 (Sheet 1 of 2) Revision 10 (10/99)

NSSS SUPPLIED ACTIVE VALVES AND PUMPS

→

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Boric Acid Makeup System (BAM)										
Boric Acid Pumps	BAMMPMP0001A BAMMPMP0001B	N/A	2	Horizontal, Centrifugal	Motor	Crane-Deming	3	143 gpm 231 ft TDH 250 ^{0}F	-	Operate
Boric Acid Makeup Tank Gravity Feed Valves	BAMMVAAA113A BAMMVAAA113B	3CH-V106A 3CH-V107B	2	Gate	Motor	Powell	3	200 psig 250 ^{0}F	3”	Open
Boric Acid Makeup Pump Recirculation Valves	BAMMVAAA126A BAMMVAAA126B	3CH-F170A 3CH-F171B	2	Globe	Diaphragm	Fisher	3	200 psig 250 ^{0}F	1”	Close
Emergency Boration Valve	BAMMVAAA133	3CH-V112A/B	1	Gate	Motor	Powell	3	200 psig 250 ^{0}F	3”	Open
Chemical and Volume Control System (CVC)										
CVC Charging Pumps	CVCMPMP0001A CVCMPMP0001B CVCMPMP0001AB	N/A	3	Horizontal, Positive Displacement	Motor	Gaulin Corp	2	44 gpm 2,410 psig 250 ^{0}F	-	Operate
Letdown Stop Valve	CVCMVAAA101	1CH-F1516A/B	1	Globe	Diaphragm	Fisher	1	2,485 psig 650 ^{0}F	2”	Close
Volume Control Tank Discharge Valve	CVCMVAAA183	2CH-V123A/B	1	Gate	Motor	Powell	2	200 psig 250 ^{0}F	4”	Close
Volume Control Tank Makeup Isolation Valve	CVCMVAAA510	3CH-F117A/B	1	Globe	Diaphragm	Fisher	3	200 psig 250 ^{0}F	3”	Close

←

TABLE 3.9-8 (Sheet 2 of 2) Revision 305 (11/11)

NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Safety Injection System (SI)										
LPSI Pumps	SI MPMP0001A SI MPMP0001B	N/A	2	Vertical, Centrifugal	Motor	Ingersoll-Rand	2	4,150 gpm 342 ft TDH 400 ^{0}F	-	Operate
HPSI Pumps	SI MPMP0002A SI MPMP0002B SI MPMP0002AB	N/A	3	Horizontal, Centrifugal	Motor	Ingersoll-Rand	2	405 gpm 2,830 ft TDH 400 ^{0}F	-	Operate
LPSI Pump Discharge FCVs	SI MVAAA129A SI MVAAA129B	2SI-FM317A 2SI-FM348B	2	Butterfly	Piston	Fisher	2	650 psig 400 ^{0}F	10”	Close
→(EC26496, R305) LPSI Header Isolation Valves	SI MVAAA138A SI MVAAA138B SI MVAAA139A SI MVAAA139B	2SI-V1543B2 2SI-V1539B1 2SI-V1541A2 2SI-V1549A1	4	Globe	Motor	Target Rock	2	2,485 psig 650 ^{0}F	6”	Open/Close
←(EC26496, R305) HPSI Header Isolation Valves	SI MVAAA225A SI MVAAA225B SI MVAAA226A SI MVAAA226B SI MVAAA227A SI MVAAA227B SI MVAAA228A SI MVAAA228B	2SI-V1550A1 2SI-V1545B1 2SI-V1546A2 2SI-V1540B2 2SI-V1542A3 2SI-V1547B3 2SI-V1548A4 2SI-V1544B4	8	Globe	Motor	Target Rock	2	2,485 psig 650 ^{0}F	2”	Open
SIT Leakage Drain Valves	SI MVAAA303A SI MVAAA303B SI MVAAA304A SI MVAAA304B	1SI-F1551TK1A 1SI-F1552TK1B 1SI-F1553TK2A 1SI-F1554TK2B	4	Gate	Diaphragm	Anchor Darling	1	2,485 psig 650 ^{0}F	1”	Close

TABLE 3.9-9 (Sheet 1 of 24) Revision 307 (07/13)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Auxiliary Component Cooling Water System (ACC)										
ACC Water Pumps	ACCMPMP0001A ACCMPMP0001B	N/A	2	Horizontal, Centrifugal	Motor	Babcock & Wilcox	3	6,500 gpm 145 ft TDH 125 ^{0}F	-	Operate
→(DRN 98-1951, R11)										
ACC Pump Discharge Check	ACCMVAAA108A ACCMVAAA108B	3CC-V286A 3CC-V287B	2	Check	-	TRW Mission/ Atwood & Morrill	3	125 psig 125 ^{0}F	16”	Open/ Close
←(DRN 98-1951, R11)										
ACC Pump Discharge Line Isolation	ACCMVAAA110A ACCMVAAA110B	3CC-B288A 3CC-B289B	2	Butterfly	Motor	Jamesbury	3	125 psig 125 °F	16”	Open/ Close
ACC Header to Essential Chillers	ACCMVAAA112A ACCMVAAA112B	3CC-F276A 3CC-F277B	2	Butterfly	Piston	Jamesbury	3	125 psig 125 ^{0}F	6”	Open/ Close
ACC Header to Essential Chillers	ACCMVAAA113A ACCMVAAA113B	3CC-V280A 3CC-V281B	2	Check	-	TRW Mission	3	125 psig 125 ^{0}F	6”	Open
ACC Header to EFW Isolation	ACCMVAAA114A ACCMVAAA114B	3CC-B311A 3CC-B312B	2	Butterfly	Manual	Enertech	3	125 psig 125 ^{0}F	6”	Open
ACC Header to EFW Drain	ACCMVAAA115A ACCMVAAA115B	3CC-V605-1 3CC-V604-24	2	Globe	Manual	Velan	3	125 psig 125 ^{0}F	1”	Close
ACC Header to EFW Isolation	ACCMVAAA116A ACCMVAAA116B	3CC-V210A 3CC-V211B	2	Gate	Manual	Pacific	3	125 psig 125 ^{0}F	6”	Open
→(EC-41355, R307)										
ACC Header CCW HX Temp Control	ACCMVAAA126A ACCMVAAA126B	3CC-TM290A 3CC-TM291B	2	Butterfly	Piston / Manual	Fisher Controls	3	125 psig 125 ^{0}F	12”	Open/ Close
←(EC-41355, R307)										
Wet Cooling Tower Cross-connect Isolation	ACCMVAAA138A ACCMVAAA138B	3CC-F284A 3CC-F285B	2	Butterfly	Diaphragm	Jamesbury	3	75 psig 125 ^{0}F	4”	Open
ACC Header Rtn from Essential Chillers Isol	ACCMVAAA139A ACCMVAAA139B	3CC-F278A 3CC-F279B	2	Butterfly	Piston	Jamesbury	3	125 psig 125 ^{0}F	6”	Open/ Close

TABLE 3.9-9 (Sheet 2 of 24) Revision 12 (10/02)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
ACC Header Rtn from Essential Chillers Check	ACCMVAAA140A ACCMVAAA140B	3CC-V282A 3CC-V283B	2	Check	-	TRW Mission	3	125 psig 125 ^{0}F	6”	Open
ACC Jockey Pump Discharge Check	ACCMVAAA1045A ACCMVAAA1045B	N/A	2	Check	-	BNL Industries	3	125 psig 125 ^{0}F	2”	Close
ACC Pump Recirc Check	ACCMVAAA1051A ACCMVAAA1051B	N/A	2	Check	-	BNL Industries	3	125 psig 125 ^{0}F	3”	Open
Annulus Negative Pressure System (ANP)										
ANP Upstream/ Downstream Containment Isolation	ANPMVAAA101 ANPMVAAA102	3HV-B175A 3HV-B176B	2	Butterfly	Piston	Fisher Controls	3	-26 in. w g 150 ^{0}F	6”	Close
Atmosphere (Containment) Radiation Monitoring System (ARM)										
ARM Containment Isolation	ARMISV0103 ARMISV0109 ARMISV0110	2CA-E606A 2CA-E604B 2CA-E605A	3	Globe	Solenoid	Valcor	2	50 psig 400 ^{0}F	3/4”	Close
	ARMMVAAA104	2CA-V607	1	Check	-	Velan	2	50 psig 400 ^{0}F	1”	Close
Boric Acid Makeup System (BAM)										
Boric Acid Gravity Feed Check	BAMMVAAA115	2CH-V128A/B	1	Check	-	Anchor Darling	2	150 psig 250 ^{0}F	3”	Open/ Close
BAM Pump Discharge Check	BAMMVAAA129A BAMMVAAA129B	3CH-V108A 3CH-V110B	2	Check	-	Anchor Darling	3	150 psig 200 ^{0}F	3”	Open
Emergency Boration Header	BAMMVAAA135	2CH-V130A/B	1	Check	-	Anchor Darling	2	150 psig 200 ^{0}F	3”	Open
BAM Header Flow Control	BAMMVAAA141	3CH-FM172A/B	1	Globe	Diaphragm	Fisher Controls	3	200 psig 250 ^{0}F	1”	Close

TABLE 3.9-9 (Sheet 3 of 24) Revision 14 (12/05)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Blowdown System (BD)										
S/G Blowdown Inside Cntmt Isol.	BDMVAAA102A BDMVAAA102B	2BD-F603 2BD-F605	2	Gate	Piston	Anchor Darling	2	1,085 psig 555 ^{0}F	4”	Close
S/G Blowdown Outside Cntmt Isol	BDMVAAA103A BDMVAAA103B	2BD-F604 2BD-F606	2	Globe	Diaphragm	Masoneilan	2	1,085 psig 555 ^{0}F	4”	Close
Boron Management System (BM)										
Reactor Drain Tank Cntmt Isolation	BMMVAAA109 BMMVAAA110	2BM-F108A/B 2BM-F109A/B	2	Diaphragm	Diaphragm	ITT Grinnell	2	150 psig 263 ^{0}F	3”	Close
Containment Atmospheric Purge System (CAP)										
→(DRN 05-1063, R14)										
Containment Purge Inlet Isolation Valves	CAPMVAAA103 CAPMVAAA104	2HV-B151A 2HV-B152A	2	Butterfly	Piston	Fisher Controls	2	44 psig 300 ^{0}F	48”	Close
Containment Purge Exhaust Isolation Valves	CAPMVAAA203 CAPMVAAA204	2HV-B153B 2HV-B154B	2	Butterfly	Piston	Fisher Controls	2	44 psig 300 ^{0}F	48”	Close
←(DRN 05-1063, R14)										
Containment Atmospheric Release System (CAR)										
CAR Supply Header Containment Isolation	CARMVAAA101A CARMVAAA101B	2HV-B188A 2HV-B187B	2	Butterfly	Manual	Jamesbury	2	44 psig 300 ^{0}F	4”	Open
	CARMVAAA102A CARMVAAA102B	2HV-B184A 2HV-B185B	2	Check	-	GPE Controls	2	150 psig 269 ^{0}F	4”	Open/ Close
CAR Exhaust Header Pressure Control Inlet	CARMVAAA200B	2HV-F228A	1	Ball	Diaphragm	ITT Grinnell	2	44 psig 300 ^{0}F	4”	Close
→ (DRN M99-1090, R11; 05-1133, R14)										
CAR Exhaust Header Inlet Isolation	CARMVAAA201A CARMVAAA201B	2HV-F253A 2HV-F254B	2	Butterfly	Motor	Fisher Controls	2	44 psig 150 ^{0}F	4”	Open/ Close
← (DRN M99-1090, R11; 05-1133, R14)										
CAR Exhaust Hdr Upstream Isol.	CARMVAAA202A CARMVAAA202B	2HV-B190A 2HV-F229B	2	Butterfly Ball	Manual Diaphragm	Jamesbury ITT Grinnell	2	44 psig 300 ^{0}F	4”	Open Open/ Close

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
CAR Exhaust Hdr Downstream Isol.	CARMVAAA203A CARMVAAA203B	2HV-B189A 2HV-B191B	3	Butterfly	Manual	Jamesbury	2	44 psig 300 ^{0}F	4”	Open
CAR Exhaust Header Discharge	CARMVAAA204A CARMVAAA204B	2HV-B167A 2HV-B168B	2	Butterfly	Motor	Fisher Controls	2	-26 in. wg 150 ^{0}F	4”	Open/ Close
Component Cooling Water System (CC)										
Component Cooling Water Pumps	CCMPMP0001A CCMPMP0001B	N/A	3	Horizontal Centrifugal	Motor	Babcock and Wilcox	3	6,800 gpm 145 ft TDH	-	Operate
	CCMPMP0001AB							162 ^{0}F		
→(EC-41355, R307)										
CCW Pump Suction Header	CCMVAAA114A CCMVAAA114B CCMVAAA115A CCMVAAA115B	3CC-F113A/B 3CC-F116A/B 3CC-F114A/B 3CC-F115A/B	4	Butterfly	Piston / Manual	Fisher Controls	3	125 psig 175 ^{0}F	20”	Close
←(EC-41355, R307)										
CCW Pump Discharge	CCMVAAA123A CCMVAAA123B	3CC-V101A 3CC-V102B	3	Check	-	TRW Mission	3	125 psig 175 ^{0}F	20”	Open/ Close
	CCMVAAA123A/B	3CC-V103A/B								Open
→(EC-41355, R307)										
CCW Pump Discharge Header	CCMVAAA126A CCMVAAA126B CCMVAAA127A CCMVAAA127B	3CC-F109A/B 3CC-F112A/B 3CC-F110A/B 3CC-F111A/B	4	Butterfly	Piston / Manual	Fisher Controls	3	125 psig 175 ^{0}F	20”	Close
←(EC-41355, R307)										
Dry Cooling Tower Bypass	CCMVAAA134A CCMVAAA134B	3CC-B265A 3CC-B262B	2	Butterfly	Piston	Jamesbury	3	125 psig 175 ^{0}F	16”	Open/ Close
Dry Cooling Tower Isolation Valves	CCMVAAA135A CCMVAAA135B	3CC-B201A 3CC-B203B	2	Butterfly	Piston	Jamesbury	3	125 psig 175 ^{0}F	20”	Open/ Close
Dry Cooling Tower Tube Bundle Isolation	CCMVAAA137A CCMVAAA137B CCMVAAA139A CCMVAAA139B CCMVAAA141A CCMVAAA141B	3CC-B81A 3CC-B71B 3CC-B82A 3CC-B72B 3CC-B83A 3CC-B73B	16	Butterfly	Manual	Jamesbury	3	150 psig 175 ^{0}F	8”	Close

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
	CCMVAAA143A CCMVAAA143B CCMVAAA173A CCMVAAA173B CCMVAAA175A CCMVAAA175B CCMVAAA177A CCMVAAA177B CCMVAAA179A CCMVAAA179B	3CC-B84A 3CC-B74B 3CC-B94A 3CC-B64B 3CC-B93A 3CC-B63B 3CC-B92A 3CC-B62B 3CC-B91A 3CC-B61B								
Dry Cooling Tower Outlet Header	CCMVAAA181A CCMVAAA181B	3CC-V202A 3CC-V204B	2	Check	-	TRW Mission	3	125 psig 175 ^{0}F	20”	Open/ Close
→(EC-41355, R307)										
CCW to Fuel Pool Heat Exchanger	CCMVAAA200A CCMVAAA200B	3CC-F122A 3CC-F123B	2	Butterfly	Piston / Manual	Fisher Controls	3	125 psig 125 ^{0}F	16”	Close/ Open*
←(EC-41355, R307)										
CCW to Essential Chillers Isol.	CCMVAAA301A CCMVAAA301B	3CC-F272A 3CC-F273B	2	Butterfly	Piston	Jamesbury	3	125 psig 175 ^{0}F	6”	Open/ Close
CCW to Essential Chillers Check	CCMVAAA302A CCMVAAA302B	3CC-V176A 3CC-V177B	2	Check	-	TRW Mission	3	125 psig 175 ^{0}F	6”	Open
CCW from Essential Chillers	CCMVAAA322A CCMVAAA322B	3CC-F274A 3CC-F275B	2	Butterfly	Piston	Jamesbury	3	125 psig 175 ^{0}F	6”	Open/ Close
CCW from Essential Chillers	CCMVAAA323A CCMVAAA323B	3CC-V178A 3CC-V179B	2	Check	-	TRW Mission	3	125 psig 175 ^{0}F	6”	Open
EDG CCW Flow Control	CCMVAAA413A CCMVAAA413B	3CC-F268A 3CC-F269B	2	Butterfly	Piston	Jamesbury	3	125 psig 175 ^{0}F	6”	Open
CCW NNS Supply Header Isol.	CCMVAAA501	3CC-F133A/B	1	Butterfly	Piston	Fisher Controls	3	125 psig 125 ^{0}F	12”	Close
CCW NNS Return Header Isol.	CCMVAAA562	3CC-F132A/B	1	Butterfly	Piston	Fisher Controls	3	125 psig 175 ^{0}F	12”	Close
→(EC-41355, R307)										
CCW Header Return Isol.	CCMVAAA563	3CC-F121B	1	Butterfly	Piston / Manual	Fisher Controls	3	125 psig 175 ^{0}F	16”	Close/ Open*
←(EC-41355, R307)										

→(EC-41355, R307)
***Open function is important to safety for restoring Spent Fuel Pool Cooling**
←(EC-41355, R307)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

→(EC-41355, R307)

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
FP Hx Temp. Cont.	CCMVAAA620	3CC-FM138A/B	1	Butterfly	Diaphragm / Manual	Fisher Controls	3	125 psig 175 ^{0}F	12”	Close / Open*
←(EC-41355, R307)										
Letdown HX Temp. Control	CCMVAAA636	3CC-TM169A/B	1	Globe	Diaphragm	Valtek	3	150 psig 175 ^{0}F	6”	Close
CCW to Outside Containment	CCMVAAA641	2CC-F146A/B	1	Butterfly	Piston	Jamsebury	2	125 psig 125 ^{0}F	10”	Close
→(DRN 06-499, R14-B; 05-1572, R15)										
CCW to Inside Containment	CCMVAAA644	2CC-V242A/B	1	Check	-	TRW Mission	2	125 psig 125 ^{0}F	10”	Open / Close
←(DRN 06-499, R14-B; 05-1572, R15)										
CCW Return Containment Isolation	CCMVAAA710 CCMVAAA713	2CC-F243A/B 2CC-F147A/B	2	Butterfly	Piston	Jamsebury	2	125 psig 175 ^{0}F	10”	Close
→(EC-41355, R307)										
CCW Header Return Isol.	CCMVAAA727	3CC-F120A	1	Butterfly	Piston / Manual	Fisher Controls	3	125 psig 175 ^{0}F	16”	Close / Open*
←(EC-41355, R307)										
PASS Chlr CCW Outlet Check	CCMVAAA80312B	3CC-V693	1	Check	-		3	125 psig 125 ^{0}F	1”	Close
→(DRN 99-1081, R11)										
←(DRN 99-1081, R11)										
PASS HX CCW Outlet Check	CCMVAAA8068	3CC-V710	1	Check	-	Anchor Darling	3	125 psig 175 ^{0}F	1”	Close
Containment Fan Coolers CCW Inlet Isol.	CCMVAAA807A CCMVAAA807B CCMVAAA808A CCMVAAA808B	2CC-F154A1 2CC-F157B2 2CC-F155A2 2CC-F156B1	4	Butterfly	Piston	Jamesbury	2	200 psig 263 ^{0}F	8”	Open
Containment Fan Coolers CCW Outlet Isol.	CCMVAAA822A CCMVAAA822B CCMVAAA823A CCMVAAA823B	2CC-F159A2 2CC-F160B1 2CC-F158A1 2CC-F161B2	4	Butterfly	Piston	Jamesbury	2	200 psig 263 ^{0}F	8”	Open
Containment Fan Coolers Temp. Control	CCMVAAA835A CCMVAAA835B	3CC-TM148A 3CC-TM149B	2	Butterfly	Diaphragm	Fisher Controls	3	125 psig 175 ^{0}F	8”	Open

→(EC-41355, R307)
*Open function is important to safety for restoring Spent Fuel Pool Cooling
←(EC-41355, R307)

TABLE 3.9-9 (Sheet 7 of 24) Revision 12 (10/02)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Shutdown HX CCW FCVs.	CCMVAAA963A CCMVAAA963B	3CC-F130A 3CC-F131B	2	Butterfly	Piston	Jamesbury	3	125 psig 175 ^{0}F	10”	Open
DCT Pressure Equalization Isol.	CCMVAAA1811A CCMVAAA1811B	N/A	2	Globe	Manual	Velan	3	135 psig 175 ^{0}F	2”	Open
Essential Chilled Water System (CHW)										
Chilled Water Pumps	CHWMPMP0001A CHWMPMP0001B CHWMPMP0001AB	P1(3A-SA) P1(3B-SB) P1(3C-SA/B)	3	Horizontal Centrifugal	Motor	Ingersoll- Dresser	3	525 gpm 140 ft TDH 104 ^{0}F	-	Operate
CHW Pump Discharge	CHWMVAAA114A CHWMVAAA114B CHWMVAAA114AB	3CC-V114A 3CC-V115B 3CC-V116AB	3	Check	-	TRW Mission	3	120 psig 104 ^{0}F	6”	Open
SVSMAHU0002A CHW Outlet FCV	CHWMVAAA578	3AC-TM137A	1	Globe	Hydra-motor	Masoneilan	3	120 psig 104 ^{0}F	2”	Open
SVSMAHU0001A CHW Outlet FCV	CHWMVAAA591	3AC-TM188A	1	Globe	Hydra-motor	Masoneilan	3	120 psig 104 ^{0}F	4”	Open
HVCMAHU0001A CHW Outlet FCV	CHWMVAAA603	3AC-TM163A	1	Globe	Hydra-motor	Masoneilan	3	120 psig 104 ^{0}F	4”	Open
HVCMAHU0002B CHW Outlet FCV	CHWMVAAA887	3AC-TM161B	1	Globe	Hydra-motor	Masoneilan	3	120 psig 104 ^{0}F	2”	Open
HVCMAHU0001B CHW Outlet FCV	CHWMVAAA900	3AC-TM189B	1	Globe	Hydra-motor	Masoneilan	3	120 psig 104 ^{0}F	4”	Open
HVCMAHU0001B CHW Outlet FCV	CHWMVAAA919	3AC-TM172B	1	Globe	Hydra-motor	Masoneilan	3	120 psig 104 ^{0}F	4”	Open
Essential Chiller Oil Cooler Pumps	RFRMPMP0001A RFRMPMP0001AB RFRMPMP0001B	N/A	3	Rotary Vane	Motor	Carrier	3	40 gpm 29 psig	-	Operate

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Condensate Makeup and Storage System (CMU)										
CCW Makeup Pumps	CMUMPMP0004A CMUMPMP0004B	N/A	2	Horizontal Centrifugal	Motor	Babcock and Wilcox	3	600 gpm 150 ft TDH 175 ^{0}F	-	Operate
CCW Makeup Pump Discharge Alternate Supply	CMUMVAAA21312A CMUMVAAA21312B	3CC-V670-1 3CC-V670-2	2	Check	-	Velan	3	125 psig 175 ^{0}F	1”	Close
→(DRN 04-515, R13-B) Condensate Makeup to Containment ←(DRN 04-515, R13-B)	CMUMVAAA245	2DW-V643	1	Check	-	Velan	2	150 psig 125 ^{0}F	1.5”	Open/ Close
CCW Makeup Pump Recirc Check	CMUMVAAA508A CMUMVAAA508B	3CC-V629-1 3CC-V629-2	2	Check	-	Velan	3	125 psig 120 ^{0}F	1’	Open
CCW Makeup Pump Discharge	CMUMVAAA510A CMUMVAAA510B	3CC-V229A 3CC-V231B	2	Check	-	TRW Mission	3	125 psig 120 ^{0}F	4”	Open/ Close
EDG Standpipe LCV	CMUISV0524A CMUISV0524B	3CC-E640A 3CC-E641B	2	Globe	Solenoid	Valcor	2	125 psig 120 ^{0}F	1”	Open/ Close
CHW Expansion Tank LCV	CMUISV0532A CMUISV0532AB CMUISV0532B	3AC-E611A 3AC-E613A/B 3AC-E612B	3	Globe	Solenoid	Valcor	2	1454 psig 120 ^{0}F	1”	Open/ Close
CCW Makeup Pump to EDG Isol	CMUMVAAA523A CMUMVAAA523B	3CC-V608-6 3CC-V608-7	2	Globe	Manual	Velan	3	125 psig 175 ^{0}F	1”	Close
CCW Makeup Pump to CHW Expansion Tank	CMUMVAAA527A CMUMVAAA527B	3CC-V647-1 3CC-V647-2	2	Globe	Manual	Velan	3	125 psig 175 ^{0}F	2”	Close
CCW Return Hdr Makeup Isol	CMUMVAAA536A CMUMVAAA536B	3CC-V234A 3CC-V236B	2	Gate	Manual	Pacific	3	125 psig 120 ^{0}F	4”	Close
CCW Return Hdr Makeup CV	CMUMVAAA538A CMUMVAAA538B	3CC-F240A 3CC-F241B	2	Globe	Diaphragm	Masoneilan	3	125 psig 120 ^{0}F	4”	Open

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Containment Spray System (CS)										
Containment Spray Pumps	CSMPMP0001A CSMPMP0001B	N/A	2	Horizontal Centrifugal	Motor	Babcock and Wilcox	2	1,810 gpm 485 ft TDH 240 ^{0}F	-	Operate
CS Pump Discharge	CSMVAAA111A CSMVAAA111B	2CS-V301A 2CS-V302B	2	Stop Check	Manual	Anchor Darling	2	650 psig 400 ^{0}F	10”	Open/ Close
Shutdown Cooling HX Outlet	CSMVAAA117A CSMVAAA117B	2CS-V303A 2CS-V304B	2	Stop Check	Manual	Anchor Darling	2	650 psig 350 ^{0}F	10”	Open/ Close
CS Header Containment Isolation	CSMVAAA125A[(a)] CSMVAAA125B	2CS-F305A 2CS-F306B	2	Gate	Piston	WKM	2	300 psig 190 ^{0}F	10”	Open
CS Header Riser Check	CSMVAAA128A CSMVAAA128B	2CS-V103A 2CS-V104B	2	Check	-	Anchor Darling	2	175 psig 260 ^{0}F	10”	Open/ Close
CS Header Riser Bypass	CSISV0129A CSISV0129B	2CS-E608A 2CS-E609B	2	Globe	Solenoid	Valcor	2	175 psig 265 ^{0}F	1/2”	Close
Chemical and Volume Control System (CVC)										
Letdown Inside Containment Isol	CVCMVAAA103	1CH-F2501A/B	1	Globe	Diaphragm	Masoneilan	1	2,485 psig 650 ^{0}F	2”	Close
Letdown Outside Containment Isol	CVCMVAAA109	2CH-F1518A/B	1	Globe	Diaphragm	Masoneilan	2	2,485 psig 650 ^{0}F	2”	Close
Volume Control Tank Outlet	CVCMVAAA184	2CH-V124A/B	1	Check	-	Anchor Darling	2	150 psig 250 ^{0}F	4”	Close
→(DRN 99-868, R13-A; 06-1183, R15)										
Charging Pump Discharge Check	CVCMVAAA194A CVCMVAAA194A/B CVCMVAAA194B	2CH-V1502-1 2CH-V1502-2 2CH-V1502-3	3	Check	-	Anchor Darling Anchor Darling Anchor Darling	2	3,125 psig 250 ^{0}F	2”	Open
←(DRN 99-868, R13-A; 06-1183, R15) →(DRN 05-1133, R14) ←(DRN 05-1133, R14)										
Pressurizer Aux Spray	CVCISV0216A CVCISV0216B	1CH-E2505A 1CH-E2505B	2	Globe	Solenoid	Target Rock	1	2,485 psig 650 ^{0}F	2”	Open/ Close
Pressurizer Aux Spray Line Check	CVCMVAAA217A CVCMVAAA217B	1CH-V2502-1 1CH-V2502-4	2	Check	-	Velan	1	2,485 psig 650 ^{0}F	2”	Open/ Close

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Charging Line to RCS Stop Isol.	CVCISV0218A CVCISV0218B	1CH-E2503A 1CH-E2504B	2	Globe	Solenoid	Target Rock	1	2,485 psig 650 ^{0}F	2"	Open/ Close
Charging to RC Loop Bypass Check	CVCMVAAA219	1CH-V2506	1	Check	-	Velan	1	2,485 psig 650 ^{0}F	2"	Open
→(DRN 04-515, R13-B; 05-1063, R14)										
Charging to RC Loop Check	CVCMVAAA221A CVCMVAAA221B	1CH-V2502-3 1CH-V2502-2	2	Check	-	Velan	1	2,485 psig 650°F	2"	Open/ Close
←(DRN 04-515, R13-B)										
RCP Control Bleedoff Isol	CVCMVAAA401	2CH-F1512A/B	1	Globe	Diaphragm	WKM	2	2,485 psig 650°F	¾"	Close
←(DRN 05-1063, R14)										
RWSP to CVC Pumps Suction	CVCMVAAA507	3CH-V121A/B	1	Gate	Motor	Powell	3	150 psig 200 ^{0}F	3"	Open
RWSP to CVC Pumps Suction	CVCMVAAA508	2CH-V129A/B	1	Check	-	Anchor Darling	2	150 psig 250 ^{0}F	3"	Open/ Close
Containment Vacuum Relief System (CVR)										
Containment Vac Relief Control	CVRMVAAA101 CVRMVAAA201	2HV-B157B 2HV-B156A	2	Butterfly	Piston	Fisher Controls	2	44 psig 300 ^{0}F	24"	Open
Containment Vac Relief Cont Valve	CVRMVAAA102 CVRMVAAA202	2HV-V181B 2HV-V180A	2	Check	-	GPE Controls	2	150 psig 269 ^{0}F	24"	Open/ Close
Containment/ Annulus DP Switches LP Check	CVRMVAAA302A CVRMVAAA302B	N/A	2	Excess Flow Check	-	Dragon	2	50 psig 400 ^{0}F	1/2"	Open
CVR Cont. Isol	CVRISV0400 CVRISV0401	2HV-E633B 2HV-E634A	2	Globe	Solenoid	Valcor	2	998 psig 350 ^{0}F	1/4"	Close
Emergency Feedwater System (EFW)										
Emergency Feedwater Pumps	EFWMPMP0001A EFWMPMP0001B	N/A	2	Horizontal Centrifugal	Motor	Bingham Willamette	3	395 gpm 2,673 ft TDH	-	Operate
	EFWMPMP0001A/B		1	Horizontal Centrifugal	Turbine	Bingham Willamette	3	780 gpm 2,673 ft TDH	-	Operate
Motor Driven EFW Pump Recir.	EFWMVAAA204A EFWMVAAA204B	3FW-V1507-2 3FW-V1507-1	2	Check	-	Velan	3	1,400 psig 115 ^{0}F	1"	Open
Turbine Driven EFW Pump Recir.	EFWMVAAA204A/B	3FW-V1517	1	Check	-	Velan	3	1,400 psig 115 ^{0}F	11/2"	Open

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
→(DRN 99-0691; EC-19716, R304)										
EFW Pump Discharge Check	EFWMVAAA207A EFWMVAAA207B	3FW-V601A 3FW-V602B	2	Check	-	TRW Mission	3	1,400 psig 115 ^{0}F	4"	Open/ Close
	EFWMVAAA207A/B	3FW-V603A/B	1	Check	-	TRW Mission	3	1,400 psig 115 ^{0}F	6"	Open/ Close
←(DRN 99-0691; EC-19716, R304)										
EFW Pump Header Discharge	EFWMVAAA2191A EFWMVAAA2191B	3FW-V1541A 3FW-V1542B	2	Check	-	Anchor Darling	3	1,400 psig 115 ^{0}F	6"	Open/ Close
→(DRN 05-40, R14; EC-41355, R307)										
EFW Header to SG Backup FCVs	EFWMVAAA223A EFWMVAAA223B	2FW-V852A 2FW-V854B	2	Globe	Diaphragm/ Manual	Masoneilan	2	1,400 psig 480 ^{0}F	4"	Open/ Close Modulate
EFW Header to SG Primary FCVs	EFWMVAAA224A EFWMVAAA224B	2FW-V851B 2FW-V853A	2	Globe	Diaphragm/ Manual	Masoneilan	2	1,400 psig 480 ^{0}F	4"	Open/ Close Modulate
←(DRN 05-40, R14)										
EFW to SG Primary Isol.	EFWMVAAA228A EFWMVAAA228B	2FW-V848A 2FW-V850B	2	Globe	Diaphragm/ Manual	Masoneilan	2	1,400 psig 480 ^{0}F	4"	Open/ Close
EFW to SG Backup Isol.	EFWMVAAA229A EFWMVAAA229B	2FW-V847B 2FW-V849A	2	Globe	Diaphragm/ Manual	Masoneilan	2	1,400 psig 480 ^{0}F	4"	Open/ Close
←(EC-41355, R307)										
Emergency Diesel Generator Air System (EGA)										
EDG Air Dryer Return Check	EGAMVAAA136A EGAMVAAA136B EGAMVAAA137A EGAMVAAA137B	3EA-V604A 3EA-V608B 3EA-V602A 3EA-V606B	4	Check	-	BNL	3	265 psig 125 ^{0}F	1"	Close
EDG Starting Air Check	EGAMVAAA161A EGAMVAAA161B EGAMVAAA162A EGAMVAAA162B	KSV-23-12 (drawing #)	4	Check	-	Crane	3	500 psig 150 ^{0}F	2.5"	Open
Emergency Diesel Generator Cooling System (EGC)										
Engine Driven Jkt Water Circ Pumps	EGCMPMP0002A EGCMPMP0002B	N/A	2	Horizontal Centrifugal	Engine	Allis-Chalmers	3	1,080 gpm 70 ft TDH	-	Operate

TABLE 3.9-9 (Sheet 12 of 24) Revision 309 (06/16)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
EDG Jacket Water Circ Pump Disch.	EGCMVAAA102A EGCMVAAA102B	KSV-58-4 (drawing #)	2	Check	-	Anderson Greenwood	3	150 psig 225 ^{0}F	3”	Open/ Close
EDG Jacket Water Pump Disch.	EGCMVAAA105A EGCMVAAA105B	KSV-58-4 (drawing #)	2	Check	-	Anderson Greenwood	3	150 psig 225 ^{0}F	6”	Open
EDG Jacket Water Temp. Control	EGCMVAAA106A EGCMVAAA106B	KSV-58-4 (drawing #)	2	3-way	Thermo-static	Amot Control	3	150 psig 500 ^{0}F	6”	Operate
Motor Driven Circ Pumps	EGCMPMP0001A EGCMPMP0001B	N/A	2	Horizontal Centrifugal	Engine	Crane Deming	3	175 gpm 40 ft TDH	-	Operate
Emergency Diesel Generator Fuel Oil System (EGF)										
Diesel Oil Transfer Pumps	EGFMPMP0001A EGFMPMP0001B	N/A	2	Horizontal Centrifugal	Motor	Goulds Pumps	3	50 gpm 186 ft TDH 125 ^{0}F	-	Operate
→(LBDCR 15-010, R309)										
Fuel Oil Transfer Pump Suction	EGFMVAAA106-A	3EG-V614A/B	1	Gate	-	Velan	3	150 psig 125 degf	2”	Open
Fuel Oil Transfer Pump Suction	EGFMVAAA106-B	3EG-V617A/B	1	Gate	-	Velan	3	150 psig 125 degf	2”	Open
Fuel Oil Transfer Tank Alt Fill	EGFMVAAA1061	3EG-V653A/B	1	Globe	-	Velan	3	150 psig 125 degf	2”	Open
←(LBDCR 15-010, R309)										
Fuel Oil Transfer Pump Disch.	EGFMVAAA109A EGFMVAAA109B	3EG-V601A 3EG-V602B	2	Check	-	Velan	3	150 psig 125 degf	2”	Open
Emergency Diesel Generator Lube Oil System (EGL)										
Engine Driven Lube Oil Pumps	EGLMPMP0003A EGLMPMP0003B	N/A	2	Positive Displacement	Engine	Roper	3	530 gpm 90 psig	-	Operate
EDG Lube Oil Heater Outlet	EGLMVAAA204A EGLMVAAA204B	KSV-58-4 (drawing #)	2	Check	-	Anderson Greenwood	3	150 psig 225 ^{0}F	3”	Open/ Close
EDG Engine Driven LO Pump Suction Check	EGLMVAAA206A EGLMVAAA206B	KSV-28-3 (drawing #)	2	Check	-	CES	3	100 psig 225 ^{0}F	6”	Open
EDG Standby LO Pump Discharge Check	EGLMVAAA208A EGLMVAAA208B	KSV-58-4 (drawing #)	2	Check	-	Anderson Greenwood	3	150 psig 225 ^{0}F	6”	Open
EDG Lube Oil Temp. Control	EGLMVAAA209A EGLMVAAA209B	KSV-58-4 (drawing #)	2	3-way	Thermo-static	Amot Control	3	150 psig 500 ^{0}F	6”	Operate

TABLE 3.9-9 (Sheet 13 of 24) Revision 12-B (04/03)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Fire Protection System (FP)										
FP Header Containment Isol.	FPMVAAA601A FPMVAAA601B	2FP-F127 2FP-F129	2	Globe	Diaphragm	WKM	2	175 psig 125 °F	3"	Close
RCB FP Header Check	FPMVAAA602A FPMVAAA602B	2FP-V128 2FP-V130	2	Check	-	TRW Mission	2	175 psig 125 °F	3"	Close
Feedwater System (FW)										
SG Startup Feedwater Reg.	FWMVAAA166A FWMVAAA166B	5FW-FM835 5FW-FM836	2	Gate	Diaphragm	Fisher Controls	4 (Note 1)	1,400 psig 480 °F	6"	Close
SG Main Feedwater Reg.	FWMVAAA173A FWMVAAA173B	5FW-FM833 5FW-FM834	2	Angle	Diaphragm	Fisher Controls	4 (Note 1)	1,400 psig 480 °F	16"	Close
FW to EFW Hdr Pressurizing	FWMVAAA1763A FWMVAAA1763B	3FW-V633 3FW-V634	2	Check	-	Velan	3	1,400 psig 200 °F	1"	Open/ Close
Feedwater to SG Check	FWMVAAA181A FWMVAAA181B	2FW-V821A 2FW-V822B	2	Check	-	Anchor Darling	2	1,400 psig 480 °F	20"	Close
SG Main Feedwater Isol.	FWMVAAA184A FWMVAAA184B	2FW-V823A 2FW-V824B	2	Gate	Hydraulic	Anchor Darling	2	1,400 psig 480 °F	20"	Close
→(DRN 03-291, R12-B)										
Fuel Pool Cooling & Purification (FS)										
RWSP Purification Pump Suction Isol. From RWSP	FSMVAAA423	3FS-V135A/B	1	Gate	Manual	Anchor Darling Valve Co.	3	50 psig 125 °F	3"	Close
Fuel Pool Ion Exchanger to RWSP Isolation	FSMVAAA404	3FS-V123	1	Diaphragm	Manual	ITT Grinnell	3	150 psig 200 °F	3"	Close
←(DRN 03-291, R12-B)										
Gaseous Waste Management System (GWM)										
Containment Vent Header Isol.	GWMMVAAA104 GWMMVAAA105	2WM-F157A/B 2WM-F158A/B	2	Diaphragm	Diaphragm	ITT Grinnell	2	150 psig 200 °F	1"	Close

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Hydrogen Recombiner and Analyzer System (HRA)										
H_2 Analyzer Sample Pumps	HRAMPMP0001A HRAMPMP0001B	N/A	2	Diaphragm	Motor	Air Dimensions	2	0.5 CFM 350 °F	-	Operate
Cntmt Dome Area HRA Sample Isol.	HRAISV0101A HRAISV0101B	2HA-E601A 2HA-E621B	2	Globe	Solenoid	Valcor	2	100 psig 350 °F	3/8"	Open
HRA Cntmt Smpl Header Inside Cntmt Isol.	HRAISV0109A HRAISV0109B	2HA-E608A 2HA-E628B	2	Globe	Solenoid	Valcor	2	100 psig 350 °F	3/8"	Open/ Close
HRA Cntmt Smpl Header Outside Cntmt Isol.	HRAISV0110A HRAISV0110B	2HA-E609A 2HA-E629B	2	Globe	Solenoid	Valcor	2	100 psig 350 °F	3/8"	Open/ Close
HRA Cntmt Smpl Return Header Isol.	HRAISV0126A HRAISV0126B	2HA-E610A 2HA-E630B	2	Globe	Solenoid	Valcor	2	100 psig 350 °F	3/8"	Open/ Close
Annulus Dome Area HRA Smpl Isol.	HRAISV0201A HRAISV0201B	2HA-E602A 2HA-E622B	2	Globe	Solenoid	Valcor	2	100 psig 350 °F	3/8"	Open/ Close
HRA Annulus Smpl Return Header Isol	HRAISV0202A HRAISV0202B	2HA-E633A 2HA-E634B	2	Globe	Solenoid	Valcor	2	100 psig 350 °F	3/8"	Open/ Close
→(DRN 05-1063, R14) HRA Cntmt Smpl Return Header Isol ←(DRN 05-1063, R14)	HRAMVAAA128A HRAMVAAA128B	2HA-V637A 2HA-V637B	2	Check		Velan	2	100 psig 350 °F	3/8"	Open/ Close
Control Room HVAC System (HVC)										
CR Normal OAI Isol.	HVCMVAAA101 HVCMVAAA102	3HV-B170B 3HV-B169A	2	Butterfly	Piston	Fisher Controls	3	4 in. wg 150 °F	16"	Close
CR Emerg. Fltr North OAI Isol.	HVCMVAAA201A HVCMVAAA201B	3HV-B196A 3HV-B197B	2	Butterfly	Motor	Fisher Controls	3	-5 in. wg 100 ° F	8"	Open/ Close
CR Emerg. Fltr Unit North OAI Downstream Isol.	HVCMVAAA202A HVCMVAAA202B	3HV-B198A 3HV-B199B	2	Butterfly	Motor	Fisher Controls	3	-5 in. wg 100 ° F	8"	Close
CR Emerg. Fltr South OAI Isol.	HVCMVAAA203A HVCMVAAA203B	3HV-B201A 3HV-B200B	2	Butterfly	Motor	Fisher Controls	3	-5 in. wg 100 °F	8"	Open/ Close

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
CR Emerg. Fltr Unit South OAI Downstream Isol.	HVCMVAAA204A HVCMVAAA204B	3HV-B203A 3HV-B202B	2	Butterfly	Motor	Fisher Controls	3	-5 in. wg 100 °F	8”	Close
CR Toilet Exh. Fan Isol.	HVCMVAAA306 HVCMVAAA307	3HV-B178B 3HV-B177A	2	Butterfly	Piston	Fisher Controls	3	4 in. wg 150 °F	12”	Close
Kitchen/Conf. Rm Exh. Fan Isol.	HVCMVAAA313 HVCMVAAA314	3HV-B172B 3HV-B171A	2	Butterfly	Piston	Fisher Controls	3	4 in. wg 150 °F	12”	Close
Reactor Auxiliary Building HVAC System (HVR)										
RAB Normal Sply to RB-4 Isol.	HVRMVAAA104 HVRMVAAA105	3HV-B224A 3HV-B223B	2	Butterfly	Piston	Fisher Controls	3	4 psig 100 °F	30”	Close
RAB Normal Sply to CVAS Isol.	HVRMVAAA106 HVRMVAAA107	3HV-B226A 3HV-B227B	2	Butterfly	Piston	Fisher Controls	3	-5 in. wg 100 0 °F	36”	Close
CVAS to RAB Normal Exh. Isol.	HVRMVAAA108 HVRMVAAA109	3HV-B218A 3HV-B217B	2	Butterfly	Piston	Fisher Controls	3	-5 in. wg 120 °F	42”	Close
CCW Hx to RAB Normal Exh. Isol.	HVRMVAAA110 HVRMVAAA111	3HV-B215B 3HV-B216A	2	Butterfly	Piston	Jamesbury	3	4 psig 120 °F	12”	Close
CVAS Exh. to CVAS Filter B	HVRMVAAA301	3HV-B225B	1	Butterfly	Piston	Fisher Controls	3	-5 in. wg 100 °F	18”	Open
CVAS Exh. to CVAS Filter A	HVRMVAAA302	3HV-B210A	1	Butterfly	Piston	Jamesbury	3	-10 in. wg 120 °F	14”	Open
CVAS Filter Train Inlet Valve	HVRMVAAA304A HVRMVAAA304B	3HV-B208A 3HV-B209B	2	Butterfly	Motor	Fisher Controls	3	-10 in. wg 120 °F	18”	Open
CVAS Exh. Fan Inlet Valve	HVRMVAAA313A HVRMVAAA313B	3HV-B206A 3HV-B207B	2	Butterfly	Motor	Fisher Controls	3	-15 in. wg 120 °F	20”	Open

TABLE 3.9-9 (Sheet 16 of 24) Revision 309 (06/16)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Instrument Air System (IA)										
→(DRN 05-1133, R14) IA Check to CVC-209 VALVOP ←(DRN 05-1133, R14)	IAICHK0209	N/A	1	Check	-	Circle Seal Controls	3	275 psig 100 °F	¼”	Close
IA to N_2 ACC Headers Check Valves	IAMVAAA520212 IAMVAAA520242 IAMVAAA552182 IAMVAAA552202 IAMVAAA57212 IAMVAAA573132 IAMVAAA902112 IAMVAAA90232	3IA-V612-3 3IA-V612-4 3IA-V612-5 3IA-V612-6 3IA-V612-2 3IA-V612-1 3IA-V612-8 3IA-V612-7	8	Check	-	Velan	3	125 psig 125 °F	1”	Close
IA Hdr Flow Control	IAMVAAA909	2IA-F601A/B	1	Globe	Diaphragm	WKM	2	125 psig 125 °F	2”	Close
IA to RCB-4 Header Check	IAMVAAA910	2IA-V602A/B	1	Check	-	Velan	2	125 psig 125 °F	2”	Close
→(LBDCR 15-010, R309) N2 Accum I BU AS Main Isol 1	IA MVAAA9542	-	1	Globe	-	Dragon	SR	2500 psig 100 degf	½”	Open
N2 Accum I BU AS Vent	IA MVAAA9543	-	1	Globe	-	Dragon	SR	2500 psig 100 degf	½”	Closed
N2 Accum I BU AS Main Isol 2	IA MVAAA9544	-	1	Globe	-	Dragon	SR	2500 psig 100 degf	½”	Open
N2 Accum II BU AS Main Isol 1	IA MVAAA9552	-	1	Globe	-	Dragon	SR	2500 psig 100 degf	½”	Open
N2 Accum II BU AS Vent	IA MVAAA9553	-	1	Globe	-	Dragon	SR	2500 psig 100 degf	½”	Closed
N2 Accum II BU AS Main Isol 2	IA MVAAA9554	-	1	Globe	-	Dragon	SR	2500 psig 100 degf	½”	Open
IA HP Supply Station HP Side Isolation	IAMVAAA965 IAMVAAA975 IAMVAAA985 IAMVAAA995	N/A	4	Globe	Manual	Dragon	3	3000 psig 125 °F	3/8”	Open
←(LBDCR 15-010, R309)										

TABLE 3.9-9 (Sheet 17 of 24) Revision 309 (06/16)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
IA HP Supply Station LP Side Isolation	IAMVAAA968 IAMVAAA978 IAMVAAA988 IAMVAAA998	N/A	4	Globe	Manual	Dragon	3	125 psig 125 °F	3/8”	Open
IA HP Supply to LP Side Pressure Regulating Valves	IAMVAAA966 IAMVAAA976 IAMVAAA986 IAMVAAA996	N/A	4	Press Reg	-	Target Rock	3	3000 psig 125 °F	1”	Operate
IA HP Supply Station Cross-connects	IAMVAAA969 IAMVAAA979 IAMVAAA989 IAMVAAA999	N/A	4	Globe	Manual	Dragon	3	125 psig 125 °F	3/8”	Open
→(EC-18248, R304) SI-405A(B) Accum EIA check valve ←(EC-18248, R304)	IAMVAAA98842B IAMVAAA98852A	N/A	2	Check	-	BNL	3	150 psig 125°F	1/2”	Open/ Close
Main Steam System (MS)										
→(EC-41355, R307) SG MS Atm. Dump ←(EC-41355, R307)	MSMVAAA116A MSMVAAA116B	2MS-PM629A 2MS-PM630B	2	Angle (8x12)	Piston / Manual	Control Components	2	1,085 psig 555 °F	8”	Open/ Close
MSIV Upstream Startup Drain	MSMVAAA119A MSMVAAA119B	2MS-V671 2MS-V664	2	Globe	Motor	Velan	2	1,085 psig 555 °F	2”	Close
MSIV Upstream Normal Drain	MSMVAAA120A MSMVAAA120B	2MS-V670 2MS-V663	2	Globe	Motor	Velan	2	1,085 psig 555 °F	2”	Close
Main Steam Isolation Valve	MSMVAAA124A MSMVAAA124B	2MS-V602A 2MS-V604B	2	Gate	Hydraulic	WKM	2	1,085 psig 555 °F	40”	Close
→(DRN 00-538, R12) EFW Pump AB Turbine Steam Supply from SG ←(DRN 00-538, R12)	MSMVAAA401A MSMVAAA401B	2MS-V611A 2MS-V612B	2	Gate	Motor	GE Sentinel	2	1,085 psig 555 °F	6"x4"x6"	Open
EFW Pump AB Turbine Steam Supply Check	MSMVAAA402A MSMVAAA402B	3MS-V676A 3MS-V677B	2	Check	-	TRW Mission	3	1,085 psig 555 °F	6”	Open/ Close

TABLE 3.9-9 (Sheet 18 of 24) Revision 309 (06/16)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Nitrogen Gas System (NG)										
Cntmt Nitrogen Supply Hdr Flow Control Isol.	NGMVAAA157	2NG-F604	1	Globe	Diaphragm	Velan	2	800 psig 300 °F	1”	Close
Cntmt Nitrogen Supply Check Hdr	NGMVAAA158	2NG-V666	1	Check	-	Velan	2	800 psig 300 °F	1”	Close
Safety Injection Tank Nitrogen Supply PCV	NGMVAAA161A NGMVAAA161B NGMVAAA162A NGMVAAA162B	2SI-F605TK1A 2SI-F606TK1B 2SI-F607TK2A 2SI-F608TK2B	4	Ball	Diaphragm	WKM	2	700 psig 150 °F	1”	Close
→(DRN 01-370, R11) N_2 Accumulators 1&2 Outlet Isol. ←(DRN 01-370, R11)	NGISV0609 NGISV0610	3NG-E671-1 3NG-E671-2	2	Globe	Solenoid	Target Rock	3	800 psig 200 °F	1”	Open / Close
N_2 Accumulators 3&4 Inlet Check	NGMVAAA703 NGMVAAA704	3NG-V668-3 3NG-V668-4	2	Check	-	Velan	3	800 psig 200 °F	1”	Close
N_2 Accumulators 1&2 Inlet Check →(DRN 01-370, R11)	NGMVAAA603 NGMVAAA604	3NG-V668-1 3NG-V668-2	2	Check	-	Velan	3	800 psig 200 °F	1”	Close
N_2 Accumulators 3&4 Outlet Isol. ←(DRN 01-370, R11)	NGISV0709 NGISV0710	3NG-E671-3 3NG-E671-4	2	Globe	Solenoid	Target Rock	3	800 psig 200 °F	1”	Open / Close
N_2 Accumulators 5&6 Inlet Check	NGMVAAA803 NGMVAAA804	3NG-V668-5 3NG-V668-6	2	Check	-	Velan	3	800 psig 200 °F	1”	Close
→(DRN 01-370, R11) N_2 Accumulators 5&6 Outlet Isol. ←(DRN 01-370, R11)	NGISV0809 NGISV0810	3NG-E671-5 3NG-E671-6	2	Globe	Solenoid	Target Rock	3	800 psig 200 °F	1”	Open / Close
N_2 Accumulators 7&8 Inlet Check →(DRN 01-370, R11)	NGMVAAA903 NGMVAAA904	3NG-V668-7 3NG-V668-8	2	Check	-	Velan	3	800 psig 200 °F	1”	Close
N_2 Accumulators 7&8 Outlet Isol. ←(DRN 01-370, R11)	NGISV0909 NGISV0910	3NG-E671-7 3NG-E671-8	2	Globe	Solenoid	Target Rock	3	800 psig 200 °F	1”	Open / Close
N_2 Accumulators Outlet Pressure Regulating Valves	NG MVAAA611 NG MVAAA612 NG MVAAA711 NG MVAAA712 NG MVAAA811 NG MVAAA812 NG MVAAA911 NG MVAAA912	3NG-P670-1 3NG-P670-2 3NG-P670-3 3NG-P670-4 3NG-P670-5 3NG-P670-6 3NG-P670-7 3NG-P670-8	8	Press Reg	-	Target Rock	3	800 psig 200 °F	1”	Operate

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Primary Makeup Water System (PMU)										
Boric Acid Dilution Header	PMUMVAAA146	3CH-V116A/B	1	Check	-	Anchor Darling	3	150 psig 200 °F	3”	Close
→(DRN 04-515, R13-B)										
PMU Inside Cntmt Header Check	PMUMVAAA152	2DW-V610	1	Check	-	Velan	2	150 psig 125 °F	2”	Open/ Close
PMU to Quench Tank Header Check	PMUMVAAA154	7RC-V601	1	Check	-	Velan	4	130 psig 350°F	2”	Open
←(DRN 04-515, R13-B)										
Primary Sampling System (PSL										
RCS Hot Legs Sample Cntmt Isol	PSLMVAAA105 PSLMVAAA107	2SL-F1501A/B 2SL-F1504A/B	2	Globe	Diaphragm	WKM	2	2,485 psig 650 °F	1/2”	Close
PZR Surge Line Sample Cntmt Isol	PSLMVAAA203 PSLMVAAA204	2SL-F1502A/B 2SL-F1505A/B	2	Globe	Diaphragm	WKM	2	2,485 psig 650 °F	1/2”	Close
PZR Steam Space Sample Cntmt Isol	PSLMVAAA303 PSLMVAAA304	2SL-F1503A/B 2SL-F1506A/B	2	Globe	Diaphragm	WKM	2	2,485 psig 650 °F	1/2”	Close
Reactor Coolant System (RC)										
Reactor Vessel Vent to Quench Tank Isol.	RCISV1014 RCISV1015	2RC-E2560B 2RC-E2559A	2	Globe	Solenoid	Target Rock	2	2,485 psig 700 °F	1”	Open/ Close
Reactor/PZR Vent to Quench Tank Isol.	RCISV1017 RCISV3186	2RC-E2562B 2RC-E2561A	2	Globe	Solenoid	Target Rock	2	2,485 psig 700 °F	1”	Open/ Close
PZR Normal Spray Control	RCMVAAA301A RCMVAAA301B	1RC-F1501A 1RC-F1502B	2	Globe	Diaphragm	Fisher Controls	1	2,485 psig 650 °F	3”	Close
Pressurizer Vent to Quench Tank Isol.	RCISV3183 RCISV3184	2RC-E2558B 2RC-E2557A	2	Globe	Solenoid	Target Rock	2	2,485 psig 700 °F	1”	Open/ Close
RCP Control Bleedoff Isol.	RCMVAAA606	2CH-F1513A/B	1	Globe	Diaphragm	WKM	2	2,485 psig 650 °F	¾”	Close
Station Air System (SA)										
IA to CC-710 Cont. Isolation	SAISV9082	N/A	1	Gate	Solenoid	Valcor	2	125 psig 125 °F	1”	Open/ Close
CC-710 IA Backup Supply Inside Cont. Isol.	SAMVAAA9085	N/A	1	Check	-	BNL	2	125 psig 263 °F	1/2”	Open/ Close

TABLE 3.9-9 (Sheet 20 of 24) Revision 309 (06/16)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Shield Building Ventilation System (SBV)										
SBV Filter Train Inlet Isol.	SBVMVAAA101A SBVMVAAA101B	2HV-B160A 2HV-B161B	2	Butterfly	Motor	Fisher Controls	2	-34 in. wg 200 °F	30”	Open
SBV Exh. Fan Suction Isol.	SBVMVAAA110A SBVMVAAA110B	2HV-B158A 2HV-B159B	2	Butterfly	Motor	Fisher Controls	2	-34 in. wg 200 °F	30”	Open
SBV Exh. Fan Recirc Check	SBVMVAAA112A SBVMVAAA112B	2HV-V182A 2HV-V183B	2	Check	-	GPE Controls	2	150 psig 150 °F	30”	Open
SBV Filter Train Inlet Isol.	SBVMVAAA101A SBVMVAAA101B	2HV-B160A 2HV-B161B	2	Butterfly	Motor	Fisher Controls	2	-34 in. wg 200 °F	30”	Open
SBV Exh. Fan Suction Isol.	SBVMVAAA110A SBVMVAAA110B	2HV-B158A 2HV-B159B	2	Butterfly	Motor	Fisher Controls	2	-34 in. wg 200 °F	30”	Open
SBV Exh. Fan Recirc Isol.	SBVMVAAA113A SBVMVAAA113B	2HV-B164A 2HV-B165B	2	Butterfly	Motor	Fisher Controls	2	-34 in. wg 200 °F	30”	Open/ Close
SBV Exh. Fan to Plant Stack.	SBVMVAAA114A SBVMVAAA114B	2HV-B162A 2HV-B163B	2	Butterfly	Motor	Fisher Controls	2	-34 in. wg 200 °F	30”	Open/ Close
Safety Injection System (SI)										
RWSP Outlet Header Isol.	SIMVAAA106A SIMVAAA106B	2SI-L103A 2SI-L104B	2	Butterfly	Piston	Fisher Controls	2	160 psig 250 °F	24”	Open
LPSI Pump Suction Check	SIMVAAA1071A SIMVAAA1071B	2SI-V354A 2SI-V355B	2	Check	-	Anchor Darling	2	440 psig 400 °F	20”	Open/ Close
RWSP Outlet Header Check	SIMVAAA107A SIMVAAA107B	2SI-V107A 2SI-V108B	2	Check	-	TRW Mission	2	160 psig 250 °F	24”	Open/ Close
LPSI Pump Suction Check	SIMVAAA108A SIMVAAA108B	2SI-V331A 2SI-V332B	2	Check	-	TRW Mission	2	440 psig 400 °F	20”	Open/ Close
LPSI Pump Min. Flow Recirc	SIISV1161A SIISV1161B	2SI-E1587A 2SI-E1588B	2	Globe	Solenoid	Target Rock	2	1,950 psig 400 °F	2”	Close
LPSI Pump Min. Flow Recirc	SIMVAAA116A SIMVAAA116B	2SI-V1589A 2SI-V1590B	2	Stop Check	Manual	Anchor Darling	2	2735 psig 680 °F	2”	Open
SI Recirc Hdr to RWSP Upstrm Isol	SIMVAAA120A SIMVAAA120B	2SI-V810A 2SI-V802B	2	Gate	Motor	Anchor Darling	2	1,950 psig 400 °F	4”	Close
SI Recirc Hdr to RWSP Dnstrm Isol	SIMVAAA121A SIMVAAA121B	2SI-V809A 2SI-V801B	2	Gate	Motor	Anchor Darling	2	1,950 psig 400 °F	4”	Close
LPSI Pump Discharge Check	SIMVAAA122A SIMVAAA122B	2SI-V333A 2SI-V334B	2	Check	-	Anchor Darling	2	650 psig 400 °F	8”	Open

TABLE 3.9-9 (Sheet 21 of 24) Revision 309 (06/16)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Shutdown Cooling HX Inlet	SIMVAAA125A SIMVAAA125B	2SI-V306A 2SI-V305B	2	Gate	Motor	Anchor Darling	2	650 psig 400 °F	10"	Open
RC Loop Shdn Cooling Warmup →(DRN 02-387, R11-B)	SIMVAAA135A SIMVAAA135B	2SI-V353A 2SI-V346B	2	Gate	Motor	Anchor Darling	2	650 psig 350 °F	8"	Close
LPSI Hdr to RC Loop Inside Cntmt Isolation ←(DRN 02-387, R11-B) →(DRN 02-1398, R12-A)	SIMVAAA142A SIMVAAA142B SIMVAAA143A SIMVAAA143B	1SI-V1520RL2B 1SI-V1518RL1B 1SI-V1519RL2A 1SI-V1517RL1A	4	Check	-	Anchor Darling or Flowserve	1	2,485 psig 650 °F	8"	Open/ Close
LPSI Hdr to RC Loop Auto-Vent Isolation ←(DRN 02-1398, R12-A)	SI MVAAA 14023A SI MVAAA 14024A	N/A N/A	2	Globe	Solenoid	Target Rock	2	2,485 psig 650 °F	3/4"	Close
HPSI Pump Suction Check	SIMVAAA201A SIMVAAA201B	2SI-V315A 2SI-V316B	2	Check	-	TRW Mission	2	160 psig 250 °F	10"	Open
HPSI Pump Min. Flow Recirc Stop Check	SIMVAAA205A SIMVAAA205A/B SIMVAAA205B	2SI-V1562-2 2SI-V1562-1 2SI-V1562-3	3	Stop Check	Manual	Velan	2	1,950 psig 400 °F	2"	Open
HPSI Pump Discharge Check	SIMVAAA207A SIMVAAA207A/B SIMVAAA207B	2SI-V821A 2SI-V805A/B 2SI-V822B	3	Check	-	Anchor Darling	2	1,950 psig 250 °F	4"	Open/ Close
HPSI Discharge Header Check	SIMVAAA216	2SI-V1521A	1	Check	-	Anchor Darling	2	2,485 psig 250 °F	4"	Open
HPSI Disch. Hdr A Orifice Bypass	SIMVAAA219A	2SI-V1534	1	Gate	Motor	Anchor Darling	2	2,485 psig 250 °F	4"	Close
HPSI Disch. Hdr B Orifice Bypass	SIMVAAA219B	2SI-V811B	1	Gate	Motor	Anchor Darling	2	1,950 psig 250 °F	4"	Close
HPSI Header to RC Loop Inside Cntmt Check	SIMVAAA241 SIMVAAA242 SIMVAAA243 SIMVAAA244	1SI-V1522RL1A 1SI-V1523RL1B 1SI-V1524RL2A 1SI-V1525RL2B	4	Check	-	Anchor Darling	1	2,485 psig 650 °F	3"	Open/ Close
HPSI Pump AB Mini Flow to Recirc Line B Stop Check	SIMVAAA245	2SI-V1562-4	1	Stop Check	Manual	Velan	2	1,950 psig 400 °F	2"	Open
RC Loop Hot Leg Inj. Leakage Drain	SIMVAAA301 SIMVAAA302	1SI-V2504 1SI-V2505	2	Gate	Diaphragm	Anchor Darling	1	2,485 psig 650 °F	1"	Close

TABLE 3.9-9 (Sheet 22 of 24) Revision 305 (11/11)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Safety Injection Tank Fill/Drain	SIMVAAA307A SIMVAAA307B SIMVAAA308A SIMVAAA308B	2SI-F1564TK1A 2SI-F1565TK1B 2SI-F1566TK2A 2SI-F1567TK2B	4	Globe	Diaphragm	Fisher Controls	2	700 psig 150 °F	2”	Close
Safety Injection Tank Vent	SIISV0323A SIISV0323B SIISV0324A SIISV0324B SIISV0325A SIISV0325B SIISV0326A SIISV0326B	2SI-E636 2SI-E633 2SI-E638 2SI-E639 2SI-E632 2SI-E637 2SI-E634 2SI-E635	8	Globe	Solenoid	Valcor	2	700 psig 150 °F	1”	Open
Safety Injection Tank Outlet Check	SIMVAAA329A SIMVAAA329B SIMVAAA330A SIMVAAA330B	1SI-V1510TK1A 1SI-V1512TK1B 1SI-V1514TK2A 1SI-V1516TK2B	4	Check	-	Anchor Darling	1	2,485 psig 650 °F	12”	Open/ Close
→(EC-13981, R304)										
Safety Injection Header Check	SIMVAAA335A SIMVAAA335B SIMVAAA336A SIMVAAA336B	1SI-V1509RL1A 1SI-V1511RL1B 1SI-V1513RL2A 1SI-V1515RL2B	4	Check	-	Anchor Darling / Flowserve	1	2,485 psig 650 °F	12”	Open/ Close
←(EC-13981, R304)										
SIT Drain Header to RWSP Isol.	SIMVAAA343	2SI-F1561A/B	1	Gate	Diaphragm	Anchor Darling	2	1,950 psig 150 °F	2”	Close
RC Loop Shdn Cooling Upstream Suction Isol.	SIMVAAA401A SIMVAAA401B	1SI-V1504A 1SI-V1502B	2	Gate	Motor	Lunkenheimer	1	2,485 psig 650 °F	14”	Open
→(EC-935, R302; EC-14765, R305)										
RC Loop SDC Suction Inside Containment Isol.	SIMVAAA405A SIMVAAA405B	1SI-V1503A 1SI-V1501B	2	Gate	Pneumatic	Lunkenheimer	1	2,485 psig 650 °F	14”	Open
←(EC-935, R302)										
SI-405 Bypass Fill Valves	SIMVAAA4052A SIMVAAA4052B	NA NA	2	Globe	Solenoid	Valcor	1	2,485 psig 650 °F	3/4”	Open/ Close
←(EC-14765, R305)										
RC Loop SDC Suction Outside Containment Isol.	SIMVAAA407A SIMVAAA407B	2SI-V327A 2SI-V326B	2	Gate	Motor	Anchor Darling	2	440 psig 400 °F	14”	Open
Shutdown Cooling HX Outlet Isol.	SIMVAAA412A SIMVAAA412B	2SI-V307A 2SI-V308B	2	Gate	Motor	Anchor Darling	2	650 psig 400 °F	10”	Open

TABLE 3.9-9 (Sheet 23 of 24) Revision 14 (12/05)

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Shutdown Cooling HX Temp. Control	SIMVAAA415A SIMVAAA415B	2SI-FM318A 2SI-FM349B	2	Butterfly	Motor	Fisher Controls	2	650 psig 350 °F	10”	Open/ Close
RC Loop Hot Leg Injection Isol.	SIMVAAA502A SIMVAAA502B	2SI-V1557 2SI-V1558	2	Gate	Motor	Anchor Darling	2	2,485 psig 250 °F	3”	Open
RC Loop Hot Leg Injection Flow Control	SIMVAAA506A SIMVAAA506B	2SI-V1556 2SI-V1559	2	Globe	Motor	Anchor Darling	2	2,485 psig 250 °F	3”	Open
RC Loop Hot Leg Injection Inside Cntmt Check	SIMVAAA510A SIMVAAA510B	1SI-V2506 1SI-V2508	2	Check	-	Anchor Darling	1	2,485 psig 650 °F	3”	Open/ Close
RC Loop Hot Leg Injection Check	SIMVAAA512A SIMVAAA512B	1SI-V2507 1SI-V2509	2	Check	-	Anchor Darling	1	2,485 psig 650 °F	3”	Open/ Close
→(DRN 02-1039, R12; 05-1133, R14)										
LPSI Header Auto Vent Isol.	SIISV6011 SIISV6012	2SI-E655 2SI-E654	2	Globe	Solenoid	Target Rock	2	650 psig 400 °F	1.5 ”	Close
←(DRN 02-1039, R12; 05-1133, R14)										
Safety Injection Sump Outlet Header Isol.	SIMVAAA602A SIMVAAA602B	2SI-L101A 2SI-L102B	2	Butterfly	Motor	Fisher Controls	2	160 psig 250 °F	24”	Open/ Close
Safety Injection Sump Outlet Header Check	SIMVAAA604A SIMVAAA604B	2SI-V105A 2SI-V106B	2	Check	-	TRW Mission	2	160 psig 250 °F	24”	Open
RWSP Vacuum Breaker	SIMVAAA717A SIMVAAA717B	3SI-V118A 3SI-V117B	2	Check	-	TRW Mission	3	10 psig 120 °F	16”	Open
Sump Pump System (SP)										
Containment Sump Header Cntmt Isol.	SPMVAAA105 SPMVAAA106	2WM-F104A/B 2WM-F105A/B	2	Diaphragm	Diaphragm	ITT Grinnell	2	100 psig 150 °F	1.5”	Close

NON-NSSS SUPPLIED ACTIVE VALVES AND PUMPS

Equipment	UNID	Ebasco No.	Qnty	Type	Operator	Manufacturer	Safety Class	Design Rating	Size	Function
Secondary Sampling System (SSL)										
Main Steam Line Sample Isol.	SSLMVAAA301A SSLMVAAA301B	2MS-F714 2MS-F715	2	Globe	Diaphragm	Velan	2	1,085 psig 555 °F	1”	Close
SG Blowdown Sample Inside Cntmt Isol.	SSLMVAAA8004A SSLMVAAA8004B	2SL-F601 2SL-F603	2	Globe	Diaphragm	Velan	2	1,085 psig 555 °F	1/2”	Close
SG Blowdown Sample Outside Cntmt Isol.	SSLMVAAA8006A SSLMVAAA8006B	2SL-F602 2SL-F604	2	Globe	Diaphragm	Velan	2	1,085 psig 555 °F	1/2”	Close

(a) Component contains replacement parts which were supplied per NRC Generic Letter 89-09. This letter allows the use of replacement parts which are identical to the original part except the manufacturer cannot supply an ASME Code Data Sheet or provide the ASME Section III “N” stamp. NRC Generic Letter requires that the part be built to the original ASME Section III Code and also requires an Authorized Nuclear Inspector’s approval.

NOTES:

1. These valves are NNS but they are actuated by MSIS. Therefore, they are included in the Table.

→(DRN 05-1133, R14)

2. Deleted

←(DRN 05-1133, R14)

→(DRN 04-515, R13-B)

3. NNS valve (PMU 154) is included in this Table due to credit as providing a containment penetration pressure flow path in the resolution of Generic Letter 96-06.

←(DRN 04-515, R13-B)

TABLE 3.9-10 (Sheet 1 of 5) Revision 15 (03/07)

CODE CLASS 1 AND 2 SAFETY RELIEF VALVES

Equipment	UNID	Ebasco No.	Size (in)	Safety Class	Design Conditions	Pipng Analysis Stress Calculation No.
→(DRN 01-476, R12)						
Boron Management System						
Reactor Drain Tank Pump Suction Thermal Relief	BM MVAAA1091	N/A	1/2 x 1	2	50 psig 250 ^{0}F	2717
←(DRN 01-476, R12)						
Component Cooling Water System (CC)						
Containment Fan Cooler C CCW Inlet Relief	CCMVAAA811A	2CC-R21	1 x 1	2	125 psig 125 ^{0}F	2685
Containment Fan Cooler B CCW Inlet Relief	CCMVAAA811B	2CC-R24	1 x 1	2	125 psig 125 ^{0}F	2693
Containment Fan Cooler A CCW Inlet Relief	CCMVAAA812A	2CC-R22	1 x 1	2	125 psig 125 ^{0}F	2683
Containment Fan Cooler D CCW Inlet Relief	CCMVAAA812B	2CC-R23	1 x 1	2	125 psig 125 ^{0}F	2692
→(DRN 01-405; 01-476, R12)						
CCW Return Line Thermal Relief	CCMVAAA7102	N/A	1/2 x 1	2	125 psig 175°F	2520-1
←(DRN 01-405; 01-476, R12)						
→(DRN 06-499, R14-B; 05-1572, R15)						
CCW Supply Line Thermal Relief	CCMVAAA6443	N/A	1/2 x 1	2	125 psig 175°F	2521-1
←(DRN 06-499, R14-B; 05-1572, R15)						
Containment Spray System (CS)						
Containment Spray Pump Suction Header Relief	CSMVAAA1031B CSMVAAA1041A	2CS-R614B 2CS-R615A	3/4 x 1	2	200 psig 250 ^{0}F	2727 2726
Chemical and Volume Control System (CVC)						
Letdown Heat Exchanger Inlet Relief to Holdup Tanks	CVCMVAAA115	2CH-R626A/B	2 x 3	2	650 psig 550 ^{0}F	1137-2
Letdown Back PCVs Outlet Relief to Holdup Tanks	CVCMVAAA126	2CH-R629A/B	2 x 3	2	200 psig 250 ^{0}F	1137-2
Volume Control Tank Outlet Relief to Holdup Tanks	CVCMVAAA182	2CH-R182A/B	3 x 4	2	75 psig 250 ^{0}F	2869-4

CODE CLASS 1 AND 2 SAFETY RELIEF VALVES

Equipment	UNID	Ebasco No.	Size (in)	Safety Class	Design Conditions	Pipng Analysis Stress Calculation No.
Volume Control Tank Outlet Header Relief to Holdup Tanks	CVCMVAAA185	2CH-R184A/B	3/4 x 1	2	150 psig 250 ^{0}F	2869-6
Charging Pump A Discharge Relief	CVCMVAAA192A	2CH-R1526A	1-1/2 x 2	2	3,125 psig 250 ^{0}F	1224-2
Charging Pump AB Discharge Relief	CVCMVAAA192AB	2CH-R1527A/B	1-1/2 x 2	2	3,125 psig 250 ^{0}F	1224-1
Chemical and Volume Control System (CVC) (Continued)						
Charging Pump B Discharge Relief	CVCMVAAA192B	2CH-R1528B	1-1/2 x 2	2	3,125 psig 250 ^{0}F	1224-1
→ (DRN 01-405; 01-476)						
Letdown Line Thermal Relief	CVCMVAAA1081	N/A	1 x 1	2	2,485 psig 650°F	1205
Reactor Coolant Pump Controlled Bleed Off Thermal Relief	RCMVAAA6061	N/A	1 x 1	2	2,485 psig 650°F	2660-1
← (DRN 01-405; 01-476)						
Main Steam System (MS)						
Main Steam Line 1 Safety #1	MSMVAAA106A	2MS-R613A	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 2 Safety #1	MSMVAAA106B	2MS-R619B	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 1 Safety #2	MSMVAAA108A	2MS-R614A	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 2 Safety #2	MSMVAAA108B	2MS-R620B	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 1 Safety #3	MSMVAAA110A	2MS-R615A	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 2 Safety #3	MSMVAAA110B	2MS-R621B	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030

TABLE 3.9-10 (Sheet 3 of 5) Revision 12 (10/02)

CODE CLASS 1 AND 2 SAFETY RELIEF VALVES

Equipment	UNID	Ebasco No.	Size (in)	Safety Class	Design Conditions	Pipng Analysis Stress Calculation No.
Main Steam Line 1 Safety #4	MSMVAAA112A	2MS-R616A	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 2 Safety #4	MSMVAAA112B	2MS-R622B	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 1 Safety #5	MSMVAAA113A	2MS-R617A	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 2 Safety #5	MSMVAAA113B	2MS-R623B	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam System (MS) (Continued)						
Main Steam Line 1 Safety #6	MSMVAAA114A	2MS-R618A	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Main Steam Line 2 Safety #6	MSMVAAA114B	2MS-R624B	8 x 10 x 10	2	1,085 psig 555 ^{0}F	1030
Reactor Coolant System (RCS)						
Pressurizer Relief A	RCMVAAA317A	1RC-R2573A	6 x 8	1	2,485 psig 700 ^{0}F	1163
Pressurizer Relief B	RCMVAAA317B	1RC-R2574B	6 x 8	1	2,485 psig 700 ^{0}F	1163
RCP Control Bleedoff Relief	RCMVAAA603	2CH-R1515A/B	1 x 1-1/2	2	2,485 psig 650 ^{0}F	2660-1
Safety Injection System (SI)						
LPSI Pump A Discharge Header Relief	SIMVAAA132A	2SI-R613A/B	1 x 1-1/2	2	650 psig 400 ^{0}F	1029-2
LPSI Pump B Discharge Header Relief	SIMVAAA132B	2SI-R350B	1 x 1-1/2	2	650 psig 400 ^{0}F	1029-1

TABLE 3.9-10 (Sheet 4 of 5) Revision 12 (10/02)

CODE CLASS 1 AND 2 SAFETY RELIEF VALVES

Equipment	UNID	Ebasco No.	Size (in)	Safety Class	Design Conditions	Pipng Analysis Stress Calculation No.
HPSI Discharge Header A Relief	SIMVAAA214	2SI-R823A	1 x 1-1/2	2	2,485 psig 250 ^{0}F	1056-7
HPSI A to RC Cold Legs Header Relief	SIMVAAA220A	2SI-R1530A	1-1/2 x 2	2	2,485 psig 300 ^{0}F	1056-4
HPSI B to RC Cold Legs Header Relief	SIMVAAA220B	2SI-R824B	1 x 1-1/2	2	2,485 psig 300 ^{0}F	1056-2
Safety Injection Tank 1A Relief to Cntmt Sump	SIMVAAA327A	2SI-R617TK1A	3/4 x 1	2	700 psig 150 ^{0}F	2609
Safety Injection Tank 1B Relief to Cntmt Sump	SIMVAAA327B	2SI-R619TK1B	3/4 x 1	2	700 psig 150 ^{0}F	2611
Safety Injection Tank 2A Relief to Cntmt Sump	SIMVAAA328A	2SI-R618TK2A	3/4 x 1	2	700 psig 150 ^{0}F	2612
Safety Injection System (SI) (Continued)						
Safety Injection Tank 2B Relief to Cntmt Sump	SIMVAAA328B	2SI-R620TK2B	3/4 x 1	2	700 psig 150 ^{0}F	2610
HPSI Pumps A, B, A/B Suction Relief	SIMVAAA2034A SIMVAAA2034B SIMVAAA2034A/B	2SI-R124A 2SI-R125B 2SI-R126A/B	3/4 x 1	2	200 psig 250 ^{0}F	2899 2903 2904
RC Loop 2 SDC Suction Relief to RDT	SIMVAAA404A	1SI-R2501A	1 x 1	1	2,485 psig 650 ^{0}F	1024
RC Loop 1 SDC Suction Relief to RDT	SIMVAAA404B	1SI-R2502B	1 x 1	1	2,485 psig 650 ^{0}F	1020
RC Loop 2 SDC Suction LTOP Relief to Cntmt Smp	SIMVAAA406A	2SI-R339A	6 x 8	2	440 psig 400 ^{0}F	1024
RC Loop 1 SDC Suction LTOP Relief to Cntmt Smp	SIMVAAA406B	2SI-R340B	6 x 8	2	440 psig 400 ^{0}F	1020

CODE CLASS 1 AND 2 SAFETY RELIEF VALVES

Equipment	UNID	Ebasco No.	Size (in)	Safety Class	Design Conditions	Pipng Analysis Stress Calculation No.
RC Loop 2 SDC Suction Relief to Holdup Tanks	SIMVAAA408A	2SI-R614A	3/4 x 1	2	440 psig 400 ^{0}F	1125-4
RC Loop 1 SDC Suction Relief to Holdup Tanks	SIMVAAA408B	2SI-R612B	3/4 x 1	2	440 psig 400 ^{0}F	1125-6
RC Loop 1 Hot Leg Injection Relief to Waste Tanks	SIMVAAA503A	2SI-R1529	1-1/2 x 2	2	2,485 psig 300 ^{0}F	2593
RC Loop 2 Hot Leg Injection Relief to Waste Tanks	SIMVAAA503B	2SI-R1531A	1-1/2 x 2	2	2,485 psig 300 ^{0}F	1056-4
→(DRN 01-476)						
Safety Injection Tanks Drain Line to RWSP Thermal Relief	SIMVAAA3434	N/A	1 x 1	2	1,950 psig 150°F	1134
Sump Pump System						
Containment Sump Pump Discharge Line Thermal Relief	SP MVAAA1051	N/A	1/2 x 1	2	100 psig 150°F	2710
←(DRN 01-476)						
→ (DRN 99-1081)						
Nitrogen Gas System	NGMVAAA1523		3/4 x 1	2	800 psig 200°F	2822
← (DRN 99-1081)						

TABLE 3.9-11

PIPE STRESSES DUE TO SAFETY RELIEF VALVE LOADS

Safety Class 1 Pressure Relief Valves

Component Operating Conditions		Equation	Stress Level in Connecting Pipe
1)	Normal & Upset	NB-3652, Eq. 9	1.5 Sm*
2)	Emergency	NB-3652, Eq. 9	2.25 Sm**
3)	Faulted	NB-3652, Eq. 9	3.0 Sm***

Safety Class 2 Pressure Relief Valves

Component Operating Conditions		Equation	Stress Level in Connecting Pipe
1)	Upset	NC-3652, Eq. 9	Longitudinal stress due to press. & wt + other mechanical loads + OBE: 1.2 S_h
2)	Emergency	NC-3652, Eq. 9	Max. pressure (1.5 design pressure) and longitudinal stress due to press. & wt + other mechanical loads + SSE: 2.4 S_h
3)	Faulted	NC-3652, Eq. 9	Max. Pressure (2.0 design pressure and longitudinal stress due to press. & wt + other mechanical loads + SSE: 2.4 S_h

Notes:

* Stress due to design pressure & weight + other mechanical loads + OBE.

** Stress due to emergency pressure & weight + other mechanical loads + OBE.

*** Stress due to design pressure & weight + other mechanical loads + SSE.

TABLE 3.9-12 Revision 307 (07/13)

STRESS LIMITS FOR CEDM PRESSURE HOUSINGS

Loading Conditions and Categories	Stress Categories and Limits of Stress Intensities [(a)]
Normal and Upset: Normal operating loading plus operating basis earthquake forces	Figures NB-3221-1 and 3222-1 including notes
→(DRN03-2056, R14) Faulted: Normal operating loadings plus SSE forces plus BLPB loadings ←(DRN03-2056, R14)	Table F-1322.2-1, Appendix F, "Rules for Evaluating Faulted Conditions"
Emergency: Normal Operating Loadings plus SSE forces	Figure NB-3224-1 including notes
Testing: Testing plant transients	Paragraph NB-3226

For the above listed conditions and categories, the following limits regarding function apply:

Normal and upset: The CEDMs are designed to function normally during and after exposure to these conditions.

→(DRN 01-1102, R12)

Faulted: The deflections of the CEDM are limited so that the CEAs can be inserted after exposure to these conditions. Those deflections which could influence the ability to move CEAs are limited to less than 80 percent of the deflections required to prevent CEA movement.

←(DRN 01-1102, R12)

Emergency: The CEDMs shall be designed to permit scramming of the CEAs during and after exposure to these conditions.

→(EC-2800, R307)

(a) References listed are taken from Section III of the ASME Boiler and Pressure Vessel Code, 1998 Edition up to and including 2000 addenda.

←(EC-2800, R307)

TABLE 3.9-13 Revision 14 (12/05)

STRESS LIMITS FOR REACTOR INTERNALS

Operating Category	Limits of Stress Intensities
→(DRN 03-2056, R14) Normal and Upset ←(DRN 03-2056, R14)	Figure NG 3221-1 including notes.
Faulted	Appendix F, Rules for Evaluating Faulted Conditions, Section F-1380 and Table F-1322 including notes.

TABLE 3.9-14 Revision 11 (05/01)

NSSS TANKS AND HEAT EXCHANGERS

Equipment	Explanation Why Pipe Break Loads are not Specified for Equipment Design (See Notes)
Letdown Heat Exchanger	1
Shutdown Heat Exchanger	2
Fuel Pool Heat Exchanger	2
Regenerative Heat Exchanger	1
→ (DRN 99-1003)	
← (DRN 99-1003)	
Purification Ion Exchanger	2
Safety Injection Tanks	1
Volume Control Tank	2
Gas Surge Tank	2
Gas Decay Tanks	3
Boric Acid Make-up Tanks	2

Notes:

1) Pipe breaks in lines attached to this equipment renders the system inoperative. Operation of the system under these conditions is not required.

2) Equipment located in moderate energy system.

3) Piping attached to this component is 1 inch or smaller for which pipe breaks are not postulated.

TABLE 3.9-15 Revision 11-B (06/02)

NSSS PUMPS

Pump	Explanation why pipe break loads are not specified for pump design (see Notes)
Boric Acid	2
Reactor Drain	2
High Pressure Safety Injection	2
Low Pressure Safety Injection	2
→ (DRN 00-804)	
Flash Tank	2 3
← (DRN 00-804)	
Fuel Pool	2
Charging	1
Fuel Pool Purification	2

Notes:

(1) Pipe breaks in lines attached to this equipment renders the system inoperative. Operation of the system under these conditions is not required.

(2) Equipment located in moderate energy system.

→ (DRN 00-804)

(3) The BMS Flash Tank and Flash Tank pumps have been made inactive per ER-W3-00-0225-00-00.

← (DRN 00-804)

TABLE 3.9-16 (1 of 4)

PIPE BREAK LOAD CONSIDERATIONS
FOR ACTIVE VALVES LOCATED IN
HIGH ENERGY SYSTEMS

Valve	Tag No.	Pipe Break Load Specified For Valve Operability	Explanation (See Note)
Charging Pump Check Valves	2CH-V1502 (3)	No	4
Aux. Spray Valves	1CH-E2505A	No	1
	1CH-E2505B	No	1
Charging Valves	1CH-E2503A	No	1
	1CH-E2504B	No	1
Letdown-Primary Loop Isolation	1CH-F2501A/B	No	1
	1CH-F1516A/B	No	1
Letdown-Containment Isolation	2CH-F1518A/B	No	2
Main Steam Isolation	2MS-V602A/B 2MS-V604A/B	Yes	8
MS to EM FW Pump Turbine	2MS-V611A	No	3
	2MS-V612B	No	3
MS to EM FW Pump Turbine-Check	3MS-V676A	No	3
	3MS-V677B	No	3
Feedwater Isolation	2FW-V823A	No	2
	2FW-V822B	No	2
Feedwater Isolation-Check	2FW-V821A	No	1
	2FW-V822B	No	1
Blowdown Isolation Inside Containment	2BD-F603	No	1
	2BD-F606	No	1
Blowdown Isolation Outside Containment	2BD-F604	No	2
	2BD-F606	No	2

TABLE 3.9-16 (2 of 4)

PIPE BREAK LOAD CONSIDERATIONS
FOR ACTIVE VALVES LOCATED IN
HIGH ENERGY SYSTEMS

Valve	Tag No.	Pipe Break Load Specified For Valve Operability	Explanation (See Note)
EM FW Isolation	2FW-V847B	No	3
	2FW-V848A	No	3
	2FW-V849A	No	3
	2FW-V850B	No	3
	2FW-V851B	No	3
	2FW-V852A	No	3
	2FW-V853A	No	3
	2FW-V854B	No	3
Drain Valves	2MS-V663	No	3
	2MS-V664	No	3
	2MS-V670	No	3
	2MS-V671	No	3
Atmospheric Dump Valves	2MS-PM629A	No	3
	2MS-PM630B	No	3
Feedwater Control Valves	5FW-FM835	No	6
	5FW-FM836	No	6
	5FW-FM833	No	6
	5FW-FM834	No	6
SI-Primary Loop Isolation Valves	1SI-V1501B	No	7
	1SI-V1502B	No	7
	1SI-V1503A	No	7
	1SI-V1504A	No	7
	1SI-V1509RL1A	No	7
	1SI-V1510TK1A	No	7
	1SI-V1511RL1B	No	7
	1SI-V1512TK1B	No	7
	1SI-V1513RL2A	No	7
	1SI-V1514TK2A	No	7
	1SI-V1515RL2B	No	7
	1SI-V1516TK2B	No	7
	1SI-V1517RL1A	No	7

PIPE BREAK LOAD CONSIDERATIONS
FOR ACTIVE VALVES LOCATED IN
HIGH ENERGY SYSTEMS

Valve	Tag No.	Pipe Break Load Specified For Valve Operability	Explanation (See Note)
	1SI-V1518RL1B	No	7
	1SI-V1519RL2A	No	7
	1SI-V1520RL2B	No	7
	1SI-V1522RL1A	No	7
	1SI-V1523RL1B	No	7
	1SI-V1524RL2A	No	7
	1SI-V1525RL2B	No	7
	1SI-V2506	No	7
	1SI-V2507	No	7
	1SI-V2508	No	7
	1SI-V2509	No	7
Nitrogen Accumulator Isolating Valves	3NG-E671 (8)	No	5
	3NG-P670 (8)	No	5

Notes to Table 3.9-16:

1. Piping Break outside the containment does not transmit additional pipe loads to the components located inside at the containment, because the process pipe is anchored of the containment. Failure of the valve to operate following a pipe break inside the containment does not violate any criteria affecting safe shutdown or release of uncontrolled radiation, since there is a redundant valve located outside containment which is not affected by the pipe break.

2. Piping break inside the containment does not transmit additional pipe loads to the components located inside of the containment, because the process pipe is anchored at the containment. Failure of the valve to operate following a pipe break inside the containment does not violate any criteria affecting safe shutdown or release of uncontrolled radiation, since there is a redundant valve located inside containment which is not affected by the pipe break.

3. Located in break exclusion area. Pipe breaks are postulated at sufficient distance from the break exclusion area, and the Main Steam and Feedwater piping is restrained by a system of pipe rupture restraints which limit displacement of the piping. Therefore, the pipe break loads transmitted to these valves are insignificant.

TABLE 3.9-16 (4 of 4)

PIPE BREAK LOAD CONSIDERATIONS
FOR ACTIVE VALVES LOCATED IN
HIGH ENERGY SYSTEMS

Notes to Table 3.9-16: (Cont'd)

4. Pipe break in these lines renders the charging system inoperative and operation of the system under this condition is not required.

5. Valves located in 1" or smaller pipe lines for which pipe breaks are not postulated.

6. The only safety related function for these valves is to provide redundant closure in case the Feedwater Isolation valve fails to close.

7. Pipe rupture of those lines where this valve is located is considered LOCA, and under these conditions operation of this valve is not required.

8. See Table 3.9-17 for MSLB Flow conditions as applicable to the MSIV's. The MSIV's were evaluated by the manufacturer to operate under MSLB conditions. These conditions are given in Table 3.9-17. The manufacturer's evaluation is in W-K-M Report No. 15-0068-06," 40"x30"x40" Class Model D-2 Gas Spring Operated Main Steam Line Break Gate and Seat Analysis," Revision P204, dated October 17, 1980.

TABLE 3.9-17

MSLB FLOW CONDITIONS AS APPLICABLE TO THE MSIV's

Time (sec)	Pressure (psia)	Flow Rate (lbm/sec)	Quality
0.0	1000.0	2198	1.0
0.25	943.	5847	1.0
0.50	913	5651	1.0
0.75	884	5418	1.0
1.0	859	5253	1.0
1.5	726	4580	1.0
2.0	770	21015	0.0349
2.5	789	21000	0.0322
3.0	782	20524	0.0409
3.5	776	19999	0.0510
4.0	763	18653	0.0617
5.0	741	18217	0.0769
6.0	714	16888	0.0991
7.0	687	15804	0.1168
8.0	659	14913	0.1307
9.0	631	13992	0.1434
10.0	607	13140	0.1551
11.0	583	11578	0.1737
12.0	559	10700	0.1897
13.0	536	9767	0.2094
14.0	519	8789	0.2356
15.0	491	7922	0.2673

TABLE 3.9-18 (Sheet 1 of 2) Revision 2 (12/88)

STRESS LIMITS AND LOADING COMBINATIONS FOR PIPE SUPPORTS

STRESS LIMITS

Allowable stresses for Waterford SES Unit 3 a 650 degrees F maximum for pipe supports. All values in psi.

Reference: MSS SP-58 AISC Manual, and B31.1

Note: Pipe supports are as built using the stress limits below or the higher allowables given in the AISC Manual.

Shape & Use	Fb Bending	Ft Tension	Fv Shear	Fp Bearing	Tension at Pin Hole
Supplementary Steel (Structural Shapes)	Ref. Std. 602 18,000 (Beams & Channels) 20,000 (Ang)	N/A	N/A	N/A	N/A
Hanger Components					
(Incl. cantilever & trapeze members) Structural (Shapes)	15,000	15,000	12,000	24,000	11,000
Plates and Bars	14,500	14,500	11,600	23,200	10,850
Rods at Threads	N/A	9,000	N/A	N/A	N/A
Rods - Plain	N/A	14,500	N/A	N/A	N/A
Pins	15,950	N/A	11,600	23,200	N/A
Pipe	15,000	15,000	12,000	See tables for compression values per AISC 1.5.1.3.1	N/A
Bars & Plates 304 Steel	11,200	11,200	8,950	17,900	8,400
Pipe 304 Steel	11,200	11,200	8,950	Calculated per AISC 1.5.1.3.1	N/A
Bolts	13,700	13,700	10,960	21,920	N/A
Tube Steel A500 GRB Square or Rectangular Tubine	27,600	27,600	22,000	Calculated per AISC 1.5.1.3.1	20,700

TABLE 3.9-18 (Sheet 2 of 2)

STRESS LIMITS AND LOADING COMBINATIONS FOR PIPE SUPPORTS

LOADING COMBINATIONS

→
OBE + Thermal Loads + Dead Weight + Operating Loads + PVC, PVO or FVC ≤ The Above Allowable Stress

SSE + Thermal Loads + Dead Weight + Operating Loads + PVC, PVO or FVC ≤ 90% Minimum Allowable Yield Stress

NOTE: The SRSS Combination for earthquake (OBE or SSE) and valve opening and closure (RVC, RVO or FVC) may be used.
←

TABLE 3.9-19 (Sheet 1 of 2)

VIBRATION TESTING MODES

Flow Modes for Preoperational Vibration Testing

Piping Systems	Steady State	Test Level	Transient	Test Level	Instrumentation Required
Main Steam from Steam Generators	100% Power		Turbine Trip at 100% power	4	(Will be identified in system test procedure)
	Full flow through atmospheric dump valves, all valves open	1	None		None
Main Steam to Auxiliary Feed-water Pump Tur-bine	Run at full pump flow	1	AFW turbine trip at full pump flow	1	None
Feedwater and Auxiliary Feed-water	Single AFW Pump Operation for Pumps 2A, 2B, 2C; recircu-lation	1	Pump start, recircu-	1	None
Intake Cooling Water Pumps Dis-charge Piping	Pump(s) Operating	1	None		None
Component Cooling Water	Pump(s) Operating	1	None		None
Diesel Oil Trans-fer Pump Dis-charge Piping	Pump(s) Operating	1	None		None
Steam Generator Blowdown	Flow at normal rate	1	Initiate flow, system cold	1	None
	Flow at maximum rate	1	None		None

TABLE 3.9-19 (Sheet 2 of 2)

VIBRATION TESTING MODES

Flow Modes for Preoperational Vibration Testing

Piping Systems Required	Steady State	Test Level	Transient	Test Level	Instrumentation
Reactor Coolant Main Loop	Single and Multiple Pump Operation	1	Pump(s) starts and stops	1	None
			Pressurizer Spray Valve Cycling	2	Hand-Held Vibration Amplitude Meter
Relief Valve Discharge Piping			RV operation	4	(Will be identified in system test procedure)
Chemical & Volume Control System	Letdown flow modes	1			None
	Boric acid makeup pumps A and B	1			None
	Charging Pumps A, B and A/B Single and Multiple pump operation	2	Single and Multiple Pumps starts and stops	2	Hand-Held Vibration Amplitude Meter
Low Pressure Safety Injection	LPSI Pumps A and B operating in minimum recirculation mode	1	None		None
	Shutdown cooling mode	1	None		None
High Pressure Safety Injection	HPSI Pumps A, B and A/B operating in minimum recircula- tion mode	1	None		None
	Safety injection mode	1	None		None
Fuel Pool Cooling	Pump(s) A and B operating	1	None		None
Containment Spray	Pumps A and B in minimum recirculation mode	1	None		None

TABLE 3.9-20

INSTRUMENTATION REQUIREMENTS FOR LEVEL 2 TEST

To perform the Level 2 Test, the following instruments are needed:

1. Transducers (Accelerometers)

   Range of amplitudes up to 600 mils
   Range of frequencies: 2 - 1000 Hz
   Maximum operating temperature: 653 F

   Hand-held transducers may be used for steady state vibration measurements if testing temperature is less than 250F. Temporarily mounted transducers are recommended for: (a) steady state tests, if piping temperature exceeds 250F; (b) all Level 2 transient tests. Clamped brackets or magnetic bases may be used to mount transducers.

2. Hand-Held Vibration Indicators for Steady State Testing

   Each instrument shall have the following range of scales:

(a) 0 - 10 mils	(b) 0 - 20 mils	(c) 0 - 40 mils
(d) 0 - 100 mils	(e) 0 - 300 mils	(f) 0 - 600 mils

3. Hand-Held Peak Hold Indicator for Transient Test

   Required ranges of displacement are the same as for steady state testing in Item (2) above.

TABLE 3.9-21 (1 of 2)

CLASS 1 RANGE VALUES

FIGURE NUMBER	CALCULATION NUMBER	POINT NUMBER	STRESS RANGE (psi)	STRESS RATIO 3Sm/STRESS RANGE	m	n	Ke
3.6A-17	1133	37	40,452	1.243	1.7	0.3	1.0
3.6A-17	1133	71	55,698	1.007	1.7	0.3	1.0
3.6A-8	1135	24	48,596	1.035	1.7	0.3	1.0
3.6A-37	1163	2400	52,757	0.914	1.7	0.3	1.0
3.6A-37	1163	4400	48,264	1.000	1.7	0.3	1.0
3.6A-31	1271-1	23	49,189	1.140	1.7	0.3	1.0
3.6A-31	1271-1	101	49,221	1.139	1.7	0.3	1.0
3.6A-31	1271-1	104	50,163	1.118	1.7	0.3	1.0
3.6A-31	1271-1	32	51,345	1.092	1.7	0.3	1.0
3.6A-34	1271-2	116	53,454	1.049	1.7	0.3	1.0
3.6A-27	1206	20	41,909	1.207	1.7	0.3	1.0
3.6A-27	1206	22	43,456	1.164	1.7	0.3	1.0
3.6A-27	1206	23	44,984	1.125	1.7	0.3	1.0
3.6A-27	1206	25	44,627	1.134	1.7	0.3	1.0
3.6A-27	1206	27	48,403	1.045	1.7	0.3	1.0
3.6A-27	1206	28	48,538	1.243	1.7	0.3	1.0
3.6A-27	1206	31	40,712	1.042	1.7	0.3	1.0
3.6A-38	2606	2741	42,346	1.190	1.7	0.3	1.0
3.6A-7	2696	2001	63,029	0.803	1.7	0.3	1.818
3.6A-7	2696	3	51,972	0.973	1.7	0.3	1.09
3.6A-7	2696	4	51,346	0.986	1.7	0.3	1.049
3.6A-7	2696	5	49,784	1.016	1.7	0.3	1.0
3.6A-7	2696	50	42,827	1.182	1.7	0.3	1.0

TABLE 3.9-21 (Sheet 2 of 2) Revision 14 (12/05)

CLASS 1 RANGE VALUES

FIGURE NUMBER	CALCULATION NUMBER	POINT NUMBER	STRESS RANGE (psi)	STRESS RATIO 3Sm/STRESS RANGE	m	n	Ke
3.6A-21	1020	4	43,066	1.137	1.7	0.3	1.0
3.6A-21	1020	7	47,706	1.027	1.7	0.3	1.0
3.6A-21	1020	75	45,802	1.069	1.7	0.3	1.0
3.6A-21	1020	95	41,538	1.227	1.7	0.3	1.0
3.6A-21	1020	96	39,302	1.246	1.7	0.3	1.0
3.6A-21	1020	12	42,169	1.202	1.7	0.3	1.0
3.6A-21	1020	125	44,480	1.139	1.7	0.3	1.0
3.6A-21	1020	2205	56,649	0.865	1.7	0.3	1.0
3.6A-20	1020	938	41,396	1.183	1.7	0.3	1.0
3.6A-20	1020	946	52,034	1.153	1.7	0.3	1.0
3.6A-26	1024	4	40,780	1.249	1.7	0.3	1.0
3.6A-25	1024	19	45,580	1.118	1.7	0.3	1.0
3.6A-24	1024	827	61,022	0.830	1.7	0.3	1.0
3.6A-14	1131	525	40,331	1.247	1.7	0.3	1.0
3.6A-11	1132	5061	54,930	1.021	1.7	0.3	1.0
3.6A-11	1132	516	44,709	1.125	1.7	0.3	1.0
3.6A-11	1132	5221	44,290	1.142	1.7	0.3	1.0
3.6A-11	1132	5243	49,215	1.028	1.7	0.3	1.0
3.6A-11	1132	525	48,246	1.049	1.7	0.3	1.0
3.6A-11	1132	529	41,393	1.222	1.7	0.3	1.0
3.6A-11	1132	5290	43,279	1.169	1.7	0.3	1.0
3.6A-11	1132	530	42,733	1.184	1.7	0.3	1.0
3.6A-11	1132	5374	48,926	1.226	1.7	0.3	1.0
3.6A-17	1133	31	44,210	1.137	1.7	0.3	1.0
3.6A-17	1133	34	42,249	1.197	1.7	0.3	1.0
3.6A-17	1133	35	42,972	1.177	1.7	0.3	1.0

→(DRN 03-2056, R14)

NOTE:

Stress ranges and stress ratios represent an historic summary of data developed to illustrate high stress locations. For the current values of stress range and stress ratio in the piping system, refer to the latest revision of the pipe stress analysis calculations.

←DRN 03-2056, R14)