3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section shows the transients which are used in the design of the ASME Boiler and Pressure Vessel Code (ASME Code) Class 1 core support, reactor internals, and control rod drive (CRD) components. The number of cycles or events for each transient are included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Code if applicable. (The first four conditions correspond to Service Levels A, B, C, and D, respectively for those components constructed to the 1976 or later Edition of the ASME Code.)

3.9.1.1.1 Control Rod Drive Transients

The normal and test service load cycles used for design purposes for the 40 year life of the Control Rod Drive (CRD) are as follows:

Transient		Category	Cycles
1.	Reactor startup/shutdown	Normal/upset	120
2.	Vessel pressure tests	Normal/upset	130
3.	Vessel overpressure tests	Normal/upset	10
4.	Scram test plus startup scrams	Normal/upset	300
5.	Operational scrams	Normal/upset	300
6.	Jog cycles	Normal/upset	30,000
7.	Shim/drive cycles	Normal/upset	1,000

In addition to the above cycles, the following have been considered in the design of the CRD.

Transient		Category	Cycles
8.	Operating Basis Earthquake* (OBE)	Normal/upset	10
9.	Safe Shutdown Earthquake** (SSE)	Faulted	1
10.	Scram with inoperative buffer	Normal/upset	10
11.	Scram with stuck control blade	Normal/upset	1

All ASME Code Class 1 components of the CRD have been analyzed according to the ASME Code.

The capability of the CRD's to withstand the emergency and faulted conditions is verified by test rather than analysis.

3.9.1.1.2 CRD Housing and Incore Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

Transient		Category	Cycles
1.	Normal startup & shutdown	Normal/upset	120
2.	Vessel pressure tests	Normal/upset	130
3.	Vessel overpressure tests	Normal/upset	10
4.	Interruption of feedwater flow	Normal/upset	80
5.	Scram	Normal/upset	200
6.	OBE	Normal/upset	10
7.	SSE	Faulted	1

* The frequency of occurrence of this transient would indicate emergency category. However, for conservatism, this OBE condition was analyzed as an upset condition. Ten peak OBE cycles for each occurrence are postulated.

** SSES is a faulted condition; however, in the stress analysis report, it was treated as emergency with lower stress limits.

Transient	Category	Cycles
8. Stuck rod scram	Normal/upset	1
9. Scram no buffer	Normal/upset	10

3.9.1.1.3 Hydraulic Control Unit Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40 year life and the Hydraulic Control Unit (HCU) are as follows:

Transient	Category	Cycles
1. Normal startup and shutdown	Normal/upset	120
2. Vessel pressure tests	Normal/upset	130
3. Vessel overpressure tests	Normal/upset	10
4. Scram tests (cold)	Normal/upset	300
5. Operational scrams (hot)	Normal/upset	300
6. Jog cycles	Normal/upset	30,000
7. Drive cycles	Normal/upset	1,000
8. Scram with stuck scram discharge valve	Normal/upset	1
9 OBE	Normal/upset	10
10. SSE	Faulted	1

3.9.1.1.4 Core Support and Reactor Internals Transients

The events and number of cycles used for the design and fatigue analysis for the 40-year life of the core support and internals are shown in Table 3.9-1.

3.9.1.1.5 Main Steam System Transients

The transients considered in the stress analysis of the main steam piping are included in Table 3.9-4.

3.9.1.1.6 Recirculation System Transients

The transients considered in the stress analysis of the recirculation piping are included in Table 3.9-4.

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes the reactor pressure vessel, support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9-1 were specified in the reactor assembly design and fatigue analysis.

Reactor design cycle or transient limits are as follows:

Transient	Design Cycle
Heatup and Cooldown (Envelopes items 3, 10, 13, 16b, 16d from Table 3.9-1)	70°F to 551°F to 70°F
Reactor Trip cycle (Item 9 from Table 3.9-1)	100% to 0% Rated Thermal Power
Hydrostatic Pressure and Leak Tests (Item 2 from Table 3.9-1)	Pressurized ≥930 psig and ≤1250 psig

3.9.1.1.8 Main Steam Isolation Valve Transients

The main steam isolation valves are designed for the following service conditions and thermal cycles:

Transient			Category	Cycles
1.	Pre-op @100°F/hr		Normal/upset	150
2.	Startup (heating 100°F/hr)		Normal/upset	120
3.	Shutdown			
	a.	cooling cycles @100°F/hr 540°F to 375°F	Normal/upset	120
	b.	cooling cycles @270°F/hr 375°F to 330°F	Normal/upset	120
	c.	cooling cycles @100°F/hr 330°F to 100°F	Normal/upset	120

Transient			Category	Cycles
4.	Scram cooling cycles @100°F/hr		Normal/upset	180
5.	Emergency and faulted transients			
	a.	546°F to 281°F in 15 sec	Emergency/faulted	1
	b.	546°F to 375°F in 3.3 min	Emergency/faulted	1
		375°F to 281°F @300°F/hr	Emergency/faulted	1
	c1.	546°F to 375°F in 10 min	Emergency/faulted	8
	c2.	375°F to 281°F @100°F/hr	Emergency/faulted	8
	d1.	546°F to 583°F in 2 sec	Emergency/faulted	1
	d2.	583°F to 538°F in 30 sec	Emergency/faulted	1
	d3.	538°F to 400°F @100°F/hr	Emergency/faulted	1
	d4.	400°F to 546°F @100°F/hr	Emergency/faulted	1
	e1.	561°F to 500°F in 7 min	Emergency/faulted	10
	e2.	500°F to 400°F @100°F/hr	Emergency/faulted	10
	e3.	400°F to 546°F @100°F/hr	Emergency/faulted	10

3.9.1.1.9 Safety/Relief Valves Transients

The transients used in the analysis of the safety/relief valves are as follows:

Transient		Category	Cycles
1.	Pre-op and in-service testing (100°F/hr).	Normal/upset	150
2.	Startup (100°F/hr) and pressure increase (0 psig to 1000 psig).	Normal/upset	120
3.	Shutdown (100°F/hr, pressure decrease to 0 psig).	Normal/upset	120
4.	Scram.	Normal/upset	180
5.	System pressure and temperature decay from 1000 psig and 546°F to 35 psig and 281°F within 15 sec.	Emergency/faulted	1

Transient		Category	Cycles
6.	System temperature change from 546°F to 375°F within 3.3 mins and from 375°F to 281°F at a rate of 300°F/hr. Pressure change from 1000 psig to 35 psig	Emergency/faulted	1
7.	System temperature change from 546°F to 375°F within 10 min. and from 375°F to 281°F at a rate of 100°F/hr. Pressure change from 1000 psig to 35 psig.	Emergency/faulted	8
8.	System temperature change from 546°F to 583°F within 2 sec, from 583°F to 538°F within 30 sec, and from 538°F to 400°F and return to 546°F at a rate of 100°F/hr. Pressure change from 1000 psig thence to 1350 psig , thence to 240 psig and return to 1000 psig.	Emergency/faulted	1
9.	System temperature changes, greater than 30°F, from 561°F to 500°F within 7 min. and from 500°F to 400°F and return to normal operating temperature at 546°F at a rate of 100°F/hr. Pressure change from 1000 psig to 1180 psig to 240 psig and return to normal operating of 1000 psig.	Emergency/faulted	10

Paragraph NB3552 of ASME III code excludes various transients and provides means for combining those which are not excluded. Review and approval of the equipment supplier's certified calculation provides assurance of proper accounting of the specified transients.

3.9.1.1.10 Recirculation Flow Control Valve Transients

Not applicable since Susquehanna SES has no flow control valve.

3.9.1.1.11 Recirculation Pump Transients

The following transients are listed in the design specification as a requirement for design considerations. However, a submitted certified analysis considering thermal stresses was not required. The vendor was required to submit a certification of compliance. The submitted certified design calculations only considered pressure transient. Nozzle piping loads were considered in accordance with the following paragraph:

> "The pump case was designed to withstand secondary stresses due to piping reactions in accordance with Paragraph 452.4b of the ASME Standard Code for Pumps and Valves for Nuclear Power (1968 Draft)."

Transient	Category	Cycles
1. Heatup and cooldown at 100°F/hr	Normal/upset	300
2. ±29°F temperature changes	Normal/upset	600
3. ±50°F temperature changes	Normal/upset	200
4. RPV pressure transients to 110% design pressure	Normal/upset	1
5. SRV blowdowns	Emergency	8
6. Improper pump startup, 100°F to 552°F in 15 sec	Emergency	1
7. Cooling transient 552°F to 281°F in 15 sec	Faulted	1
8. Hydrotest to 1300 psig	Testing	130
9. Hydrotest to 1670 psig	Testing	3

3.9.1.1.12 Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves.

Transient	Cycles
1. 50°F - 575°F - 50°F at 100°F/hr	300
2. ±29°F between limits of 50°F and 575°F, instantaneous	600
3. +50°F between limits of 50°F and 546°F, instantaneous	200
4. 546°F to 375°F, instantaneous	30
5. 546°F to 281°F, instantaneous	2
6. 130°F to 546°F, instantaneous	1
7. 110% design pressure at 575°F	1
8. 1300 psi at 100°F installed hydrostatic test	130
9. 1670 psi at 100°F installed hydrostatic test	3

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific NSSS components. (Computer programs were not used in the analysis of all components, thus, not all components are listed.) The NSSS programs can be divided into two categories.

The computer programs discussed in this section are those programs used for the original plant design. Changes to later versions of these programs or the addition of entirely new computer programs for safety related applications is controlled by procedures under our Operational Quality Assurance Program.

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

(a) MASS
(b) SNAP (MULTISHELL)
(c) HEATER
(d) SAP4
(e) ANSI7
(f) LUGSTR
(g) PISYS
(h) RVFOR
(i) TSFOR
(j) EZPYP
(k) PDA
(l) SAP4G
(m) FTFLG01
(n) ANSYS
(o) POSUM
(p) BILRD
(q) DYSEA
(r) SPECA
(s) SEISM

Vendor Programs

The verification of the following CB&I programs is assured by contractual requirements between GE and the vendor. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10CFR50, Appendix B.

(a) 711 GENOZZ
(b) 948 NAPALM
(c) 1027
(d) 846
(e) 781 KALNINS
(i) 928 TGRV
(j) 962 E0962A
(k) 984
(l) 992 GASP
(m) 1037 DUNHAM'S

(f)	979 ASFAST	(n)	1335
(g)	766 TEMAPR	(o)	1606 & 1647 HAP
(h)	767 PRINCESS	(p)	1634 N

A list of computer programs used in the BOP system components is provided in Table 3.9-5. This list consists of computer programs that are developed and/or owned by Bechtel Power Corporation, and computer programs that are recognized and widely used in industry.

The Bechtel developed and/or owned computer programs are documented, verified, and maintained by Bechtel and meet the requirements of 10CFR50, Appendix B. A brief description of each of these programs is provided in Appendix 3.9A.

3.9.1.2.1 Reactor Vessel and Internals

The computer programs used in the preparation of the reactor vessel stress report are identified and their use summarized in the following paragraphs.

3.9.1.2.1.1 Reactor Vessel

3.9.1.2.1.1.1 CB&I Program 711 "GENOZZ"

The GENOZZ computer program is used to proportion barrel and double taper type nozzle configurations to comply with the specifications of the ASME Code, Section III and contract documents. The program will either design such a configuration or analyze the configuration input into it. If the input configuration will not comply with the specifications, the program will modify the design and redesign it to yield an acceptable result.

3.9.1.2.1.1.2 CB&I Program 948 - "NAPALM"

The basis for the program NAPALM, Nozzle Analysis Program-All Loads Mechanical, is to analyze nozzles for mechanical loads and find the maximum stress intensity and location. The program analyzes at specified locations from the point of application of the mechanical loads. At each location the maximum stress intensity is calculated for both the inside and outside surfaces of the nozzle. The program gives the maximum stress intensity for both the inside and outside surfaces of the nozzle as well as its angular location around the circumference of the nozzle from the 0° reference location. The principle stresses are also printed. The stresses resulting from each component of loading (bending, axial, shear, and torsion) are printed, as well as the loadings which caused these stresses.

3.9.1.2.1.1.3 CB&I Program 1027

This program is a computerized version of the analysis method contained in the "Welding Research Council Bulletin F107," December, 1965.

Part of this program provides for the determination of the shell stress intensities (S) at each of four cardinal points at both the upper and lower shell plate surfaces (ordinarily considered outside and inside surfaces) around the perimeter of a loaded attachment on a cylindrical or spherical vessel. With the determination of each S, there is also determined the components of that S (2 normal stresses, and one shear stress). This program provides the same information

as the manual computation and the input data is essentially the input of the geometry of the vessel and attachment.

3.9.1.2.1.1.4 CB&I Program 846

This program computes the required thickness of a hemispherical head with a large number of circular parallel penetrations by means of the area replacement method in accordance with the ASME Code, Section III.

In cases where the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

3.9.1.2.1.1.5 CB&I Program 781 - "KALNINS"

This program is a thin elastic shell program for shells of revolution. This program was developed by Dr. A. Kalnins of Lehigh University. Extensive revisions and improvements have been made by Dr. J. Endicott to yield the CB&I version of this program.

The basic method of analysis was published by Professor Kalnins in the Journal of Applied Mechanics, Volume 31, September, 1964, pages 467 through 476. The KALNINS thin shell program (Program 781) is used to establish the shell influence coefficient and to perform detail stress analysis of the vessel. The stresses and the deformations of the vessel can be computed for any combination of the following axi-symmetric loading:

a) Preload condition

b) Internal pressure

c) Thermal load

3.9.1.2.1.1.6 CB&I Program 979 - "ASFAST"

ASFAST Program (Program 979) performs the stress analysis of axi-symmetric, bolted closure flanges between head and cylindrical shell.

3.9.1.2.1.1.7 CB&I Program 766 - "TEMAPR"

This program will reduce any arbitrary temperature gradient through the wall thickness to an equivalent linear gradient. The resulting equivalent gradient will have the same average temperature and the same temperature-moment as the given temperature distribution. Input consists of plate thickness and actual temperature distribution. The output contains the average temperature and total gradient through the wall thickness. The program is written in FORTRAN IV language.

3.9.1.2.1.1.8 CB&I Program 767 - "PRINCESS"

The PRINCESS computer program calculates the maximum alternating stress amplitudes from a series of stress values by the method in Section III of the ASME Pressure Vessel Code.

3.9.1.2.1.1.9 CB&I Program 928 - "TGRV"

The TGRV program is used to calculate temperature distributions in structures or vessels. Although it is primarily a program for solving the heat conduction equations, some provisions have been made for including radiation and convection effects at the surfaces of the vessel. The TGRV program is a greatly modified version of the TIGER heat transfer program written about 1958 at Knolls Atomic Power Laboratory, by A. P. Bray. There have been many versions of TIGER in existence including TIGER II, TIGER II B, TIGER IV, and TIGER V, in addition to TGRV. This manual has been written for use with CB&I's version of TGRV.

This program utilizes an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments and continually summed. Linear mathematics are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer, conduction, radiation and convection, as well as internal heat generation.

Given any odd-shaped structure, which can be represented by a three dimensional field, its geometry and physical properties, boundary conditions, and internal heat generation rates, TGRV will calculate and give as output the steady state or transient temperature distributions in the structure as a function of time.

3.9.1.2.1.1.10 CB&I Program 962 - "EO962A"

Program E0962A is one of a group of programs (E0953A, E1606A, E0962A, E0992N, E1037N, and E0984N) which are used together to determine the temperature distribution and stresses in pressure vessel components by the finite element method.

Program E0962A is primarily a plotting program. Using the nodal temperatures calculated by program E1606A or Program E0928A, and the node and element cards for the finite element model, it calculates and plots lines of constant temperature (isotherms). These isotherm plots are used as part of the stress report to present the results of the thermal analysis. They are also very useful in determining at which points in time the thermal stresses should be determined.

In addition to its plotting capability the program can also determine the temperatures of some of the nodal points by interpolation. This feature of the program is intended primarily for use with the compatible TGRV and finite element models that are generated by program E0953A.

3.9.1.2.1.1.11 CB&I Program 984

Program 984 is used to calculate the stress intensity of the stress differences, on a component level, between two different stress conditions. The calculation of the stress intensity of stress component differences (the range of stress intensity) is required by Section III of the ASME Code.

3.9.1.2.1.1.12 CB&I Program 992 - GASP

The GASP computer program, originated by Prof. E. L. Wilson of the University of California at Berkeley, uses the finite element method to determine the stresses and displacements of plane

or axi-symmetric structures of arbitrary geometry and is written in FORTRAN IV. For a detailed account, see the following reference document.

> Wilson, E. L.; "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties," Aerojet General Corporation, Sacramento, California. Technical Memorandum No. 23, July 1965.

As mentioned above, the program determines the stresses and displacements of plane or axi-symmetric structures using the finite element method. The structures may have arbitrary geometry and have linear or non-linear material properties. The loadings may be thermal, mechanical, accelerational, or a combination of these.

The structure to be analyzed is broken up into a finite number of discrete elements or "finite-elements" which are interconnected at finite number of "nodal-points" or "nodes." The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes. The basic input to the program consists of the geometry of the stress-model and the boundary conditions. The program then gives the stress components at the center of each element and the displacements at the nodes, consistent with the prescribed boundary conditions.

3.9.1.2.1.1.13 CB&I Program 1037 - "DUNHAM'S"

DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacements of axi-symmetric structures of arbitrary geometry subjected to either axi-symmetric loads or non-axi-symmetric loads represented by a Fourier series.

This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle non-axi-symmetric loads (which requires that each node have three degrees of freedom) and the material properties for DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.1.14 CB&I Program 1335

To obtain stresses in the shroud support, the baffle plate must be made a continuous circular plate. The program makes this modification and allows the baffle plate to be included in CB&I program 781 as two isotropic parts and an orthotropic portion at the middle (where the diffuser holes are located).

3.9.1.2.1.1.15 CB&I Programs 1606 and 1657 - "HAP"

The HAP program is an axi-symmetric nonlinear heat analysis program. It is a finite element program and is used to determine nodal temperatures in a two-dimensional or axi-symmetric body subjected to transient disturbances. Programs 1606 and 1657 are identical except that 1606 has a larger storage area allocated and can thus be used to solve larger problems. The model for program 1606 is compatible with CB&I stress programs 992 and 1037.

3.9.1.2.1.1.16 CB&I Program 1634N

This program is used to analyze thin cylindrical shells subjected to local loading beyond the range where Bijloard's curves are directly applicable, i.e., R/t >300.

This program computes stresses and displacements in thin walled elastic cylindrical shells subjected to mechanical loading such as radial loads, longitudinal and circumferential moments.

3.9.1.2.1.2 Reactor Internals

The following programs are used in the analysis of core support structures and other safety related reactor internals: MASS, SNAP (MULTISHELL), and HEATER. These programs are described in detail in Section 4.1.

3.9.1.2.2 Piping

The computer programs used in the analysis of NSSS piping systems are identified and summarized below:

3.9.1.2.2.1 Structural Analysis Program SAP 4

The Structural Analysis Program SAP 4, for the static and dynamic analysis of linear structural system is the result of several years research and development experience. The program has proven to be a very flexible and efficient analysis tool. The first version of the SAP Program was published in September, 1970. An improved static analysis program, namely SOLID SAP, or SAP 2, was presented in 1971. Work was then started on a new static and dynamic analysis program. The program SAP 3 was released towards the end of 1972. SAP 4 has the additional analysis capability of out-of-core direct integration for the time history analysis.

The structural systems to be analyzed may be composed of combinations of a number of different structural elements. The program presently contains the following element types:

a) three-dimensional truss element

b) three-dimensional beam element

c) plane stress and plane strain element,

d) two-dimensional axisymetric solid,

e) three-dimensional solid,

f) thick shell element,

g) thin plate or thin shell element,

h) boundary element,

i) pipe element (tangent and bend).

These structural elements can be used in a static or dynamic analysis. The capacity of the program depends mainly on the total number of nodal points in the system, the number of eigenvalues needed in the dynamic analysis and the computer used. There is practically no restriction on the number of elements used, the number of load cases or the order and bandwidth of the stiffness matrix. Each nodal point in the system can have from zero to six

displacement degrees of freedom. The element stiffness and mass matrices are assembled in condensed form; therefore, the program is equally efficient in the analysis of one-, two-, or three-dimensional systems.

The formation of the structure matrices is carried out in the same way in a static or dynamic analysis. The static analysis is continued by solving the equations of equilibrium followed by the computation of element stresses. In a dynamic analysis the choice is between:

a) frequency calculations only,

b) frequency calculations followed by response history analysis,

c) frequency calculations followed by response spectrum analysis,

d) response history analysis by direct integration.

To obtain the frequencies and vibration mode shapes, solution routines are used which calculate the required eigenvalues and eigenvectors directly without a transformation of the structure stiffness matrix and mass matrix. This way the program operation and necessary input data for a dynamic analysis is a simple addition to what is needed for a static analysis.

3.9.1.2.2.2 Component Analysis/ANSI 7

Application. The ANSI 7 Computer Program determines stress and accumulative usage factors for thermal weight, seismic relief valve lift and turbine stop valve closure (as applicable) conditions of loadings derived from the Structural System Analysis in accordance with NB-3600 of ASME Code Section III.

Program Organization. For Class 1 ASME Code stress analysis, the program generates and prints hoop, bending, thermal discontinuity, linear temperature gradient and nonlinear temperature gradient components of stress for each equation of subarticle NB-3600 of Section III. Load combination results from possible load sets for Class 1 stress equations. The total stress (sum of component stresses) and the stress ratio (total stress divided by appropriate stress intensity limit) is printed for each Class 1 equation. The total stress (sum of the component stresses) and the stress ratio (total stress divided by the appropriate stress intensity limit) is printed for each one of the equations 9, 10, 12, and 13 of NB-3600. The alternating stresses and usage factor are calculated per NB-3653.6.

3.9.1.2.2.3 Integral Attachment/LUGSTR

The computer program "LUGSTR" was prepared to evaluate the stress in the pipe wall that are produced by loads applied to the integral attachments. The program was prepared based on the Welding Research Council Bulletin 198 including the evaluation due to stress range and fatigue analysis.

3.9.1.2.2.4 Piping Analysis Program/PISYS

PISYS is a computer code specialized for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes

called pipe joints. It is through these joints that the model interacts with the environment, and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading, and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option-of-response-spectrum analysis and the results are documented in a report to the Commission, "PISYS Analysis of NRC Problem," NEDO-24210, August, 1979.

3.9.1.2.2.5 Relief Valve Discharge Pipe Forces/RVFO R

The relief valve discharge pipe connects the relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time dependent forces to develop in the pipe wall. This computer program computes the transient fluid mechanics and the resultant pipe forces using the method of characteristics.

3.9.1.2.2.6 Turbine Stop Valve Closure/TSFOR

The TSFOR program computes the time history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

3.9.1.2.2.7 Piping Analysis Program/EZPYP

EZPYP links the ANSI-7 and SAP program together. The EZPYP program can be used to run several SAP cases by making user specified changes to a biasic SAP pipe model. By controlling files and SAP runs the EZPYP program gives the analyst the capability to perform a complete piping analysis in one computer run.

3.9.1.2.2.8 Pipe Whip Analysis/PDA

The pipe whip analysis was performed using the PDA computer program. PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust-force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress-strain relations are used for the pipe and the restraint. Similar to the popular plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Shear deformation is also neglected. The pipe bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using an energy consideration and the equations are numerically integrated in small time steps to yield time-history information of the deformed pipe.

3.9.1.2.3 Recirculation and ECCS Pumps and Motors

3.9.1.2.3.1 Recirculation Pumps

No computer programs were used in the design of the recirculation pumps.

3.9.1.2.3.2 ECCS Pumps and Motors

3.9.1.2.3.2.1 Structural Analysis Program/SAP4G

SAP4G is used to analyze the structural and functional integrity of the ECCS pump/motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.3.2.2 Effects of Flange Joint Connections/FTFLG01

The flange joints connecting the pump bowl castings are analyzed using FTFLG01. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix II and Section III of the ASME Code.

3.9.1.2.3.2.3 Structural Analysis of Discharge Head/ANSYS

ANSYS is used to analyze the pump discharge head flange and bolting taking into account of the prying action developed by the flag face contact surface. The program is described in detail in 3.12.

3.9.1.2.3.2.4 Beam Element Data Processing/POSUM

POSUM is a computer code designed to process SAP generated beam element data for pump or heat exchanger models. The purpose is to determine the load combination that would produce the maximum stress in a selected beam element. It is intended for use on RHR heat exchangers with four nozzles or ECCS pumps with two nozzles.

3.9.1.2.4 RHR Heat Exchangers

3.9.1.2.4.1 Structural Analysis Programs/SAP4G

SAP4G is used to analyze the structural and functional integrity of the RHR heat exchangers. The description of this program is provided in Subsection 3.9.1.2.3.2.1.

3.9.1.2.4.2 Shell Attachment Parameters and Coefficients/BILRD

BILRD is used to calculate the shell attachment parameters and coefficients utilized in the stress analysis of the support to shell junction. The method, per Welding Research Council Bulletin No. 107, is implemented to calculate local membrane stress due to the support reaction loads on the heat exchanger shell.

3.9.1.2.4.3 Beam Element Data Processing/POSUM

POSUM is used to process SAP generated beam element data. The description of this program is provided in Subsection 3.9.1.2.3.2.4.

3.9.1.2.5 Dynamic Loads Analysis

3.9.1.2.5.1 Dynamic Analysis Program/DYSEA

DYSEA simulates a beam model in the annulus pressurization dynamic analysis. A detailed description of DYSEA is provided in Section 4.1. DYSEA employs a preprocessor program names GZAPL. GZAPL converts pressure time histories into time varying loads and forcing functions for DYSEA. The overall resultant forces and moments time histories at specified points of resolution can also be obtained from GZAPL.

3.9.1.2.5.2 Acceleration Response Spectrum Program/SPECA

SPECA generates acceleration response spectrum for an arbitrary input time history of piece-wise linear accelerations, i.e., to compute maximum acceleration responses for a series of single-degree-of-freedom systems subjected to the same input. It can accept acceleration time histories from a random file. It also has the capability of generating the broadened/enveloped spectra when the special points are generated equally spaced on a logarithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analyses.

3.9.1.2.5.3 Fuel Support Loads Program/SEISM

SEISM02 computes the vertical fuel support loads using the component element methods in dynamics. The methodology is based on the publication "The Component Element Method in Dynamics," by S. Levy and J. P. D. Wilkinson, McGraw Hill Co., New York, 1976.

3.9.1.3 Experimental Stress Analysis

When experimental stress analysis is used in lieu of analytical methods for Seismic Category I ASME Code items, the requirements for experimental testing enumerated in the ASME Code applicable for the specific components under test are applied. When testing is required for Seismic Category I non-ASME Code items, account is taken of the effects of differences in size, dimensional tolerances, and ultimate strength (or other governing material properties) between the actual and tested parts to assure that the loads obtained from tests are realistic or conservative representation of the capability of the actual structure.

The following subsections in this section list the components upon which experimental stress analysis was used.

3.9.1.3.1 Experimental Stress Analysis of NSSS Piping Components

The following components have been tested to verify their design adequacy:

a) Snubbers

b) Pipe whip restraints

Descriptions of the snubber and whip restraint tests are contained in Subsections 3.9.3.4 and 3.6, respectively.

3.9.1.3.2 Seismic Category I Items Other Than NSSS

Experimental testing is performed in the qualification and acceptance of snubbers, compensating struts, and honeycomb material used in energy absorbing components for pipe break.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

All Seismic Category I equipment in the NSSS is evaluated for the faulted loading conditions. However, emergency stress limits rather than faulted stress limits were used in many cases. In essentially all cases, calculated stresses are within allowable limits. The following paragraphs in subsection 3.9.1.4 show examples of the treatment of faulted conditions for the major components on a component by component basis. Additional discussion of faulted analysis can be found in Subsections 3.9.3 and 3.9.5, and Table 3.9-2.

Subsection 3.9.2.2 and Section 3.7 discuss the treatment of dynamic loads resulting from the postulated seismic and hydrodynamic events. Section 3.9.2.5 discusses the dynamic analysis of loads affected on reactor internals equipment resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas and no cases are identified where design limits, such as clearance limits, are violated.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Control Rod Drives

The ASME Code components of the CRD have been analyzed for abnormal conditions shown in Subsection 3.9.1.1.1.

The load criteria, calculated and allowable stresses for various operating conditions are summarized in Table 3.9-2(u).

The design adequacy of non ASME code components of the CRD has been verified by analysis and extensive testing programs on both component parts, specially instrumented prototype drives and production drives. The testing has included postulated abnormal events as well as the service life cycle listed in Subsection 3.9.1.1.1.

3.9.1.4.1.2 Hydraulic Control Unit

The Hydraulic Control Unit (HCU) was analyzed for the faulted condition.

The analysis of the HCU under faulted condition loads establishes the structural integrity of the system.

Section 3.9.2.2a.2.4 discusses the dynamic qualification of the HCU.

3.9.1.4.2 Standard Reactor Internal Components

3.9.1.4.2.1 CR Guide Tube

The maximum calculated stress on the CR Guide Tube occurs in the base during the faulted condition. The loading criteria and calculated and allowable stresses are summarized in Table 3.9-2aa.

3.9.1.4.2.2 Incore Housing

The maximum calculated stress on the Incore Housing occurs at the outer surface of the vessel penetration during the faulted condition. The loading criteria and calculated and allowable stresses are shown in Table 3.9-2ab.

3.9.1.4.2.3 Jet Pump

The maximum stress in the jet pump occurs in the faulted condition due to impulse loading of the diffuser during a pipe rupture and blowdown. Table 3.9-2w summarizes loading criteria, and calculated and allowable stresses.

3.9.1.4.2.4 Orificed Fuel Support

A series of vertical and horizontal load tests were performed on the orificed fuel support (OFS) in order to verify the design. Results from these tests indicate that the component and seismic loading of the OFS are well below the stress limit allowables with a safety margin of 1.26 for the normal and upset conditions, and 1.5 for the faulted condition. The allowable stress limits were arrived at by applying a .65 quality factor to the ASME Code allowables of 1.5 S for the upset condition, and 1.5 x .7 Su for the faulted condition.

3.9.1.4.2.5 Control Rod Drive Housing

The CRD Housing is analyzed for the faulted condition considering SSE and hydrodynamic loads. Table 3.9-2v shows that the calculated stresses are within the allowable stresses.

3.9.1.4.3 Reactor Pressure Vessel Assembly

The reactor pressure vessel, support skirt, and the shroud support were evaluated using elastic analysis methods for the faulted conditions. For the support skirt and shroud support an elastic analysis was performed and buckling was evaluated for the compressive load. Table 3.9-2a lists the loading criteria and calculated and allowable stresses for the various loading combinations.

3.9.1.4.4 Core Support Structure

The evaluations for faulted conditions for the core support structure are discussed in Subsection 3.9.5. The loading criteria and calculated and allowable stresses are summarized in Table 3.9-2b.

3.9.1.4.5 Main Steam Isolation, Recirculation Gate, and Safety/Relief Valves

Tables 3.9-2g, 3.9-2h, and 3.9-2j provide a summary of the analyses of the safety/relief, main steam isolation, and recirculation gate valves, respectively.

Standard design rules, as defined in the ASME Code, Section III, are utilized in the analysis of pressure boundary components of Seismic Category I valves. Conventional elastic stress analysis is used to evaluate components not defined in the ASME code. The code allowable stresses are applied to determine acceptability of structure under applicable loading conditions including faulted condition.

3.9.1.4.6 Main Steam and Recirculation Piping

For Main Steam and Recirculation System piping, elastic analysis methods are used for evaluating faulted loading conditions. The equivalent allowable stresses using elastic techniques are obtained from ASME Code Section III, Appendix F, "Rules for Evaluation of Faulted Conditions," and these are above elastic limits. Additional information on the main steam and recirculation piping and pipe-mounted equipment is in Table 3.9-2d and 3.9-2e.

3.9.1.4.7 Nuclear Steam Supply System Pumps, Heat Exchanger, and Turbines

The recirculation, ECCS, RCIC, and SLC pumps, RHR heat exchangers and RCIC turbine have been analyzed for the faulted loading conditions identified in Subsection 3.9.1.1. In all cases, stresses were within the elastic limits. The analytical methods, stress limits, and allowable stresses are discussed in Subsections 3.9.2.2 and 3.9.3.1.

3.9.1.4.8 Control Rod Drive Housing Supports

Design adequacy of the CRD Housing Supports is shown by comparing the total static and dynamic loads to the original design loads. The comparison shows that the hydrodynamic loads and other dynamic loads combined by SRSS are less than the original design loads.

3.9.1.4.9 Fuel Storage Racks

The stress criteria, loadings, calculated stresses, and stress limits for the faulted conditions for the new fuel storage racks are shown in Table 3.9-2s. No inelastic stress analyses were used on these components.

Similar information for the spent fuel storage racks was provided by the rack vendor.

3.9.1.4.10 Fuel Assembly (Including Channels)

Seismic/ LOCA loading evaluations for channeled ATRIUM™-10 fuel and channeled Lead Use Assemblies are located in Section 4.2. (Fuel System)

3.9.1.4.11 Refueling Equipment

Refueling and servicing equipment that is important to safety is classified as essential equipment per the requirements of 10 CFR 50, Appendix A. This equipment and other equipment whose failure would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra and combines loads at frequencies up to 33 Hz for seismic and up to 60 Hz for the hydrodynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC, allowables. The calculated stresses and allowable limits for the faulted loads for the fuel preparation machine are provided in Table 3.9-2s. The refueling platform has also been examined; it can withstand the faulted loads due to seismic hydrodynamic events.

3.9.1.4.12 Seismic Category I Items Other than NSSS

For statically applied loads, the stress allowables of Appendix F of the ASME Code, Section III, Winter 1972 were used for code components. For non-code components, allowables were based on tests or accepted standards consistent with those in Appendix F of the code.

Dynamic loads for components loaded in the elastic range were calculated using dynamic load factors, time history analysis, or any other method that assumes elastic behavior of the component.

The limits of the elastic range are defined in Paragraph 1323 of Appendix F for the code components. The local yielding due to stress concentration is assumed not to affect the validity of the assumptions of elastic behavior. The stress allowables of Appendix F for elastically analyzed components were used for code components. For non-code components, allowables were based on tests or accepted material standards consistent with those in Appendix F for elastically analyzed components.

The methods used in evaluating the pipe break effects are discussed in Section 3.6.

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1a Preoperational Vibration and Dynamic Effects Testing on NSSS Piping

The test program is divided into three phases: preoperational vibration, startup vibration, and operational transients.

3.9.2.1a.1 Preoperational Vibration Testing

The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation piping are acceptable. This phase of the test uses visual observation.

3.9.2.1a.2 Small Attached Piping

There is no small attached piping in the NSSS scope of supply.

3.9.2.1a.3 Startup Vibration

The purpose of this phase of the program is to verify that the main steam and recirculation piping vibration are within acceptable limits. Because of limited access due to high radiation levels, no visual observation is made during this phase of the test. Remote measurements were made during the following steady state conditions:

a) Main steam flow at 25% of rated;

b) Main steam flow at 50% of rated;

c) Main steam flow at 75% of rated;

d) Main steam flow at 100% of rated.

3.9.2.1a.4 Operating Transient Loads

The purpose of the operating transient test phase is to verify that pipe stresses are within Code Limits. The amplitude of displacements and number of cycles per transient of the main steam and recirculation piping were measured and the displacements compared with acceptance criteria. The deflections are correlated with the calculated deflections to assure that the stresses remain within Code limits. Remote vibration and deflection measurements were taken during the following transients:

a) Recirculation pump starts;

b) Recirculation pump trip at 100% of rated flow;

c) Turbine stop valve closure at 100% power;

d) Manual discharge of representative S/R valves at 1,000 psig and at planned transient tests that result in S/R valve discharge.

3.9.2.1a.5 Test Evaluation and Acceptance Criteria

The piping response to test conditions shall be considered acceptable if the organization responsible for the stress report reviews the test results and determines that the tests verify that the piping responded in a manner consistent with the predictions of the stress report and/or that the tests verify that piping stresses are within Code limits. To insure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Level 1 and 2, are described in the following paragraphs.

3.9.2.1a.5.1 Level 1 Criterion

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory.

If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1a.5.2 Level 2 Criterion

Level 2 specifies the level of pipe motion which, if exceeded, requires that the responsible piping design engineer be advised. If the Level 2 limit is not satisfied, plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

3.9.2.1a.6 Corrective Actions

During the course of the tests, the remote measurements shall be regularly checked to determine compliance with the Level 1 criterion. If trends indicate that the Level 1 criterion may be violated, the measurements shall be monitored at more frequent intervals. The test will be held or terminated as soon as the criterion is violated. As soon as possible after the test hold or termination, the following corrective actions will be taken:

a) Installation Inspection. A walkdown of the piping and suspension will be made to identify any obstruction or improperly operating suspension components. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action will be taken to correct any discrepancies before repeating the test.

b) Instrumentation Inspection. The instrumentation installation and calibration will be checked and any discrepancies corrected. Additional instrumentation will be added, if necessary.

c) Repeat Test. If actions (a) and (b) identify discrepancies that could account for failure to meet the Level 1 criterion, the test will be repeated.

d) Resolution of Findings. If the Level 1 criterion is violated on the repeat test or no relevant discrepancies are identified in (a) and (b), the organization responsible for the stress report shall review the test results and criteria to determine if the test can be safely continued.

If the test measurements indicate failure to meet the Level 2 criterion, the following corrective actions will be taken after completion of the test:

a) Installation Inspection. A walkdown of the piping and suspension shall be made to identify any obstruction or improperly operating suspension components. If vibration exceeds limits, the source of the vibration must be identified. Action, such as suspension adjustment, will be taken to correct any discrepancies.

b) Instrumentation Inspection. The instrumentation installation and calibration will be checked and any discrepancies corrected.

c) Repeat Test. If (a) and (b) above identify a malfunction or discrepancy that could account for failure to comply with Level 2 criterion and appropriate corrective action has been taken, the test may be repeated.

d) Documentation of Discrepancies. If the test is not repeated, the discrepancies found under actions (a) and (b) above shall be documented in the test evaluation report and correlated with the test condition. The test will not be considered complete until the test results are reconciled with the acceptance criteria.

3.9.2.1a.7 Measurement Locations

Remote shock and vibration measurements were made in the three orthogonal directions near the first downstream S/R valve on each steam line; and in the three orthogonal directions on the piping between the recirculation pump discharge and the first downstream valve. During preoperational testing prior to fuel load, visual inspection of the piping was made, and any visible vibration measured with a handheld instrument.

For each of the selected remote measurement locations, Levels 1 and 2 deflection and acceleration limits were prescribed in the startup test specification. Level 2 limits were based on the results of the stress report adjusted for operating mode and instrument accuracy; Level 1 limits were based on maximum allowable Code stress limits.

3.9.2.1b Preoperational Thermal Vibration and Dynamic Effects of Testing of Piping other than NSSS

The dynamic effects on all safety-related ASME Class 1, 2, and 3 piping systems, including their supports and restraints, are considered as required by NB-3622.3, NC-3622, and ND-3622 of Section III of ASME B&PV Code. The structural and functional integrity of the piping system is ensured under a postulated seismic event by dynamic analysis only. Piping systems having significant anticipated transients loads, e.g., main stop valve closure or relief valve blow for example, are analyzed for the time-dependent forces. In addition, piping steady state vibration and dynamic transient tests were performed as summarized below, to ensure that

a) Excessive steady state vibration is not present in the piping that would result in piping stresses and restraint loads above the allowables.

b) The piping is adequately restrained to withstand the dynamic transient loads.

Cognizant design personnel familiar with the systems to be tested developed the test plans, and evaluate the test results. Also the cognizant design personnel witnessed the test. The data acquired from the tests was compared with the expected results to determine the acceptability of the total system response.

A list of all piping systems in the BOP is provided in Table 3.9-20. ASME Section III Class 1, 2 and 3 piping systems, high energy piping systems, moderate energy piping systems, seismic Category I and seismic Category II systems are identified in the Table. The Table also identifies the tests to be performed for each system.

Piping thermal expansion tests are performed for the safety-related piping systems with normal operating temperature exceeding 300°F. Safety-related piping systems with normal operating temperature less than 300°F do not have enough significant thermal expansion to warrant thermal expansion tests. Engineering review of all seismic Category I piping systems, including their supports, restraints or snubbers, is performed after completion of construction and prior to fuel load to ensure that no restraint of normal thermal movement occurs due to interferences and obstructions, and that the support and restraints are in accordance with the design intent.

For the systems receiving thermal expansion tests, the pipe movements are monitored to ensure that no restraint of normal thermal movement occurs at locations other than at the designed restraint locations.

The thermal expansion test program verifies that the free thermal expansion of piping systems takes place at the snubbers by monitoring the thermal movement. Performance of the snubbers designed for transient loads such as due to Main Stop Valve Closure or Main Steam Relief Valve Discharge are verified by measuring the load in the snubber during the dynamic transient tests. The snubbers are qualified by dynamic testing for cyclic loading as described in Subsection 3.9.3.4.1.

The acceptance criterion for thermal expansion tests and dynamic transient tests is that the measured pipe displacements or restraint loads shall be below the calculated or design values.

The acceptance criterion for the steady-state vibration tests is:

Either

The maximum measured amplitude of the piping vibration shall not induce a stress in the pipe more than half the endurance limit of the material for B31.3 piping. The maximum stress in the pipe due to steady state vibration for class 1, 2 & 3 piping is limited to one-half of the endurance limit (allowable stress corresponding to 10^6 cycles in Appendix I of ASME Section III), the steady-state vibration-induced stress will not contribute to the reduction of fatigue life of piping.

Or

Acceptance criterion are divided into two categories, i.e., Level 1 and Level 2. If the Level 1 criterion is violated, the test must be placed on hold. If the Level 2 criterion is violated, the test can continue, but the measurements must be evaluated to verify that continued test operation will not result in exceeding piping fatigue requirements.

For steady state vibration the piping peak stress zero to peak due to vibration only (neglecting pressure) will not exceed 10,000 psi for the Level 1 criterion and 5,000 psi for the Level 2 criterion. These limits are below the piping material fatigue endurance limits as defined for 10^6 cycles in Appendix I of the ASME Code, Section III.

The Table 3.9-20 provides cross reference between the FSAR Section 3.9 and the appropriate test description in FSAR Chapter 14.

<u>3.9.2.1b.1 Piping Dynamic Transient Tests</u>

During the preoperational and/or startup testing, dynamic transient tests will be performed on the following piping for the indicated modes of operation.

a) Main steam piping outside the containment for main steam turbine stop valve closure at 30 percent ±10%, 75 percent ±10%, and 100 percent +0 -10% power.

b) Main steam bypass piping to the anchor near the bypass valves for the turbine stop valve closure.

c) Selected main steam safety/relief valve discharge piping for the main steam relief valve opening. The selected SRV discharge piping brackets all the SRV discharge piping.

d) HPCI turbine steam supply piping for HPCI turbine trip.

e) Feedwater system reactor feed pump trip/coastdown under operating conditions; Pump A only, B only, and C only and at normal pump flowrate.

From past experience, the dynamic transients in other piping systems are not significant.

Dynamic transient analysis of the subject lines has been performed to determine the response of the piping system and the restraint loads. During the test the displacement of the pipe, loads in the snubbers and restraints and pressure at representative locations will be measured.

Acceptance criteria for this test are that the measured loads in the snubbers and restraints shall be below the design values of the snubbers and restraints. In the case (e) the acceptance criteria is that the measured response shall be less than the acceptable response determined by analysis.

3.9.2.1b.2 Piping Steady State Vibration Testing

The piping system associated with the following components' operation will be observed for steady state vibration during preoperational test programs or power ascension:

a) RHR pump

b) HPCI pump and turbine

c) RCIC pump and turbine

d) Core spray pump

e) Main Steam

f) Feedwater

g) Reactor Water Cleanup

From experience on other nuclear power plants, the steady state vibration in other piping systems is not critical. However, abnormal vibrations of other systems during system walkdown on initial startup or power escalation will be noted and instrumented if necessary to determine the acceptability of such vibration.

Steady-state vibration in the subject piping systems is primarily induced by the flow in the pipe and the equipment motion. In general, the specific causes of the steady-state vibration is not known beforehand; therefore, design engineers with stress analysis experience and familiarity with the subject piping system will visually observe the lines or monitor inaccessible lines with suitable instrumentation during all significant modes of system operation and classify each line as acceptable if the vibration is not significant, or questionable if vibration is significant. The lines with questionable steady-state vibration will be monitored by suitable instrumentation to determine the system response.

The type of the instrumentation, if necessary, will be determined by the design engineer so that the maximum amplitude and frequency response of the piping system can be determined. The instrumentation will not screen out the significant frequencies.

For lines with questionable steady-state vibration, the acceptance criterion is as discussed in Subsection 3.9.2.1b.

When required, additional restraints will be provided to reduce the steady-state vibration and to keep the stresses below the acceptance criterion levels.

3.9.2.2a Seismic and Hydrodynamic Qualification of Safety-Related NSSS Mechanical Equipment

This subsection describes the criteria for dynamic qualification of mechanical safety-related equipment and also describes the qualification testing and/or analysis applicable to this plant for all the major components on a component-by-component basis. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit, for example, motor powered pumps. These modules are generally discussed in this paragraph rather than providing discussion of the separate electrical parts in Sections 3.10 and 3.11. Dynamic qualification testing for pumps and valves is also discussed in Subsection 3.9.3.2. Electrical supporting equipment such as control consoles, cabinets, and panels which are part of the NSSS are discussed in Subsection 3.10.

All safety related NSSS mechanical equipment located in the Containment and the Reactor and Control Buildings are qualified for the combined seismic and hydrodynamic vibratory loadings. Procedures for the assessment and requalification of safety-related NSSS mechanical equipment for the additional hydrodynamic loads are described in Section 7.1.6 of the Design Assessment Report (DAR.)

3.9.2.2a.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety-related function during and after the application of dynamic loads is demonstrated by tests and/or analysis. Selection of testing, or analysis, or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, equipment operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or testing is also used to show there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for hydrodynamic loads. If a natural frequency is discovered, dynamic tests may be conducted and in conjunction with mathematical analysis used to verify operability and structural integrity at the required dynamic input conditions. When the equipment is qualified by dynamic test, either the response spectrum or time history of the attachment point is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Dynamic loading conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) at frequencies through 35 Hz. Whichever method is used, the input amplitude during testing envelopes the actual input amplitude expected during dynamic loading conditions.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a test is used to determine spring constant and operational capability at maximum equivalent dynamic loading conditions. Pipe-mounted equipment is analyzed in the piping system dynamic analysis.

3.9.2.2a.1.1 Test Input Motion

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input, such as sine beats, can be used provided one of the following conditions are met:

a) The characteristics of the required input motion are dominated by one frequency.

b) The anticipated response of the equipment is adequately represented by one mode.

c) The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

3.9.2.2a.1.2 Application of Input Motion

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2a.1.3 Fixture Design

The fixture design will simulate the actual service mounting and cause no dynamic coupling to the equipment.

3.9.2.2a.1.4 Prototype Testing

Equipment testing has been conducted on prototypes of the equipment installed in this plant.

3.9.2.2a.2 Seismic and Hydrodynamic Qualification of Specific Mechanical Components

The following sections discuss the testing or analytical qualification of NSSS equipment. Seismic and hydrodynamic qualification is also described in Subsections 3.9.1.4, 3.9.3.1, and 3.9.3.2.

3.9.2.2a.2.1 Jet Pumps

A dynamic analysis of the jet pumps was performed. The stresses resulting from the analysis are below the design allowables.

3.9.2.2a.2.2 CRD and CRD Housing

The dynamic qualification of the CRD housing (with enclosed CRD) was done analytically. The results of this analysis established the structural integrity of these components. Preliminary dynamic tests have been conducted to verify the operability of the Control Rod Drive during a dynamic event. A test was performed in which the CRD was shown to function satisfactorily, while a static bow in the fuel channels was used to simulate dynamic loading.

3.9.2.2a.2.3 Core Support (Orificed Fuel Support and CR Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and hydrodynamic events has shown that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2a.2.4 Hydraulic Control Unit (HCU)

The seismic and hydrodynamic load adequacy of the HCU for the faulted condition is demonstrated by test and analysis. With the HCU's mounted on a seismic support structure, the dynamic loading results from application of 3.0g vertical at the natural frequency of 7 to 30 Hz, and 1.0g horizontal at 2 to 6 Hz, and 5.0g horizontal at 10 Hz. At these frequencies, the maximum HCU capability demonstrated for dynamic loading is 20g vertical at 7 to 30 Hz, and greater than 4g horizontal at 2 to 6 Hz, and 8g horizontal at 10 Hz.

3.9.2.2a.2.5 Fuel Channels

Fuel channel loading is discussed in Section 3.9.1.4.10.

3.9.2.2a.2.6 Recirculation Pump and Motor Assembly

Calculations are made to determine that the recirculation pump and motor assembly are designed to withstand the specific static to seismic and hydrodynamic forces. The flooded assembly was analyzed as a free body supported by constant support hangers from the brackets on the motor mounting member with mechanical snubbers attached to brackets located on the pump case and the top of the motor frame.

Primary stresses due to horizontal and vertical dynamic forces were considered to act simultaneously and are conservatively added directly. Horizontal and vertical dynamic forces were applied at mass centers and equilibrium reactions determined for motor and pump brackets.

3.9.2.2a.2.7 ECCS Pump and Motor Assembly

Pump/motor assemblies were analyzed with static loading equivalent to seismic acceleration under faulted conditions since the natural frequencies are above 33 Hz. The maximum specified vertical and horizontal accelerations were applied simultaneously in the worst case

combination and the results of the analysis indicate that the pump is capable of sustaining the loadings without overstressing the pump components.

A motor of similar design has been dynamically qualified by a combination of static analysis and dynamic testing. The motor has been seismically qualified via dynamic testing in accordance with IEEE 344, 1975. The qualification test program included demonstration of startup and shutdown capabilities, as well as no load operability during seismic and hydrodynamic loading conditions.

3.9.2.2a.2.8 RCIC Pump Assembly

The barrel type RCIC pump is mounted on a large cross-section pedestal.

The RCIC pump assembly is qualified by analysis using static loading equivalent to seismic and hydrodynamic loading with the design operating loads and temperature. The results of this analysis confirm that the calculated stresses are substantially less than the allowable stresses.

3.9.2.2a.2.9 RCIC Turbine Assembly

The RCIC turbine is qualified by analysis using static loading equivalent to dynamic loading. The turbine assembly and its components were considered to be supported as designed. Horizontal/vertical accelerations were applied to the mass centers of gravity. The magnitude of the acceleration coefficients was 3.0g horizontal and 1.0g vertical. The results of the analysis indicate that the turbine assembly is capable of sustaining the above loadings without overstressing any components.

The turbine assembly is qualified by dynamic testing, in accordance with IEEE 344-1975. The qualification test program demonstrated start-up, steady-state operability, and shutdown capabilities.

3.9.2.2a.2.10 Standby Liquid Control Pump and Motor Assembly

The SLC positive displacement pump and motor are mounted on a common base plate which is qualified by static analysis using static loading equivalent to the dynamic loading with the design operating loads and temperature.

The results of this analysis confirm that the calculated stresses are substantially less than the allowable stresses.

3.9.2.2a.2.11 RHR Heat Exchangers

A three-dimensional finite element model of the RHR heat exchanger and its support was developed and analyzed using the response spectrum method to verify that the heat exchanger can withstand seismic and hydrodynamic loads. The same model was statically analyzed to evaluate the effects of the external piping loads and dead weight to ensure that the nozzle load criteria and stress limits are met. Critical location stresses were evaluated and found to be lower than the corresponding allowable values.

3.9.2.2a.2.12 Safety Relief Valves (SRV)

Three SRVs of the Susquehanna design were subjected to the following qualification test programs in order to demonstrate compliance with the performance requirements under the specified conditions.

1. Life Cycle Tests - These tests consisted of subjecting each of the prequalification production units to approximately 300 safety and relief actuations in order to verify acceptability of the design to meet the requirements for (a) set pressure, (b) opening and closing response time, (c) blowdown, (d) seat tightness, (e) achievement of rated-capacity flow lift (ASME) during each actuation, (f) proper reclosure after each actuation without a tendency to stick open, chatter, or resulting in disc oscillation, and (g) opening without any inlet pressure applied which requirement (h) simulates an emergency operability condition.

   Conditions such as operating temperature, pressure ramp rate, dynamic and static back-pressures, pneumatic operating pressure and solenoid voltage were varied to assure valve operability under normal and transient operating conditions. Upon completion of the tests, test units were disassembled and inspected. This test program established the qualified service life of the safety relief valve.

2. Seismic Tests - The test units were subjected to seismic tests to simulate the normal, upset, emergency, and faulted conditions.

   Post-OBE and post-SSE reference frame tests were performed to determine the operability effects due to repeated combinations of seismic simulations, nozzle loadings, temperature and pressure. These reference frame tests consisted of set pressure determination during safety actuation, response time determination during relief actuation, valve leakage, and an emergency operability test. These reference frame tests were performed with induced nozzle loads applied.

   In order to evaluate the design capability of the test unit, the OBE and SSE tests were repeated using a higher input level. The test conditions during these tests are shown in Table 5.2-3.

   After the seismic tests, the electro-pneumatic actuator assembly was removed from the test unit and subjected to post seismic reference frame tests, a negative pressure test, post-negative pressure reference frame tests, a postulated Loss of Coolant Accident (LOCA) environmental test, and a post-LOCA reference frame test and inspection.

3.9.2.2a.2.13 Standby Liquid Control Tank

The standby liquid control tank is a cylindrical tank 9 feet in diameter and 12 feet high bolted to the concrete floor. The Standby Liquid Control Tank is qualified by analysis for:

a) Stresses in the tank bearing plate

b) Belt stresses

c) Sloshing loads imposed at the natural frequency of sloshing, which is 0.58 Hz

d) Minimum wall thickness

e) Buckling

The results of the analysis confirm that the calculated stresses at the investigated locations are below the allowable stresses.

3.9.2.2a.2.14 Main Steam Isolation Valves

The main steam isolation valves were analyzed; representative models were statically tested to demonstrate operability at the specified faulted condition. Static testing consisted of loading the valve actuator mechanically to equivalent specified dynamic loading while valve closure was performed. Operation of the valve under simulated faulted conditions was demonstrated by this test.

3.9.2.2a.2.15 Main Steam Safety/Relief Valves

Due to the complexity of the SRVs and the performance requirements, the total assembly of the safety/relief valve (including electrical, pneumatic devices) was dynamically tested at accelerations equal to or greater than the combined specified SSE and hydrodynamic loading. Satisfactory operation of the valves was demonstrated during and after the test.

3.9.2.2a.2.16 HPCI Turbine

The HPCI turbine was qualified by static analysis equivalent seismic acceleration. The turbine assembly and its components were considered to be supported as designed, and loading equivalent to horizontal/vertical accelerations was applied to the center of mass. The results of the analysis indicate that the turbine assembly is capable of sustaining the loadings without overstressing any components. The turbine electronic governor assembly has been seismically qualified via dynamic testing, in accordance with IEEE 344-1975. The qualification test program demonstrated startup, steady state operability, and shutdown capabilities.

3.9.2.2a.2.17 HPCI Pump

The HPCI pump is a split body type comprising a booster pump and a main pump mounted on a common base plate. The pump assembly behaves as a rigid body; therefore, qualification by analysis was performed. Results are obtained by using acceleration forces acting simultaneously in two directions, one vertical and one horizontal and calculated using the square root of the sum of the squares method. The pump mass, support system, and accessory piping are shown by analysis to have a natural frequency less than 33 Hz.

3.9.2.2b Seismic Qualification Testing of Safety Related Non-NSSS Mechanical Equipment

All Non-NSSS Seismic Category I equipment has been designed to withstand simultaneously the horizontal and vertical accelerations caused by the OBE and the SSE, in conjunction with other applicable loads. All equipment classified as active have demonstrated through qualification that they will perform their design function before, during and after a design basis accident.

The criteria for the seismic qualification of non-NSSS mechanical and electrical equipment, with the exception of valves, valve operators other than relief valves and the equipment found in the

Diesel Generator 'E' Building, is contained in project specification No. 8856-G-10 for a seismic environment complemented by No. 8856-G-22 for a combined seismic and hydrodynamic environment. For the Diesel Generator 'E' facility, the criteria for seismic qualification of mechanical and electrical equipment is contained in project specification No. C-1041 and Cooper Energy Services Standard No. SD-140. The standard IEEE-344, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations", is referenced in the G-10 and C-1041 Specifications and is being used as a supplement to the G-10, G-22, and C-1041 Specifications in the individual equipment procurement documentation package. Specifications G-10, G-22, and C-1041 and Standard IEEE-344 address the requirements of the demonstration of the seismic adequacy of equipment by analysis and/or tests. NRC Regulatory Guide 1.100 Revision 1, August 1977 accepts the use of standard IEEE-344 with a few modifications. Table 3.9-18 shows the comparison of the specification G-10 with IEEE-344-1975.

Non-NSSS motor-and air-operated valves are addressed in Subsection 3.9.3.2b.2. Control valves are addressed in Section 3.10b.

The assessment and requalification of safety-related non-NSSS mechanical equipment for the additional hydrodynamic loads are described in Section 7.1.7 of the Design Assessment Report (DAR).

3.9.2.2b.1 Safety-Related and Safety-Impacted Mechanical Equipment Other than for the NSSS

3.9.2.2b.1.1 Dynamic Analysis Without Testing

Structural analysis without testing was used if structural integrity alone could ensure the intended design function. Equipment that falls into this category includes:

Safety-Related

a) Diesel oil storage tanks

b) Containment instrument gas accumulators

c) Suppression pool suction strainers

d) Nuclear safety/relief valves

e) Vacuum breakers

Safety-Impacted

f) Supports for air handling units

g) Diesel building supports for cranes

h) Reactor building supports for cranes

i) Fuel pool skimmer surge tanks

Rotational analysis without testing was used to qualify heavy rotating machinery where it had to be verified that deformations from seismic loading would not bind the rotating element so that the component could not perform its intended design function. Components that fall into this category include:

a) Diesel generators

b) Diesel oil transfer pumps

c) RHR service water pumps

d) Emergency service water pumps

e) Control room centrifugal water chiller pumps

Refer to Tables 3.9-16 and 3.9-17 for listings of dynamically qualified equipment.

3.9.2.2b.1.2 Dynamic Testing

The equipment subjected to dynamic testing are the hydrogen recombiners (NSD-E-JFW 1003 March 4, 1975) and rupture discs (Black Sivalls Bryson, January 3, 1977). The rupture discs are installed in the exhaust of the HPCI and RCIC turbines.

3.9.2.2b.2 Criteria

For dynamic analysis without testing the equipment listed under Subsection 3.9.2.2b.1.1, and for dynamically testing the rupture discs under Subsection 3.9.2.2b.1.2, the criteria are as follows.

Response Spectrum Curves

The appropriate response spectrum curves for the equipment in question were issued with the material requisition or the equipment specification, for OBE, SSE, LOCA and SRV (LOCA & SRV only when applicable). Response spectrum curves are based upon the seismic analysis of the supporting structure and represent the maximum seismic response, as a function of oscillator frequency, of an array of single degree of freedom damped oscillators at a particular location within the structure. Response spectrum curves, plotted in terms of acceleration versus frequency, correspond to various locations within the buildings and are identified with respect to the points noted on the mathematical model for each direction of vibration to be considered. This may include the vertical as well as both the north-south and the east-west horizontal directions. In addition, each response spectrum curve corresponds to a particular damping ratio, i.e., the ratio of damping of the single degree of freedom oscillators to critical damping. See Section 3.7 for the appropriate response spectrum curves.

Load Combinations and Allowable Stress Limits

Seismic Category I equipment has been designed to withstand the more severe of the following load combinations:

a) OBE Conditions

Gravity loads and operating loads (or Design Basis Accident loads, if applicable) including associated temperature and pressures combined by absolute sums with the dynamic seismic loading of the OBE. Allowable stresses in the structural steel portions may be increased to 125 percent of the allowable working stress limits as set forth in ASME Boiler and Pressure Vessel Code Section III, or other applicable industrial codes. The customary increase in normal allowable working stress due to an earthquake shall be used if, according to the appropriate code, it is less than 25 percent. Resulting deflections, misalignment or binding of parts, or effects on electrical performance (microphones, contact bounce, etc.) do not prevent operation of the equipment during or after the seismic disturbance.

b) SSE Conditions

Gravity loads and operating loads (or Design Basis Accident loads, if applicable), including associated temperatures and pressures combined by absolute sums with the dynamic seismic loading of the SSE. Allowable stresses in the structural portions may be increased to 150 percent of allowable working stress limits in accordance with the appropriate codes listed in (a); however, the stresses may not exceed 0.9 Fy in bending, 0.85 Fy for axial tension, and 0.5 Fy in shear, where Fy equals the material minimum yield stress at the design temperature. For equipment designed by the maximum shear stress theory, the difference between the maximum and minimum principal stresses will not exceed 0.9 Fy. The resulting deflections, misalignment, or binding of parts, or effects on electrical performance (microphones, contact bounce, etc.) will not prevent operation of the equipment during or after the seismic disturbance.

Prevention of Overturning and Sliding

Stationary equipment is designed to prevent overturning or sliding by using anchor bolts or other suitable mechanical anchoring devices. The effect of friction on the ability to resist sliding is neglected. The effect of upward vertical seismic loads on reducing overturning resistance is considered. Anchoring devices are designed in accordance with the requirements of Items a) and b) and the AISC Manual of Steel Construction. The proposed anchoring system is shown on the Seller's drawings so that the Buyer can provide the proper foundation.

Dynamic Testing

Seismic adequacy was established for the rupture discs by dynamically testing them to meet the criteria defined under a and b above. Actual testing of equipment was done with base connections simulating the actual installation in accordance with one of the following methods:

a) The equipment was subjected to an input excitation such that the measured response was equal to or greater than the specified design response.

b) The equipment was subjected to an input excitation whose response spectrum equaled or exceeded the specified response spectrum for that location.

Criteria for the Diesel Fuel Oil Storage Tanks

These tanks are buried below grade under a cover of 16 ft (8 ft for diesel 'E' fuel tank) of earth. Equivalent fluid pressure of soil is 110 lb/ft^3 (100 lb/ft^3 for diesel 'E' fuel tank).

Tanks and tank supports are designed to withstand an H-20 loading, according to AASHTO, applied above 16 ft (8 ft for diesel 'E' fuel tank) of saturated overburden. The H-20 loading acts simultaneously with normal fluid pressure. Tank walls and ends will not deflect more than 3 percent maximum under the most unfavorable loading conditions.

The diesel fuel oil storage tanks are designed, fabricated, tested, and stamped in accordance with the ASME Code, Section III, Class 3. The tanks, including vents and openings, are designed as underground atmospheric tanks in accordance with OSHA Section 1910.106.

Tanks and their supports are designated Seismic Category I, and are designed to resist the increased earth pressure from the OBE and the SSE. For the OBE, the lateral earth pressure is 90 psf (180 psf for diesel 'E' fuel tank), for the SSE, 180 psf (330 psf for diesel 'E' fuel tank). When combined with other normal operating conditions, the stresses are limited to 125 percent of the ASME Code, Section III allowable stresses for the OBE condition, and are limited to 90 percent of the material's yield stress for the SSE condition.

Tanks are designed to withstand external pressure resulting from being buried in ground having a water table surface at ground level when the tanks are empty. Hydraulic uplift forces on buried tanks are resisted by the weight of the empty tank and the foundation mat plus 16 ft (8 ft for diesel 'E' fuel tank) of saturated overburden.

3.9.2.3 Dynamic Response of Reactor Internals under Operational Flow Transients and Steady State Conditions

The major reactor internal components within the vessel are subjected to extensive testing. In addition, dynamic system analyses are conducted to describe and evaluate the flow-induced vibration phenomena resulting from normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady state conditions are not predetermined by detailed analysis. Special analysis of the response signals measured from reactor internals of many similar designs are performed to obtain the parameters which determine the amplitude and modal contributions in the vibration responses. These studies are useful for extrapolating the results from tests to components of similar design. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

1) Dynamic analysis of major components and subassemblies is performed to identify natural vibration modes and frequencies. The analysis models used for Seismic Category 1 structures are similar to those outlined in subsection 3.7.2.

2) Data from previous plant vibration measurements is assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWRs of differing size and design.

3) Parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates, and structural parameters such as natural frequency and significant dimensions.

4) Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode.

5) Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions, based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of paragraph 1 above.

The dynamic modal analysis also forms the basis for interpretation of the prototype plant preoperational and initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component

3.9.2.4 Confirmatory Flow-Induced Vibration Testing of Reactor Internals

Reactor internals were tested in accordance with provisions of Regulatory Guide 1.20, Revision 2, for Non-prototype Category I plants. The test procedure required operation of the recirculation system at rated flow with internals important to safety installed. Inspection for evidence of vibration, wear, or loose parts followed. Blade guides, incore instruments, neutron sources, dryer and fuel were not installed. Control rods were either not installed or fully withdrawn and prevented from inserting. The test duration was sufficient to subject critical components to at least 10^6 cycles of vibration during two-loop and single-loop operation of the recirculation system. At the completion of the flow test, the vessel head and shroud head were removed, the vessel was drained and major components will be inspected on a selected basis. The inspection covered all components which were examined on the prototype design, including the shroud, shroud head, core support structures, jet pumps, peripheral control rod drive guide tubes and peripheral in-core guide tubes. Access will be provided to the reactor lower plenum.

Reactor internals for Susquehanna are substantially the same as the internals design configurations that have been tested in prototype BWR/4 plants. Results of the prototype tests are presented in a Licensing Topical Report (Ref. 3.9-7). This report also contains additional information on the confirmatory inspection program.

A labyrinth seal, consisting of five circumferential grooves on each jet pump mixer at the slip joint interface with the jet pump diffuser collar, reduces leakage at the slip joint. Tests performed by General Electric Company (Reference 3.9-10) demonstrated that the labyrinth seals reduce leakage through the slip joints. However, PPL no longer credits the labyrinth seals with this function. Each jet pump is equipped with a slip joint clamp to reduce jet pump vibration.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison was made of the periods of the applied

forces and the natural periods of the core support structures being acted upon by the applied forces. These periods were determined from a 12-node vertical dynamic model of the RPV and internals. Besides the real masses of the RPV and core support structures, account was made for the water inside the RPV.

The time-varying pressures are applied to the dynamic model of the reactor internals described above. Except for the dynamic model and the nature and locations of the forcing functions, the dynamic analysis method is identical to that described for seismic analysis and is detailed in Subsection 3.7.2.1.

Dynamic loads are combined by SRSS. The results are then combined with other static and steady state loads on an ABS basis to confirm the adequacy of design loads. The results of the dynamic analysis are summarized in Tables 3.9-2, 3.9-2b, 3.9-2w, and 3.9-2aa.

3.9.2.6 Correlations of Reactor Internals Vibration Test Results with the Analytical Results

Prior to initiation of the instrumented vibration test program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test are analyzed in detail. The results of the data analysis, vibration amplitudes, natural frequencies and mode shapes, are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained is utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are the same as those used for the vibration analysis of the prototype plant.

The vibration test data are supplemented by data from forced oscillation tests of reactor internal components to provide the analysis with additional information concerning the dynamic behavior of the reactor internals.

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

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and hydrodynamic events for the design of safety-related ASME code components (except containment components, which are discussed in Section 3.8.)

This section also lists the major ASME Class 1, 2, and 3 equipment and associated pressure retaining parts on a component by component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Design transients for ASME Class 2 equipment are not addressed in this section. They are covered in Subsection 3.9.1.1. Seismic and hydrodynamic related loads are discussed in Subsections 3.9.2.2a, 3.9.2.2b and Section 3.7.

Table 3.9-2 is the major part of this section; it presents the loading combination, analytical methods (by reference or example) and also the calculated stress or other design values for the

most critical areas of the ASME Code Class 1, 2 and 3 components, supports, and core support structures. These design values are also compared to applicable code allowables.

3.9.3.1.1 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in the ASME Boiler and Pressure Vessel Code, Section III.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of System startup, operation in the design power range, normal hot standby (with main condenser available), and System shutdown other than Upset, Emergency, Faulted, or Testing.

3.9.3.1.1.2 Upset Condition

Any deviations from Normal Conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The Upset Conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power, or an operating basis earthquake. Hot standby with the main condenser isolated is an Upset Condition.

3.9.3.1.1.3 Emergency Condition

Those deviations from Normal Conditions which require shutdown for correction of the conditions or repair of damage in the RCPB. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, transients caused by one of the following: a multiple valve blowdown of the reactor vessel; loss of reactor coolant from a small break or crack which does not depressurize the reactor system nor result in leakage beyond normal makeup system capacity, but which requires the safety functions of isolation of containment, and reactor shutdown; improper assembly of the core during refueling; and seizure of one recirculation pump.

3.9.3.1.1.4 Faulted Condition

Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated events are the most drastic that must be designed against and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: a control rod drop accident, a fuel-handling accident, a main steam line break, a recirculation loop break, the combination of any pipe break plus the seismic motion associated with SSE and hydrodynamic loading plus a loss of offsite power, or the safe shutdown earthquake.

3.9.3.1.1.5 Correlation of Plant Conditions with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation can be used to identify the appropriate plant condition for any hypothesized event or sequence of events.

Plant Conditions	Event Encountered Probability per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	1.0 > p > 10-2
Emergency (low probability)	10-2 > p > 10-4
Faulted (extremely low probability)	10-4 > p > 10-6

3.9.3.1.1.6 Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could impair its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment shall be capable of accomplishing its safety functions as required by the event but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

Functional capability of safety-related essential piping components will be assured using the criteria given in Enclosure 110-1 to NRC questions and the Rodabaugh criteria.

3.9.3.1.1.7 Compliance with Regulatory Guide 1.48

Regulatory Guide 1.48 was issued after the design of this plant was established. Compliance with this Regulatory Guide is addressed in Section 3.13.

3.9.3.1.2 Reactor Vessel Assembly

The reactor vessel assembly consists of the reactor pressure vessel support skirt, shroud support and shroud plate.

The reactor pressure vessel, vessel support skirt, and shroud support are constructed in accordance with Section III of the ASME Code. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The reactor pressure vessel is an ASME Code Class I component. Complete stress reports on these components have been prepared in accordance with ASME requirements. Table 3.9-2a provides a summary of the stress criteria, load combinations, calculated and allowable stresses. The stress analysis performed for the reactor vessel assembly, including the faulted condition, were completed

using elastic methods. The stress Load combinations and stress analyses for the core support structures and other reactor internals are discussed in Subsection 3.9.5.

3.9.3.1.3 Main Steam Piping

The main steam piping discussed in this paragraph includes that piping extending from the reactor pressure vessel to the outboard main steam isolation valve. This piping is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3600. The load combinations and stress criteria for the main steam piping and pipe-mounted equipment are shown in Table 3.9-2d.

The rules contained in Appendix F of ASME Code Section III will be used in evaluating faulted loading conditions independently of all other design and operating conditions. Stresses calculated on an elastic basis will be evaluated in accordance with F-1360.

3.9.3.1.4 Recirculation Loop Piping

This piping is designed in accordance with the ASME for the recirculation piping and pipe-mounted equipment Code Section III, Subsection NB-3600. The load combinations and allowables are shown in Table 3.9-2e. The rules contained in Appendix F of ASME Code Section III are used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses calculated on an elastic basis are evaluated in accordance with F-1360.

3.9.3.1.5 Recirculation System Valves

The recirculation system valves are designed in accordance with the ASME Code, Section III, Class I, Subsection NB-3500. The discharge gate valves are required to close for LPCI flow injection. Loading combinations and other stress analysis information are presented in Table 3.9-2(j).

3.9.3.1.6 Recirculation Pump

In the design of the recirculation pumps, the ASME Code, Section VIII, Division 1, 1971 Edition with latest addenda was used as a guide in calculations made for determining the thickness of pressure-retaining parts, and in sizing the pressure-retaining bolting.

The pump vendor made calculations for the design of the pressure-containing components to include the determination of minimum wall thickness, allowable stress and pressures. The design calculations are shown in Table 3.9-2i.

Load, shear, and moment diagrams were constructed to scale, using live loads, dead loads, and calculated snubber reactions. Combined bending, tension and shear stresses were determined for each major component of the assembly, including the pump driver mount, motor flange bolting, and pump case.

Replacement pump cover gaskets have been upgraded by the pump Original Equipment Manufacturer (OEM) to eliminate the use of asbestos and to improve reliability. The replacement pump cover gaskets require a higher bolt preload than the original gaskets. The OEM prepared a Gasket Upgrade Design Report that concludes the pump subcomponents are acceptable and meet the ASME Code requirements.

The maximum combined tensile stress in the cover bolting was calculated using tensile stress from design pressure.

Combined primary stresses did not exceed 150 percent of the code allowable stress shown in Section VIII of the ASME Code, 1971 Edition.

These methods and calculations demonstrate that the pump will maintain pressure integrity at all times.

3.9.3.1.7 Standby Liquid Control (SLC) Tank

The SLC tank is designed in accordance with the ASME Code, Section III. A summary of the design calculations and stress criteria used are shown in Table 3.9-2m.

3.9.3.1.8 Residual Heat Removal Heat Exchangers

The RHR heat exchanger is designed in accordance with the ASME B&PV Code, Section III. The loading combinations considered and other stress analysis are presented in Table 3.9-2o.

3.9.3.1.9 RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine has been designed and fabricated following the basic guidelines for an ASME Code Section III, Class 2 component.

The RCIC Turbine is surveillance tested according to current Technical Specifications.

Design conditions for the RCIC turbine include:

a) Auto Quick start per Technical Specification surveillance requirements.

b) Turbine Inlet - 1250 psig at saturated temperature

c) Turbine Exhaust - 165 psig at saturated temperature

Table 3.9-9 contains a summary of the RCIC turbine components calculated and allowable loads.

3.9.3.1.10 RCIC Pump

The RCIC pump has been designed and fabricated to the requirements for an ASME Code Section III Class 2 component.

The RCIC pump is surveillance tested in conjunction with the RCIC turbine. Surveillance testing is performed according to current Technical Specification surveillance requirements. Design conditions for the RCIC pump include:

a)	Maximum NPSHR -		21.3 feet
b)	Total head, maximum	High speed	3060 feet
		Low speed	525 feet

c) Constant flow rate: 625 gpm

d) Normal ambient operating temperature - 60°F to 100°F

e) Normal/Upset conditions which control the pump design include:

Design pressure - 1500 psig

Design temperature - 40°F - 140°F

Seismic loads - 2/3 of SSE

Table 3.9-2r contains a summary of the design calculation for the RCIC pump components.

3.9.3.1.11 ECCS Pumps

The RHR, CS and HPCI pumps are designed in accordance with the ASME Code, Section III. The stress criteria and calculated and allowable stresses are summarized in Table 3.9-2n.

Design condition for RHR and core spray pumps are as follows:

	RHR	CORE SPRAY
Design pressure		
Suction	220 psig	125 psig
Discharge	500 psig	500 psig
Design Temperature	40-360°F	40-212°F

3.9.3.1.12 Standby Liquid Control Pump

The standby liquid control pump has been designed and fabricated to the 1968 P&V Code for Class 2 component.

The SLC pumps and motors are functionally tested by pumping demineralized water through a closed test loop. The SLC pumps are capable of injecting the net contents of the storage tank into the reactor in less than an hour. The pumps are capable of injecting flow into the reactor against pressure up to the second lowest spring set pressure (1195) of the reactor safety relief valves.

Design conditions for the SLC pump include:

a) Flow rate 43 gpm

b) Maximum operating discharge pressure 1250 psig

c) Ambient conditions:

Temperature	70°F - 120°F
Relative Humidity	20% - 95%

d) Normal/upset conditions which control the pump design include:

Design pressure	1500 psig
Design temperature	150°F
Seismic Loads	2/3 of SSE

Stress limits for the pressure boundary are the ASME Code allowable stress (1.0S) for general membrane

e) Faulted or emergency conditions include:

Design pressure	1500 psig
Design temperature	150°F
Safe shutdown earthquake	Horizontal 1.5g Vertical 0.14g

A summary of the design calculations for the SLC pump components is contained in Table 3.9-2l.

3.9.3.1.13 Main Steam Isolation Valves and Safety/Relief Valves

The main steam isolation and safety relief valves are designed in accordance with the requirements of the ASME Code Section III, Subsection NB-3500 for Class 1 components.

Load combination, analytical methods, calculated stresses, and allowable limits are shown in Table 3.9-2g and 3.9-2(h), respectively.

3.9.3.1.14 Safety Relief Valve Piping

See Subsection 3.9.3.1.19.

3.9.3.1.15 High Pressure Coolant Injection (HPCI) Turbine

Although not under the jurisdiction of the ASME Code, the HPCI turbine has been designed and fabricated to the basic guidelines for an ASME Code Section III as a Class 2 component.

Surveillance testing is performed according to current Technical Specification surveillance requirements.

Design conditions for the HPCI turbine include:

a) Auto-Startup -- 30 cycles per year with reactor pressure at 1150 psig peak and saturated temperature, turbine exhaust pressure at 50 psig peak and saturated temperature.

b) Turbine Inlet - 1250 psig at saturated temperature.

c) Turbine Exhaust - 200 psig at saturated temperature

d) Upset conditions, which control the turbine design include:

Design pressure

Design temperature

Operating basis earthquake

Inlet and exhaust piping nozzle loads

Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane and 1.5S for bending (local membrane).

e) Faulted, or emergency conditions include:

Design pressure

Design temperature

Safe shutdown earthquake

Inlet and exhaust piping nozzle loads

Stress limits for pressure boundary are 120% of ASME Code Section III allowable stress (1.2S) for general membrane and 1.8S for bending (local membrane).

f) Nozzle loading definition includes:

Upset	-	Inlet F	=	(20,000 - M)/2.5, but < 5,000 lbs.
	-	Exhaust F	=	(20,000 - M)/0.8, but < 11,500 lbs.
Faulted (or - Emergency)		Inlet F	=	(30,000 - M)/2.5, but < 7,500 lbs.
		Exhaust F	=	(30,000 - M)/0.8, but < 17,250 lbs.

Where F (lbs) and M (ft-lb) are the resultant force and moment on the respective nozzle.

A summary of the design calculations for the HPCI turbine components are shown in Table 3.9-2(ae).

3.9.3.1.16 High Pressure Coolant Injection (HPCI) Pump

The HPCI pump has been designed and fabricated to the requirements for an ASME Code Section III Class 2 component.

The HPCI pump is surveillance tested in conjunction with the HPCI turbine. The HPCI pump is surveillance tested according to the current Technical Specification surveillance requirements. The HPCI pump takes condensate from the above-ground storage tank and at design flow discharges condensate back to the above-ground storage tank via a closed test loop.

Design conditions for HPCI pump include:

a) Total head , maximum- High speed: 3060 feet

Low speed: 525 feet

b) Constant flow rate - 5000 gpm

c) Normal ambient operating temperature - 60°F to 100°F

d) Normal plus Upset conditions which control the pump design include:

Design pressure	-	1500 psig
Design temperature	-	40°F - 140°F
Seismic Loads	-	2/3 of SSE
Suction nozzle loads	-	F = 1940 lb M = 2460 ft-lbs
Discharge nozzle loads	-	F = 3715 lbs M = 4330 ft-lbs

Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane and 1.5S for bending (local membrane).

e) Faulted, or Emergency conditions include:

Design pressure	-	1500 psig
Design temperature	-	40°F - 140°F
Safe shutdown earthquake	-	Horizontal - 1.50g Vertical - 0.14g

Stress limits for pressure boundary are 120% of ASME Code allowable stress (1.2S) for general membrane and 1.8S for bending (local membrane).

f) Nozzle loading

Pump nozzles are subject to loading from the connecting pipe. The method of analysis shows the maximum resultant moment is due to pipe reaction. The maximum resultant force shall not exceed the allowable. Allowable nozzle forces and moments are expressed as:

Normal plus Upset

Suction	- F = 33,000-0.79M
Discharge	- F = 32,000-1.54M

Emergency:

Suction	- F = 43,000-0.74M
Discharge	- F = 47,000-1.23M

The calculated stress values are compared to allowable stresses for critical components in Table 3.9-2t.

3.9.3.1.17 Reactor Water Cleanup (RWCU) System

The RWCU pump and heat exchangers are not part of a safety system and are not designed to Seismic Category I requirements.

However, the requirements for ASME Code Section III, Class 3 components are used as guidelines in evaluating the RWCU system components.

The design loading combinations and limits for the pump include the following:

a) Normal plus upset loads: This includes the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads from connected piping, dead weight loads, seismic loads, plus torsional loads due to rotating parts.

b) Seismic loading: This equipment and supports are designed to withstand the static seismic forces applied at the mass center, assuming that the pump is flooded.

c) Stresses in the supports and the anchor bolts due to seismic loads are combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the ASME Code Section III.

d) Equipment operates between 70°F and 532.3°F. Transient analysis is not required for Class III components in this temperature range.

Tables 3.9-2(p) and 3.9-2(c) show the calculated stress values and allowable stress limits for the pump and heat exchangers, respectively.

3.9.3.1.18 This Section Has Been Intentionally Deleted

3.9.3.1.19 ASME Code Constructed Items Not Furnished with the NSSS

The design loading combinations categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted for ASME code constructed items are presented in Table 3.9-6.

The method of combining the peak loads on components and supports resulting from different dynamic events was addressed by the Mark II Owners Subgroup on SRSS. The generic resolution has been reviewed and applies to Susquehanna SES.

The design criteria and stress limits associated with each of the plant operating conditions for each type of ASME code constructed item are presented in Tables 3.9-7, 3.9-8, 3.9-9, 3.9-10, 3.9-11 and 3.9-12.

The component operating condition will be the same as the plant operating condition, except for active pumps or valves for which, the emergency or faulted plant condition is considered a normal operating condition.

3.9.3.2a NSSS Pump and Valve Operability Assurance

The active NSSS pumps and valves are listed in Table 3.9-3.

Active mechanical equipment classified as Seismic Category I are designed to perform their function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include "active" (active equipment must perform a mechanical motion during the course of accomplishing a safety function) pumps and valves in fluid systems such as the emergency core cooling system. Operability is assured by satisfying the requirements of the following programs. Safety-related valves are qualified by prototype testing and analysis satisfying stress and deformation criteria at all critical locations and safety-related active pumps by analysis with suitable stress limits and nozzle loads. The content of these programs is detailed below.

3.9.3.2a.1 ECCS Pumps

All active pumps are qualified for operability by first being subjected to rigid tests before installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts to 125% of the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, (2) seal leakage tests, (3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, Net Positive Suction Head (NPSH) requirements and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures (except water cooled bearings) and vibration levels. Both are shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic in-service inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the safety-related active pumps are analyzed for operability during a faulted condition by assuring that (1) the pump will not be damaged during the seismic and hydrodynamic event, and (2) the pump will continue operating despite the faulted loads.

3.9.3.2a.1.1 Analysis of Loading, Stress, and Acceleration Conditions

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the material elastic limit, as indicated in Table 3.9-2. A three-dimensional finite element model of the pump/motor and its supports is developed using the response spectrum method of dynamic analysis. The same model is analyzed for static nozzle loads, pump thrust loads, and deadweight. Critical location stresses are evaluated and compared with the allowable criteria. The average membrane stress σ_m) for the faulted conditions loads are maintained at 1.2S, or approximately 0.75 σ_y(σ_y - yield stress), and the maximum stress in local fibers (σ_m + bending stress o) is limited to 1.8S, or approximately 1.1 σ_y. The maximum dynamic nozzle loads are considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9-2 as allowables, will assure that critical parts of the pump will not be damaged during the faulted condition; therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic and hydrodynamic events.

A dynamic analysis is made to determine the seismic load from the applicable floor response spectra. Analysis is made to check that faulted condition nozzle loads and dynamic accelerations will not impair the operability of the pumps during or following the faulted condition. Components of the pump, when having a natural frequency above 33 Hz, are considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas.

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations will be determined using the same conservatisms contained in the horizontal and vertical accelerations used for "rigid" structures. The static analysis is performed using the adjusted accelerations; the stress limits stated in Table-3.9-2 must still be satisfied.

3.9.3.2a.1.2 Pump Operation During and Following Faulted Condition Loading

Active pump/motor rotor combinations are designed to rotate at a constant speed under all design conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic and hydrodynamic event, will prevent the rotor from becoming seized. In actuality, the dynamic loadings will cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore the pump will not shutdown during the faulted event and will operate at the design speed despite the faulted loads.

The functional ability of the active pumps after a faulted condition is assured since only normal operating loads and steady state nozzle loads exist. For the active pumps, the faulted condition is more severe than the normal condition only due to seismic and hydrodynamic loads on the equipment itself and the increase in nozzle loads due to the SSE on the connecting pipe. The SSE event is infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse then the normal plant

operating limits. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions be limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2a.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies, with natural frequencies well above 33 Hz. With this fact verified, each equipment assembly is qualified by static analysis only. This static qualification verifies operability under seismic and hydrodynamic conditions, and assures structural loading stresses within Code limitations.

3.9.3.2a.3 RCIC Turbine

Analysis and testing done on the RCIC turbine is covered by Subsections 3.9.2.2, 3.9.3.1, and Table 3.9-2q.

3.9.3.2a.4 ECCS Motors

Qualification of the Class 1E motors used for the ECCS motors is in compliance with IEEE Standard 323-1971 or 1974. The qualification of motors of all sizes is based on completion of a type test, followed up with review and comparison of design and material details and seismic analysis of production units, ranging from 500 to 3500 Bhp. The motor is used in the type test. All manufacturing, inspection, and routine tests performed by motor manufacturers on production units are performed on the test motor.

The type test has been performed on a 1250 HP vertical motor in accordance with IEEE Standard 323-1971. Normal operation during the design life is first simulated, then the motor is subjected to a number of seismic events. Then the abnormal environmental condition possible during and after a loss of coolant accident (LOCA) is simulated. The test plan for the type test was as follows:

a) Thermal aging of the motor electrical insulation system (which is a part of the stator only) was based on IEEE Standard 275-1966. The amount of aging equaled the total estimated operating days at maximum insulation surface temperature.

b) Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma radiation during normal and abnormal conditions.

c) The normal operating inducted vibration effect on the insulation system has been simulated by 1.5g horizontal vibration acceleration of 60 Hz current frequency for one hour duration.

d) Motor bearings are selected and their operating life is established based on bearing manufacturer's test and operating data using the loads calculated to act on the bearing.

e) The dynamic load deflection analysis on the rotor shaft, performed to ensure adequate rotation clearance, has been verified by static loading and deflection of the rotor for the type-test motor.

f) Dynamic load aging and testing has been performed on a biaxial test table in accordance with IEEE 344-1975. During this type test the shake table was activated simulating the maximum design limit of the safe shutdown earthquake and hydrodynamic loads with motor starts and operation combination as may possibly occur during a plant life.

g) An environmental test simulating a LOCA condition with 100 days duration time has been performed with the test motor fully loaded to simulate pump operation. The test consisted of startup and six hours operation at 212°F ambient temperature and 100% steam environment. Another startup and operation of the test motor after one hour stand-still in the same environment was followed by sufficient operation at high humidity and temperature. The operation was based on the temperature-life characteristic curve from IEEE 275-1966 for the insulation type used on the ECCS motors.

3.9.3.2a.5 NSSS Valves

The Class 1 Active Valves are the Main Steam Isolation Valves, Safety/Relief Valves, Recirculation Discharge and Bypass Gate Valves, Standby Liquid Control Valves and Control Rod Drive Scram Discharge Volume Vent and Drain Valves. Each of these valves is dynamically qualified for operability in a manner unique to its design. Therefore, each method of qualification is detailed individually below.

3.9.3.2a.5.1 Main Steam Isolation Valve

The MSIV's are evaluated for operability during dynamic acceleration by both analysis and test. This analysis for MSIV operability is completed in two separate ways. First the valve body is designed in accordance with the ASME Code Section III Class 1 which limits deformation to be within the elastic limit of the material by limiting pressure and pipe reaction input loads (including seismic and hydrodynamic loads). This assures only small deformation in the operating area of the valve body, hence, there is no interference with valve operability.

A static deflection test was conducted on a MSIV of similar design to assure operability under maximum deformation from seismic loading. A maximum static load equivalent to 8 g's applied perpendicular to the actuator axis centerline resulted in no significant change in valve closure rate and no change in measured seat leakage following termination of the load.

To assure that design limits are not exceeded for both piping input loads and actuator dynamic loads, the MSIV is mathematically modeled in the Main Steam Line System Analysis. The valve input loads, amplified accelerations, and resonance frequencies are determined based on site excitation input to the system for MSIV as a part of the overall steamline analysis. Pipe anchors and restraints are applied as required to limit pipe system resonance frequencies and amplified accelerations to within acceptable limits for the MSIV's. Additional details on the analysis of these valves is shown on Table 3.9-2(h).

The main steam isolation valve operability during LOCA conditions was demonstrated as defined in the report APED-5750 (March 1969). The test specimen was a 20" valve of a design representative of the MSIV's.

3.9.3.2a.5.2 Main Steam Safety/Relief Valves

The safety/relief valves are qualified for operability during seismic and hydrodynamic events by both design and test.

The valve is designed for the largest moments that can occur in service. These are 400,000 in-lbs and 300,000 in-lbs at the inlet and outlet, respectively. These moments are resultants due to dead weight plus dynamic seismic effects of 3 g's horizontal and 1 g vertical of both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge. A production S/R valve demonstrated operability while being dynamically (shake table) tested at loads greater than equipment design limit loads.

A mathematical model of this valve is included in the main steam line system analysis. This analysis assures the design limits are not exceeded.

The safety relief valves are generically qualified by testing for seismic and hydrodynamic loads. The natural frequencies are determined to be greater than 33 Hz for seismic and 60 Hz for hydrodynamic loads.

Additional details on the analysis of these valves are shown in Table 3.9-2g.

3.9.3.2a.5.3 Recirculation (Discharge and Bypass) Gate Valves

Recirculation discharge and bypass gate valves are evaluated for operability during seismic and hydrodynamic events by both analysis and test.

Motor operators were generically qualified to IEEE 382-1980 which requires a dynamic test to verify the absence of any natural frequencies below 33 Hz and then a demonstration of operability during dynamic testing. The operators have been qualified to acceleration levels of 10 g from 2 Hz to 100 Hz.

The valve are designed in accordance with the ASME Code, Section III Class 1 design rules. The discharge valves are designed to seismic accelerations of 9.8 g’s horizontal and 2.188 g’s vertical including gravity. Both valves’ extended structures were analyzed to show that they could withstand both compressive stresses and bending stresses imposed by the seismic accelerations. The valve fundamental frequencies were determined by frequency analysis to be less than the seismic cut-off frequency of 33 Hz. This required dynamic analysis considering multinode response. However, since the valves are pipe-mounted and the required response spectra at the valve location were not available, it was necessary to perform a dynamic analysis on the entire piping system. A simple lumped-mass model of the valve and its actuator was developed based on the valve fundamental frequency, and was used to represent the valve dynamic characteristics in the piping analysis.

Dynamic piping analysis indicated that the loads imposed on these valves are less than the design allowable loads. Additional details on analysis of the discharge valves are shown in Table 3.9-2(j).

3.9.3.2a.5.4 Explosive Valves

The SLC explosive valve has been qualified to IEEE 344-1975. The qualification test included a demonstration of the absence of natural frequencies below 35 Hz and the ability to remain operable under a horizontal seismic coefficient of 6.5g and a vertical seismic coefficient of 4.5g at 33 Hz.

3.9.3.2a.5.5 CRD Scram Discharge Volume Vent and Drain Valves

The CRD Scram Discharge Vent and Drain Valves are evaluated for operability during dynamic acceleration by test. The testing consisted of a combination of vibration aging testing. SRV cycling induced fatigue load testing and seismic testing at acceleration levels based upon both upset and faulted required response spectra (RRS). The valve successfully passed all qualification testing.

3.9.3.2a.6 HPCI Turbine

The HPCI turbine is dynamically qualified by static analysis. The turbine assembly and its components were considered to be supported as designed, and horizontal/vertical accelerations were applied to the mass's center of gravity. The magnitude of the acceleration coefficients was 1.50 horizontal and 0.48 vertical. The results of the analysis indicate that the turbine assembly is capable of sustaining the above loadings without overstressing any component.

The turbine was dynamically qualified via dynamic testing by the 1st quarter 1983 in accordance with IEEE 344-1975. The qualification test program demonstrated start-up, steady state operability, and shutdown capabilities.

3.9.3.2b Non-NSSS Pump and Valve Operability Assurance

The pumps under this category are:

a) Diesel oil transfer pumps

b) RHR service water pumps

c) Emergency service water pumps

d) Control structure chiller - cooling water pumps.

All the above pumps are Class 3.

3.9.3.2b.1 Pumps

The pumps listed above are subjected to testing both in the manufacturer's shop and following their installation to verify that they meet the criteria required by the respective specifications.

During manufacture, nondestructive test procedures including liquid penetrant examination, radiographic examination, magnetic particle inspection, and ultrasonic inspection are applied to the pumps. All these procedures are performed in accordance with ASME Code, Section III.

After the pumps have been assembled they are hydrostatically tested and performance tested in the manufacturer's shop in accordance with the Hydraulic Institute's standards.

After the pumps are installed they will undergo functional tests. Provisions will be made for in-service inspection and operational testing.

All these tests demonstrate that the pumps are reliable and will function as specified.

3.9.3.2b.1.1 Analysis of Loading, Stress, and Acceleration Conditions

In addition to the tests and procedures referred to above, the pumps are seismically analyzed to ensure that they will be capable of operating both during and after the OBE and DBE.

In performing these analyses, conservative seismic accelerations and stress criteria were used; this ensures that critical parts of the pump are not damaged during a seismic event and that the pump can still operate following such an event.

3.9.3.2b.1.2 Pump Operation During and Following SSE Loading

Each pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., with no rotation. Motors are designed to withstand short periods of severe overload and, typically, the rotor can be seized five full seconds before a circuit breaker shuts down the pumps. However, the high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, will prevent the rotor from becoming seized. In actuality, the seismic loadings will cause only a slight increase in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump will not shut down during the event and will operate at the design speed despite the seismic loads.

From previous discussions, it is evident that the pump/motor units will withstand seismic loadings and, therefore, will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and ensure the seismic operability of these pumps. Post-seismic condition operating loads will be no worse than the normal plant operating limits.

3.9.3.2b.2 Valves

Active ASME Class 1, 2, and 3 valves are identified in the Plant’s ISI program manual.

Safety related active valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test in accordance with ASME Code Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, functional tests which verify that the valve will open and close within the specified time limits, and operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation, etc). in accordance with IEEE 382-1972. After installation, cold hydrostatic construction tests, functional tests in accordance with the requirements of Chapter 14, and periodic in-service operation in accordance with the requirements of Chapter 16, are performed to verify and ensure the functional ability of the valve.

The valves are designed using either stress analyses or the pressure-containing minimum wall thickness requirements. On all active valves with extended topworks, an analysis is also performed for static equivalent SSE loads applied at the center of gravity of the extended structure. The maximum stress limits allowed in the analyses demonstrate structural integrity and are equal to the limits recommended by the ASME Code for the particular ASME class of valve analyzed as listed in Tables 3.9-7 and 3.9-12. Loading combinations are listed in Tables 3.9-6 and 3.9-14. In addition to these tests and analyses, a representative valve of each design type is tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits.

A. Selection of Representative Valve

The valves requiring operability qualification are divided into different groups: by valve manufacturer, valve type, size, pressure class, material type (carbon steel, stainless steel, and alloy steel) and actuator type (AC electric, DC electric, air, hydraulic, etc.) Valve sizes that cover the range of sizes in service are qualified as shown in Table 3.9-15 by tests, and the results are used to qualify all valves within the intermediate range of sizes. A tabulation is made of the weight of the valve actuator, the actuator thrust margin (a ratio of the maximum thrust available from the actuator divided by the design thrust required for the valve), and the yoke configuration (as related to stiffness) for each valve assembly. For a range of qualified valve sizes, as defined by the qualification table, the valve assembly with the heaviest actuator, lowest thrust margin, and least stiff yoke is picked as the test unit. In those cases where a test unit is not readily apparent, more than one unit is tested to provide a conservative test position. This procedure is repeated within each group until all listed units are represented by a test unit, and for each group until all the necessary valves are represented by a test unit.

In addition to the tests, the stress calculations for each valve assembly are reviewed. A tabulation is made for all qualified valve assemblies comparing the yoke stress for all valve classes, the yoke-flange to body and the yoke-flange to actuator-bolting stresses, as applicable, for all classes of valves, and the body stress for Class 1 valves. This is done to provide further analytical justification for the qualification of non-tested valves by tested valves.

B. Qualification Testing Procedures

The valve is mounted in a manner that will conservatively represent typical valve installations. The valve unit includes the actuator and all appurtenances usually attached to the valve in service. The operability of the valve during a SSE is demonstrated by satisfying the following criteria:

a) All the active valves with topworks are designed to have a first natural frequency greater than 33 Hz. This is shown by suitable test or analysis.

b) The extended topworks of the valve are subjected to a statically applied equivalent seismic load. Load is applied at the center of gravity of the topworks in the direction of the weakest axis of the yoke. The design pressure of the valve or the design pressure of the system is simultaneously applied to the valve during the static load tests.

c) The valve is then operated at minimum specified actuation supply voltage or pressure with equivalent seismic static load applied. The valve must perform its safety-related function within the specified operating time limits.

d) Motor operators and other electrical appurtenances necessary for operation are qualified as operable during the SSE in accordance IEEE-344-1975 prior to their installation on the valve.

For valves with and without the topworks supported, the statically applied load envelopes the specified G-force times the weight of the topworks. **This load is generally greater than would result from 3.0 g horizontal and 3.0 g vertical**. For valves with the topworks supported, additional loading from thermal and/or anchor movements may be imposed upon the valve through the support(s). If the loads due to thermal and/or anchor movements are greater than the statically applied load (which envelopes inertial forces), **stress and critical deflection analyses are** performed on the valve (considering the maximum applied loading) **as an acceptable qualification alternative.**

An exception to the above described seismic qualification approach is the RHR throttle valves (HV151F017A/B and HV251F017A/B) which were not tested with a static seismic load but instead were qualified by a combination of static seismic analysis and static deflection operability analysis. The static deflection operability analysis verified that adequate internal clearances exist to insure the binding does not occur within the valve during and after a design basis event.

The piping designer limits the valve accelerations and support loads to allowable values as determined by the qualification test and analysis.

The valve is leak tested following the test described above to show that the valve has not been damaged. The leak rates must not exceed the original allowable leakage rate specified for the valve.

The above testing program applies only to valves with overhanging structures, i.e., the motor operator or air actuator assembly. Because of their simple characteristics, check and other compact valves are not affected by seismic acceleration. Check valves have no extended structures to distort the valve and cause a malfunction. Check valve discs are designed to allow sufficient clearance around the disc to prevent distortions from nozzle or other imposed loads. Accordingly, check valves are qualified by a combination of the following tests and analysis:

a) The air-operated check valves are analyzed to ensure that the air cylinder cannot impair the ability of the valve to operate as a simple check valve during seismic loading. No functional test simulating seismic loading is performed. Air operators on check valves do not perform a safety function.

b) In-shop hydrostatic test

c) In-shop seat leakage test

d) Periodic valve exercise and inspection to ensure the functional ability of the valve in accordance with the requirements of Chapter 16.

Using the methods described, all the safety-related active valves in the systems are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and ensure that the active valves will perform their safety-related functions when necessary.

3.9.3.3a Design and Installation of NSSS Supplied Pressure Relief Devices

3.9.3.3a.1 Main Steam Safety/Relief Valves

Safety/relief valve lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period from opening of the safety/relief valve until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the safety/relief valve cause the safety/relief valve discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a stepwise time history solution of the fluid flow equation to generate a time-history of the fluid properties at numerous locations along the pipe. Simultaneously, reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid friction terms. Figure 3.9-2 shows a set of fluid property and pipe section load transients typical of those produced by relief valve discharge.

The method of analysis applied to determine piping system response to relief valve operation is time history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the safety/relief valve, the main steam line, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. The Code stress limits corresponding to load combinations classification as normal, upset, emergency and faulted, are applied to the steam and discharge pipe.

3.9.3.3b Design and Installation Details for Mounting of Pressure Relief Devices in ASME Code Class 1 and 2 Systems

The design of pressure relieving devices can be grouped into two categories: open discharge and closed discharge.

a) Open Discharge

There are no open discharge pressure relieving devices mounted on ASME Code Class 1 and 2 systems.

b) Closed Discharge

A closed discharge system is characterized by piping between the valve and a tank, or some other terminal end. Under steady-state conditions, there are no net unbalanced forces. The initial transient response and resulting stresses are determined by using either a time-history computer solution or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included. Water slug effects are also considered.

Time history dynamic analysis is performed for the discharge piping and its supports. The effect of the loading on the header is also considered. The design load combinations for a given transient are shown in Table 3.9-2, and the design criteria and stress limits for the relief valve are shown in Table 3.9-2g.

3.9.3.4 Component Supports

3.9.3.4.1 Recirculation Piping Supports

The NSSS-designed recirculation piping supports are designed in accordance with Subsection NF of ASME Code Section III. (Non-NSSS designed pipe supports on recirculation piping are in accordance with Subsection 3.9.3.4.6.) Supports are either designed by load rating per paragraph NF-3260, or to the stress limits for linear supports per paragraph NF-3231. In general, the load combinations for the various operating conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet the Code requirements.

The design criteria and dynamic testing requirements for component supports are as follows:

Component Supports

All components supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed.

Hangers

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

Snubbers

The design load on snubbers includes those loads caused by dynamic forces (OBE and SSE) and hydrodynamic loads, system anchor movements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

The snubbers are designed and load rated in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. Faulted condition design uses the criteria outlined in Appendix F of the ASME code. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

The snubbers are also tested dynamically to ensure that they can perform as required in the following manner:

a) The snubber will be subjected to either force or displacement that varies approximately as the sine wave.

b) The frequency (Hz) of the input motion or force will be verified at small increments within the specified range.

c) The resulting relative displacements and corresponding loads across the working components, including end attachments, will be recorded.

d) The test will be conducted with the snubber at various temperatures representative of operating conditions.

e) The rated load in both tension and compression will be equal to or higher than the peak load.

f) The duration of the test at each frequency will be specified.

Dynamic Testing

The criterion used to demonstrate the operability of the snubber under dynamic loading conditions is that the total travel of the unit, including lost motion and deflection during dynamic load cycling, shall not exceed ±.060 inches (.120 inch total).

The dynamic testing on a prototype snubber consists of the following:

(a) The snubber is subjected to load cycling at 100%, 75%, 50% of the rated load and this load varies approximately as the sine wave function.

(b) The frequency of the input force is varied from 3 to 33 Hz in 3 Hz steps.

(c) The duration of the test (load cycles can be determined by this time interval) at each frequency is 10 seconds as a minimum.

Struts

The design load on struts includes those loads caused by dead weight, thermal expansion, primary dynamic forces, i.e., (OBE) and SSE, system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The permissible compressive load on the reactor vessel support skirt cylinder (modeled as plate and shell type component support) was limited by the GE design specification to 90 percent of the load which produces yield stress, divided by the safety factor for the condition being evaluated. The effects of fabrication and operational eccentricity was included. The safety factor for faulted conditions was 1.125.

An analysis of reactor pressure vessel support skirt buckling for faulted conditions shows that the support skirt has the capability to meet ASME Code Section III, Paragraph F-1370(c) faulted condition limits of 0.67 times the critical buckling strength of the support at temperature. The faulted condition analyzed included the compressive loads due to the design basis maximum earthquake, the overturning moments and shears due to the jet reactor load resulting from a severed pipe, and the compressive effects on the support skirt due to the thermal and pressure

expansion of the reactor vessel. The expected maximum earthquake loads for the reactor vessel support skirts are less than 60% of the maximum design basis loads used in the buckling analysis described; therefore, the expected faulted loads are well below the critical buckling limits of Paragraph F-1370(c) for the vessel support skirt. The expected earthquake loads were determined using the seismic dynamic analysis methods described in Section 3.7.

The loading condition, stress criterion, calculated and allowable stresses are summarized in Table 3.9-2a.

3.9.3.4.3 NSSS Floor-Mounted Equipment (Pumps, Heat Exchangers and RCIC Turbine)

The RHR pump, core spray pump, RHR heat exchangers, RCIC pump, SLC pump, and RCIC turbine are all analyzed to verify the adequacy of their support structures under various plant operating conditions. In all cases, the calculated loads in the critical support areas are within the ASME Code allowables.

3.9.3.4.4 Supports for ASME Code Class 1, 2 and 3 Active Components

ASME Code Class 1, 2 and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Seismic Category I active pumps supports are qualified for seismic and hydrodynamic loads by testing when the pump supports along with the pumps are fulfilling the following conditions:

1) Simulate actual mounting conditions.

2) Simulate all static and dynamic loadings on the pump.

3) Monitor pump operability during testing.

4) The normal operation of the pump during and after the test indicates that the supports are adequate. Any deflection or deformation of the pump supports which precludes the operability of the pump, is not accepted; and,

5) Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Seismic and hydrodynamic qualification of component supports by analysis is generally accomplished as follows:

1) Stresses at all support elements and parts such as pumps holddown, and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to be within the allowable limits as specified in ASME Subsection NF.

2) For Normal and Upset plant conditions, the deflections and deformations of the supports are assured to be within the elastic limits and not to exceed the values permitted by the designer based on his design verification tests to ensure the operability of the pumps.

3) For Emergency and Faulted plant conditions, the deformations must not exceed the values permitted by the designer to ensure the operability of the pumps. Elastic/plastic analysis will be performed if the deflections are above the elastic limits.

3.9.3.4.5 HPCI Turbine

This section has been intentionally deleted.

3.9.3.4.6 Non-NSSS Supports

The design loading combinations for supports for ASME Code Class 1, 2, and 3 components, categorized with respect to plant operating conditions identified as normal, upset, emergency, and faulted, are given in Table 3.9-14. This table also provides the stress limits for each plant operating condition.

The loads imposed on the ASME Class 1, 2 and 3 active valves and pumps are limited to values meeting both the manufacturer's and code allowables to insure operability of the active components by the design of the supports. The supports are designed to remain elastic under the maximum loads. The minor local deformations associated with the elastic deformation of the support will not impair operability of the active components.

3.9.3.4.6.1 Snubbers

Snubbers will be used in seismic Category I systems. Snubbers will be purchased with load ratings appropriate for the design conditions and load combinations.

3.9.3.4.6.1.1 Snubber Design Specification

The specification for the purchase of shock suppressors (snubbers) covers the following related to snubber design, supplier's performance qualification tests and load tests. Mechanical snubbers specified for Susquehanna SES are addressed below. Design specification information pertaining to hydraulic snubbers is also provided below.

Design Criteria

Mechanical Snubbers

a) The frictional resistance of purchased suppressors shall not exceed 2% of the rated load. However, for suppressors tested during in-service inspection, the frictional resistance may exceed 2% of the rated load provided that the effects of this increase on the Seismic Category I systems on which the suppressors are used have been evaluated and shown to be acceptable.

b) All purchased snubbers shall be designed such that they limit the acceleration of the pipe to a maximum of 0.02g when subjected to any load up to rated load. An evaluation of installed snubbers and the Seismic Category I systems on which they are used indicates that the permissible acceleration can be increased to 0.04g maximum, with some exceptions that are limited to 0.02g as defined in the snubber program.

c) The total pipe movement a long the axis of the suppressor shall not exceed ±.060 inches due to any applied dynamic cycle load from 3 to 33 cps up to the rated load at the unit.

d) The suppressor shall be designed for an exposure to a temperature of 40°F prior to initial startup and 200°F during continuous operations and to a relative humidity of 55 percent normally and 90 percent during shutdown. The radiation exposure shall be 100 Roentgen/hour.

Performance Test: Two types of tests are required.

Production Test: This type of test is required to be performed on each unit.

a) Check unit to confirm acceleration level is less than specified maximum.

b) Check unit to confirm that it operates freely over the total stroke.

c) Measure and record the force required to initiate motion over the stroke in tension and compression.

d) Measure and record lost motion of the snubber mechanism.

Qualification Tests: These types of tests are to be performed on randomly selected production models. These tests are used to demonstrate the required load performance (load rating) and specified displacement when subjected to dynamic load cycling. Also included in these tests are low temperature, high temperature, humidity, salt/sand and dust spray test, life test and faulted load test.

Hydraulic Snubbers

The following environmental conditions apply to each hydraulic snubber:

	Normal Operation	Emergency	Faulted
Temperature range	32°F to 176°F 0°C to 80°C	32°F to 302°F 0°C to 150°C	32°F to 302°F 0°C to 150°C
Humidity	Up to 100%	Up to 100%	Up to 100%
Pressure	1 bar	1 bar	5 bar
Radiation (*total accum)	10^7 rad	10^7 rad	10^7 rad
Max. no. cycles	20000	40	1

The following performance testing is required for each hydraulic snubber:

The frictional resistance, including breakaway, in compression and tension shall not exceed 1.5% of normal load for a normal load > 4500 lbs.

The standard lock-up velocity in compression and tension shall be between 4.7 ipm to 14.2 ipm.

The standard bleed velocity shall be between 0.47 imp to 4.7 ipm at normal load.

The piston rod travel during a dynamic functional test under rated load from 2 to 35 cps shall be kept at < $\pm$ 0.16”.

3.9.3.4.6.1.2 Snubber Analysis Model

A piping system is idealized as a mathematical model consisting of lumped masses connected by massless elastic members. The elastic members are given the properties of the piping system being analyzed. The lumped masses are carefully located to adequately represent the dynamic and elastic properties of the piping system. A lumped mass is located at the beginning and end of every elbow, valve, at the extended valve operator, and at the intersection of every tee. On straight runs, lumped masses are located at spacings no greater than the span length corresponding to 33 cps. A mass point is located at every extended mass to account for torsional effects on the piping system. In addition, the increased stiffness and mass of valves is considered in the modeling of a piping system.

The three-dimensional stiffness matrix of the mathematical model is determined by the direct stiffness method. Axial, shear, flexural and torsional deformations of each member are included. For curved members and branch connectors a decreased stiffness is used in accordance with ASME Section III. The mass matrix is also calculated.

Snubbers are considered to be rigid members in the dynamic model. Differences in tension and compression spring rates will not effect design calculations; similarly entrapped air and temperature do not effect mechanical snubbers. Hydraulic snubbers manufactured by LISEGA have a pressurized reservoir that precludes air entrapment and environmental temperature range limits are provided in the design specification.

The load conditions and combinations are being addressed as a generic issue and are included in the SSES plant Design Assessment Report (DAR). The transients analyzed will include:

1. Seismic

2. Hydrodynamic

    a) LOCA induced
    b) SRV induced

3. Flow disruption transients (e.g., fast valve closure)

4. Normal Operating Loads.

Snubber locations and sizes are chosen to maintain the stresses due to the above listed loads to below the ASME code allowable stresses. Since all of the loads described above will be maintained below the code allowable stresses for the plant condition indicated in the DAR, the snubber with the appropriate load rating will be used.

3.9.4 CONTROL ROD DRIVE SYSTEM

This plant is equipped with a hydraulic control rod drive system. The discussion in this section includes the Control Rod Drive Mechanism (CRDM), the Hydraulic Control Unit (HCU), the Condensate Supply System and the Scram discharge volume and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information on CRDS

Descriptive information on the control rod drives as well as the entire control and drive system is contained in Section 4.6.

3.9.4.2 Applicable CRDS Design Specifications

The Control Rod Drive System (CRDS) is designed to meet the functional design criteria as outlined in Section 4.6 and consists of the following:

a) Locking piston control rod drive;

b) Hydraulic control unit;

c) Hydraulic power supply (pumps),

d) Interconnecting piping,

e) Flow and pressure and isolation valves,

f) Instrumentation and electrical controls.

Those CRD components forming part of the primary pressure boundary are designed according to ASME Code Section III.

The quality group classification of the CRD hydraulic system is outlined in Table 3.2-1, and the components are designed according to the codes and standards governing the individual quality groups.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in Subsection 3.9.1.1, faulted conditions in Subsection 3.9.1.4, and dynamic testing in Subsection 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRDs and CRD Housings have been evaluated analytically and the design load combinations and stress limits for the CRD housing are listed in Table 3.9-2v. For the non-code components, experimental testing was used to assure the CRD performance under all possible conditions as described in Subsection 3.9.4.4.

Deformations are not a limiting factor in the analysis of the CRDs components based upon the results of numerous tests on the drive.

3.9.4.3.1 Control Rod Drive Housing Supports

The Control Rod Drive (CRD) housing support system functions are described in Section 4.6.

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90% of yield and the allowable shear stress used was 60% of yield. These design stresses are 1.5 times the AISC allowable stresses (60% and 40% of yield, respectively).

The CRD housing supports are designed as Seismic Category I equipment.

3.9.4.4 CRD Performance Assurance Program

The CRD test program consists of the following tests:

a) Development tests

b) Factory Quality Control Tests

c) 5 year Maintenance Life tests
d) 1.5X Design Life tests

e) Operational tests

f) Acceptance tests

g) Surveillance tests

All of the above tests except c) and d) are discussed in Subsections 4.6.3 through 4.6.3.1.1.5. Tests c) and d) are discussed below:

c) "5 Year Maintenance Life" Tests

Four Control Rod Drives are picked at random from the production stock each year and subjected to various tests under simulated reactor conditions and 1/8 of the cycles specified in Subsection 3.9.1.1. Upon completion of the test program, the control rod drive parts are checked to the drawings and all parts must meet or surpass the minimum specified requirements.

d) 1.5X Design Life Tests

When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the life test cycles specified in Subsection 3.9.1.1.

Two CRDs were tested in 1976. Upon completion of the test program, the CRDs were disassembled and the parts checked to the drawing for wear and/or damage. All parts met or surpassed the minimum specified requirements.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel internals.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, CRDs, and incore nuclear instrumentation) are identified below:

Core Support Structures

Shroud

Shroud support

Core plate and holddown bolts

Top guide (including bolts and keepers)

Fuel supports

Control rod guide tubes

Control rod drive housing

Reactor Internals

Jet Pump assemblies and instrumentation

*Feedwater spargers

Vessel head spray nozzle

Differential pressure and liquid control lines

In-core flux monitor tubes

*Initial startup neutron sources

*Surveillance sample holders

Core spray lines and spargers

*In-Core instrument housings

*Steam dryer

*Shroud head and steam separator assembly

*Guide rods

CRD thermal sleeves

* Non-safety class components

A general assembly drawing of the important reactor components is shown in Figure 3.9-3.

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-4. It is the volume inside the core shroud up to the level of the jet pump suction inlet.

The design arrangement of the reactor internals, such as the jet pumps, steam separators and guide tube, is such that one end is unrestricted and thus free to expand.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in Subsection 3.9.5.1. These structures form partitions within the reactor vessel, to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Figure 3.9-4 shows the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the Downcomer annulus, thus providing a floodable region following a recirculation line break. The volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide's grid plate below. The central portion and the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the bottom by the core support. The lower portion, surrounding part of the lower plenum, is welded to the reactor pressure vessel shroud support.

3.9.5.1.1.2 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the top guide to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the Downcomer annulus.

3.9.5.1.2 Core Plate

The core plate consists of a circular stainless steel plate with bored holes stiffened with a rim and beam structure. The plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core support plate.

The entire assembly is bolted to a support ledge on the lower portions of the shroud.

3.9.5.1.3 Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings and fastened to a peripheral rim. Each opening provides lateral support and guidance for four fuel assemblies or in the case of peripheral fuel, one or two fuel assemblies. Sockets are provided in the bottom of the beam intersections to anchor the in-core flux monitors and startup neutron sources. The rim of the top guide rests on a ledge between the upper and central portions of the shroud. The top guide has alignment pins that engage and bear against slots in the shroud which are used to correctly position the assembly before it is secured. Lateral restraint is provided by wedge blocks between the top guide and the shroud wall.

3.9.5.1.4 Fuel Support

The fuel supports, shown in Figure 3.9-5 are of two basic types; namely, peripheral supports and four-lobed orificed fuel supports. The peripheral fuel support is located at the outer edge of the active core and is not adjacent to control rods. Each peripheral fuel support will support one fuel assembly and contains a single orifice assembly designed to assure proper coolant flow to the peripheral fuel assembly. Each four-lobed orificed fuel support will support four fuel assemblies and is provided with four orifice plates to assure proper coolant flow distribution to each rod-controlled fuel assembly. The four-lobed orificed fuel supports rest in the top of the control rod guide tubes which are supported laterally by the core support. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell. (see Subsection 4.4.2).

3.9.5.1.5 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings up through holes in the core support plate. Each tube is designed as the guide for a control rod and as the vertical support for a four-lobed orificed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing, which in turn transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the control rod drive housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the control rod drive housing to hold the thermal sleeve in position.

3.9.5.1.6 Jet Pump Assemblies

The jet pump assemblies are located in two semi-circular groups in the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the jet pump is covered in detail in References 3.9-1 and 3.9-2. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (see Figure 3.9-6). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a hold- down clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and

diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

3.9.5.1.7 Steam Dryers

The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe, below the water level. This skirt forms a seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly, which would occur only under accident conditions, is restricted by steam dryer hold-down brackets attached to the reactor vessel top head.

3.9.5.1.8 Feedwater Spargers

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle and is shaped to conform to the curve of the vessel wall. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.9 Core Spray Lines

The core spray lines are the means for directing flow to the core spray nozzles which distribute coolant during accident conditions.

Two core spray lines enter the reactor vessel through the two core spray nozzles. (see Section 5.4). The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the upper shroud. The two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the shroud and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the shroud. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers (see Section 6.3).

3.9.5.1.10 Vessel Head Spray Nozzle

When reactor coolant is returned to the reactor vessel part of the flow can be diverted to a spray nozzle in the reactor head. This spray maintains saturated conditions in the reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. The spray also decreases thermal stratification in the reactor vessel coolant. This ensures that the water level in the reactor vessel can rise. The higher water level provides conduction cooling to more of the mass of metal of the reactor vessel and, therefore, helps maintain the cooldown rate.

The vessel head spray nozzle is mounted to a short length of pipe and a flange, which is bolted to a mating flange on the reactor vessel head nozzle (see Subsection 5.4.7).

3.9.5.1.11 Differential Pressure and Liquid Control Line

The differential pressure and liquid control line serves a dual function within the reactor vessel - to provide a path for the injection of the liquid control solution into the coolant stream and to sense the differential pressure across the core support plate (described in Section 5.4). This line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support plate. It is used to sense the pressure below the core support plate during normal operation and to inject liquid control solution if required. This location facilitates good mixing and dispersion. The inner pipe also reduces thermal shock to the vessel nozzle should the standby liquid control system be actuated. The outer pipe terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.12 In-Core Flux Monitor Guide Tubes

In-core flux monitor guide tubes provide a means of positioning fixed detectors in the core as well as provide a path for calibration monitors (TIP System).

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing (see Section 5.4) in the lower plenum to the top of the core support plate. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring and intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes. A lattice work of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded, after assembly, to prevent loosening during reactor operation.

3.9.5.1.13 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules (see Section 5.4). The baskets hang from the brackets that are attached to the inside wall of the reactor vessel and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while avoiding jet pump removal interference or damage.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied reveals four significant faulted events:

a) Recirculation Line Break: a break in a recirculation line between the reactor vessel and the recirculation pump suction.

b) Steam line break accident: a break in one main steam line between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across some of the structures within the reactor.

c) Earthquake: subjects the core support structures and reactor internals to significant forces as a result of ground motion.

d) Safety relief valve discharge in combination with an SSE.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other engineered safety feature reactor internals are less severe than these three postulated events.

The faulted conditions for the reactor pressure vessel internals are discussed in Subsection 3.9.1.4. Loading combination and analysis for the reactor pressure vessel internals are discussed in Subsection 3.9.3.l, Tables 3.9-1 and 3.9-2.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes, which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressure in the various regions of the reactor. Figure 3.9-7 shows the nine reactor nodes. The computer code used is the General Electric Short-Term Thermal-Hydraulic Model described in Reference 3.9-3. This model has been approved for use in ECCS conformance evaluation under 10CFR50, Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are, therefore, not described in Reference 3.9-3. These additional features are discussed below:

a) The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

b) The flow path between the bypass region and the shroud head is more accurately modeled since the fuel assembly pressure differential is influenced by flashing in the guide tubes and bypass region for a steamline break. In the ECCS analysis, the momentum equation is solved in this flow path, but its irreversible loss coefficient is conservatively set at an arbitrary low value.

c) The enthalpies in the guide tubes and the bypass are calculated separately, since the fuel assembly ΔP is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Recirculation Line and Steam Line Break

3.9.5.2.3.1 Accident Definition

Both a recirculation line break (the largest liquid break) and an inside steam line break (the largest steam break) are considered in determining the design basis accident for the engineered safety feature reactor internals. The recirculation line break is the same as the design basis loss-of-coolant accident described in Section 6.3. A sudden, complete circumferential break is assumed to occur in one recirculation loop. The pressure differentials on the reactor internals and core support structures are in all cases lower than for the main steam line break.

The analysis of the steam line break assumes a sudden, complete circumferential break of one main steam line between the reactor vessel and the main steam line restrictor. A steam line break upstream of the flow restrictors produces a larger blowdown area and thus a faster depressurization rate than a break downstream of the restrictors. The larger blowdown area results in greater pressure differentials across the reactor internal structures.

The steam line break accident produces significantly higher pressure differentials across the reactor internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steam line break. Therefore, the steam line break is the design basis accident for internal pressure differentials.

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. The core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and consequently the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the reactor pressure vessel and thus the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that the calculated pressure differences bound those which could be expected if a steam line break should occur, an analysis is conducted at a low power-high recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (that is, the drive flow necessary to achieve rated core flow at rated power.)

This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

3.9.5.2.4 Seismic and Hydrodynamic Loads

The seismic and hydrodynamic loads acting on the structures within the reactor vessel are based on a dynamic analysis as described in Section 3.7. Seismic analysis is performed by coupling the lumped mass model of the reactor vessel and internals, as described in Section 3.7, with the building model to determine the acceleration force and moment time histories in the reactor vessel and internals. This is accomplished by using the modal superpostion method. Acceleration response spectra are also generated for subsystem analyses of selected components.

3.9.5.3 Design Loading Categories

Loading combinations for the core support structures are shown in Table 3.9-2. The basis for determining faulted loads on the reactor internals is shown for dynamic loads in Section 3.7 and for pipe rupture loads in Subsection 3.9.5.2.3 and 3.9.5.4.3. Table 3.9-2b shows allowable and calculated stress values for highly stressed core support structures and selected reactor internal components. Table 3.9-2aa provides this same type of information for the CRD guide tubes.

Stress intensity and other design limits are discussed in Subsection 3.9.5.4.4. The core support structures which are fabricated as part of the reactor pressure vessel assembly are discussed in Subsection 3.9.1.3 in conjunction with the reactor pressure vessel.

The design requirements for equipment classified as "other" e.g., steam dryers and shroud heads, were specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where possible, design requirements are based on applicable industry codes and standards. If these are not available, the designer relies on accepted industry or engineering practices.

3.9.5.4 Design Bases

3.9.5.4.1 Safety Design Bases

The reactor core support structures and internals shall meet the following safety design bases:

1) They shall be arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

2) Deformation shall be limited to assure that the control rods and core standby cooling systems can perform their safety functions.

3) Mechanical design of applicable structures shall assure that safety design bases (1) and (2), above, are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.4.2 Power Generation Design Bases

The reactor core support structures and internals shall be designed to the following power generation design bases:

1) They shall provide the proper coolant distribution during all anticipated normal operating conditions to full power operation of the core without fuel damage.

2) They shall be arranged to facilitate refueling operations.

3) They shall be designed to facilitate inspection.

3.9.5.4.3 Response of Internals Due to Inside Steam Break Accident

It is concluded that the maximum pressure loads acting on the reactor internal components result from an inside steam line break, and on some components the loads are greatest with operation at the minimum power associated with the maximum core flow. This has been substantiated by the analytical comparison of liquid versus steam breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that, although possible, it is not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be less.

3.9.5.4.4 Stress, Deformation, and Fatigue Limits for Reactor Internals (Except Core Support Structure)

These limits are summarized in Table 3.9-2(b).

Design Condition	SF
Normal	2.25
Upset	2.25
Emergency	1.5
Faulted	1.125

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined to determine if adequate clearances exist during emergency and faulted conditions. No mechanical clearance problems have been identified. No plastic deformation occurs in the reactor internal components during emergency or faulted conditions as shown in Subsections 3.9.4 and 3.9.3.1, and Table 3.9-2. This is used in demonstrating that no mechanical interferences exist. No fatigue analysis is required under the faulted conditions due to the low encounter frequency of faulted events and the low number of cycles. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.5.

3.9.5.4.5 Stress, Deformation, and Fatigue Limits for Core Support Structures

These limits are summarized in Tables 3.9-2a, 3.9-2b, and 3.9-2v.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

The construction permits for the Susquehanna Steam Electric Station were issued in November 1973. Relating this date to the requirements of 10CFR50.55a(g), the preservice examination

program, including provisions for design and access to enable operational readiness testing of pumps and valves, complied, as a minimum, with the 1971 Edition of the ASME B&PV Code Section XI including Addenda through Summer 1972.

This ASME Code Edition does not require preservice and in-service testing of pumps and valves to ensure operational readiness. The requirements for in-service testing of pumps and valves were added as Subsections IWV and IWP to ASME B&PV Code Section XI, Summer 1973 Addenda, effective December 30, 1973. By then, design and procurement for SSES was under way; however, the preservice testing program for pumps and valves for assessing operational readiness was conducted, to the extent practical, so that it complied with requirements of the 1974 Edition, ASME B&PV Code through Winter 1975 Addenda.

The first 120 months' in-service tests will assess operational readiness of pumps and valves. These tests complied, to the extent practical within design limitations, with the requirements of 10CFR50.55a.

During successive 120-month periods, in-service tests of pumps and valves for assessing operational readiness will comply, to the extent practical within design limitations, with the requirements of 10CFR50.55a.

3.9.7 REFERENCES

3.9-1 "Design and Performance of G.E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

3.9-2 Moen, H.H., "Testing of Improved Jet Pumps for the BWR/6 Nuclear System," General Electric Company, Atomic Power Equipment Department, NEDO-10602, June 1972.

3.9-3 General Electric Company, "Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K," Proprietary Document, General Electric Company, NEDE-20566.

3.9-4 Not Used

3.9-5 Not Used

3.9-6 Seismic Analysis of Piping Systems, BP-TOP-1, Bechtel Power Corporation, San Francisco, California, Rev. 2, January, 1975.

3.9-7 "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants," NEDE-24057-P (Class III) and NEDO-24057 (Class I), November, 1977.

3.9-8 "Functional Capability Criteria for Essential Mark II Piping," NEDO-21985, 78 NED174 (Class I), September, 1978.

3.9-9 "Power Uprate Engineering Report for Susquehanna Steam Electric Station, Units 1 and 2," NEDC-32161P, As Revised by PP&L Calculation EC-PUPC-1001, Revision 0, March, 1994.

3.9-10 “BWR Jet Pump Assembly Maintenance Issues,” General Electric Company, San Jose, CA, June 2002

TABLE 3.9-1

TRANSIENTS AND THE NUMBER OF ASSOCIATED CYCLES CONSIDERED IN THE DESIGN AND FATIGUE ANALYSIS OF THE RPV ASSEMBLY AND INTERNAL TRANSIENTS

		No. of Cycles
Normal, Upset, and Testing Conditions		
1.	Bolt Up*	123
2.	Design Hydrostatic Test	130
3.	Startup (100°F/hr Heatup Rate)**	117
4.	Daily Reduction to 75% Power*	10,000
5.	Weekly Reduction to 50% Power*	2,000
6.	Control Rod Pattern Change*	400
7.	Loss of Feedwater Heaters (80 Cycles Total):	80
8.	50% Safe Shutdown Earthquake Event at Rated Operating Conditions	10****
9.	Scram:	
	a. Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	40
	b. Other Scrams	140
10.	Reduction to 0% Power, Hot Standby with main condenser available, Shutdown (100°F/hr Cooldown Rate)**	111
11.	Unbolt	123
12.	Pre-op Blowdown	10
13.	Natural Circulation Startup	3
14.	Loss of AC Power, Natural Circulation Restart	5
Emergency Conditions		
15.	Scram:	
	a. Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stay Open	1***

TABLE 3.9-1 (Continued)

Transients	No. of Cycles
Emergency Conditions (Continued)	
16. Scram	
a. --	--
b. Automatic Blowdown	1***
c. Loss of Feedwater Pumps, Isolation Valves Closed	5
d. Single Safety or Relief Valve Blowdown	8
17. Improper Start of Cold Recirculation Loop	1***
18. Sudden Start of Pump in Cold Recirculation Loop	1***
19. Improper Startup with Reactor Drain Shut Off	1***
Faulted Condition	
20. Pipe Rupture and Blowdown	1***
21. Safe Shutdown Earthquake at Rated Operating Conditions	1***

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* Applies to reactor pressure vessel only.

** Bulk average vessel coolant temperature change in any 1-hour period.

*** The annual encounter probability for the one cycle events is $< 10^{-2}$ for emergency and 10 for faulted events.

**** Includes 10 maximum load cycles per event. Not required to be considered in fatigue analysis due to low encounter frequency ($<10^{-2}$) and low number of cycles.

TABLE 3.9-2

INTRODUCTION

This table lists the major mechanical safety related mechanical components in the plant. Various parts of the table are referenced in Section 3.9. The format in the various parts of the table is consistent, since variations exist on analytical methods and depth of detail necessary to demonstrate the safety aspects of various components.

TABLE 3.9-2 INDEX

TABLE	CONTENTS
3.9-2	LOAD COMBINATION AND ACCEPTANCE CRITERIA FOR ASME CODE CLASS 1, 2, AND 3 PIPING AND COMPONENTS
3.9-2a	REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY (i) Vessel Support Skirt (ii) Shroud Support (iii) RPV Feedwater Nozzle (iv) CRD Penetration - CRD Housing (v) CRD Penetration - Stub Tube
3.9-2b	REACTOR INTERNALS & ASSOCIATED EQUIPMENT (i) Top Guide - Highest Stressed Beam (ii) Core Plate (Ligament in Top Plate) (iii) Vessel Head Spray Nozzle
3.9-2c	REACTOR WATER CLEANUP (REGENERATIVE & NON-REGENERATIVE) HEAT EXCHANGERS
3.9-2d	ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT
3.9-2e	ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT
3.9-2f	NOT USED
3.9-2g	MAIN STEAM SAFETY/RELIEF VALVES
3.9-2h	MAIN STEAM ISOLATION VALVE
3.9-2i	RECIRCULATION PUMP
3.9-2j	REACTOR RECIRCULATION SYSTEM GATE VALVES
3.9-2k	CLASS III SAFETY RELIEF VALVE DISCHARGE PIPING
3.9-2l	STANDBY LIQUID CONTROL PUMP
3.9-2m	STANDBY LIQUID CONTROL TANK
3.9-2n	ECCS PUMPS (i) RHR Pumps (ii) Core Spray Pumps
3.9-2o	RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGER

TABLE 3.9-2 INDEX - (Continued)

TABLE 3.9-2

LOAD COMBINATION AND ACCEPTANCE CRITERIA
FOR ASME CODE CLASS 1, 2, AND 3
NSSS PIPING AND EQUIPMENT

Load Combination	Design Basis	Evaluation Basis	Service Level
$N + SRV_{(ALL)}$	Upset	Upset	(B)
N + OBE	Upset	Upset	(B)
$N + OBE + SRV_{(ALL)}$	Emergency	Upset	(B)
$N + SSE + SRV_{(ALL)}$	Faulted	Faulted*	(D)
N + SBA + SRV	Emergency	Emergency*	(C)
N + IBA + SRV	Faulted	Faulted*	(D)
$N + SBA + SRV_{(ADS)}$	Emergency	Emergency*	(C)
$N + SBA + OBE + SRV_{(ADS)}$	Faulted	Faulted*	(D)
$N + IBA + OBE + SRV_{(ADS)}$	Faulted	Faulted*	(D)
$N + SBA/IBA + SSE + SRV_{(ADS)}$	Faulted	Faulted*	(D)
N + LOCA** + SSE	Faulted	Faulted*	(D)

NOTE; All dynamic loads are combined by SRSS

LOAD DEFINITION LEGEND

Normal(N) - Normal and/or abnormal loads depending on acceptance criteria.

OBE - Operational basis earthquake loads.

SSE - Safe Shutdown earthquake loads.

SRV_{ALL} - The loads induced by the actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).

SRV_{ADS} - The loads induced by the actuation of safety/relief valves associated with Automatic Depressurzation System which actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.

SRV - Safety/relief-valve-discharge-induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling).

LOAD COMBINATION TABLE 3.9-2 (Cont'd)

LOCA	- The loss of coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping).
$LOCA_1$	- Pool swell drag/fallout loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
$LOCA_2$	- Pool swell impact loads on piping and components located above the suppression pool water upper surface.
$LOCA_3$	- Oscillating pressure induced loads on submerged piping and components during condensation oscillations.
$LOCA_4$	- Building motion induced loads from chugging.
$LOCA_5$	- Building motion induced loads from main vent air clearing.
$LOCA_6$	- Vertical and horizontal loads on main vent piping.
$LOCA_7$	- Annulus pressurization loads.
SBA	- The abnormal transients associated with a Small Break Accident.
IBA	- The abnormal transients associated with an Intermediate Break Accident.

* All ASME Code Class 1, 2, and 3 piping systems that are required to function for safety shutdown under the postulated events shall meet the requirements of NRC's "Interim Technical Position-Function Capability of passive components" - by MEB.

** The most severe load combination among LOCA.

TABLE 3.9-3

NSSS SEISMIC CATEGORY I ACTIVE PUMPS AND VALVES

COMPONENT NAME	IDENTIFICATION AS SHOWN ON APPLICABLE FIGURES
Main Steam Isolation Valves	B 21 F 022 B 21 F 028
Safety Relief Valves	B 21 F 013
Control Rod Drive Globe Valves	C 12 F 010/180 (Unit 1) C 12 F 011/181 (Unit 1) C 12 F 010A/B (Unit 2) C 12 F 011A/B (Unit 2)
Recirculation System (Discharge and Bypass) Gate Valves	B 31 F 031 B 31 F 032
Standby Liquid Control Pump	C 41 C 001
Standby Liquid Control Valve	C 41 F 004
RHR Pump	E 11 C 002
Core Spray Pump and Motor	E 21 C 001
RCIC Pump	E 51 C 001
RCIC Turbine	E 51 C 002
HPCI Pump	E 41 C 001
HPCI Turbine	E 41 C 002

TABLE 3.9-4 (Page 1 of 21)

APPLICABLE THERMAL TRANSIENTS

(PRE STARTUP LEAK TEST)
130 CYCLES
CONDITION - TEST

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	70	100	30 min	60	30
Head Spray (RHR)	70	100	30 min	60	30
Recirculation Suction	70	100	30 min	60	30
Recirculation Discharge	70	100	30 min	60	30
Bottom Drain	70	100	30 min	60	30
Standby Liquid Control	70	100	30 min	60	30
	100	50	Step	10 min Duration	50
	50	100	Step		50
Core Spray	70	100	30 min	60	30
Feedwater	70	100	30 min	60	30
CRDHS Return	70	100	30 min	60	30

Remarks: After temperature is raised to 100°F, reactor pressure is increased to 1250 psig and then decreased to 0 psig.

TABLE 3.9-4 (Page 2 of 21)

APPLICABLE THERMAL TRANSIENTS

(STARTUP)
120 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	100	551	-	100	451
Head Spray (RHR)	100	551	-	100	451
Recirculation Suction	100	551	-	100	451
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray only occurs 10 times	100	406	-	100	306
	406	50	Step	-	356
	50	406	Step	-	356
	406	551	-	100	145
Feedwater	100	551	-	100	451
	551	90	Step	-	461
	90	420	30 min	660	330
CRDHS Return	100	50	Step	-	50

Remarks: Reactor pressure increases from 0 to 1038 psig at rate of temperature increase.

TABLE 3.9-4 (Page 3 of 21)

APPLICABLE THERMAL TRANSIENTS

(DAILY POWER REDUCTION AND ROD PATTERN CHANGE)
10,400 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	527			
Bottom Drain	527	527			
Standby Liquid Control	527	527			
Core Spray	527	527			
Feedwater	420	354	15 min	264	66
	354	420	15 min	264	66
CRDHS Return	50	50			
Clean Return	435	435			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 4 of 21)

APPLICABLE THERMAL TRANSIENTS

(WEEKLY POWER REDUCTION)
2000 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	527			
Bottom Drain	527	527			
Standby Liquid Control	527	527			
Core Spray	527	527			
Feedwater	420	324	30 min	192	96
	324	420	30 min	192	96
CRDHS Return	50	50			
Cleanup Return	435	435			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 5 of 21)

APPLICABLE THERMAL TRANSIENTS

(TURBINE TRIP 100 PERCENT BYPASS)
10 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	495	1.5 min	1280	32
	495	527	4 min	480	32
Recirculation Discharge	495	527	4 min	480	32
Bottom Drain	495	527	4 min	480	32
Standby Liquid Control	495	527	4 min	480	32
Core Spray	495	527	4 min	480	32
Feedwater	420	100	1.5 min	12,800	320
	100	420	4 min	4800	320
CRDHS Return	50	50			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 6 of 21)

APPLICABLE THERMAL TRANSIENTS

(PARTIAL FEEDWATER HEATER BYPASS)
70 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	517	2 min	300	10
	517	527	4 min	150	10
Recirculation Discharge	517	527	4 min	150	10
Bottom Drain	517	527	4 min	150	10
Standby Liquid Control	517	527	4 min	150	10
Core Spray	517	527	4 min	150	10
Feedwater	420	265	1.5 min	6200	155
	265	420	3 min	3100	155
CRDHS Return	50	50			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 7 of 21)

APPLICABLE THERMAL TRANSIENTS

(SCRAM - T/G TRIP FEEDWATER ON - MSIV OPEN)
40 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	565	10 sec	5040	14
	565	543	15 sec	5280	22
	543	400	-	100	143
	400	551	-	100	151
Head Spray (RHR)	400	551	-	100	151
Recirculation Suction	527	400	-	100	127
	400	551	-	100	151
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray	543	527	Step	-	16
Feedwater	420	275	1 min	8700	145
	275	100	15 min	700	175
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	50	50			

Remarks: Reactor pressure increases to 1163 psig all relief valves open. Pressure decreases to 240 psig and then increases to 1038 psig.

TABLE 3.9-4 (Page 8 of 21)

APPLICABLE THERMAL TRANSIENTS

(ALL OTHER SCRAMS)
140 CYCLES
CONDITION - UPSET

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	543	15 sec	1920	8
	543	400	-	100	143
	400	551	-	100	151
Head Spray (RHR)	400	551	-	100	151
Recirculation Suction	527	400	-	100	127
	400	551	-	100	151
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray	543	527	Step	-	16
Feedwater	420	275	1 min	8700	145
	275	100	15 min	700	175
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	50	50			

Remarks: Reactor pressure decreases to 240 psig and then increases to 1038 psig.

TABLE 3.9-4 (Page 9 of 21)

APPLICABLE THERMAL TRANSIENTS

(RATED POWER)
SEE BELOW
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	527			
Bottom Drain *240 Times	527	150	1 hr	377	377
	150	527	Step	-	377
Standby Liquid Control *10 Times	527	60	Step	-	467
	60	527	60 min	467	467
Core Spray	527	527			
Feedwater	420	420			
CRDHS Return	50	50			
Cleanup Return	435	435			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 10 of 21)

APPLICABLE THERMAL TRANSIENTS

(SHUT DOWN)
111 CYCLES
CONDITION - NORMAL

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	375	-	100	176
	375	330	10 min	270	45
	330	100	-	100	230
Head Spray (RHR) and RHR Return	551	375	-	100	176
	375	50	Step	15 sec Duration	325
	50	300	Step		250
	300	100	-	100	200
Recirculation Suction	527	551	Step	-	24
	551	375	-	100	176
	375	330	10 min	270	45
	330	100	-	100	230
Bottom Drain	330	100	-	100	230
Standby Liquid Control	330	100	-	100	230
Core Spray	330	100	-	100	230
Recirculation Discharge	527	551	Step	-	24
	551	375	-	100	176
	375	300	Step	-	75
	300	260	10 min	240	40
	260	100	-	100	160
Feedwater *5 step changes to 100°F and back during cooldown	420	265	30 min	310	155
	265	420	Step	-	155
	420	551	-	100	131
	551	100	-	100	451
CRDHS Return	50	50			

Remarks: Reactor pressure decreases from 1038 psig to 0 psig.

TABLE 3.9-4 (Page 11 of 21)

APPLICABLE THERMAL TRANSIENTS

(LOSS OF FEEDWATER PUMPS - MSIV CLOSED)
10 CYCLES
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	571	3 sec	24,000	20
	571	565	10 sec	2,160	6
	565	536	3 min	580	29
	536	565	73 min	24	29
	565	505	7 min	514	60
	505	400	-	100	105
	400	551	-	100	151
Head Spray (RHR)	400	551	-	100	151
Recirculation Suction	527	300	30 min	454	227
	300	551	-	100	251
	551	543	Step	-	8
	543	527	Step	-	16
Recirculation Discharge	543	527	Step	-	16
Bottom Drain	527	300	3.7 min	3,681	227
	300	505	23 min	535	205
	505	300	3 min	4,100	205
	300	505	73 min	169	205
	505	300	7 min	1,757	205
	300	551	-	100	251
	551	543	Step	-	8
	543	527	Step	-	16
Standby Liquid Control	543	527	Step	-	16
Core Spray	543	527	Step	-	16

TABLE 3.9-4 (Page 12 of 21)

APPLICABLE THERMAL TRANSIENTS

(LOSS OF FEEDWATER PUMPS - MSIV CLOSED) **(continued)**

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Feedwater	420	551	Step	-	131
	551	40	Step	-	511
	40	551	23 min	1,333	511
	551	40	Step	-	511
	40	551	51 min	601	511
	551	40	Step	-	511
	40	300	5 min	3120	260
	300	551	-	100	251
	551	100	Step	-	451
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	50	545	10 min	2,970	495

Remarks: Reactor pressure increases to 1218 psig. All relief valves open. Pressure decreases to 1163 psig and relief valves close. RCIC initiates and pressure decreases to 913 psig. RCIC trips off on high level and pressure increases to 1163 psig and one relief valve opens and then closes as pressure decreases at rate of 100°F/hr.

TABLE 3.9-4 (Page 13 of 21)

APPLICABLE THERMAL TRANSIENTS

(REACTOR OVERPRESSURE DELAYED SCRAM)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	583	2 sec	57,600	32
	583	543	30 sec	4,800	40
	543	400	-	100	143
Head Spray (RHR)	543	400	-	100	143
Recirculation Suction	527	562	11 sec	11,455	35
	562	400	-	100	162
Recirculation Discharge	562	400	-	100	162
Bottom Drain	562	400	-	100	162
Standby Liquid Control	562	400	-	100	162
Core Spray	562	400	-	100	162
Feedwater	420	276	1 min	8640	144
	276	100	15 min	704	176
	100	250	Step	-	150
	250	420	30 min	340	170
CRDHS Return	100	50	Step	-	50

Remarks: Reactor pressure increases to 1350 psig. All relief valves and safety valves open. Pressure decreases to 240 psig.

TABLE 3.9-4 (Page 14 of 21)

APPLICABLE THERMAL TRANSIENTS

(SINGLE SAFETY/RELIEF VALVE BLOWDOWN)
8 CYCLES
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	375	10 min	1,056	176
	375	100	-	100	275
Head Spray (RHR)	375	100	-	100	275
Recirculation Suction	527	375	10 min	912	152
	375	100	-	100	275
Recirculation Discharge	375	100	-	100	275
Bottom Drain	275	100	-	100	275
Standby Liquid Control	375	100	-	100	275
Core Spray	375	100	-	100	275
Feedwater	420	276	1 min	8640	144
	276	100	15 min	704	176
CRDHS Return	50	50			

Remarks: Reactor pressure decreases to 0 psig with one relief valve or safety valve open.

TABLE 3.9-4 (Page 15 of 21)

APPLICABLE THERMAL TRANSIENTS

(AUTOMATIC DEPRESSURIZATION)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	375	3.3 min	3,200	176
	375	281	-	300	94
Head Spray (RHR)	375	281	-	300	94
Recirculation Suction	527	375	3.3 min	2,764	152
	375	281	-	300	94
Recirculation Discharge	375	281	-	300	94
Bottom Drain	375	281	-	300	94
Standby Liquid Control	375	281	-	300	94
Core Spray	375	281	-	300	94
Feedwater	420	276	1 min	8640	144
	276	100	15 min	704	176
CRDHS Return	50	50			

Remarks: Reactor pressure decreases to 35 psig with auto-blowdown relief valves open.

TABLE 3.9-4 (Page 16 of 21)

APPLICABLE THERMAL TRANSIENTS

(IMPROPER START OF COLD RECIRCULATION LOOP)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	130	Step	26 sec Duration	397
	130	527	Step		397
Recirculation Discharge	527	130	Step	34 sec Duration	397
	130	527	Step		397
Bottom Drain	527	527			
Standby Liquid Control	527	527			
Core Spray	527	268	Step	34 sec Duration	259
	268	527	Step		259
Feedwater	420	420			
CRDHS Return	50	50			

Remarks: Reactor pressure remains at 1038 psig.

TABLE 3.9-4 (Page 17 of 21)

APPLICABLE THERMAL TRANSIENTS

(SUDDEN PUMP START IN COLD LOOP)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	551			
Head Spray (RHR)	551	551			
Recirculation Suction	527	527			
Recirculation Discharge	527	130	Step	34 second Duration	397
	130	527	Step		397
Bottom Drain	527	350	Step		177
	350	527	Step		177
Standby Liquid Control	350	527	Step		177
Core Spray	527	527			
Feedwater	420	420			
CRDHS Return	50	50			
Remarks: Reactor pressure remains at 1038 psig.					

TABLE 3.9-4 (Page 18 of 21)

APPLICABLE THERMAL TRANSIENTS

(IMPROPER START WITH RECIRCULATION PUMPS OFF)
1 CYCLE
CONDITION - EMERGENCY

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	100	551	-	100	451
Head Spray (RHR)	100	551	-	100	451
Recirculation Suction	100	551	-	100	451
Recirculation Discharge	100	551	-	100	451
	551	130	Step	34 sec Duration	421
	130	551	Step		421
Bottom Drain	100	551	5 min	5,412	451
Standby Liquid Control	100	551	5 min	5,412	451
Core Spray	100	551	-	100	451
Feedwater	90	551	-	100	461
	551	90	Step	-	461
	90	420	30 min	660	330
CRDHS Returns	50	50			

Remarks: Reactor pressure increases to 1038 psig as temperature increases.

TABLE 3.9-4 (Page 19 of 21)

APPLICABLE THERMAL TRANSIENTS

(PIPE RUPTURE AND BLOWDOWN)
1 CYCLE
CONDITION - FAULTED

Pipeline	Initial Temp. °F	Final Temp. °F	Time	Temp. Rate °F/Hour	Temp. °F
Main Steam Line	551	281	15 sec	64,800	270
Head Spray (RHR)	551	281	15 sec	64,800	270
Recirculation Suction	527	281	15 sec	59,040	246
Recirculation Discharge	527	281	15 sec	59,040	246
	281	223	35 sec	5,966	58
	223	50	Step	90 sec Duration	173
	50	130	Step		80
Bottom Drain	527	281	15 sec	59,040	246
	281	273	35 sec	822	8
	273	50	Step	90 sec Duration	223
	50	130	Step		80
Standby Liquid Control	50	130	Step		80
Core Spray	527	406	10 sec	43,560	121
	406	50	Step	90 sec Duration	356
	50	130	Step		80
Feedwater	420	281	15 sec	33,400	139
CRDHS Return	50	50			

Remarks: Reactor pressure decreases from 1038 psig to 35 psig in 15 seconds.

TABLE 3.9-4 (Page 20 of 21)

APPLICABLE THERMAL TRANSIENTS

(BECHTEL CRITERIA)

1/2 SSE Cycles (Operating Basis Earthquake) Condition-Upset	
Expected number of equivalent 1/2 SSE in life of pipe system	5
Average duration of strong motion vibration 1/2 SSE	15 sec
Average number of maximum seismic load cycles of pipe system for each 1/2 SSE	10
Total lifetime number of maximum seismic load cycles of piping system	50
SSE Cycles (Design Basis Earthquake) Condition-Faulted	
Expected number of equivalent SSE in life of pipe system	1
Average duration of strong motion vibration SSE	15 sec

TABLE 3.9-4 (Page 21 of 21)

APPLICABLE THERMAL TRANSIENTS

(GENERAL ELECTRIC CRITERIA)
FOR NSSS PIPING

1. 1/2 SSE Cycles Condition - Upset

Expected number of equivalent 1/2 SSE in life of pipe system	1
Average duration of strong motion vibration 1/2 SSE	30 sec
Average number of maximum seismic load cycles of pipe system for each 1/2 SSE	10
Total lifetime number of maximum seismic load cycles of piping system	10

2. SSE Cycles Condition-Faulted

Expected number of equivalent SSE in life of pipe system	1
Average duration of strong motion vibration SSE	30 sec
Average number of maximum seismic load cycles of pipe system for each SSE	1
Total lifetime number of maximum seismic load cycles of piping system	1

3. Turbine Stop Valve* Closure (TSVC) Condition-Upset 120 cycles

4. Relief Valve Lift* (RVL) Condition-Upset 20,100 cycles

* not applicable to recirculation piping due to negligible effect.

TABLE 3.9-5

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 3.9-6

DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 1, 2, AND 3 COMPONENTS (NON-NSSS)

Condition		Design Loading Combinations(1)
Design		PD
Normal		PD + DW
Upset	(a)	PO + DW + $(OBE^2 + SRV\ x^2)1/2$
	(b)	PO + DW + $(RVC^2 + OBE^2)1/2$
	(c)	PO + DW + FV
	(d)	PO + DW + OBE + RVO
Emergency	(a)	PO + DW + $(OBE^2 + SRV^2_{ADS} + SBA^2)1/2$
	(b)	PO + DW + $(FV^2 + OBE^2)1/2$
Faulted	(a)	PO + DW + $(OBE^2 + SRV^2_{ADS} + IBA^2)1/2$
	(b)	PO + DW + $(SSE^2 + SRV^2_{ADS} + IBA^2)1/2$
	(c)	PO + DW + $(SSE^2 + DBA^2)1/2$

NOTE: (1) As required by the appropriate subsection, i.e. NB, NC, or ND, of ASME Section III Division I. Other loads, such as loads from thermal transient, thermal gradients, and the anchor point displacement portion of the OBE or SRV may require consideration in addition to those primary stress-producing loads listed.

Legend:

PD	-	Design Pressure
PO	-	Operating Pressure
DW	-	Dead weight
OBE	-	Operating basis earthquake (inertia portion)
SSE	-	Safe shutdown earthquake (inertia portion)
SRV_X	-	Loads due to Safety Relief Valve Blow - Axisymmetric or Asymmetric
SRV_{ADS}	-	Loads due to Automatic Depressurization SRV Blow-Axisymmetric
SBA	-	Small Break Accident
IBA	-	Intermediate Break Accident
DBA	-	Design Basis Accident
FV	-	Transient response of the piping system associated with fast valve closure. Transients associated with valve closure time less than 5 seconds are considered.
RVC	-	Transient response of the piping system associated with relief valve opening in a closed system.
RVO	-	Sustained load or response of the piping system associated with relief valve opening in an open system or last segment of the closed system with steady state load.

SBA, IBA, and DBA include all event induced loads, as applicable, such as chugging, condensation oscillation, pool swell, drag loads, annulus pressurization, etc.

For the NSSS load combination, see Table 3.9-2.

TABLE 3.9-7

DESIGN CRITERIA FOR ASME CODE CLASS 1 VALVES
(NON-NSSS)

Condition	Stress Limits
Design	NB-3521[1]
Normal and Upset	NB-3200 or NB-3500[1] (Standard Design Rules)
Emergency[2]	NB-3526[3]
Faulted[2]	NB-3527[3]

(1) As specified by ASME Code Section III, 1971 thru Winter 1972 Addenda.
(2) Where valve function must be ensured (active valve) during the emergency or faulted conditions, the specified emergency or faulted condition for the plant shall be considered the normal condition for the valve.
(3) As specified by ASME Code Section III, 1971, Winter 1973 Addenda.

TABLE 3.9-8

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 VESSELS DESIGNED TO NC-3300 AND ND-3300

Condition	Stress Limits[1][2]
Design and Normal	The vessel shall conform to the requirements of NC-3300 and ND-3300.
Upset, Emergency, and Faulted	The vessel shall conform to the requirements of ASME Code Case 1607-1.

(1) As specified by ASME Code Section III, 1971 thru 1972 Winter Addenda.

(2) For the Diesel Generator E Facility, the design criteria for class 3 vessels is governed by ASME code, Section III, 1980 edition through summer 1981 addenda.

TABLE 3.9-9

DESIGN CRITERIA FOR ASME CODE CLASS 2 VESSELS DESIGNED TO ALTERNATE RULES OF NC-3200

Page 1 of 2

Condition	Stress Limits[1][2]
Design and Normal	The vessel shall conform to the requirements of NC-3200.
Upset[3]	$P_e \le 3\ S_m$ $P_m \le 1.1\ S_m$ $(P_m \text{ or } P_L) + P_b \le 1.65\ S_m$
Emergency	$P_m \le$ greater of $1.2\ S_m$ or $1.0\ S_y$ $(P_m \text{ or } P_L) + P_b \le$ greater of $1.8\ S_m$ or $1.5\ S_y$
Faulted [4]	$P_m \le 2.0\ S_m$ $(P_m \text{ or } P_L) + P_b \le 2.4\ S_m$

(1) Definition of symbols:

P_m	=	General primary membrane stress intensity. This stress intensity is derived from the average value across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by pressure and other mechanical loads.
P_L	=	Local primary membrane stress intensity. Same as P_m except that discontinuities are considered.
P_b	=	Primary bending stress intensity. Component of primary stress intensity proportional to distance from centroid of solid section. Excludes discontinuities and concentrations. Produced only by pressure and other mechanical loads.
P_e	=	Secondary stress intensity range. Developed by constraint of adjacent parts or by self-constraint of a structure. Considers discontinuities but not concentrations. Produced by mechanical loads and by thermal expansion.
S_m	=	Design stress intensity value, Appendix I, Table I-1.0.
S_y	=	Yield strength value, Appendix I, Table I-2.0.

(2) These limits do not take into account either local or general buckling that might occur in thin-wall vessels. Such buckling shall be considered for upset conditions, but need not be considered for emergency or faulted conditions unless required by the design specification.

TABLE 3.9-9

DESIGN CRITERIA FOR ASME CODE CLASS 2 VESSELS DESIGNED TO ALTERNATE RULES OF NC-3200

Condition	Stress Limits(1)(2)

(3)	Fatigue analysis requirements of NC-3219 and Appendix XIV shall also be considered.
(4)	As an alternative to satisfying these limits, the faulted condition stress limits of Appendix F may be applied provided that a complete analysis in accordance with NC-3211.1(c) is performed.

TABLE 3.9-10

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 PIPING

Condition	Stress Limits[1]
Design, Normal, Upset, and Emergency [2]	The piping shall conform to the requirements of Section III, paragraphs NC-3600 and ND-3600.
Faulted[2]	The piping shall conform to the requirements of ASME Code Case 1606.

(1) As specified by ASME Code Section III, 1971 through 1972 Winter Addenda. For diesel generator 'E' facility piping on the auxiliary and air start skids the applicable code is ASME Section III, 1980 through Summer 1981 Addenda. See Table 3.2-3 for later code editions used for snubber elimination or other piping modifications.

(2) Functional capability of essential piping will be assured per Rodabaugh Criteria for emergency and faulted conditions only, Ref. 3.9-8.

TABLE 3.9-11

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 PUMPS

Condition	Stress Limits[2][3]
Design and Normal	The pump shall conform to the requirements of Section III, Paragraphs NC-3400 and ND-3400.
Upset, Emergency[1] and Faulted[1].	The pump shall conform to the requirements of ASME Code Case 1636-1.

(1) Where pump function must be ensured (active pumps) during the emergency or faulted condition, the pumps nozzle loads due to the specified emergency or faulted plant conditions shall be considered in satisfying the normal condition stress limits for the pump.

(2) As specified by ASME Code Section III, 1971 and Winter 1972 Addenda.

(3) For pumps mounted on the diesel generator 'E' auxiliary skid, the design criteria is governed by ASME Code, Section III, 1980 edition through Summer 1981 Addenda.

TABLE 3.9-12

DESIGN CRITERIA FOR ASME CODE CLASS 2 AND 3 VALVES

Condition	Stress Limits(2)(3)
Design and Normal	The valve shall conform to the requirements of Section III, Paragraphs NC-3500 and ND-3500.
Upset, Emergency(1) and Faulted(1)	The valve shall conform to the requirements of ASME Code Case 1635-1.

(1) Where valve function must be ensured (active valve) during the emergency or faulted condition, the specified emergency or faulted conditions for the plant shall be considered as the normal condition for the valve.

(2) As specified by ASME Code Section III, 1971 and Winter 1972 Addenda.

(3) For valves supplied on the diesel generator 'E' auxiliary and air start skids, the design criteria is governed by the ASME Code, Section III, 1980 edition through Summer 1981 Addenda.

TABLE 3.9-13

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TABLE 3.9-14

DESIGN LOADING COMBINATIONS FOR SUPPORTS FOR ASME CODE CLASS 1, 2, AND 3 COMPONENTS[3] (NON-NSSS)

Condition	Design Loading Combinations[1]	Allowable Stress*[2]
Hydrostatic Test	(a) HTDW	0.8 Sy*
Normal and Upset	(a) $DW + TH + (OBE^2 + SRV^2)1/2$ (b) $DW + TH + (RVC^2 + OBE^2)1/2$ (c) $DW + TH + FV$ (d) $DW + TH + OBE + RVO$	S_h
Emergency	(a) $DW + TH + (OBE^2 + SRV^2_{ADS} + SBA^2)1/2$ (b) $DW + TH + (OBE^2 + FV^2)1/2$	1.8 S_h
Faulted	(a) $DW + TH + (SSE^2 + SRV^2_{ADS} + IBA^2)1/2$ (b) $DW + TH + (OBE^2 + SRV^2_{ADS} + IBA^2)1/2$ (c) $DW + TH + (SSE^2 + DBA^2)1/2$	0.9 Sy

* Snubbers, compensating struts and struts comply with all the requirements of ASME Section III, Subsection NF. (They are not commercially available to the requirements of ANSI B31.1.)

NOTES:

(1) Loads due to OBE, SSE, SRV_x, SRV_{ADS}, SBA, IBA and DBA include both inertia portion and anchor motion portion when response spectra method is used. The loads from inertia portion and anchor motion are combined by the method of Square Root of the Sum of Squares (SRSS).

(2) The allowable stress shall be limited to two-thirds of the critical buckling stress.

(3) Supports on the diesel generator E and air start skids comply with all the requirements of ASME, Section III, Subsection JF, 1980 edition through Summer 1981.

Legend:
- HTDW - Piping dead weight due to hydrostatic test
- TH - Reaction at the support due to thermal expansion of the pipe
- Sy - Yield stress
- S_h - Allowable stress per ANSI B31.1

See Table 3.9-6 for additional nomenclature.
See Table 3.9-2 for NSSS support.

TABLE 3.9-15

VALVE QUALIFICATION TEST RANGE

Size of Qualified Valve	Qualification Extends To These Valve Sizes																			
	.5	1	1.5	2	3	4	6	8	10	12	14	16	18	20	22	24	26	28	30	36
.5	X	X																		
1	X	X	X																	
1.5		X	X	X																
2			X	X	X															
3				X	X	X														
4					X	X	X													
6						X	X	X												
8							X	X	X	X										
10								X	X	X	X									
12								X	X	X	X	X								
14									X	X	X	X	X	X						
16										X	X	X	X	X	X					
18											X	X	X	X	X	X				
20											X	X	X	X	X	X	X	X		
22												X	X	X	X	X	X	X	X	
24													X	X	X	X	X	X	X	
26														X	X	X	X	X	X	X
28														X	X	X	X	X	X	X
30															X	X	X	X	X	X
36																	X	X	X	X

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Table 3.9-18

SUMMARY COMPARISON – PROJECT SPECIFICATOIN – 10, "GENERAL PROJECT REQUIREMENTS FOR A SEISMIC DESIGN AND ANALYSIS OF CLASS 1 EQUIPMENT AND SUPPORTS"*

SPEC. G-10	IEEE-344-1975	REMARKS
1. Analysis		
Equipment is classified as (1) structurally simple and (2) structurally complex. Structurally simple equipment is one which can be adequately represented as a single degree of freedom system or equipment whose fundamental frequency is greater than 33 cps. Otherwise the equipment is structurally complex.	There are two methods of analysis; one approach is based upon equivalent static analysis and the other on dynamic analysis. For the static coefficient analysis, no determination of natural frequency is made but the response of equipment is assumed to be the peak of the RRS at a conservative value of damping multiplied by a static coefficient of 1.5 to take into account the effects of both multifrequency excitation and multifrequency response. A lower value of static coefficient can be used, if it can be shown to yield conservative results.	Reg. Guide 1.100 Rev. 1 of August 1977 indicates that the static coefficient of IEEE-344-1975 of 1.5 is acceptable for verifying structural integrity of frame type structures such as columns, beams that can be represented by a simple model. For equipment having configurations other than frame type structure justification should be provided for the use of static coefficient.
For equipment which is structurally simple due to single degree of freedom system the seismic load consists of a static load corresponding to the equipment weight times the acceleration selected from the response spectrum curve for the natural frequency of the equipment. If the equipment frequency is not known the acceleration shall correspond to the maximum value of the response spectrum.	For the dynamic analysis the equipment shall be modeled to best represent its mass distribution and stiffness characteristics, and this model is used to determine if the equipment is rigid or flexible. If there is no response in the frequency range below the high-frequency asymptote (ZPA) of the RRS, it is considered rigid. Then the seismic forces on each component of the equipment are obtained by concentrating its main at its center of the gravity and multiplying the values of main and the appropriate maximum floor acceleration. If flexible, the model can be analyzed using response spectrum model analysis technique or time history analysis. The response of interest is determined by combining each model response considering all significant modes by SRSS. The absolute sum of similar effects should be considered for closely spaced modes which are those with frequencies differing by 10% or less. In the analysis the effects of each of the two major horizontal directions and the vertical direction should be considered.	
For equipment which is structurally simple due to fundamental frequency greater than 33 cps the seismic load shall consist of a static load corresponding to the acceleration at 33 cps selected from appropriate response spectrum curve with an increase of 50%.		

* The siesmic design of the deisel generator 'E' facility confirms to project specification C-1041 or Cooper Energy Services Standard No. DS-140 and IEEE Standard 344-75 in lieu of project specification G-10.

Table 3.9-18

SUMMARY COMPARISON – PROJECT SPECIFICATOIN – 10, "GENERAL PROJECT REQUIREMENTS FOR A SEISMIC DESIGN AND ANALYSIS OF CLASS 1 EQUIPMENT AND SUPPORTS"*

SPEC. G-10	IEEE-344-1975	REMARKS
For equipment which is structurally complex for the analysis purposes, the equipment shall be idealized by a lumped mass model. Frequencies and mode shapes are determined for the vertical, and two orthogonal horizontal directions. Spectral acceleration per mode shall be obtained from appropriate response spectrum curves, the value chosen to be the largest value on the curve when the frequency is varied by ±0%. The results of the individual modes shall be combined by the square root of sum of the squares method. For closely spaced modes which have frequencies that do not differ by more than 10% the responses of all these modes are combined by sum of the absolute values before combining with other modes by SRSS method.		
2. Damping		
The damping values specified are 1/2 % for OBE and 1% for SSE in the response curves attached with the material requisition and equipment specifications. If there is evidence (such as test results) of different damping values these can be used.	The allowable damping values for the equipment are 2% for the OBE and 3% for SSE. Damping values higher than these can be used if justified by documented test data. If equipment damping is not known a value of 5% is recommended.	Reg. Guide 1.61 for equipment also allows damping values of 2% for OBE and 4% for SSE and higher values are permitted in a dynamic seismic analysis for documented test data are provided.
3. Testing		
Testing for both OBE and SSE loads must be done unless it can be shown for a particular item that the SSE is a more severe condition than the OBE.	Seismic qualifications tests designed to show adequacy of performance during and following a SSE must be preceded by one or more OBE test. The number of tests shall be justified for each site or shall produce the equivalent effect of 5 OBE's.	
4. Testing		
Perform frequency sweep at line amplitude acceleration input varying frequency (at sweep rate of 1/6 of forcing frequency per minute) and determine all resonant frequencies below 33 cps and first resonant frequency above 33 cps if below 65 cps.	Exploratory test may be run in the form of low level continuous sinusoidal sweep (such as 0.2g) at a rate no greater than 2 octaves per minute over the frequency range equal to or greater than that to which the equipment is to be qualified.	

Table 3.9-18

SUMMARY COMPARISON – PROJECT SPECIFICATOIN – 10, "GENERAL PROJECT REQUIREMENTS FOR A SEISMIC DESIGN AND ANALYSIS OF CLASS 1 EQUIPMENT AND SUPPORTS"*

SPEC. G-10	IEEE-344-1975	REMARKS
5. Testing		
Input acceleration, a, for each of horizontal and vertical directions at each reasonant (frequency is determined by, a = 1.5 x 1-(ff/fe) – Sa where, SA is spectral response acceleration for equipment frequency fe, and ff is the forcing frequency of the shaking device (ff 0.8 fe). The factor 1.5 is used to account for the possible excitation of other modes.	The maximum acceleration of shake table should be at least equal to ZPA on RRS. For equipment with more than one predominant frequency the shake motion should provide TRS acceleration at the test frequency of 1.5 times that of RRS or less if justified. The choice of the preceding factor (with largest values of 1.5) is applicable to broadband RRS. As a consequence the TRS need not envelope RRS provided proper justification is given.	Reg Guide 1.100 indicates that the use of factor 1.5 in IEEE-344, and the concept that TRS need not envelope the RRS as a consequence of using 1.5 should not, in the absence of justification, be considered acceptable.
6. Testing		
Duration of excitation of the input acceleration shall not be less than 30 seconds.	The duration of each test shall be least equal the strong motion portion of the original time history used to obtain RRS for the SSE.	The duration of design earthquake for the Susquehanna project is 20 seconds.

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TABLE 3.9-19

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TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
Main Steam	ASME III - 2 B31.1 SC II HE	Yes	Yes	Yes	Yes	Main Stop Valve Closure and SRV Opening Transients
Extraction Steam	B31.1 SC II HE	Yes	N/A(5)	N/A	N/A	
Condensate & Refueling Water Storage	B31.1 SC II ME	No	N/A	N/A	N/A	
Feedwater	ASME III - 1,2 B31.1 HE, SC I SC II	Yes	Yes	Yes	Yes	Power Ascension Test for Safety Related Piping Portion Only
Air Removal and Seal Steam	B31.1 SC II HE	Yes	N/A	N/A	N/A	
Service Water	ASME III - 2,3 B31.1 SC I & SC II ME	No	N/A	N/A	N/A	A Portion of the System has Temperature >200°F But is Less than 300°F
Raw Water Treatment	B31.1 SC II ME	No	N/A	N/A	N/A	
Lube Oil & Diesel Oil Storage & Transfer	ASME III - 3 B31.1 SC I & II ME	No	N/A	N/A	N/A	
Auxiliary Steam	B31.1 SC II, HE	Yes	N/A	N/A	N/A	

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
Fire Protection	SC II, ME	No	N/A	N/A	N/A	
Process Sampling	B31.1. SC II, ME	No	N/A	N/A	N/A	
Chlorination	B31.1, SC II ME	No	N/A	N/A	N/A	
Compressed Air	B31.1, SC II ME	No	N/A	N/A	N/A	
Instrument Gas	ASME III - 2,3 B31.1, SC I SC II, ME	No	N/A	N/A	N/A	
Feed Pump Turbine Steam	B31.1, SC II HE	Yes	N/A	N/A	N/A	
Makeup Water	B31.1, SC II ME	No	N/A	N/A	N/A	
Valve Steam Leakoff	B31.1 SC II HE	Yes	N/A	N/A	N/A	
Acid Injection	B31.1, SC II ME	No	N/A	N/A	N/A	
Hydrogen Storage	B31.1, SC II, ME	No	N/A	N/A	N/A	
Diesel Engine Auxiliaries	ASME III - 3 B31.1, SC I & II, ME	Yes	Yes(6)	N/A	N/A	Emergency Diesel Exhaust Has T >300°F and Thermal Expansion Test Performed
				(See Remarks)		
MSIV Leakage Control	ASME III - 1,2 B31.1, SC I SC II, HE	Yes	N/A	N/A	N/A	
Reactor Recirc Motor/Generator	B31.1, SC II ME	No	N/A	N/A	N/A	

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
High Pressure Coolant Injection	ASME III - 1,2 B31.1, SC I SC II, HE, ME	Yes	Yes	Yes	Yes	HPCI Turbine Stop Valve Closure Transients for Steam Supply, Steady State Vibration for Steam Supply and Turbine Exhaust HPCI Pump Suction and discharge lines under steady state vibration.
Reactor Core Isolation Cooling	ASME III - 1,2 B31.1, SC I SC II, HE, ME	Yes	Yes	N/A	Yes	Steady State Vibration for RCIC Steam Supply and Turbine Exhaust RCIC pump suction and discharge piping under steady state vibration test
Reactor Water Cleanup	ASME III - 1,2 B31.1, SC I SC II, HE, ME	Yes	Yes	N/A	Yes(6)	Steady State Vibration for RWCU Line Inside Containment
Residual Heat Removal (Includes Head Spray)	ASME III - 1,2,3 B31.1, SC I & II, HE, ME	Yes	Yes	N/A	Yes(6)	Majority of the System has Normal Operating Temperature less than 300°F. Thermal Expansion Tests are done for SCI Systems WAA T >300°F. Steady State Vibration for Inside Containment Piping and RHR Pump Discharge.
				(See Remarks)		
Cleanup Filter Demineralizer	ASME III - 2 B31.1, SC II ME	No	N/A	N/A	N/A	
Control Rod Drive	ASME III - 2 B31.1, SC I, SC II, ME	No	N/A	N/A	Yes(6)	CRD insert/withdrawal pipe.
Standby Liquid Control	ASME III - 1,2 B31.1, SC I & II, HE, ME	No	N/A	N/A	N/A	Only a small portion of the Line near RPV has Temperature >200°F.
				(See Remarks)		
Core Spray	ASME III - 1,2 B31.1, SC I & II HE, ME	Yes	Yes(7)	N/A	Yes(6)	Steady State Vibration For Core Spray Pump Discharge.

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BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks
Fuel Pool Cooling, Cleanup & Demineralizer	ASME III - 3 B31.1, SC I SC II, ME	No	N/A	N/A	N/A	
Containment Atmospheric Control	ASME III - 2 B31.1, SC I & II ME	No	N/A	N/A	N/A	
Solid Radwaste and Radwaste Solidification	B31.1 SC II ME	No	N/A	N/A	N/A	
Off-Gas Recombiner	ASME III - 3 B31.1, SC I SC II, HE	Yes	N/A	N/A	N/A	
Ambient Temperature Charcoal Off-Gas Treatment	B31.1, SC II ME	No	N/A	N/A	N/A	
Chilled Water	ASME III - 3 B31.1, SC I SC II, ME	No	N/A	N/A	N/A	

TABLE 3.9-20

BOP PIPING SYSTEMS POWER ASCENSION TESTING

Piping System	Code(s) SC/HE/ME (1)	Temp. > 200°F	Thermal Expansion Test (2)	Dynamic Transient Test (3)	Steady State Vibration Test (4)	Remarks

NOTES: (1) Code(s): ASME III Boiler and Pressure Vessel Code, -1, -2, or -3 Denotes Nuclear Class 1, 2 or 3 Piping.
SC I or II Denote Seismic Category I or II
HE: Denotes High Energy Piping System i.e. Pressure ≥275 PSI or Temperature ≥200° During Normal Plant Operation
ME: Denotes Moderate Energy Piping System

(2) Thermal Expansion Tests for the indicated systems corresponds to test description ST-38, Chapter 14.

(3) Dynamic Transient Tests for the indicated systems corresponds to test description ST-39, Chapter 14.

(4) Steady State Vibration Tests for the indicated systems corresponds to test description ST-40, Chapter 14.

(5) N/A - Denotes Not Applicable and it means test is not performed for the reasons given below:

A) For Thermal Expansion Tests: Either the system is not safety-related or the normal operating temperature is less than 300°F.
B) For Dynamic Transient Test: Either the system is not safety-related or the system does not experience any significant transients.
C) For Steady-State Vibration Tests: Either the system is not safety-related or no significant vibration is expected.

(6) Test may be done during Peroperational Test Program.

(7) For the effect of RPV expansion only. No flow in the Core Spray line.

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TABLE 3.9-2a

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS -(psi)	MAXIMUM CALCULATED STRESS (psi)
(i) VESSEL SUPPORT SKIRT				
MATERIAL: SA-516 GR-70				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 19,150 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 28,725 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	19,150 28,725	14,566 U1 < 14,566 U2 20,330 U1 < 20,330 U2
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 29,425 @ 546°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 44,138 @ 546 °F	1. Deadweight 2. Pressure 3. DBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	29,425 44,150	20,565 U1 < 20,565 U2 29,377 U1 < 29,377 U2
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 45,960 @ 546°F $P_m + P_b \leq 3.6$ Sm 3.6 Sm = 68,940 @ 546°F	1. Deadweight 2. Pressure 3. Annulus Pressurization 4. SSE 5. LOCA	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	45,960 68,940	36,410 60,570
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.913 U1 and 0.888 U2 At Skirt Base Junction These CUFs account for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS -(psi)	MAXIMUM CALCULATED STRESS (psi)
(ii) SHROUD SUPPORT				
MATERIAL: INCONEL SB-168				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 23,300 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 34,950 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	23,300 35,000	7,600 11,600
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 28,125 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 42,188 @ 575°F	1. Deadweight 2. Pressure 3. SBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	35,000 54,400	7,600 11,600
C. FAULTED CONDITION:				
$P_m \leq Sy$[1] Sy = 28,125 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 42,188 @ 575°F	1. Deadweight 2. Pressure 3. Annulus Pressurization 4. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	35,000 52,400	15,300 23,400
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.41 At Vessel - Support Plate Junction. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

(1) Emergency allowable values used.

TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(iii) RPV FEEDWATER NOZZLE				
MATERIAL: SA 508 CL. I.				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 17,700 @ 525°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 26,550 @ 525°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	17,700 26,550	14,200 19,700
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 25,900 @ 594°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 38,850 @ 594°F	1. Deadweight 2. Pressure 3. SBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	26,500 39,800	15,600 20,800
C. FAULTED CONDITION:				
$P_m \leq 3.0$ Sm 3.0 Sm = 53,100 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 38,850 @ 594°F	1. Deadweight 2. Pressure 3. LOCA 4. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	53,100 39,800	28,300[2] 20,800
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.82 At Safe End. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

(2) Without Thermal Bending.

TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(iv) IV CRD PENETRATION - CRD HOUSING				
MATERIAL: TYPE 304SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 16,600 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 24,990 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	16,600 24,900	8,506 8,480
B. EMERGENCY CONDITION:				
$P_m \leq 1.2$ Sm 1.2 Sm = 20,000 @ 575°F $P_m + P_b \leq 1.8$ Sm 1.8 Sm = 30,000 @ 575°F	1. Deadweight 2. Pressure 3. SRV 4. SBA	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	19,920 29,880	12,470 12,710
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 39,984 @ 575°F $P_m + P_b \leq 3.6$ Sm 3.6 Sm = 59,975 @ 575°F	1. Deadweight 2. Pressure 3. LOCA 4. SCRAM 5. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	39,840 59,760	12,470 12,710
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.208 at Lower CRD Housing. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

TABLE 3.9-2a (Continued)

REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS -(psi)	MAXIMUM CALCULATED STRESS (psi)
(v) CRD PENETRATION - STUB TUBE				
MATERIAL: SB-167 INCONEL				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 20,000 @ 575°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 30,000 @ 575°F	1. Deadweight 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	20,000 30,000	5,005 28,200
B. EMERGENCY CONDITION:				
$P_m \leq Sy$ Sy = 24,100 @ 575°F $P_m + P_b \leq 1.5$ Sy 1.5 Sy = 36,150 @ 575°F	1. Deadweight 2. Pressure 3. SBA 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	24,100 36,150	6,755 30,260
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 48,000 @ 575°F $P_m + P_b \leq 3.6$ Sm 3.6 Sm = 72,000 @ 575°F	1. Deadweight 2. Pressure 3. LOCA 4. SSE	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	48,000 72,000	6,755 30,260
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR: 0.36 At Bottom of Weld. This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.				

TABLE 3.9-2b

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME B&PV CODE SEC. III SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(i) TOP GUIDE - BEAM WITH HIGHEST STRESS				
MATERIAL: SA-240 TYPE 304SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 16,900 @ 550°F	1. Normal 2. Pressure	PRIMARY MEMBRANE	16,950	2,770
$P_m + P_b \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F	3. OBE 4. SRV	PRIMARY MEMBRANE PLUS BENDING	25,430	11,500
B. EMERGENCY CONDITION:				
$P_m \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F	1. Normal 2. Pressure	PRIMARY MEMBRANE	25,430	730
$P_m + P_b \leq 2.25$ Sm 2.25 Sm = 38,025 @ 550°F	3. Chugging 4. SRV	PRIMARY MEMBRANE PLUS BENDING	38,140	9,600
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 40,560 @ 550°F	1. Normal 2. Pressure	PRIMARY MEMBRANE	40,680	3,500
$P_m + P_b \leq 3$ Sm 3 Sm = 50,7000 @ 550°F	3. SRV 4. SSES 5. LOCA	PRIMARY MEMBRANE PLUS BENDING	61,020	11,800
D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.22 At Beam-Slot Location This CUF was evaluated for the period of extended operation and found to be acceptable.				

TABLE 3.9-2b (Continued)

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME B&PV CODE SEC. III SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(ii) CORE PLATE (LIGAMENT IN TOP PLATE)				
MATERIAL: TYPE 304SS				
A. NORMAL & UPSET CONDITION:				
$P_m \leq Sm$ Sm = 16,900 @ 550°F $P_m + P_b \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F	1. Normal 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	16,900 25,470	783 10,140
B. EMERGENCY CONDITION:				
$P_m \leq 1.5$ Sm 1.5 Sm = 25,350 @ 550°F $P_m + P_b \leq 2.25$ Sm 2.25 Sm = 38,025 @ 550°F	1. Normal 2. Pressure 3. Chugging 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	25,350 38,200	647 [(1)] 12,460
C. FAULTED CONDITION:				
$P_m \leq 2.4$ Sm 2.4 Sm = 40,560 @ 550°F $P_L + P_b \leq 3$ Sm 3 Sm = 50,700 @ 550°F	1. Normal 2. Pressure 3. SRV 4. SSE 5. LOCA	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	40,560 61,020	9,570 [(1)] 15,710
D. MAXIMUM CUMULATIVE USAGE FACTOR: 0.005 At Beam-Rim Junction. This CUF was evaluated for the period of extended operation and found to be acceptable.				
(1) Value is not given in New Loads Report. However, Primary and Bending Stresses are, and these are bounding.				

TABLE 3.9-2b (Continued)

REACTOR INTERNALS AND ASSOCIATED EQUIPMENT

ASME B&PV CODE SEC. III SUBSECTION NG PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
(ii) VESSEL HEAD SPRAY NOZZLE				
MATERIAL: CARBON STEEL SA508,CL1				
A. NORMAL & UPSET CONDITION:				
$P_m \leq$ * Sm = 16,670 @ 575°f $P_L + P_b \leq 3.5$ Sm 1.5 Sm = 50,000 @ 575°F	1. Normal 2. Pressure 3. OBE 4. SRV	PRIMARY MEMBRANE PRIMARY MEMBRANE PLUS BENDING	50,000	28,700
B. EMERGENCY CONDITION:				
$P_L + P_b \leq 1.8$ Sm 1.8 Sm - 30,000 @ 575°F	1. Normal 2. Pressure 3. Chugging 4. SRV	PRIMARY MEMBRANE PLUS BENDING	30,000	28,700
C. FAULTED CONDITION:				
$P_L + P_b \leq 3$ Sm 3 Sm = 50,000 @ 575°F $P_L + P_b \leq 3$ Sm 3 Sm = 50,7000 @ 550°F	1. Normal 2. Pressure 3. SRV 4. SSE 5. LOCA	PRIMARY MEMBRANE PLUS BENDING	50,000	28,700
D. MAXIMUM CUMULATIVE USAGE FACTOR: Satisfied per N-415-1				

* Not required per NB-3222.

TABLE 3.9-2(c)

REACTOR WATER CLEANUP HEAT EXCHANGERS

Part	Required Thickness (in.)	Allowable Stresses (psi)	Actual Thickness (in.)
REGENERATIVE CU HX			
Shell	.858	15,000	1.156
Shell head	.704	17,500	1.0
Channel shell	.917	15,900	1.0*
Tubesheet	2.87	15,900	2.875*
Tubes	.084	14,025	.095
Piping	.240	15,000	.337
Channel cover	3.53	17,500	3.75*
NON-REGENERATIVE CU HX			
Shell	.168	15,000	.375
Shell head	.144	17,500	.375
Channel shell	.917	15,900	1.0*
Channel cover	3.53	17,500	3.75*
Tubesheet	2.87	15,900	2.875*
Tubes	.056	11,900	0.65
Channel piping	.240	15,000	.337
Shell piping	.073	15,000	.322

Values within 10%

Table 3.9-2d

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT – HIGHEST STRESS SUMMARY – UNIT 1

ACCEPTANCE CRITERIA	L MIT NG STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENT FICATION OF LOCATIONS OF HIGHEST STRESS POINTS – NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq.9 $\leq$ 1.5 S_m	Primary	26,473 psi	26,550 psi	0.997	1. Pressure 2. Weight 3. OBE	M.S. Line ‘A’ Sweetpolet 543
Service Levels A&B (Normal & Upset) Conditions: Eq.12 $\leq$ 3.0 S_m	Secondary	27,403 psi	53,100 psi	0.516	1. Thermal	M.S. Line ‘B’ Sockolet 266
Service Levels A&B (Normal & Upset) Condition: Eq.13 $\leq$ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	52,566 psi	54,252 psi	0.969	1. Pressure 2. Weight 3. OBE 4. Operating transient 5. SRV	M.S. Line ‘D’ Sweepolet 850
Service Levels A&B (Normal & Upset) Conditions: Cumulative Usage Factor	N/A	0.8497*	1.0	0.8497	1. Pressure 2. Thermal 3. OBE 4. Operating Transient 5. SRV	M.S. Line ‘D’ TTJ @ Sweepolet 990
Service Level B (Upset) Condition: Eq.9 $\leq$ 1.8 S_m & 1.5S_y	Primary	27,770 psi	31,860 psi	0.872	1. Pressure 2. Weight 3. OBE 4. Operating transient 5. SRV	M.S. Line ‘D’ Sweepolet 850
Service Level C (Emergency) Condition: Eq.9 $<$ 2.25 S_m & 1 8 S_y	Primary	28,432 psi	39,820 psi	0.714	1. Pressure 2. Weight 3. OBE 4. Operating transient 5. SRV 6. SBA	M.S. Line ‘D’ Sweepolet 880
Service Level D (Faulted) Condition: Eq. 9 $<$ 3.0 S_m	Primary	57,585 psi	60,000 psi	0.960	1. Pressure 2. Weight 3. SSE 4. LOCA 5. Operating Transient 6. IBA 7. SRV	M.S. Line ‘C’ Sweepolet 620

Table 3.9-2d (Continued)

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT – UNIT 1

COMPONENT LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ ALLOWABLE	LOAD NG	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Snubber/ Service Level 'B' Loads	27,693	50,000	0.554	1. Pressure 2. Weight 3. OBE 4. Operating transient	M.S. Line 'A' (S035) H35
Snubber/ Service Level 'C' Loads	35,798	159,600	0.224	1. Pressure 2. Weight 3. SBA 4. Operating transient	M.S. Line 'D' (S038) H38
Snubber/ Service Level 'D' Loads	58,063	160,000	0.363	1. Pressure 2. Weight 3. IBA 4. SSE 5. Operating transient	M.S. Line 'D' (S038) H38
SRV/Horizontal Acceleration	3 981 g	5.2g	0.766	1. Pressure 2. Weight 3. LOCA 4. SSE	Line C SRR Inlet
SRV/Vertical Acceleration	2 650 g	4.4 g	0.602	1. Pressure 2. Weight 3. LOCA 4. SRV	Line B SRP Inlet
MSIV Bonnet/ Axial Force	7,682 lb	41,221 lb	0.186	1. Pressure 2. Weight 3. LOCA 4. SSE	Line A MSIV Bonnet
MSIV Bonnet/ Bending Moment	1,242,707 in-lbs	1,589,000 in-lb	0.782	1. Pressure 2. Weight 3. LOCA 4. SSE	Line B MSIV Bonnet

* This CUF accounts for the original 40-year plant design. During the period of extended operation actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.

Table 3.9-2e

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT – HIGHEST STRESS SUMMARY – UNIT 1

ACCEPTANCE CRITERIA	L MITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOAD NG	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS - NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq. 9 ≤ 1.5 S_m	Primary	24,097 psi	25,013 psi	0.963	1. Pressure 2. Weigh 3. OBE	Loop 'B' E bow-784
Service Levels A & B (Normal & Upset) Conditions: Eq. 12 ≤ 3.0 S_m	Secondary	36,019 psi	50,025 psi	0.720	1. Thermal	Loop 'B' Weldolet 508
Service Levels A & B (Normal & Upset) Condition: Eq. 13 ≤ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	49,544 psi	50,025 psi	0.990	1. Pressure 2. Weight 3. OBE 4. Operating Transient 5. SRV	Loop 'B' TTJ@Valve End 782
Service Levels A & B (Normal and Upset) Conditions: Cumulative Usage Factor	N.A.	0.5555**	1.0	0.555	1. Pressure 2. Thermal 3. OBE 4 .Operating Transient 5. SRV	Loop 'B' Half Coupling 727
Service Level B (Upset) Condition: Eq. 9 ≤ 1.8 S_m & 1.5 S_y	Primary	25,298 psi	27,750 psi	0.912	1. Pressure 2. Weight 3. OBE 4. Operating Transient 5. SRV	Loop 'B' E bow 784

Table 3.9-2e

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT – HIGHEST STRESS SUMMARY – UNIT 1

ACCEPTANCE CRITERIA	L MITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOAD NG	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS - NODG POINT NUMBERS
Service Level C (Emergency) Condition: Eq. 9 < 2.25 S_m & 1.8 S_y	Primary	28,801 psi	33,300 psi	0.865	1. Pressure 2. Weight 3. OBE 4. Operating Transient 5. SRV 6. SBA	Loop 'B' Weldolet 508
Service Level D (Faulted) Condition: Eq. 9 < 3.0 S_m	Primary	41,019 psi	50,025 psi	0.820	1. Pressure 2. Weight 3 .LOCA 4. Operating Transient 5. SSE 6. IBA 7. SRV	Loop 'B' E bow 784

Table 3.9-2e (Continued)

ASME CODE CLASS 1 RECIRCULATION AND PIPE MOUNTED EQUIPMENT – UNIT 1

COMPONENT/LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQU PMENT WITH HIGHEST LOADS
Snubber/ Service Level 'B' Loads	56,085	120,000	0.467	1. Pressure 2. Weight 3. OBE 4. Operating transient	Loop 'A' (SAI) H46
STRUT Service Level 'C' Loads	68,094	87,840	0775	1. Pressure 2. Weight 3. SBA 4. Operating transient	Loop 'A' (PO1) H50
STRUT Service Level 'D' Loads	94,493	121,896	0.775	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' (PO1) H50
Discharge Gate Valve/ Horizontal Acceleration	7.296	9.8	0.744	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'B' Operator
Discharge Gate Vale/ Vertical Acceleration	2.2	2.2	1.0	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
By-Pass Valve/ Horizontal (East/West) Acceleration	4.336	4.336	1.0	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
By-Pass Valve/ Horizontal (North South) Acceleration	2.605	2.605	1.0	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
By-Pass Valve/ Vertical Acceleration	2.514	2.62	0.960	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
Suction Gate Valve/ Horizontal & Vertical Acceleration*	5.24	9.87	0.531	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
Recirc. Pump/ Horizontal Acceleration	2.338	2.7	0.866	1. Pressure 2. Weight 3. LOCA 4. SSE	Loop 'A' Pump Motor C.G.
Recirc. Pump/ Vertical Acceleration	1.004	2.1	0.478	1. Pressure 2. Weight 3. LOCA 4. SSE 5. SRN	Loop 'B' Pump Motor C.G.

* Allowable acceleration applies to operator only; valve body is qualified based upon allowable moment and allowable axial force.

** This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.

TABLE 3.9-2 (f)

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TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	Unless otherwise specified, all references are to ASME B and PVC Section III (July 1971).		
1. Body Inlet and Outlet Flange Stresses	Based on ANSI B31.7-1969, PARA. 1-704.5.1 & ASME BPVC Criteria		
	$S_H = \frac{fM_o}{Lg^2B} + \frac{P_bB}{4g_1} < 1.5\ S_m$		
	$S_R = \frac{(4/3te + 1)M_o}{IT^2J} < 1.5\ S_m$		
	$S_T = \frac{YM_o}{t^2J} - Z\ S_R < 1.5\ S_m$		
	Where		
	S_H = Longitudinal "Hub" Wall Stress, PSI S_R = Residual "Flange" (Body Base, Inlet) Stress, PSI S_T = Tangential "Flange" Stress, PSI		
	For Inlet		
	As S_H < 27,300 PSI (1.5 S_m @ 575°F for A-105GII)	27,300 PSI	See Note 1
	S_R<27,300 PSI (1.5 S_m)	27,300 PSI	See Note 1
	S_T< 27,300 PSI (1.5 S_m) Criteria Satisfied	27,300 PSI	See Note 1
	For Outlet		

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	As S_H < 27,300 PSI (1.5 S_m)	29,100 PSI	See Note 1
	S_R <27,300 PSI (1.5 S_m)	29,100 PSI	See Note 1
	S_T < 27,300 PSI (1.5 S_m)	29,100 PSI	See Note 1
2. Inlet and Outlet Stud Area Requirements	Based on USAS B31.7-1969, PARA. 1-704.5.1 and ASME BPVC Total Cross-Sectional Area of Bolt Shall Equal or Exceed the Greater of $AM_1 = \frac{Wm_1}{S_b}$ or $AM_2 = \frac{WM_2}{S_a}$ Where: AM_1 = The total required bolt (stud) area for operating condition, in^2. AM_2 = Total required bolt (stud) area for gasket seating, in^2. For Inlet		
	AM_1 = 9,060 in^2 > AM_2 = 0.807 in^2 As AM_1 = 9,060 in^2 (Am, actual stud area Criterion Satisfied)	9.060 in^2	See Note 1

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	For Outlet		
	$Am_1 = 4.76\ in^2 > Am_2 = 1.55\ in^2$		
	As $Am_1 = 4.76\ in^2$	$4.76\ in^2$	See Note 1
	(Am, actual stud area Criterion Satisfied)		
3. Nozzle Wall Thickness	Based on PARA. NB-3530, NB-3540 and NB-3550 for non-standard primary pressure rating.		
	1. Valve Wall Thickness Criterion		
	$t_{min.} < t_A$		
	Where $t_{min.}$ = Minimum Calculated Required Thickness With Corrosion Allowable, in.		
	t_A = Actual Nozzle Wall Thickness, in.		
	i) For thinnest section near valve seat		
	As $t_{min.}$ = 0.467 in. (t_a, actual)	0.467 in.	See Note 1.
	ii) For section about mid-section of nozzle		
	As $t_{min.}$ = 0.468 in.. (t_a, actual)	0.468 in.	See Note 1.

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	2. Cyclic Rating (NB-3550) i) The thermal cyclic index criterion $I_t = \Sigma \frac{N_{ri}}{N_i}$ As $I_t = 0.032 < 1$ Criterion Satisifed ii) The fatigue requirement Criterion Na $\leq$ 2000 cycles $S_{P1} = \frac{2}{3} Q_P + \frac{P_{eb}}{2} + Q_{t2} + 1.3Q_{t1}$ $S_{P2} = 0.4\ Q_P + \frac{K}{2} (P_{eb} + Q_{t2})$ Where: S_{P1} = The fatigue stress intensity at the inside surface of the crotch, PSI S_{P2} = The fatigue stress intensity at the outside surface of the crotch, PSI S_{P1} = 9,130 PSI > S_{P2} = 545 PSI The permissible number of startup/shutdown cycles. NA > 10^6 > 2,000 cycles Criterion Satisfied		

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
4. Bonnet	Based on Nuclear Power Piping Code USAS B31.7-1969, PARA. 1-704.5 1. Body to Bonnet Stud Stress Criterion Total Cross-Sectional Area of Bolt Shall Equal or Exceed the Greater of $Am_1 = \frac{Wm_1}{S_b}$ or $Am_2 = \frac{Wm_2}{S_a}$ $Am_1 = 5.19$ in.$^2 > AM_2 = 0.849$ in.^{2}		
	As $AM_1 = 5.19$ in.^{2}	5.19 in.^{2}	See Note 1
	(Am, actual stud area criterion satisfied) 2. Bonnet Flange Strength (B31.7-1969, PARA. 1-704.5.1) Criterion $S_H < 1.5\ S_m$ $S_R < 1.5\ S_m$ $S_T < 1.5\ S_m$		
	As $S_H < 29{,}100$ psi (15 S_m @ 500°F A105 GI	29,100 PSI	See Note 1
	$S_R < 29{,}100$ psi	29,100 PSI	See Note 1
	$S_T < 29{,}100$ psi (1.5 S_m)	29,100 PSI	See Note 1

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
5. DISC Insert	Bending Stress of Disc Insert per Roark's formulas for stress and strain, 4th Edition pg. 222-223 superposition of case no. 21 and case no. 22 for flat plates. Alternatively, finite element analysis may be used to calculate stress. $S_t = S_{r_1} + S_{r_2} = \frac{3W_1}{4t^2_{avg}} \frac{4a^4(m+1)\ln(a/b) - a^4(m+3) + b^4(m-1) + 4}{a^2(m+1) + b^2(m-1)}$ $+ \frac{3W_2}{2\pi t^2} \frac{2a^2(m+1)\ln(a/b) + a^2(m-1) - b^2(m-1)}{a^2(m+1) + b^2(m-1)}$ Where: W_1 = Pressure, psig W_2 = Seat Load, lbs. a = ½ Disc. Insert Outside Dia., 2.782 in. b = Hub Radius 0.592 in. t = Average Thickness, 1.022 in. S_{r1} = Stress Due to Pressure, psi S_{r2} = Stress Due to Seat Load, psi m = Reciprocal of poisson ratio, 3.333 1. The normal operating condition of 80% of design pressure. S_t < 45,225 psi (S_m @ 575°F for ASME SA-637 TYPE 718) Criterion Satisfied Where W_1 = 1,000 psi W_2 = 5,422 lbs.	45,225 PSI	See Note 1

TABLE 3.9-2g

SAFETY/RELIEF VALVES (MAIN STEAM)

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED	CALCULATED STRESS OR ACTUAL THICKNESS
	2. The maximum possible full flow pressure at 10% above design pressure (the stress is entirely due to pressure).		
	S_t < 45,225 psi (S_m) Criterion Satisfied	45,225 PSI	See Note 1
	Where W_1 = 1,375 psi W_2 = 0		
	3. The Set Spring Load, Zero Inlet Pressure, Load Temperature (The Stress is Entirely Due to Seat Load, Zero Inlet Pressure).		
	S_t < 50,000 psi (S_m @ Room Temperature of ASME SA-637 TYPE 718)	50,000 PSI	See Note 1
	Where W_1 = 0 W_2 = 27,110 psi		

Note 1: The calculated stress was determined by calculation to be less than the allowable stress. The actual thickness is greater than the minimum thickness required as determined by calculation. The actual thickness satisfies minimum thickness requirements for the period of extended operation.

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Design of Pressure Retaining parts	All references are made to ASME Boiler & Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1971 Ed. plus addendum July 1971. Reference the same code for explanation of the symbols used.		
Body Minimum Wall Thickness	Reference paragraph NB 3543, Nonstandard Pressure-Rated Valve, Table NB 3542-1, For design condition of 1250 psig and 575°F. The primary service rating = 495 based on a core diameter of 23.00 in. t_m = 1.539 in. (including a corrosion allowance of 0.12 in).	1.58	2.12
Body Shape Rule	Reference paragraph NB 3544, Body Shape Rules		
Radius of Crotch	Reference paragraph NB 3544.1(a), Radius of Crotch criterion $r_2 \geq 0.3\ t_m$ as r_2 = 1.00 in., t_m = 1.539 + 1.00 $\geq$ 0.3 x 1.937 = 0.581 criterion satisfied.		
Corner Radii on Internal Surfaces	Reference paragraph NB 3544.1(b), Corner Radii on Internal Surfaces criterion $r_4 < r_2$ as r_4 = 1.00 in., r_2 = 1.125 in. + 1.00 < 1.125 criterion satisfied.		
Out of Roundness	Reference paragraph 3544.5, Out of Roundness, Figure NB 3545.1-2 The body is not out of round in excess of 5%, the requirements of this article are satisfied.		
Longitudinal Curvature	Reference paragraph NB 3544.6 Longitudinal Curvature criterion $\frac{1}{r_{Long.}} + \frac{1}{r_{Lat.}} \geq \frac{4}{3d_m}$ as $r_{Long.}$ = 27 in., $r_{Lat.}$ = 11.5 in., d_m = 23.00 in., + .124 $\geq$ .058 criterion satisfied.		
Flat Wall Limitation	Reference paragraph NB 3544.7, Flat Wall Limitation Since no flat sections were built into the valve body design, the requirements of this article are satisfied.		

TABLE 3.9-2(h) (page 2 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Minimum Wall at Weld End	Reference paragraph 3544.8, Minimum Wall at Weld End Actual thickness at > 1.937 in. (i.e., 1.937 in.) measured along. The run direction is 1.492 in. Actual run distance is $\geq$ 9.5 in.		
Primary Crotch Stress Due to Internal Pressure	Reference paragraph NB 3545.1 criterion $p_m = \left(\frac{A_f}{A_m} + 0.5\right) P_s < S_m$ where $A = 503$ in.^{2}, $A_m = 57$ in.^{2}, $P_s = 1375$ psig, $P_m = 12,821$ psi, $S_m = 19,400$ psi, since $S_m > P_m$ criterion satisfied		
Valve Body Secondary Stress	Reference paragraph NB 3545.2		
Primary Plus Secondary Stress Due to Internal Pressure	Reference paragraphs NB 3545.2(a)(1), NB 3545.2(a)(2) $Q_p = C_p \left(\frac{r_i}{t_e} + 0.5\right) P_s$ Ca where $C_p = 3$, $r_i = 11.625$ in., $P_s = 1375$ psi, $t_e = 2.75$ for wye-type valve $C_a = 1.33 + Q_p = 25.965$ psi		
Secondary Stress Due to Pipe Reaction	Reference paragraph NB 3545.2(b), Figures NB 3545.2-3, NB 3545.2-5, NB 3545.2-6		
Direct or Axial Load Effect	$P_{ed} = \frac{F_d S}{G_d}$ where $S = 30,750$, $F_d = 30$ in.^{2}, $G_d = 183$ in.^{2} $P_{ed} = 5041$ psi		
Bending Load Effect	$P_{eb} = C_b \frac{F_b S}{G_b}$ where $S = 30,750$, $F_b = 340$ in.^{3},	29,100	5,041
	i.d. = 23.25 in., $r_i = 11.625$ $t_e = 2.75$ $r = 13.91$ in. as $\frac{t_e}{r} = 0.197$ 0.19 + C_b = 1 $C_b = \frac{I}{r_i + t_e}$ where $I = 15,028$ in.^{4}, $r_i = 11.625$ in., $t_e = 2.75$ in. + $G_b = 1048.2$ in.^{3}		
	+ P_{eb} = = 9.974 psi	29,100	9,974

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Torsion Load Effect	Reference paragraphs 3545.2(b)(1), 3545.2(b)(6)(c)		
	$P_{et} = 2 \frac{F_b S}{G_t}$ where $F_b = 340$ in.^{3}, $S = 30,750$ psi		
	$G_t = 2161$ in.^{3}		
	$\rightarrow P_{et} = 9,676$ psi	29,100	9,676
	For special requirement of $S = 2\ S_m = 41,000$ psi		
	$P_{ed} = 6,722 \leq 29,100$		
	$P_{eb} = 13,299 \leq 29,100$	29,100	13,299
	$P_{et} = 12,896 \leq 29,100$		
	$S_h = 33,733 \leq 58,200$ psi	58,200	33,733
Thermal Secondary Stress at Crotch Region	Reference paragraph NB 3545.2(c), Figures NB 3545.2(c)-2, NB 3545.2(c)(2), NB 3545.2(C)-3, NB 3545.2(c)-3, NB 3545.2(c)-4		
	$Q_T = Q_{T_1} + Q_{T_2}$		
	where $Te_1 = 3$ in., $Q_{T_1} = 1,100$		
	$Q_{T_2} = C_6 C_2 \Delta T_2$ where $C_2 = 0.53$ $C_6 = 210$ and $\Delta T_2 = 4.7$		
	$Q_{T_2} = 523$ psi, $Q_T = 1,623$ psi		
	criterion $S_N = Q_P + P_{ed} + 2Q_{T_2} \leq 3\ S_m$		
	where $Q_P = 25,965$ $P_{ed} = 5,041$ $Q_{T_2} = 523$		
	as $32,052 \leq 58,200$ criterion satisfied	58,200	32,052
Normal Duty Valve Fatigue Requirements	Reference paragraphs 3545.3, NB 3545.3(a), NB 3545.3a,		
	Figure 1-9-1		
	criterion $N_a \geq 2,000$ cycles		

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Normal Duty Valve Fatigue Requirements (cont'd.)	$S_{P_1} = \frac{2}{3} Q' \ p + \frac{P_{eb}}{2} + Q_{T_3} + 1.3 Q_{T_1}$ $S_{P_2} = 0.4 Q' \ p + \frac{K}{2} (P_{eb} + 2Q_{T_3})$		
	where $Q' \ P = 25,965$ $P_{eb} = 9,974$ $K=2$, $Q_{T_1} = 1,100$ $Q_{T_3} = 622$ psi		
	$\rightarrow S_{P_1} = 24,349$ $S_{P_2} = 21,604$ S_a equal to the larger of S_{P_1} and S_{P_2} $\rightarrow$ $S_a = 24,349$ psi		
	$\rightarrow N_a = 50,000 \geq 2,000$ criterion satisfied		
Cyclic Loading Requirements at Valve crotch	Reference paragraph 3550 For the largest temperature change range criterion $Q_p + P_{ed} + C_6 C_2 C_4 \ \Delta T_f \max \leq 3 S_m$		
	Where $Q'_p = 25,965$ psi, $P_{ed} = 5,041$ $C_6 = 210$ at ΔT_f max of 171 F,		
	$C_2 = 0.53$ $C_4 = 0.03$, S_m, 19,400		
	$\rightarrow 31,577 \leq 58,200$ criterion satisfied	58,200	31,577
	Thermal Transients Not Excluded by Code criterion $\frac{EN_{ri}}{EN_i} < 1$		
	Calculate the fatigue usage factor (I_t) as follows:		
	$S_n \text{ Max} = Q' \ p + P_{eb} + C_6 C_3 C_4 \ \Delta T_f \max \rightarrow S_n \max = 36,618$ psi	58,200	36,618
	Since $S_n \max < 3 S_m$ (= 58,200) the following equation is used:		
	$S_i = \frac{4}{3} Q' \ p + P_{eb} + C_6 (C_3 C_4 + C_5) \ \Delta T_{fi}$		
	for $\Delta T_{fi} + 45°F$, $N_{ri} = 120$ $S_i = 54,719$ psi, $N_i = 25,000$,		
	$N_{ri}/N_i = .0048$		
	$\Delta T_{fi} = 121°F$ $N_{ri} = 8$, $S_i = 83,069$ psi, $N_i = 6,000$		
	$N_{ri}/N_i = .0013$		
	$\Delta T_{fi} = 61°F$, $N_{ri} = 10$, $S_i = 58,319$ psi, $N_i = 20,000$		
	$N_{ri}/Ni = .0005$		

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
	as $I_t = \Sigma \frac{N_{ri}}{N_i} = 0.0066 < 1$ criterion satisfied	1.00	.0066
Disk Design Calculation	Reference paragraph NB 3546.3, Table I-1.1, Roark, 3rd Ed., Pages 198, 200, 201		
	Disk design conditions, $P_R = 1250$ psi at 500°F, $S_m = 17{,}800$ psi at 500°F		
	Case No. 13 $S = \frac{3W}{4mt^2(a^2-b^2)} (a^4 (3m + 1) + b^4 (m - 1) -$		
	$4_m\, a^2b^2 - 4(m + 1)\, a^2b^2(\ln(a/b))$		
	where W = 1250 psi, $m - \frac{10}{3}$, t = 5,875 in., a = 10.75 in., b = 1.750 inc.,		
	$s_t = 9{,}488$ psi		
	Case No. 14 $S = \frac{3W}{2\pi mt^2} \frac{(2a^2(m + 1)}{a^2-b^2} \ln \frac{(a)}{(b)} + (m - 1))$		
	where $W = 58{,}123\ lb_f$, t = 5.875 in., $m = \frac{10}{3}$,		
	a = 10.75 in., b = 1.75 in., $s_t = 4{,}460$ psi		
	$S_t = S_{t_{Case\ no.\ 13}} + S_{t_{Case\ No.\ 14}} = 13{,}949 \leq 17{,}800$ psi	17,800	13.949
	Case No. 21 $S_r = \frac{3W}{4t^2} \frac{[4a^4(m + 1) \ln \frac{(a)}{(b)} a^4 (m + 3) + b^4(m - 1) + 4a^2b^2]}{a^2 (m + 1) + b^2 (m - 1)}$		
	where W = 1250 m = 10/3, t = 3.125 in., a = 10.75 in., b = 7.25 in.		
	$\rightarrow S_r = 5{,}761$ psi		
	Case No. 22 $S_r = \frac{3W}{2\pi t^2} \frac{[2a^2 (m + 1) \ln \frac{(a)}{(b)} + a^2 (m - 1) - b^2 (m - 1)]}{a^2 (m + 1) + b^2 (m - 1)}$		
	where W = 252,755 m = 10/3, t = 3.125 a = 10.75, b = 7.25		
	$S_r = 10{,}738$ psi		

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Disk Design Calculation (cont'd.)	Total stress = $Sr_{22} + Sr_{22}$ = 16,499 psi, allowable stress 17,800 psi	17,800	16,499
	S_{shear} at inner edge disk^2		
	$S_{shear} = \frac{F}{A}$ where F = 264,536 lb, A = 71.18 in.^{2} S_{shear} = 3,716 psi		
	Allowable shear stress = 0.6 x allowable stress = 0.6 x 17,800 = 10,680 psi	10,680	3,716
Tensile Stress at thread Relief	$S_A = \frac{F}{A_t}$ where F= 46,342 lb, A_t = 2.624 in.^{2}, S = 17,661 lb/in^2		
	S_n = 17,661 psi, S_m = 30,600 psi	30,600	17,661
Bonnet Design Calculations including Seismic accelerations for SSE	Reference paragraph NB 3546.1		
Minimum Thickness	$P_{fd} = P + P_{eg}$ $P_{eg} = \frac{16M}{\pi G^3} + \frac{4F}{\pi G^2}$		
	where M = 335,253 in.-lb, F = 46,342 lb, G = 24.75 in.		
	→ P_{eg} = 209 psi, P_{fd} = 1,459 psi		
	$t - d\sqrt{\frac{CP}{S} + \frac{1.78\ W\ hg}{S_d^{\ 3}}}$		
	where C = 0.3, P = 1,459 psi, S = 17,800 psi, hg = 2.625 in., w = 910,144 lb, d = 24.75 in.		
	→ t = 4.97 in., t - 4.97 + 0.120 = 5.09 in. (Corrosion allowance is 0.120 in.)	5.09	5.344

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Reinforcement	Reference paragraph NB 3643.3		
	To account for the opening for stem in the bonnet		
	Required Reinforcement $A_r = 7.207$ in^2	8.6844	7.207
	Available Reinforcement $A = 8.6844$ in^2		
Bonnet Studs Design Calculation	Reference paragraph 3232.1 and Article E-1000		
	Bolt used 20 pieces of 2 - 8 UNC Bolts		
	Total bolt area = 53.04 in.^{2}		
Normal Operation	1. Pressure stress at Operating Condition		
	$S_1 = \frac{W_{m1}}{A_b} = 17,159$ lb/in.^{2} where $W_{m1} = 910,144$ lb.		
	$A_b = 53.04$ in.^{2} $A_{m1} = \frac{W_{m1}}{S_{Bo}} = \frac{910,144}{27,700} = 32.85$ in^2	53.04	32.85
	2. Gasket load at ambient condition with no internal pressure		
	$S_2 = \frac{W_{m2}}{A_b} = 2,019$ lb/in.^{2} where $W_{m2} = 107,065$ lb$_f$		
	$A_b = 53.04$ in.^{2} $A_{m2} = \frac{W_{m2}}{S_{Ba}} = \frac{107,065}{35,000} = 3.06$ in^2	53.04	3.06
	Maximum tensile stress = 17,159 lb/in.^{2}		
	Thermal stress is assumed negligible because the coefficients of thermal expansion of bonnet place and stud are the same.		
Body Flange Design Calculations	Reference paragraph NB 3647.1		
	Total flange moment under operating conditions		
	$M_o = M_D + M_G + M_T$		
	$M_D = H_D h_D$, $H_D = 0.785 B^2 P$, $h_D = R + 0.591$		
	where $B = 21.75$ in., $P = 1,459$ psi $\rightarrow H_D = 542,080$ lb$_f$, $h_D = 2.813$ in.,		
	$M_D = 1,524,871$ in.-lb		
	$M_G = H_G h_g$, $H_G = W - H$, $h_G = \frac{C-G}{2}$		
	where W is the higher of W_{m1} and W_{m2}		
	$W_{m1} = 0.785 G^2 P + (2b \times 3.14\ G\ m\ P)$		

TABLE 3.9-2(h) (page 8 of 9)

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Body Flance Design Calculations (cont'd.)	where $G = 24.75$ in., $b = 0.306$ in., $m = 3$, $y = 4{,}500$		
	$\rightarrow W_{m1} = 910{,}144$ lb, $W_{m2} = 107{,}065$ lb		
	$\rightarrow H_G = 208{,}210$ lb, $h_G = 2.625$ in. $\rightarrow M_G = 546{,}551$ in.-lb		
	$M_T = H_T\, h_T$, $H_T = H - H_D$, $h_T = \dfrac{R + g_{1 \cdot hG}}{2}$		
	where $H = 701{,}934$ $H_D = 542{,}080$ $R = 1.5$ in., $g_1 = 2.625$ in., $h_G = 2.625$ in.		
	$\rightarrow H_T = 159{,}854$ lb, $h_T = 3.375$ in., $M_T = 539{,}507$ in.-lb_f		
	$M_o = 2{,}610{,}929$ in.-lb. where $M_D = 1{,}524{,}871$ in.-lb,		
	$M_G = 546{,}551$ in.-lb, $M_T = 539{,}507$ in.-lb		
	Total flance moment under gasket seating condition		
	$M_o = W\left(\dfrac{C-G}{2}\right)$, $W = \left(\dfrac{A_m + B}{2}\right) s_a$		
	where $C = 30$ in., $A_b = 53.04$ in.^{2}, $G = 24.75$ in.,		
	$A_m = 32.86$ in.^{2}, $s_a = 35{,}000$ psi at 100°F		
	$\rightarrow W = 1{,}503{,}198$ lb $\rightarrow M_o = 3{,}014{,}460$ lb-in.		
Longitudinal Hub Stress	Reference Paragraph NB 3647.1(c)		
	$S_h = \dfrac{fM_o}{L_{S_1}{}^2 B} + \dfrac{PB}{4 g_o} = 21{,}794$ lb/in.^{2} $< 1.5\ S_{fo} = 26{,}700$ lb/in.^{2}	26,700	21,794
Radial Stress	Reference UA-51(1), Equation (7) of Section VIII of ASME B&PV Code, 1971 Edition		
	$S_R = \dfrac{(1.33\, t_e + 1)\, M_o}{Lt^2 B} = 12{,}303$ psi $< 1.5\ S_{fo} = 26{,}700$	26,700	12,303
Tangential Stress	$S_T = \left(\dfrac{YM_o}{t^2 B}\right) - ZS_R = 7{,}126$ psi $< 1.5\ S_{fo} = 26{,}700$	26,700	7,126
	where $Y = 4.5$, $t = 4.125$ in., $Z = 2.4$, $B = 21.75$ in.		

MAIN STEAM ISOLATION VALVE

Criteria	Method of Analysis	Allowable Stress or Minimum Thickness (in.)	Calculated Stress or Actual Thickness (in.)
Flange Stress Criteria	$\frac{S_H + S_R}{2} < S_m$, $\frac{S_H + S_R}{2} < S_m$		
	$\frac{S_H + S_R}{2} = 17,049$ psi	17,800	17,049
	$\frac{S_H + S_T}{2} = 14,460$ psi		
Stem Calculation			
Back Seated Stress	$S = \frac{F}{A}$		
	where F = 30,697 lb net upward force		
	A = 2.535 in.^{2}, the smallest cross sectional area on the stem		
	S = 12,110 psi < 26,880 psi		
Valve Close Stem Stress	$S = \frac{F}{A}$		
	where F = 46,959 lb net down force		
	A = 3.141 in.^{2}, the smallest cross sectional area on the stem		
	S = 14,950 psi < 26,880 psi		
Stem Thread Strength	Reference Federal Thread Standard		
	<u>Stem Thread Mating With Disk</u>		
	Thread 2.00 in. - 8 UN - 2 Thread		
	A_{S1} = 3.49 in.^{2}/inch engagement.		
	A_{S1} = 6.54 in.^{2}		
	$t = \frac{F}{A_{S1}}$		
	where F = 46,959 lb_f, A_{S1} = 6.54 in.^{2} + t_{Sd} = 7,177 psi		

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
1. Casing Minimum Wall Thickness $$t = \frac{PR}{SE - 0.6P} + C$$ Loads: Normal and Upset Condition — where: Design pressure & temperature — t = min. req'd thickness, in. P = design pressure, psig Primary membrane stress limit: R = max. internal radius, in. S = allowable working stress, psi Allowable working stress per ASME Sect. III, Class C — E = joint efficiency C = corrosion allowance, in.	t = 2.69 inches	$S_{all.}$ = 15,075 psi $t_{act.}$ = 3.00 inches
2. Casing Cover Minimum Thickness $$S_s = \frac{F}{A}$$ Loads: Normal and Upset Condition — F = force A = area at shear point Design pressure & temperature Primary bending & shear stress limit: $S_b = \frac{K^{2qa}}{h^2}$ 1.5 S_m per ASME Code for Pumps and Valves for Nuclear Power Class I	S_s = 3,380 psi S_b = 5,950 psi	$S_{all.}$ = 8,750 psi $t_{act.}$ = 3.5 inches $S_{all.}$ = 1.5 x 15,075 $t_{act.}$ = 7 inches

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
Allowable working stress per q = pressure load a = radius of O.D. b = radius of I.D. h = plate thickness		
3. Cover and Seal Flange Bolt Areas Bolting loads, areas and stresses shall be calculated in accordance with "Rules for Bolted Flange Connections" – ASME Sect. VIII, Para. UA-49 Loads: Normal and Upset Condition Design pressure & temperature Design gasket load Bolting Stress Limit: Allowable working stress per ASME Sect. III, Class C	Cover Flange Bolts $S_{act.}$ = 19,050 psi A_m = 90.2 sq. in. A_m = 79 sq. in. (For Upgraded Gasket) Seal Flange Bolts $S_{act.}$ = 18,000 psi A_m = 9.85 sq. in.	$S_{all.}$ = 20,000 psi $A_{all.}$ = 101 sq. in. $A_{all.}$ = 95 sq. in. (For Upgraded Gasket) $S_{all.}$ = 20,000 psi $A_{all.}$ = 11.1 sq. in.
4. Cover Clamp Flange Thickness Flange thickness and stress shall be calculated in accordance with "Rules for Bolted Flange Connections" – ASME Sect. VIII, Para. UA-51 Loads: Normal and upset Condition Design pressure & temperature Design gasket load Design bolting load	Flange Thickness Stress t = 8.9 inches	$t_{act.}$ = 9.25 inches $S_{all.}$ = 26,250 psi

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA	ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
Tangential Flange Stress Limit Allowable working stress per ASME Sect. III, Class C		
5. Seal Cover Loads: $S_b = \frac{KP}{t^2}$ Normal and Upset Condition Design pressure & temperature P = Bolt load due to pressure t = Thickness, in. K = Constant of shape factor	S_b = 2,870 psi t = 1.10	$S_{all.}$ = 15,075 psi t_{act} = 2 9/16
6. Seal Chamber Minimum Wall Thickens $t = \frac{PR}{SE - 0.6P} + C$ Loads: Normal and Upset Condition where: Design pressure & temperature Piping reactions during normal operation t = min. req'd thickness, in. P = design pressure, psig R = max. internal radius, in. S = allowable working stress, psi E = joint efficiency C = corrosion allowance, in. Combined Stress Limit: 1.5 S_m per ASME Section III Code for Pumps and Valves for Nuclear Power Class I	t= 0.741 inches	$S_{all.}$ = 1.5 x 17,075 psi $t_{act.}$ = 1.375 inches

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA		ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
7. Mounting Bracket Combined Stress Loads: Flooded weight DBE horizontal seismic force = 2.50g DBE vertical seismic force = 1.61g Combined Stress Limit: 1.5 S_m per ASME Code Section III for Pumps and Valves for Nuclear Power Class I	Bracket vertical loads shall be determined by summing the equipment and fluid weights and vertical seismic forces. Bracket horizontal loads shall be determined by applying the specified seismic force at mass center of pump-motor assembly (flooded). Horizontal and vertical loads shall be applied simultaneously to determine tensile, shear and bending stresses in the brackets. Tensile, shear and bending stresses shall be combined to determine max. combined stresses.	Lug #1 S_C = 2,270 psi Lug #2 S_C = 24,429 psi Lug #3 S_C = 14,178 psi Lug #4 S_C = 4,591 psi	1.5 S_m = 25,013 psi
8. Stresses Due to Seismic Loads Loads: Operation pressure and temperature DBE horizontal seismic force = 2.50g DBE vertical seismic force = 1.61g Combined Stress Limit: Yield stress	The flooded pump-motor assembly shall be analyzed as a free body supported by constant support hangers from the pump brackets. Horizontal and vertical seismic forces shall be applied at mass center of assembly and equilibrium reactions shall be determined for the motor and pump brackets. Loads, shear, and moment	Motor Bolt Tensile Stress: $S_{act.}$ = 22,471 psi Pump Cover Bolt Tensile Stress: $S_{act.}$ = 19,417 psi Motor Support Barrel Combined Stress: $S_{act.}$ = 3,307 psi	 $S_{all.}$ = 30,800 psi $S_{all.}$ = 32,000 psi $S_{all.}$ = 22,400 psi

TABLE 3.9-2i

RECIRCULATION PUMP

CRITERIA		ANALYTICAL RESULTS	ALLOW. STRESS OR ACTUAL THICKNESS
	diagrams shall be constructed using live loads, dead loads, and calculated snubber reactions. Combined bending, tension and shear stresses shall be determined for each major component of the assembly including motor support barrel, bolting and pump casing. The maximum combined tensile stress in the cover bolting shall be calculated using tensile stresses determined from loading diagram plus tensile stress from operating pressure.		

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *DISCHARGE*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
1.	**Body and Bonnet**			
1.1	Loads: Design pressure, design temp.	Vendor's Design calculation	1525 psi 575°F	1525 psi 575°F
1.2	Pressure Rating. lb.	Used NB-3543, Table NB-3531-4 & NB-3531-5 of ASME SECTION III	P_r = 826 lb.	P_r = 826 lb.
1.3	Minimum wall thickness, inches	Used ASME SECTION III PARA NB-3543, Table NB-3542-1	$t_m \geq 2.42$ inches	$t_m = 2.4224$ inches
1.4	Primary membrane stress, psi	Used ASME SECTION III PARA NB-3545.1	$P_m \leq S_m$ (500°F) = 19600 psi	P_m = 9954 psi
1.5	Secondary stress due to pipe reaction	Used ASME SECTION III PARA NB-3545.2 (b) (S = 30,000 psi)	$P_{ed} \leq 1.5\ S_m$ = 29400 psi $P_{eb} \leq 1.5\ S_m$ = 29400 psi $P_{et} \leq 1.5\ S_m$ = 29400 psi	P_{ed} = 4271 psi P_{eb} = 7581 psi P_{et} = 7170 psi
1.6	Primary plus secondary stress due to internal pressure	Used ASME SECTION III PARA NB-3545.2 (a)	58,000 psi (See 1.8 below)	Q_p = 24218 psi
1.7	Thermal secondary stress	Used ASME SECTION III PARA NB-3545-2 (c)	58,000 psi (See 1.8 below)	Q_{T1} = 5800 psi Q_{T2} = 1757 psi Q_{T3} = 2044 psi
1.8	Sum of primary plus secondary stress	Used ASME SECTION III PARA NB-3545.2	$S_n = Q_p + P_{ed} + 2Q_{T2}$ $S_n \leq 3S_m$ (500°F) = 58800 psi	S_n = 31674 psi
1.9	Fatigue requirements	Used ASME SECTION III PARA NB-3545.3	$N_a \geq 2000$ cycles	$N_a > 10^5$ cycles
1.10	Cyclic rating	Used ASME SECTION III PARA NB-3550	$I_t \leq 1$	I_t = .00335
2.0	**Body to Bonnet Bolting**			
2.1	Loads: Design pressure & temp., gasket loads, stem operational load, seismic load (design basis earthquake)	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII		
2.2	Bolt Area	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$A_s \geq 42.79$ sq. in. $S_b \leq 27975$ psi	A_b = 55.86 sq. in. S_b = 21288 psi
2.3	Body Bonnet Flange Stresses			
2.3.1	Operating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (575°F) = 28837 psi $S_R \leq 1.5\ S_m$ (575°F) = 28837 psi $S_T \leq 1.5\ S_m$ (575°F) = 28837 psi	S_H = 24206 psi S_R = 7307 psi S_T = 7812 psi
2.3.2	Gasket seating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (100°F) = 30000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30000 psi	S_H = 29837 psi S_R = 11050 psi S_T = 11815 psi

NOTES

(1) SECTION III = ASME BOILER AND PRESSURE VESSEL CODE, SECTION III. 1971, "NUCLEAR POWER PLANT COMPONENTS"

(2) VALVE DIFFERENTIAL PRESSURE 200 PSI

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *DISCHARGE*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
3.0	**Stress in Stem**			
3.1	Load Operator thrust and torque			
3.2	Buckling on Stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 9.	Maximum allowable load = 82,635 lbs.	Slenderness ratio = 61.1 Actual load on Stem = 34,342 lbs. Therefore no buckling.
3.3	Stem Thrust Stress	Calculate stress due to operator thrust in critical cross-section	$S_T \leq S_m = 44100$ psi	$S_T = 6968$ psi
3.4	Stem Torque Stress	Calculate shear stress due to operator torque in critical cross-section	$S_c \leq .6S_m = 26460$ psi	$S_c = 4285$ psi
4.	**Disc Analysis**			
4.1	Loads: Maximum differential pressure			
4.2	Maximum Stress	Calculate maximum stress according to chapter of R. J. Roark "Formulas for Stress and Strain"	$S_{max} \leq 1.5S_m$ (575°F) = 28500 psi	$S_{max} = 22191$ psi
5.	**Yoke and Yoke Connections**			
5.1	Loads: Stem operational load	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods		
5.2	Tensile stress in yoke legs bolts		$S_{max} \leq S_m = 23.300$ psi	Max. stress = 8869 psi
5.3	Bending stress of yoke legs		$S_b \leq 1.5\ S_m = 34{,}950$ psi	$S_b = 6359$ psi

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *SUCTION*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
1.	**Body and Bonnet**			
1.1	Loads: Design pressure, design temp.	Vendor's design calculation	1275 psi 575°F	1275 psi 575°F
1.2	Pressure Rating , lb.	Used NB-3543, Table NB-3531-4 & NB-3531-5 of ASME SECTION III	P_r = 695 lb.	P_r = 695 lb.
1.3	Minimum wall thickness inches	Used ASME SECTION III PARA NB-3543, Table NB-3542-1	$t_m \geq$ 2.03 inches	t_m = 2.0254 in.
1.4	Primary membrane stress, psi	Used ASME SECTION III PARA NB-3545.1	$P_m \leq S_m$ (500°F) = 19600 psi	P_m = 13419 psi
1.5	Secondary stress due to pipe reaction	Used ASME SECTION III[1] PARA NB-3545.2 (b) (S = 30,000 psi)	$P_{ed} \leq 1.5\ S_m$ = 29,400 psi $P_{eb} \leq 1.5\ S_m$ = 29,400 psi $P_{et} \leq 1.5\ S_m$ = 29,400 psi	P_{ed} = 4391 psi P_{eb} = 10740 psi P_{et} = 9220 psi
1.6	Primary plus secondary stress due to internal pressure	Used ASME SECTION III PARA NB-3545.2 (a)	58,800 psi (See 1.8 below)	Q_P = 23527 psi
1.7	Thermal secondary stress	Used ASME SECTION III PARA NB-3545.2 (c)	58,800 psi (See 1.8 below)	Q_{T1} = 3400 psi Q_{T2} = 1127 psi Q_{T3} = 1332 psi
1.8	Sum of primary plus secondary stress	Used ASME SECTION III PARA NB-3545.2	$S_n = Q_p + P_{ed} + 2Q_{T2}$ $S_n \leq 3S_m$ (500°F) = 58800 psi	S_n = 30332 psi
1.9	Fatigue requirements	Used ASME SECTION III PARA NB-3545.3	$N_a \geq$ 2000 cycles	$N_a > 10^5$ cycles
1.10	Cyclic rating	USED ASME SECTION III PARA NB-3550	$I_t \leq 1.0$	I_t = .002617
2.0	**Body to Bonnet Bolting**			
2.1	Loads: Design pressure & temp., gasket loads, stem operational load, seismic load (design basis earthquake)	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII		
2.2	Bolt Area	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$A_b \geq$ 35.70 sq. in. $S_b \geq$ 27975 psi	A_b = 55.86 sq. in. S_b = 17881 psi
2.3	Body Bonnet Flange Stresses	Used ASME SECTION III PARA, NB-3546.1 NB-3647.1 and Section VIII		
2.3.1	Operating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (575°F) = 28837 psi $S_R \leq 1.5\ S_m$ (575°F) = 28837 psi $S_T \leq 1.5\ S_m$ (575°F) = 28837 psi	S_H = 21000 psi S_R = 5677 psi S_T = 8049 psi
2.3.2	Gasket seating condition	Used ASME SECTION III PARA NB-3546.1 NB-3647.1 and Section VIII	$S_H \leq 1.5\ S_m$ (100°F) = 30000 psi $S_R \leq 1.5\ S_m$ (100°F) = 30000 psi $S_T \leq 1.5\ S_m$ (100°F) = 30000 psi	S_H = 27342 psi S_R = 9351 psi S_T = 13257psi

NOTES

(1) SECTION III = ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, 1971, "NUCLEAR POWER PLANT COMPONENT"

(2) VALVE DIFFERENTIAL PRESSURE 50 PSI

TABLE 3.9-2 (j)

REACTOR RECIRCULATION SYSTEM GATE VALVES, *SUCTION*

STRUCTURAL & MECHANICAL LOADING CRITERIA

0. 0.1 0.2	Component Loads Design	Design Procedure	Required Design Value	Actual Design Value
3.0	**Stress in Stem**			
3.1	Load Operator thrust and torque			
3.2	Buckling on Stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 9.	Maximum allowable load = 82,635 lbs.	Slenderness ratio = 61.1 Actual load on Stem = 17047 lbs. Therefore no buckling.
3.3	Stem Thrust Stress	Calculate stress due to operator thrust in critical cross-section	$S_T \leq S_m$ = 44100 psi	S_T = 3473 psi
3.4	Stem Torque Stress	Calculate shear stress due to operator torque in critical cross-section	$S_c \leq .6\ S_m$ = 26460 psi	S_c = 2140 psi
4.	**Disc Analysis**			
4.1	Loads: Maximum differential pressure			
4.2	Maximum Stress	Calculate maximum stress according to chapter of R. J. Roark "Formulas for Stress and Strain:	$S_{max} \leq 1.5\ S_m$ (575°F) = 28500 psi	S_{max} = 18393 psi

TABLE 3.9-2(k)

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TABLE 3.9-2L

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING			COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The standby liquid control pump has been designed and fabricated to the 1968 P&V Code for Class 2 component.						
Pressure boundary parts:						
1)	Fluid cylinder – SA182-F304	S_Y = 30,000 psi				
2)	Discharge valve stop stuffing box and cylinder head extension SA 479-304	S_Y = 30,000 psi				
3)	Discharge valve cover, cylinder head and stuffing box flange plate, SA 285 GR. C	S_Y = 30,000 psi				
4)	Stuffing box gland, ASTM A461, GR. 630	S_Y = 90,000 psi				
5)	Studs, SA 193B7	S_Y = 105,000 psi				
6)	Dowel pins[(2)] alignment, SAE 4140	S_Y = 117,000 psi				
7)	Studs, cylinder tie, SA 193-B7	S_A = 105,000 psi				
8)	Pump holddown bolts, SAE GR. 1	T_A = 15,000 psi Q_A = 12,000 psi				
9)	Power frame, foot area, cast iron	S_A = 15,000 psi				
10)	Motor holddown bolts, SAE GR. 1	T_A = 15,000 psi				
		Q_A = 12,000 psi				
11)	Motor frame foot area, cast iron	S_A = 15,000 psi				

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Normal & Upset Condition Loads:				
1. Design pressure	1. Fluid Cylinder	General Membrane	17,800	See Note (3)
2. Design temperature	2. Discharge valve stop	General Membrane	17,800	
3. Operating basis earthquake	3. Cylinder head extension	General Membrane	17,800	
4. Nozzle loads(1)	4. Discharge valve cover	General Membrane	17,800	
5. Dead weight	5. Cylinder head	General Membrane	17,800	
6. Thermal expansion	6. Stuffing box flange plate	General Membrane	17,800	
7. SRV discharge	7. Stuffing box gland	General Membrane	35,000	
	8. Cylinder head studs	Tensile	25,000	
	9. Stuffing box studs	Tensile	25,000	
Emergency or Faulted Condition:				
1. Design pressure	1. Fluid cylinder	General Membrane	21,360	4,450
2. Design temperature	2. Discharge valve stop	General Membrane	21,360	13,600
3. Weight of structure	3. Cylinder head extension	General Membrane	21,360	13,600
4. Thermal expansion	4. Discharge valve cover	General Membrane	21,360	8,150
5. Safe shutdown earthquake	5. Cylinder head	General Membrane	21,360	8,150
6. SRV	6. Stuffing box flange plate	General Membrane	21,360	10,390
7. LOCA	7. Stuffing box gland	General Membrane	42,000	11,420
8. Nozzle Loads(1)	8. Cylinder head studs	Tensile	25,000	18,820
	9. Dowel pins(2)	Shear only(2)	23,400	19,400
	10. Studs, cylinder tie	Tensile(2)	25,000	8,685
	11. Pump holddown bolts	Shear	12,000	9,415
	12. Pump holddown bolts	Tensile	15,000	11,675
	13. Power frame-foot area	Shear	15,000	1,850
	14. Power frame-foot area	Tensile	15,000	11,390
	15. Motor holddown bolts	Shear	12,000	3,020
	16. Motor holddown bolts	Tensile	15,000	5,290
	17. Motor frame-foot area	Shear	15,000	2,070
	18. Motor frame-foot area	Tensile	15,000	4,125

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE ACCELERATION (G)	CALCULATED ACCELERATION (G)
Faulted Condition				
Dynamic Loads	SLC Pump Assembly	Equivalent static acceleration	1.75g (vertical) 1.75g (horizontal)	0.45g 0.73g
1. SSE				
2. SRV				
3. LOCA				

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS	ALLOWABLE LOADS	CALCULATED LOADS
Nozzle Load Definition:				
Units: Forces – lbs.				
Moments – ft. - lbs.				
Allowable combination of forces and moments are as follows:				
Where:				
F_1 = The largest absolute value of the three actual external orthogonal forces (F_x, F_y, F_z) that may be imposed by the interface pipe, and,				
M = The largest absolute value of the three actual internal orthogonal moments (M_x, M_y, M_z) permitted from the pipe when they are combined simultaneously for a specific condition.				
			SUCTION:	
Normal and Upset Condition Loads:		F_o = Allowable value of F_1 when all	F_o = 770 lb.	207
		moments are zero.		
1. Design pressure		M_o = Allowable value of M_1 when	M_o = 490 ft.-lb.	388
2. Design temperature		all forces are zero.		
3. Dead weight			DISCHARGE:	
4. Thermal expansion				
5. Operating Basis Earthquake			F_o = 370 lb.	173
			M_o = 110 ft.-lb.	95

TABLE 3.9-2L (Continued)

STANDBY LIQUID CONTROL PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS	ALLOWABLE LOADS	CALCULATED LOADS
Emergency of Faulted Condition Loads:				
1. Design pressure				
2. Design temperature			SUCTION:	
3. Dead weight				
4. Thermal expansion			F_o = 920 lb.	207
5. Safe Shutdown Earthquake			M_o = 590 ft.-lb.	388
6. SRV				
7. LOCA			DISCHARGE:	
8. Nozzle Loads				
			F_o = 440 lb.	173
			M_o = 130 ft.-lb.	95

NOTES:

(1) Nozzle loads produce shear loads only.

(2) Dowel pins take all shear.

(3) Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition loads, therefore, the normal and upset condition is not evaluated.

(4) Operability: The sum of the plunges and rod assembly, pounds mass times 1.75, acceleration is much less than the thrust loads encountered during normal operating conditions. Therefore, the loads during the faulted condition have no significant effect on pump operability.

Table 3.9-2m

Standby Liquid Control Tank

	Criteria	Method of Analysis	Allowables		Actuals			
1.	Shell Thickness							
	Loads: Normal and Upset Design Pressure and Temperature	Brownell and Young "Process Equipment Design" $t = \frac{PR}{SE - 0.6P}$	0.01542 in.		0.1875 in.			
	Stress Limit	ASME Section III	30,000 psi		1,602 psi			
2.	Nozzle Loads							
	Loads: Normal and Upset Design Pressure and Temperature	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	F_o(lb.)	M_o (ft.-lb.)	F_o (lb.)	M_o	(ft.	- lb.)
	Overflow Nozzle		440	330	298[1]	167[2]	218[1]	203[2]
	Discharge Nozzle		440	330	298[1]	167[2]	218[1]	203[2]
	Loads: Faulted Dead Weight, Thermal Expansion and SSE Earthquake	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	F_o(lb.)	M_o (ft.-lb.)	F_o (lb.)	M_o	(ft.	- lb.)
	Overflow Nozzle		528	360	313[1]	176[2]	231[1]	219[2]
	Discharge Nozzle		528	360	313[1]	176[2]	231[1]	219[2]
3.	Anchor Bolts	ASME Section III	18,750 psi		9.617 psi			
4.	Dynamic Loads	Equivalent Static	(Horizonta) 1.5g (Vertical) 1.5g		0.41g 0.53g			
	1. SSE							
	2. SRV							
	3. LOCA							

(1) Unit 1

(2) Unit 2

TABLE 3.9-2n

(i) RHR Pumps and (ii) Core Spray Pumps

Location	Loading Condition	Criteria	Calculated Stress (psi)	Allowable Stress (psi)
(i) RHR PUMPS				
Suction Barrel Shell	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	17,672	21,000
Stuffing Box Pipe	Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	8,451	18,750
Nozzle Shell Intersection	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	15,193 (Suction) 20,392 (Discharge)	34,650 34,650
Discharge Elbow or Suction Pipe (Maximum)	FAULTED CONDITION Design Pressure Static Loads	ASME Boiler & Pressure Vessel Code, Section III	10,926 (Disch. Elbow)	21,000
Motor Stand	FAULTED CONDITION Dynamic Loads Static Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	5,717	21,000
Motor Bolting	FAULTED CONDITION Dynamic Loads Static Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	5,977	30,000
(ii) CORE SPRAY PUMPS				
Suction Barrel Shell	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	13,499	21,000
Stuffing Box Pipe	Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	6,766	18,750
Nozzle Shell Intersection	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	33,417 (Suction) 31,562 (Discharge)	34,650 34,650
Discharge Elbow or Suction Pipe (Maximum)	FAULTED CONDITION Design Pressure Static Loads	ASME Boiler & Pressure Vessel Code, Section III	11,061 (Disch. Elbow)	21,000
Motor Stand	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	4,505	21,000
Motor Bolting	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME B&PV, Section III, Subsection NF	1,960	30,000

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Min. Thickness Required	Actual Minimum Thickness
1. Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section III 1971, Winter 1972 Addenda		
Bolting Stress Limit			
Allowable working stress per ASME Section III	a. Shell-to-tube sheet bolts	1-3/8" dia.	1-3/8" dia.
	b. Channel cover bolts	1-3/8" dia.	1-3/8" dia.
2. Wall Thickness Loads: Normal and Upset Design pressure and temperature	Shell side ASME Section III Class 2 and TEMA Class C Tube side ASME Section III Class 3 and TEMA Class C		
Stress Limit			
ASME Section III	a. Shell	0.905 in.	1.0 in.*
	b. Shell cover	0.895 in.	0.895 in. min.
	c. Channel	0.932 in.	1.0 in.*
	d. Tubes 16 BWG	0.049 in.	0.065 in.*
	e. Channel cover	8.98 in.	9.00 in.*
	f. Tube sheet	7.42 in.	7.50 in.*

*Stresses within 10% of allowable.

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Nozzle Forces and Moments Force in lbs., Moment in in.-lbs.	Calculated Nozzle Forces and Moments
3. Nozzle Loads Design pressure and temperature	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits.	See below.	$\frac{Fi}{Fo} + \frac{Mi}{Mo} = 0.86$ (Inlet) $\frac{Fi}{Fo} + \frac{Mi}{Mo} = 0.53$ (Exhaust)
Dead weight, thermal expansion design basis earthquake	Primary stress less than 1.5 ASME Section III allowable	Maximum allowable piping loads for emergency conditions (including DBE) shall not exceed the following relationship for each nozzle.	
Allowable limits (design basis)		$\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$	

N1	N2	N3	N4	
Fx = 15,500	15,500	21,500	21,500	where Fi (lbs.) is the maximum of the three (3) orthogonal forces (Fx Fy Fz)
Fy = 15,500	15,500	21,500	21,500	
Fz = 15,500	15,500	21,500	21,500	
Mx = 60,000	60,000	100,000	100,000	and Mi (ft.-lbs.) is the maximum of the three (3) orthogonal moments (Mx My Mz)
My = 60,000	60,000	100,000	100,000	
Mz = 60,000	60,000	100,000	100,000	

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Min. Thickness Required (psi)	Actual (psi)
4. Support Brackets & Attachment Welds	SAP-4G Finite Element Computer Code Stress Analysis		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Stress Limits			
Allowables as per ASME Section III, Subsection NF (Upset Condition)	a. Lower Bracket Welds - Bending Stress - Shear Stress	 14,438 21,000	 5,990 3,304
	b. Upper Bracket Welds		
	- Bending Stress - Shear Stress	14,438 21,000	9,887 3,197
5. Anchor Bolts	Bolt loads calculated using SAP-4G finite Element Computer Code		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Bolting Stress Limits			
Allowable stresses as per ASME III, Appendix XVII	Lower Support Bolting		
Tension - 2461.1 Shear - 2461.2	Tension Shear	29,000 11,990	13,115 8,647

TABLE 3.9-2o

RHR HEAT EXCHANGER

Criteria	Method of Analysis	Allowable Min. Thickness Required (psi)	Actual (psi)
6. Shell Adjacent to Support Brackets	Finite Element and Localized Shell Stress Analysis		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Shell Stress Limit			
Allowables as per ASME Section III, Subsection NC	a. Maximum Principal Stress adjacent to upper support	28,875	22,432
	b. Maximum Principal Stress adjacent to lower support	28,875	23,463
7. Shell	Finite Element Dynamic Stress Analysis		
Loads: Faulted			
Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake			
Stress Limits			
Allowable as per ASME Section III, Subsection NC (Upset Condition)	- Principal Stress	19,250	18,560

TABLE 3.9-2(p)

RWCU PUMP

The following is a summary of design calculations by Hayward Tyler Fluid Dynamics Ltd. on the RWCU Pump:

Page 1 of 2

ASME Code Calculation	Required Thickness (in)	Allowable Stress (psi)	Actual Thickness (in)
Pump Part			
Body	0.8833	17,500	2.081
Drain Nozzle	0.0421	17,500	0.181
Suction Nozzle	0.2513	17,500	0.422
Suction End	3.688	17,500	3.713
Suction Bend	0.2518	15,000	0.43
Discharge Branch	0.1693	17,500	0.8615
Pump/motor Flange Studs	1.3393 (Note 1)	25,000	1.375 (Note 1)
Motor Part			
Motor Case	3.09	17,500	3.21
End Plate	1.5943	17,500	2.07
Gasket Seating	2.21	17,500	3.00
Thermal Neck	0.1871	17,500	1.3457
Motor Case Shell Body			
Pump End	0.54	17,500	0.995
Cover End	0.6058	17,500	2.8025
Motor Cover (Fill, Drain & Purge Nozzle)	0.0421	17,500	0.1986
Flange	2.585	17,500	2.826
Motor Case/Motor Cover Studs	1.3393 (Note 1)	25,000	1.375 (Note 1)

Note 1: Engagement Length

TABLE 3.9-2(p) (Continued)

ASME Code Calculation	Required Thickness (in)	Allowable Stress (psi)	Actual Thickness (in)
Heat Exchanger Part			
Heat Exchanger Shell Body	0.2404	17,500	0.331
Nozzles	0.4775	17,500	0.884
Vent Nozzle	0.0401	17,500	0.199
HP Nozzles	0.0884	17,500	0.47725
Heat Exchanger End Cover Body	0.0216	17,500	0.5575
Heat Exchanger End Cover End Plate	0.2076	17,500	1.12
Heat Exchanger LP Tubes		3,461 (Note 2)	1,450 (Note 2)

END COVER - FLANGE

ASME Code Calculation	Calculated Stress (psi)	Allowable Stress (psi)
Operating Condition		
S_H	2,354	(1.5S) 26,250
S_R	1,307	(1.0S) 17,500
S_T	962	(1.0S) 17,500
$\frac{S_H + S_R}{2}$	1,830	(1.0S) 17,500
$\frac{S_H + S_T}{2}$	1,658	(1.0S) 17,500
Gasket Seating Condition		
S_H	11,916	(1.5S) 26,250
S_R	6,617	(1.0S) 17,500
S_T	4,865	(1.0S) 17,500
$\frac{S_H + S_R}{2}$	9,267	(1.0S) 17,500
$\frac{S_H + S_T}{2}$	8,391	(1.0S) 17,500

Note 2: Pressure, (psi)

TABLE 3.9-2q

RCIC TURBINE

Criteria

The highest stressed components of the RCIC Turbine assembly are identified.
Allowable stresses for pressure retaining components are based on ASME B&PV Code, Section III.

Normal Condition:

Pressure Boundary Castings: SA216-WCB @ 500°F
S = 17,500 psi
S_A (General Membrane) = 0.8 x S
S_A (Bending) = 1.5 x 0.8 x S
Pressure Boundary Bolting: SA193-B7 @ 500°F
S_A = 1.0 x S S = 25,000 psi
Alignment Taper Pins: AISI 4037, Rc 28-35
T_a = 61,100psi S_y = 106,000 psi

Faulted Condition:

Pressure Boundary Castings: SA216-WCB @ 500°F
S = 17,500 psi
S_A = (General Membrane) = 1.2 x 0.8 x S
S_A = (Bending) = 1.8 x 0.8 x S
Pressure Boundary Bolting: SA193-B7 @ 500°F
S_A = 1.0 x S S = 25,000 psi
Alignment Taper Pins: AISI 4037, Rc 28-35
T_a = 61,100psi S_y = 106,000 psi

LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Normal Condition Loads:	Castings: 1) Stop valve	General Membrane	14,000	
1. Design Pressure	2) Governor valve	General Membrane	14,000	
2. Design Temperature	3) Turbine inlet	Local Bending	21,000	(1)
3. Deadweight	4) Turbine case	Local Bending	21,000	
4. Inlet Nozzle Loads	Pressure boundary bolts	Tensile	25,000	
5. Exhaust Nozzle Loads	Alignment taper pins	Shear	61,100	
Faulted Condition:	Castings: 1) Stop valve	General Membrane	16,800	9,800
1. Design Pressure	2) Governor valve	General Membrane	16,800	13,200
2. Design Temperature	3) Turbine inlet	Local Bending	25,200	15,300
3. Deadweight	4) Turbine case	Local Bending	25,200	18,000
4. Inlet Nozzle Loads	Pressure boundary bolts	Tensile	25,000	20,100
5. Exhaust Nozzle Loads	Alignment taper pins	Shear	61,100	46,880
6. Controlling combination of SSE, SRV, and LOCA				

TABLE 3.9-2q

RCIC TURBINE

Nozzle Load Definition:

Turbine vendor has defined allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied.

Normal Loads:
1. Design Pressure
2. Design Temperature/Thermal Expansion
3. Deadweight

Faulted Loads:(3)
1. Design Pressure
2. Design Temperature/Thermal Expansion
3. Deadweight
4. Controlling combination of SSE, SRV, and LOCA

Allowable Nozzle Load Criteria:

Inlet: $F = \frac{(2620 - M)}{3}$

Exhaust: $F = \frac{(6000 - M)}{3}$

Where: F = Resultant force (lbs.)
M = Resultant moment (ft.-lbs.)

Inlet: $F = \frac{(7000 - M)}{4.7}$

Exhaust: $F = \frac{(8500 - M)}{0.34}$ Not to exceed 7000 lbs.

Where: F = Resultant force (lbs.)
M = Resultant moment (ft.-lbs.)

NOTES:
(1) Calculated stresses for the faulted condition are lower than the allowable stresses for the normal condition. Therefore the normal, upset, and emergency conditions are not evaluated.
(2) Operability: Analysis indicates that shaft deflection with faulted loads is 0.008 inch; which is fully acceptable. And, maximum bearing loads under faulted conditions are acceptable. Furthermore, the turbine assembly has been seismically qualified via dynamic testing. This qualification included demonstration of start-up and shutdown capabilities, as well as no load operability during seismic loading conditions.

TABLE 3.9-2q

RCIC TURBINE

(3) The nozzle loads for the Upset loading condition, as determined by the piping analysis, shall satisfy the allowable criteria for the Faulted loading condition.

TABLE 3.9-2r

RCIC PUMP

Criteria

The critical components of the RCIC pump assembly are identified.
Allowable and calculated values are based on ASME Section III or VIII, as applicable.

COMPONENT	ALLOWABLE	CALCULATED (1)
Pump Hold Down Bolts: $\frac{ft^2}{Ftb^2} + \frac{fv^2}{Fvb^2}$	1	0.502
Anchor Bolt: Shear Stress Tensile Stress	10,000 psi 21,600 psi	7,422 psi 17,090 psi
Pump Outer Case	17,500 psi	7,052 psi
Pump Outer Case at Discharge Nozzle	26,250 psi	7,855 psi
Discharge Nozzle	17,500 psi	3,600 psi
Suction Elbow	18,000 psi	7,900 psi
Pump Shaft (2)	32,000 psi	5,975 psi
Impeller Key	9,000 psi	4,810 psi
Mounting Feet	17,500 psi	6,208 psi
Mounting Feet Weld Stress	9,625 psi	6,208 psi
Pump Pedestal Weld Stress	9,625 psi	1,868 psi
Base Plate & Plate Stiffener	21,600 psi	13,818 psi
Outboard Bearing – Axial (2)	17,200 lb.	1,323 lb.
Inboard Bearing – Axial (2)	7,670 lb.	376 lb.
Seal Circulation Piping - ½" - ¾"	15,000 psi 15,000 psi	10,000 psi 4,630 psi
Bypass Piping	15,000 psi	8,592 psi
Shaft: Relative Radial Displacement Between Shaft & Sleeve (2) Relative Radial Displacement Between Shaft & Mech. Seal (2) Angular Misalignment at Coupling (2)	.0055 in. .0055 in. .017 Rad.	.00383 in. .0008 in. .0022 Rad.
Impeller: Relative Radial Displacement Between Impeller & Casing (2)	.0075 in.	.00084 in.
Suction End and Discharge End Bolting	25,000 psi	20,740 psi

TABLE 3.9-2r

RCIC PUMP

Nozzle Load Definition:

Allowable nozzle loads for the pump have been defined. The above calculated stresses assume these allowable nozzle loads have been satisfied.

Units: Forces = lbs.
Moments = ft-lbs.

The allowable combinations of forces and moments are as follows:

$$\frac{F_i}{F_o} + \frac{M_i}{M_o} \leq 1$$

Where:

F_i = Largest absolute value of the three actual orthogonal forces (F_x, F_y, F_z) imposed by the interface pipe.
M_i = Largest absolute value of the three actual orthogonal moments (M_x, M_y, M_z) imposed by the interface pipe.
F_o = Allowable value of F_i when all moments are zero.
M_o = Allowable value of M_i, when all forces are zero.

Nozzle	Allowable Load (3)
Suction	F_o = 2906 M_o = 3688
Discharge	F_o = 4450 M_o = 5200

NOTES:

(1) Calculated values are due to the highest faulted condition loads and are less than 1.2 times the upset allowables. Therefore, the normal plus upset and emergency conditions are not evaluated.

TABLE 3.9-2r

RCIC PUMP

(2) Operability is addressed by the evaluation of the relative displacements between rotating and stationary components, shaft stress, and bearing loads under faulted condition loads. All criteria are met. Therefore, operability is ensured.

(3) Nozzle load allowables are applicable to all loading conditions.

TABLE 3.9-2s

NIMS Rev. 36

REACTOR REFUELING & SERVICING EQUIPMENT
(i) NEW FUEL STORAGE RACKS

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS (.7 ULT)	CALCULATED STRESS
1. NEW FUEL STORAGE RACKS Stress due to normal upset or emergency loading shall not cause a failure so as to result in a critical array.	FAULTED CONDITION "A" 1. Dead Loads 2. Full Fuel Load in rack 3. S.S.E. 4. Thermal (not applicable)	1. Beam (Axial) 2. Beam (Trans.) 3. Combined	1. 26,000#/in^2 2. 26,000#/in^2 3. 26,000#/in^2	1. 18,905#/in^2 2. 7,005#/in^2 3. 25,910#/in^2

2. SOURCE OF ALLOWABLE STRESS (.7 ULT)

a. ASTM B308 Alloy 6061-T6
b. ASME Code – Boilers and Pressure Vessels, Sect. III, NA
c. Product Safety Standards for BWR-6-Mark III, Sect. VI, A. (3)
d. ASME – Pressure Vessels and Piping: Design and Analysis, Volume One, Page 69.
e. ASTM code for Boilers and Pressure Vessels was selected on the premise that data used from this source would necessarily be on the conversative side as applied to the fuel storage rack calculations.

3. S. S. E. loads derived by dynamic analysis. Total stress refers to combined earthquake and thermal load at highest expected pool temperature. Earthquake stresses obtained by square root of the sum of the squares method for a response due to tri-axial excitation. Stress given is the highest in the total structural array. The calculated stresses are conservative since they are based on the original fuel assembly type which has the greatest mass of all fuel assemblies in use or in storage at SSES.

TABLE 3.9-2s

REACTOR REFUELING & SERVICING EQUIPMENT
(i) NEW FUEL STORAGE RACKS

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS (.7 ULT)	CALCULATED STRESS
4. NEW FUEL STORAGE RACKS Stresses due to normal upset or emergency loading shall not cause a failure so as to result in a critical array.	FAULTED CONDITION "B" (See Below, Par. 5)	(Location – See Par. 6, Below)	Not Applicable	Not Applicable
5. FAULTED CONDITION "B"	Condition "B" is an emergency condition in which the stress limit is equal to the yield strength at 0.2% offset. The racks were tested to determine their capability to safely withstand the accidental, uncontrolled, drop of a fuel bundle from its fully retracted position into the weakest portion of the rack.			
6. METHOD OF TESTING:	Four (4) rack castings were subjected to impact loads ranging from 1908 ft. lbs. To 4070 ft. lbs., which were generated by dropping simulated fuel bundles weighing 660 lbs. from heights varying from 3.0' to 6.17'. Racks were aligned in pairs and simulated bundles were dropped on both racks at the flange area. Both center impact and end impact tests were conducted. (Two (2) of the racks were X-ray examined prior to testing. Strain gauges were mounted on racks to ascertain max. strain and accelerometers were mounted on bundles to determine "G" loads.)			
7 TEST RESULTS:	A total of nineteen (19) tests were performed with drop height increased at each test. First failure occurred due to a central impact on rack No. 3 from a max. height of 6.17', (Test #13). Racks #1 and #2 both failed from a center impact caused by a load dropped from a height of 5.33', (Test #19). Accelerometer readings are not available due to the inability to adequately affix the accelerometer to the simulated fuel bundle.			

TABLE 3.9-2s

NIMS Rev. 36

REACTOR REFUELING & SERVICING EQUIPMENT
(ii) FUEL PREP MACHINE

ACCEPTABLE CRITERIA	LIMITING LOAD COMBINATION	PRIMARY LOADING	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)	OPERABILITY ASSURANCE DEMONSTRATED BY
AISC Code; ASME, Sect. III			Fu = 75,000 Fy = 30,000 S_m200 = 17,800		
Chain 302 Stainless Side Plates 17-4 PH or 17-7 PH Rollers					
Normal Condition	Static	Axial Load	17,800	12,300	Analysis
Upset Condition	N+OBE+SRV	Axial Load	24,000	18,900	Analysis
Emergency Condition	Analyzed As Upset Cond.	Axial Load	--	N/A	--
Faulted Condition	N+SSE+SRV+LOCA	Axial Load	52,500	32,200	Analysis

Note: The calculated stresses are conservative since they are based on the original fuel assembly type which has the greatest mass of all fuel assemblies in use or in storage at SSES.

TABLE 3.9-2t

HIGH PRESSURE COOLANT INJECTION PUMP

Location	Loading Condition	Criteria	Calculated Stress (psi)	Allowable Stress (psi)	Material
Pressure Boundary Parts					
Closure Bolting (Main)	Emergency/Faulted 1. Pressure 2. Design Temperature 3. Seismic 4. SRV 5. LOCA 6. Nozzle Loads	Allowables Based on Normal/Upset Condition Per ASME B&PV Code Section for Pressure Boundary Parts @ 140°F	15,498	25,000	A-193 GR.B7 Tensile Stress
Closure Bolting (Booster)			15,878	25,000	
Casing Wall (Main)			12,050	14,000	A-216 GR. WCB General Membrane
Casing Wall (Booster)			3,650	14,000	
Non Pressure Boundary Parts					
Pump Bolts (Booster) (Tensile) Booster Pump Mounting Foot	Emergency/Faulted 1. Pressure 2. Design Temperature 3. Seismic 4. Nozzle Loads 5. SRV 6. LOCA	Allowables Based on ASME B&PV Code Section IV @ 140°F 0.5 Su for Bolts 0.4 Sy for Pins	15,870	21,000	A=307 GR.B7 Su = 60,000 psi
Pump Bolts (Main) (Tensile) Main Pump Mounting Foot			19,918	21,000	
Dowel Pins (Booster) (Shear) Booster Pump Taper Pin			20,978	33,600	A-193 GR.B7 Sy =105,000 psi
Dowel Pins (Main) (Shear) Main Pump Taper Pin			22,517	33,600	

NOTE: Eight (8) anchor bolts, each carries the stresses for both units mounted on a common base plate.

TABLE 3.9-2u

CONTROL ROD DRIVE (Index Tube)

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi) (ABS)*
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiller and Pressure Vessel Code, Section III for Type 316 Stainless Steel @ 250°F S_m = 20000 psi				
For Normal and Upset Condition: S_{allow} = S_y for General Membrane ■ 1.49 for General Membrane and Bending	For Normal & Upset Condition: 1. Normal Loads (1) 2. Scram 3. OBE 4. SRV	General Membrane General Membrane and Bending	28,500 42,500	18,700 32,700
For Emergency Condition: (2)	For Emergency Condition: 1. Normal Loads (1) 2. Chugging 3. SRV 4. Scram	General Membrane and Bending	(2)	(2)
For Faulted Condition: S_{allow} = 0.80 Su for General Membrane = 2.16 S_y for General Membrane and Bending	For Faulted Condition: 1. Normal Loads (1) 2. SSE 3. Chugging 4. SRV	General Membrane General Membrane and Bending	56,500 61,560	<29,400 <32,700 (3) 29,400 (4)

NOTES:

(1) Normal loads include pressure, temperature, weight and mechanical loads.

(2) Less severe than the upset condition.

(3) Unit 1

(4) Unit 2

* The points of highest stress using the absolute sum (ABS) methodology are given. The ABS approach results in higher values than the square root sum of the squares (SRSS) methodology for corresponding load combination.

TABLE 3.9-2v

CONTROL ROD DRIVE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit – The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Section III, for Class I Vessels, for Type 304 Stainless Steel.	Normal and Upset Condition Loads: 1. Design Pressure 2. Stuck Rod Scram Loads 3. Operational Basis Earthquake, with Housing Lateral Support Installed.	Maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing for the normal, upset and emergency conditions.	16,600	15,710
For Normal and Upset Condition: S_m = 16,660 psi @ 575°F				
For Faulted Conditions: * S_{limit} = 2.4 S_m = 2.4 x 16,600 = 39,984 psi Note: Analyzed to Emergency Condition Limits.	Faulted Condition Loads: 1. Design Pressure 2. Stuck Rod Scram Loads 3. Design Basis Earthquake, with Housing Lateral Support Installed 4. Jet Reaction 5. Annulus Pressurization.		39,800	16,700

* Analyzed to Normal upset Condition Limits.

Table 3.9-2w

JET PUMPS

CRITERIA	LOADING CONDITIONS	STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Primary Membrane Plus Bending Stress Based on ASME B&PV Code Section III				
MATERIAL: TYPE 304SS				
A. For Service Levels A & B – NORMAL & UPSET CONDITION:				
S_m = 16,900 psi @ 550°F $P_m + P_b$ $S_{limit} = 1.5\ S_m$	1. Deadweight 2. Pressure 3. SRV 4. OBE	PRIMARY MEMBRANE PLUS BENDING	25,350	12,700
B. For Service Level C – EMERGENCY CONDITION:				
$P_m + P_b$ $S_{limit} = 2.25\ S_m$	1. Deadweight 2. Pressure 3. SRV 4. SBA	PRIMARY MEMBRANE PLUS BENDING	38,025	16,010
C. For Service Level D – FAULTED CONDITION:				
$P_m + P_b$ $S_{limit} = 3.0\ S_m$	1. Deadweight 2. Pressure 3. Chugging 4. SRV 5. SSE	PRIMARY MEMBRANE PLUS BENDING	60,840[(1)] 50,800[(2)]	38,416
D. MAXIMUM CUMULATIVE FATIGUE USAGE FACTOR ACCEPTANCE CRITERIA: <1.0: UNIT 1 PROJECTED CUMULATIVE USAGE FACTOR: = 0.94 UNIT 2 PROJECTED CUMULATIVE USAGE FACTOR: = 0.67 These CUFs account for the original 40-year plant design. These components are presently inspected on a regular basis. These inspections continue during the period of extended operation to ensure components continue to perform their intended functions.				
(1) Unit 1 (2) Unit 2				

TABLE 3.9-21
TABLE 3.9-22
TABLE 3.9-23
TABLE 3.9-24
TABLE 3.9-25
TABLE 3.9-26
TABLE 3.9-27
TABLE 3.9-28
TABLE 3.9-29
TABLE 3.9-30
TABLE 3.9-31
TABLE 3.9-32
TABLE 3.9-33

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TABLE 3.9-22
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TABLE 3.9-2aa

CONTROL ROD GUIDE TUBE FLANGE

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
CONTROL ROD GUIDE TUBE				
Primary Stress Limit – The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section III for Type 304				
S.S. S_m = 16,000 psi @ 575°F				
For normal and upset condition: S_{limit} = 1.5 S_m $P_m + P_b$	1. Dead weight 2. Pressure drop across guide tube 3. OBE 4. SRV 5. Scram	Primary membrane and bending (The maximum bending stress occurs at the guide tube base.)	24,000	11,563
For emergency condition: S_{limit} = 2.25 S_m $P_m + P_b$	1. Dead weight 2. Pressure 3. OBE 4. SRV 5. Chugging	Primary membrane and bending	36,000	26,103
For faulted condition: S_{limit} = 2.4 S_m $P_m + P_b$	1. Dead weight 2. Pressure 3. Chugging 4. SRV 5. SSE	Primary membrane and bending	38,400	27,010

TABLE 3.9-2ab

INCORE HOUSING

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Primary Stress Limit – The allowable primary membrane stress is based on ASME Boiler and Pressure Vessel Code, *Section III* for Class I vessels, for Type 304 stainless steel				
P_m = 16,660 psi @ 575°F				
For normal and upset conditions: $S_{limit} = P_m$	1. Design pressure 2. OBE 3. SRV	Maximum membrane stress occurs at the outer surface of the vessel penetration	16,660	6,000$^{(1)}$ 15,548$^{(2)}$
For faulted condition: $S_{limit} = 2.4\ P_m$	1. Design pressure 2. SSE 3. LOCA 4. Annulus pressurization	Maximum membrane stress occurs at the outer surface of the vessel penetration	39,840	6,450$^{(1)}$ 27,830$^{(2)}$

NOTE: The emergency condition loads are less than the normal/upset loads.

(1) Unit No. 1
(2) Unit No. 2

TABLE 3.9-2 (ac)

REACTOR VESSEL SUPPORT EQUIPMENT

CRD HOUSING SUPPORT

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TABLE 3.9-2 (ad)

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TABLE 3.9-2 (ae)

HPCI TURBINE DESIGN CALCULATIONS

TURBINE PART	CALCULATED	ALLOWABLE
Pressure Boundary Castings		
Stop Valve	8,975 psi	14,000 psi
Turbine Inlet (high pres.)	6,550 psi	14,000 psi
Turbine Wheel Case (low pres.)	6,000 psi	14,000 psi
Pressure Boundary Bolting		
Stop Valve	17,700 psi	20,000 psi
Turbine	18,290 psi	20,000 psi
Non-Pressure Boundary Components		
Turbine Shaft	5,000 psi	60,500 psi
Thrust Bearing	4,400 lbf	5,600 lbf
Journal Bearing	2,680 lbf	19,500 lbf
Stop Valve Yoke	13,500 psi	33,000 psi
Pedestal Dowel Pins	29,800 psi	61,100 psi
Pedestal Bolts	11,400 lbf	28,300 lbf

TABLE 3.9 - 2 (af), page 1 of 2

HIGH DENSITY SPENT FUEL RACKS

TYPES OF ANALYSIS PERFORMED

DYNAMIC ANALYSIS:

A dynamic modal analysis using the seismic, SRV, and LOCA response spectra was performed on a simplified model consisting of 6 racks (1 quadrant). The resulting loads on the corner module were extracted and a more detailed analysis performed.

STATIC ANALYSIS:

A detailed finite element (1364 elements) model of the corner module was developed and a static analysis performed using the loading results of the dynamic analysis. The section descriptions, allowable stresses and stress ratios for the detailed model are given on page 2 of this table.

FUEL RATTLING ANALYSIS:

A time history analysis was performed to determine local impact loads due to fuel rattling. A comparison of the support loads from the fuel rattling analysis with those of the response spectrum analysis showed that the fuel rattling results are less than or equal to the response spectrum results. Analysis of the poison can was completed using the local impact loads.

MODEL IMPACT ANALYSIS:

An equivalent static load was determined for the following drop conditions:

1) 18" fuel drop on corner of top casting

2) 18" fuel drop on middle of top casting

3) fuel drop full length through the cavity impacting bottom casting at the middle.

For the first 2 cases the equivalent static loads calculated were combined with dead load and applied to the detailed model. For the 3rd case, the ultimate load of the bundle shearing out of the fuel seat was determined and combined with dead load. This combined load was then applied to the detailed model.

TABLE 3.9-2 (af), page 2 of 2
HIGH DENSITY SPENT FUEL RACK
SUMMARY OF RESULTS FOR THE DETAILED MODEL ELEMENTS

SECT. NO.	SECTION DESCRIPTION	NORMAL ALLOWABLE STRESSES			NORMAL OPERATING CONDITION				DESIGN ACCIDENT AND EXTREME ENVIRONMENTAL CONDITIONS			
		F_a	F_{by}	F_{bx}	$\frac{f_a}{F_a}$	$\frac{f_{by}}{F_{by}}$	$\frac{f_{bx}}{F_{bx}}$	MAX STRESS RATIO (1)	$\frac{f_a}{F_a}$	$\frac{f_{by}}{F_{by}}$	$\frac{f_{bx}}{F_{by}}$	MAX STRESS RATIO(1)
1	Top Grid Outer Section	9941	15760	15760	.026	.009	.747	.78	.018	.006	.715	.74
2	Top Grid Inner Section	9420	15760	15760	.057	.055	.813	.93	.040	.039	.766	.85
3	Bottom Grid Outer Sect.	8830	15760	12120	.062	.248	.108	.42	.062	.248	.108	.42
4	Bottom Grid Inner Sect.	8550	15760	12120	.005	.831	.013	.85	.005	.831	.013	.85
5	Bottom Grid Outer Section Near Leg	9650	15760	12120	.047	.249	.269	.57	.047	.249	.269	.57
6	Bottom Grid Inner Section Near Leg	9530	15760	12120	.046	.508	.248	.80	.046	.508	.248	.80
7	Bottom Grid Foot	10250	15760	12120	.132	0	.001	.13	.160	0	.003	.16
8	Bottom Grid Foot	11020	14180	14180	.161	0	.003	.16	.195	0	.006	.20
9	1/2" Plate	3320	F_v = 1390		-	-	-	.99(2)	-	-	-	.76(2)
10	7/8" Plate	17370	F_v = 10970		-	-	-	.92(2)	-	-	-	.92(2)

(1) Stress Ratio $= \frac{f_a}{F_a} + \frac{f_{by}}{F_{by}} + \frac{f_{bx}}{F_{bx}}$

(2) Plate Stress Ratio $= \frac{f_y}{F_a} + \frac{f_x}{F_a}$

NOTE
Allowable stresses are factored up per Table 9.1-7a of the SSES-FSAR.

Table 3.9-2d.1
ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT –

HIGHEST STRESS SUMMARY – UNIT 2

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS – NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq. 9 $\leq$ 1.5 S_m	Primary	26,480 psi	26,550 psi	0.997	1. Pressure 2. Weight 3. OBE 4. SRV	M.S. Line ‘A’ Sweepolet 990
Service Levels A&B (Normal & Upset) Condition: Eq. 12 $\leq$ 3.0 S_m	Secondary	32,295 psi	54,252 psi	0.595	1. Thermal	M.S. Line ‘C’ Sweepolet 641
Service Levels A&B (Normal & Upset) Condition: Eq.13 $\leq$ 3.0 S_m	Primary Plus Secondary (Except Thermal Expansion)	54,046 psi	54,252 psi	0.996	1. Pressure 2. Weight 3. OBE 4. SRV	M.S. Line ‘A’ Sweepolet 790
Service Levels A&B (Normal & Upset) Conditions: Cumulative Usage Factor	N/A	0.683*	1.0	0.683		M.S. Line ‘B’ TTJ @ Sweepolet 932
Service Level B (Upset) Condition: Lower{Eq.9 $\leq$ 1.8 S_m Of {EQ 9 $\leq$ 1.5S_y	Primary	30,010 psi	31,860 psi	0.942	1. Pressure 2. Weight 3. OBE 4. SRV	M.S. Line ‘D’ Sweepolet 90
Service Level C (Emergency) Condition: Lower{Eq.9<2.25 S_m Of {Eq 9 < 1.8 S_y	Primary	30,110 psi	39,820 psi	0.756	1. Pressure 2. Weight 3. SBA 4. OBE 5. SRV	M.S. Line ‘D’ Sweepolet 75
Service Level D (Faulted) Condition: Eq. 9 <3.0 S_m	Primary	58,915psi	60,000 psi	0.982	1. Pressure 2. Weight 3. SSE 4. LOCA 5. IBA 6. SRV	M.S. Line ‘A’ Sweepolet 790

Table 3.9-2d.1 (Continued)

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT – UNIT 2

COMPONENT/LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Snubber/ Service Level 'B' Loads	44,848	50,000	0.897	1. Pressure 2. Weight 3. OBE 4. Operating transient	M.S. Line 'A' 120; H42
Snubber/ Service Level 'C' Loads	42,043	70,350	0.598	1. Pressure 2. Weight 3. SBA 4. Operating transient	M.S. Line 'A' 120; H42
Snubber/ Service Level 'D' Loads	74,120	91,000	0.815	1. Pressure 2. Weight 3. IBA 4. SSE 5. Operating transient	M.S. Line 'A' 120; H42
SRV/Horizontal Acceleration	4.063	5.2 g	0.781	1. Pressure 2. Weight 3. LOCA 4. SSE	Line 'C' SRV 'R'
SRV/Vertical Acceleration	2.499 g	4.4 g	0.568	1. Pressure 2. Weight 3. LOCA 4. SRV	Line 'D' SRV 'K'
MSIV Bonnet/ Axial Force	7,340	41,221 lb	0.178	1. Pressure 2. Weight 3. LOCA 4. SSE	Line 'A' MSIV Bonnet
MSIV Bonnet/ Bending Moment	1,369,899 in-lbs	1,589,000 in-lb	0.862	1. Pressure 2. Weight 3. LOCA 4. SSE	Line 'B' MSIV Bonnet

* This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.

Table 3.9-2e.1
ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT – HIGHEST STRESS SUMMARY – UNIT 2

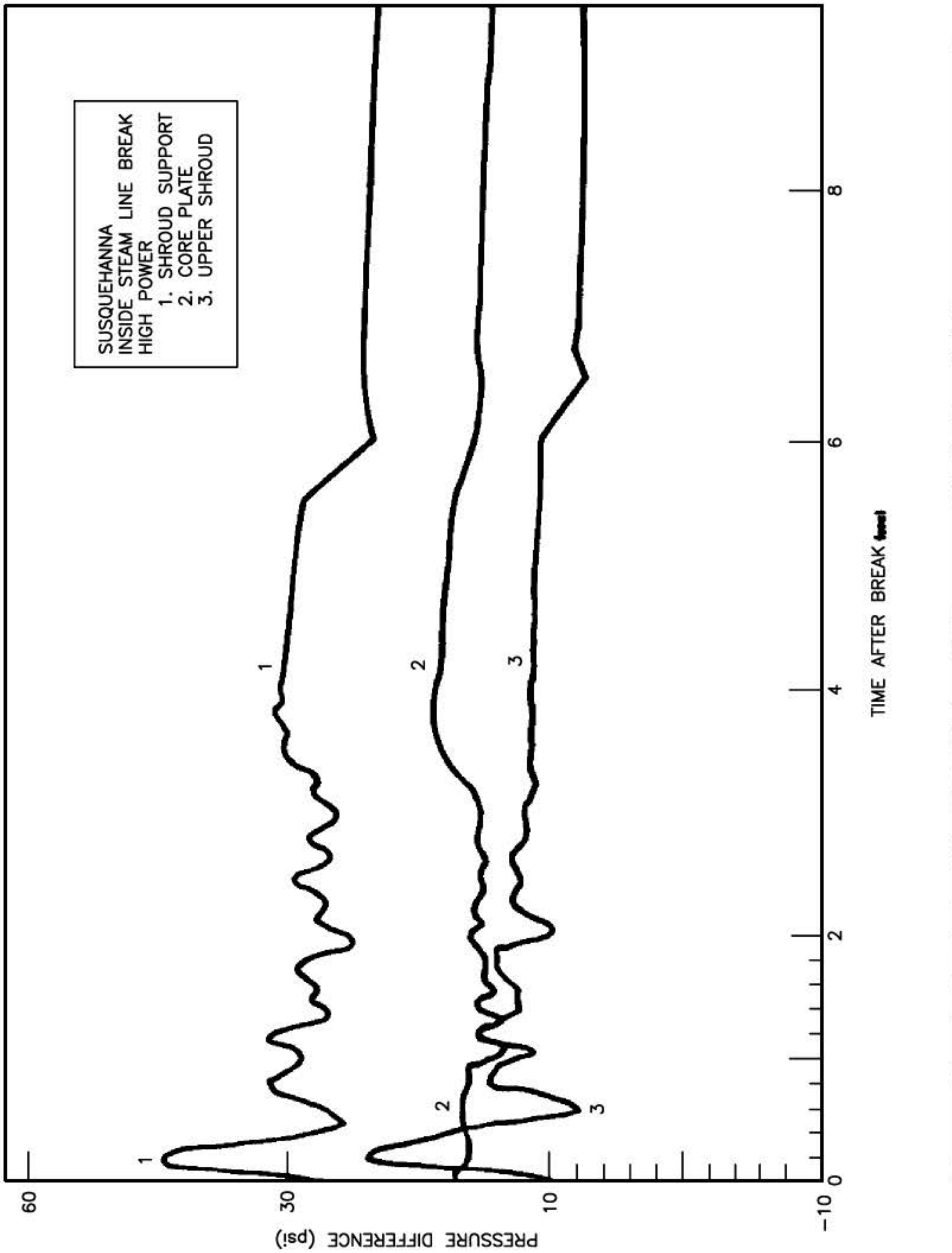
ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO ACTUAL/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATIONS OF HIGHEST STRESS POINTS – NODG POINT NUMBERS
ASME B&PV Code Section III, NB-3600						
Design Condition: Eq.9 $\leq 1.5\ S_m$	Primary	24,712 psi	25,013 psi	0.988	1. Pressure 2. Weight 3. OBE 4. SRV	Loop 'B' E bow 765
Service Levels A & B (Normal & Upset) Condition: Eq.12 $\leq 3.0\ S_m$	Secondary	35,891 psi	50,025 psi	0.717	1. Thermal	Loop 'B' Weldolet 305
Service Levels A & B (Normal & Upset) Condition: Eq.13 $\leq 3.0\ S_m$	Primary Plus Secondary (Except Thermal Expansion)	49,585 psi	50,025 psi	0.991	1. Pressure 2. Weight 3. OBE 4. SRV	Loop 'B' Half Coupling 300
Service Levels A & B (Normal & Upset) Conditions: Cumulative Usage Factor	N/A	0.7938**	1.0	0.7938		Loop 'B' Half Coupling 380
Service Level B (Upset) Condition: Lower{Eq. 9 $\leq 1.8\ S_m$ Of {EQ 9 $\leq 1.5\ S_y$	Primary	26,160 psi	27,750 psi	0.943	1. Pressure 2. Weight 3. OBE 4. SRV	Loop 'A' E bow 575
Service Level C (Emergency) Condition: Lower{Eq. 9 $< 2.25\ S_m$ Of {Eq. 9 $< 1.8\ S_y$	Primary	26,650 psi	33,300 psi	0.800	1. Pressure 2. Weight 3. SRV 4. SBA 5. OBE	Loop 'B' Weldolet 275
Service Level D (Faulted) Condition: Eq. 9 $< 3.0\ S_m$	Primary	41,120 psi	50,025 psi	0.822	1. Pressure 2. Weight 3. LOCA 4. SSE 5. SRV 6. IBA	Loop 'A' 4 inch Weldolet 315

Table 3.9-2e.1
ASME CODE CLASS 1 RECIRCULATION AND PIPE MOUNTED EQUIPMENT – UNIT 2

COMPONENT/ LOAD TYPE	HIGHEST CALCULATED LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Strut/ Service Level 'B' Loads	57,612	62,800	0.92	1. Pressure 2. Weight 3. OBE 4. Operating transient	Loop 'A' PO1; H50
Strut/ Service Level 'C' Loads	79,132	86,250	0.92	1. Pressure 2. Weight 3. SBA 4. Operating transient	Loop 'A' PO1; H50
Strut/ Service Level 'D' Loads	96,643	105,340	0.92	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' PO1; H50
Discharge Gate Valve/Horizontal Acceleration	7.678	9.8	0.783	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop A' Operator
Discharge Gate Valve/ Vertical Acceleration	2.118	2.188	0.968	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'B' Operator
By-Pass Valve/ Horizontal Acceleration (East-West)	5.438	5.468	0.995	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'B' Operator
By-Pass Valve/ Horizontal (North-South)	2.539	2.539	1.000	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop A'
Vertical Acceleration	2.376	2.376	1.000	5. Pressure 6. Weight 7. LOCA 8. Operating transient	Loop 'A' Body
Suction Valve/ Horizontal & Vertical Acceleration*	7.714	9.87	0.782	1. Pressure 2. Weight 3. LOCA 4. Operating transient	Loop 'A' Operator
Recirc. Pump/Horizontal Acceleration	2.386	2.7	0.884	1. Pressure 2. Weight 3. LOCA 4. SSE	Loop 'A' Pump Motor C.G.
Vertical Acceleration	1.388	2.1	0.661	1. Pressure 2. Weight 3. LOCA 4. SSE 5. SRN	Loop 'B' Pump Motor C.G.

* Allowable acceleration applies to operator only; valve body is qualified based upon allowable moment and allowable axial force.

** This CUF accounts for the original 40-year plant design. During the period of extended operation, actual fatigue design margins are periodically evaluated in accordance with the Plant Transient and Fatigue Monitoring Program.



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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TRANSIENT PRESSURE DIFFERENTIALS
FOLLOWING A STEAMLINE BREAK AT
105% RATED STEAM FLOW
100% RECIRCULATION FLOW

FIGURE 3.9-1, Rev. 48

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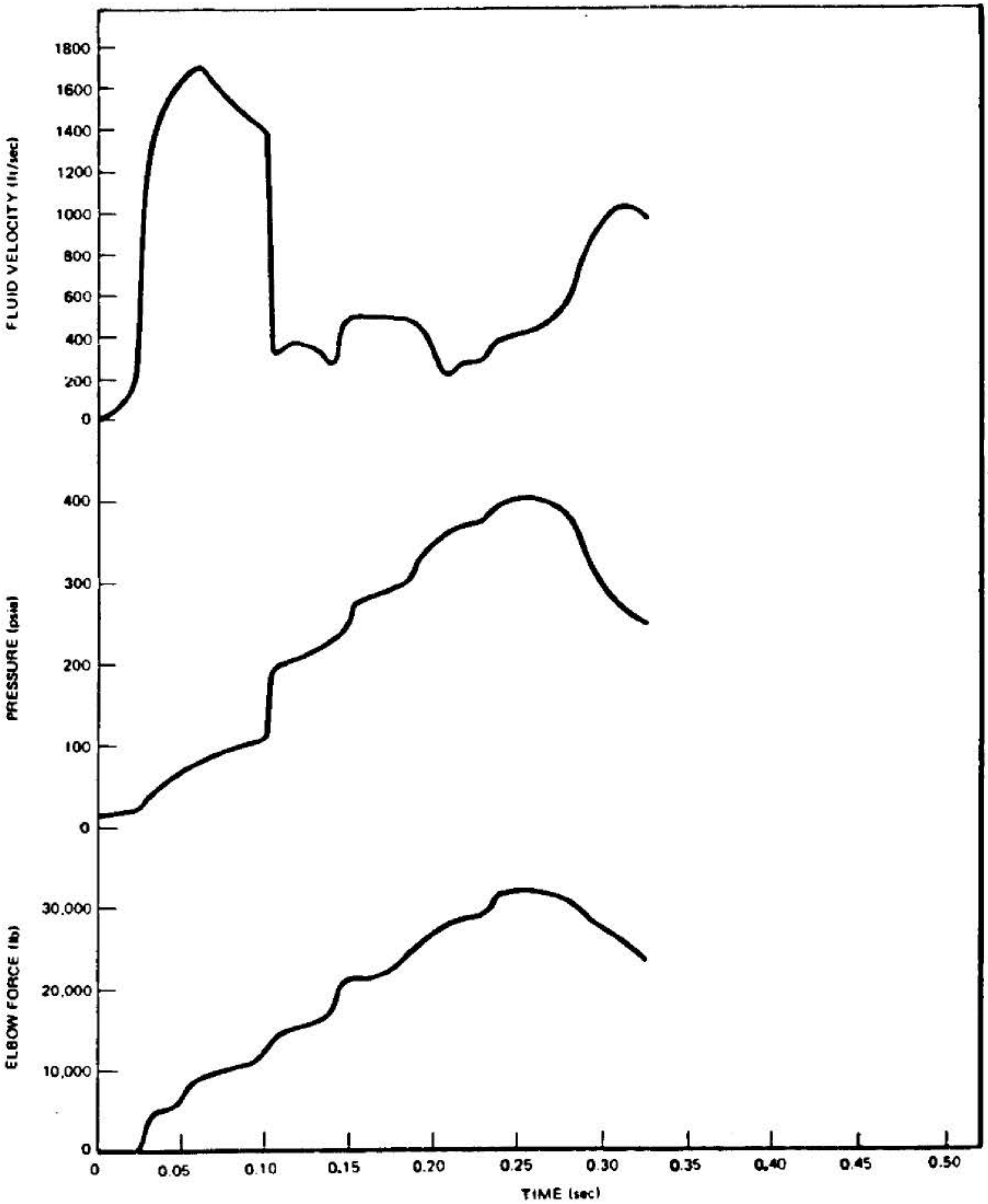

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TYPICAL RELIEF VALVE TRANSIENT

FIGURE 3.9-2, Rev. 48

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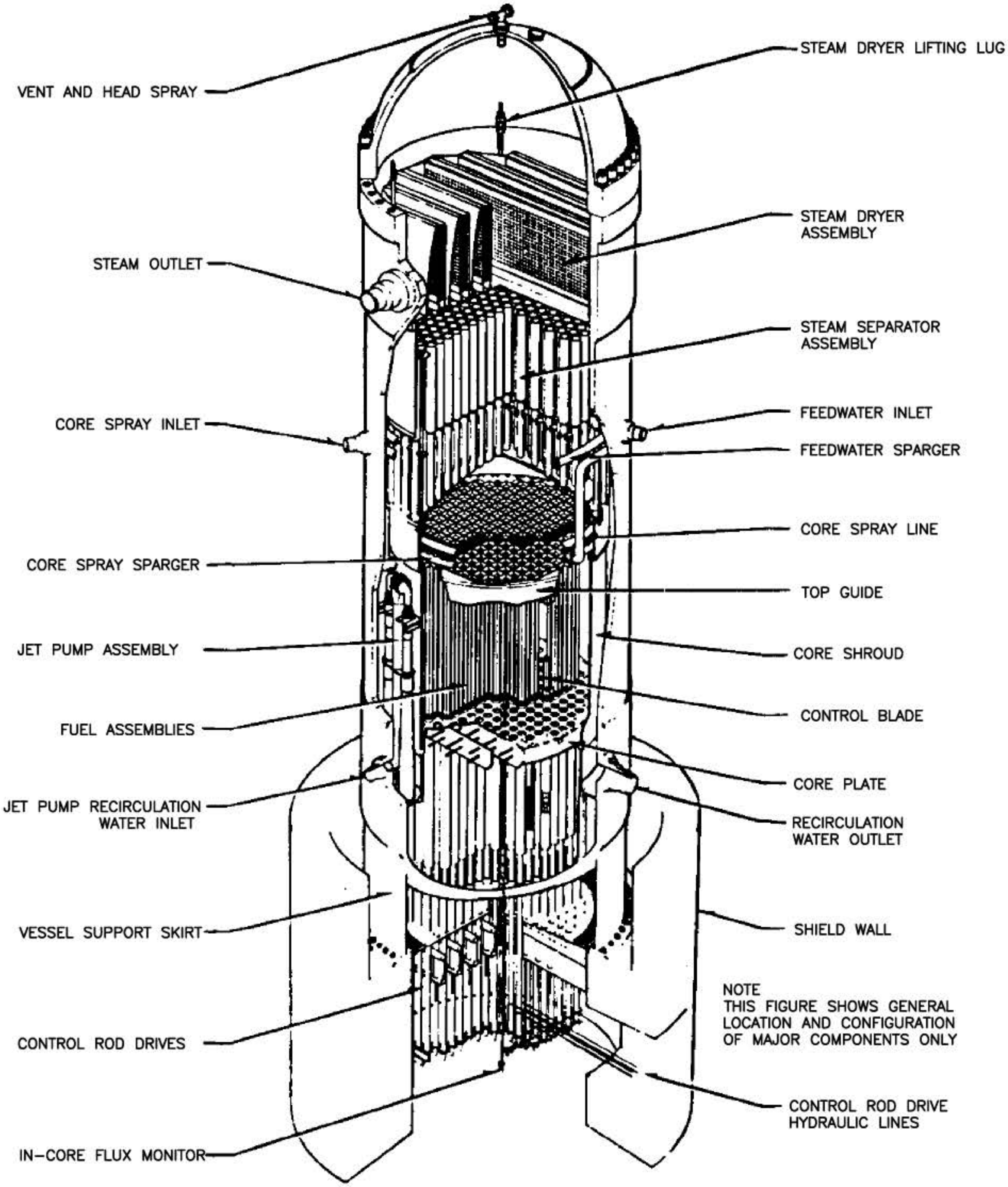

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REACTOR VESSEL CUTAWAY

FIGURE 3.9-3, Rev. 48

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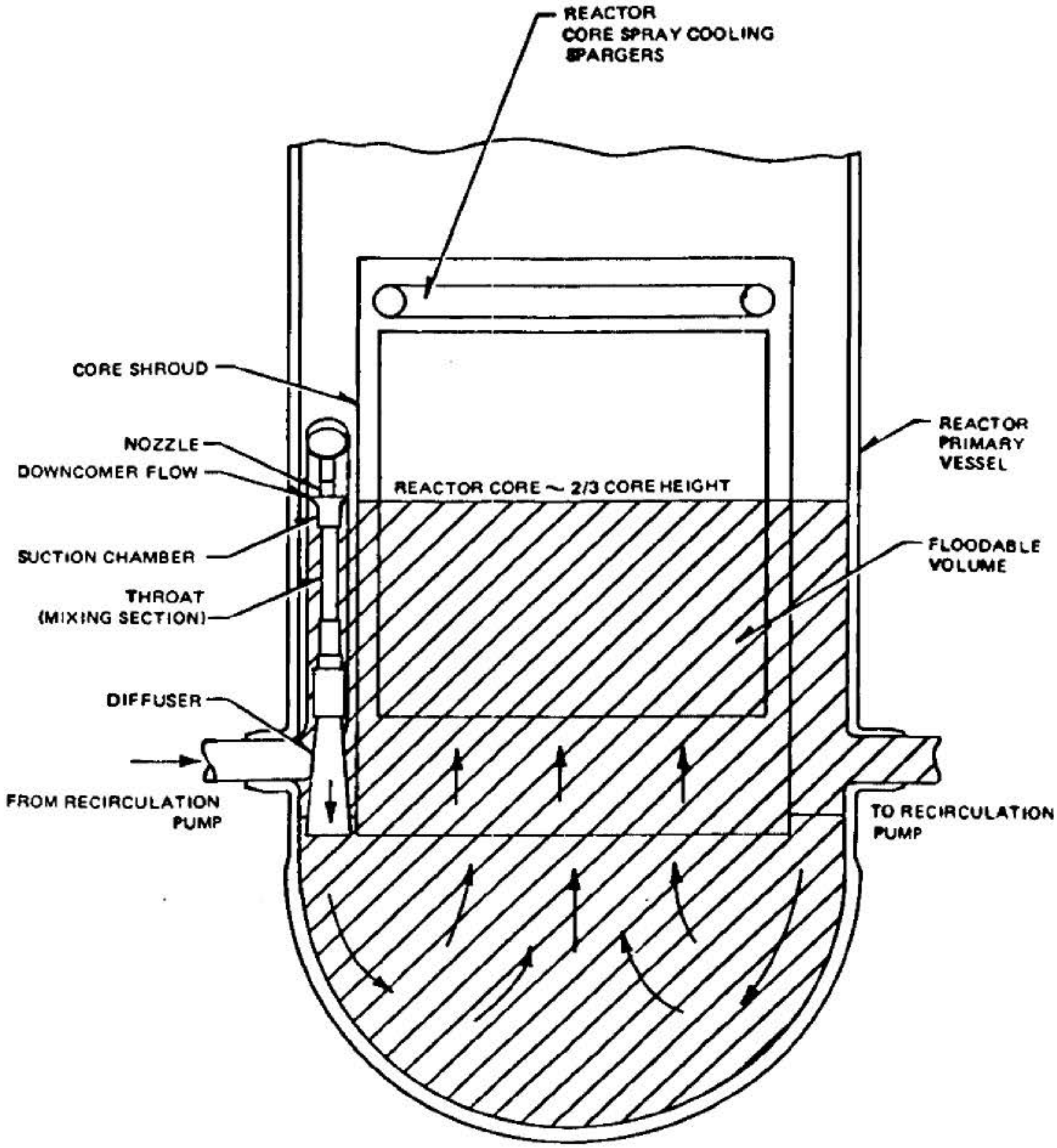

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR INTERNALS FLOW PATHS FIGURE 3.9-4, Rev. 48

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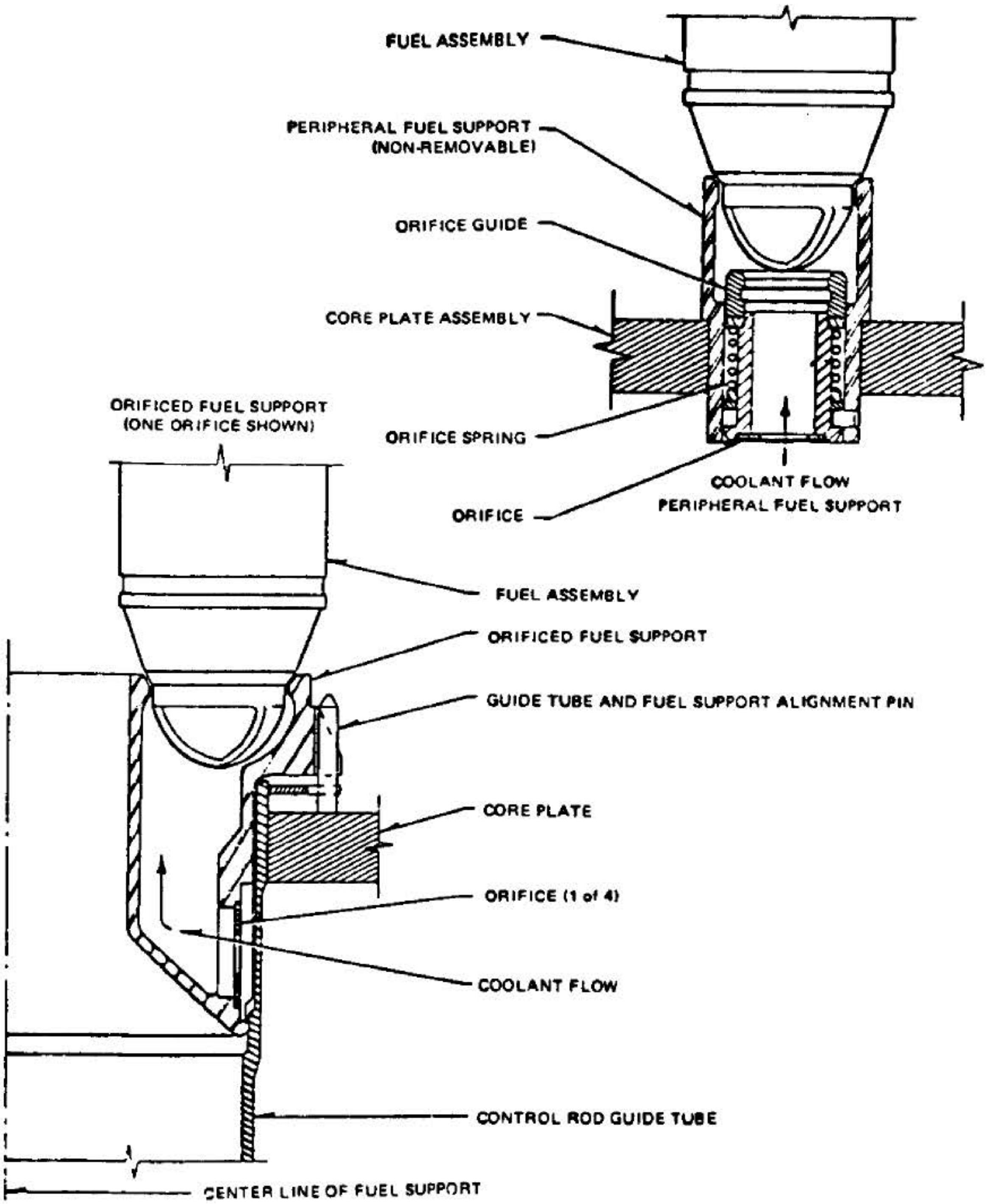

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UNITS 1 & 2
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FUEL SUPPORT PIECES

FIGURE 3.9-5, Rev. 48

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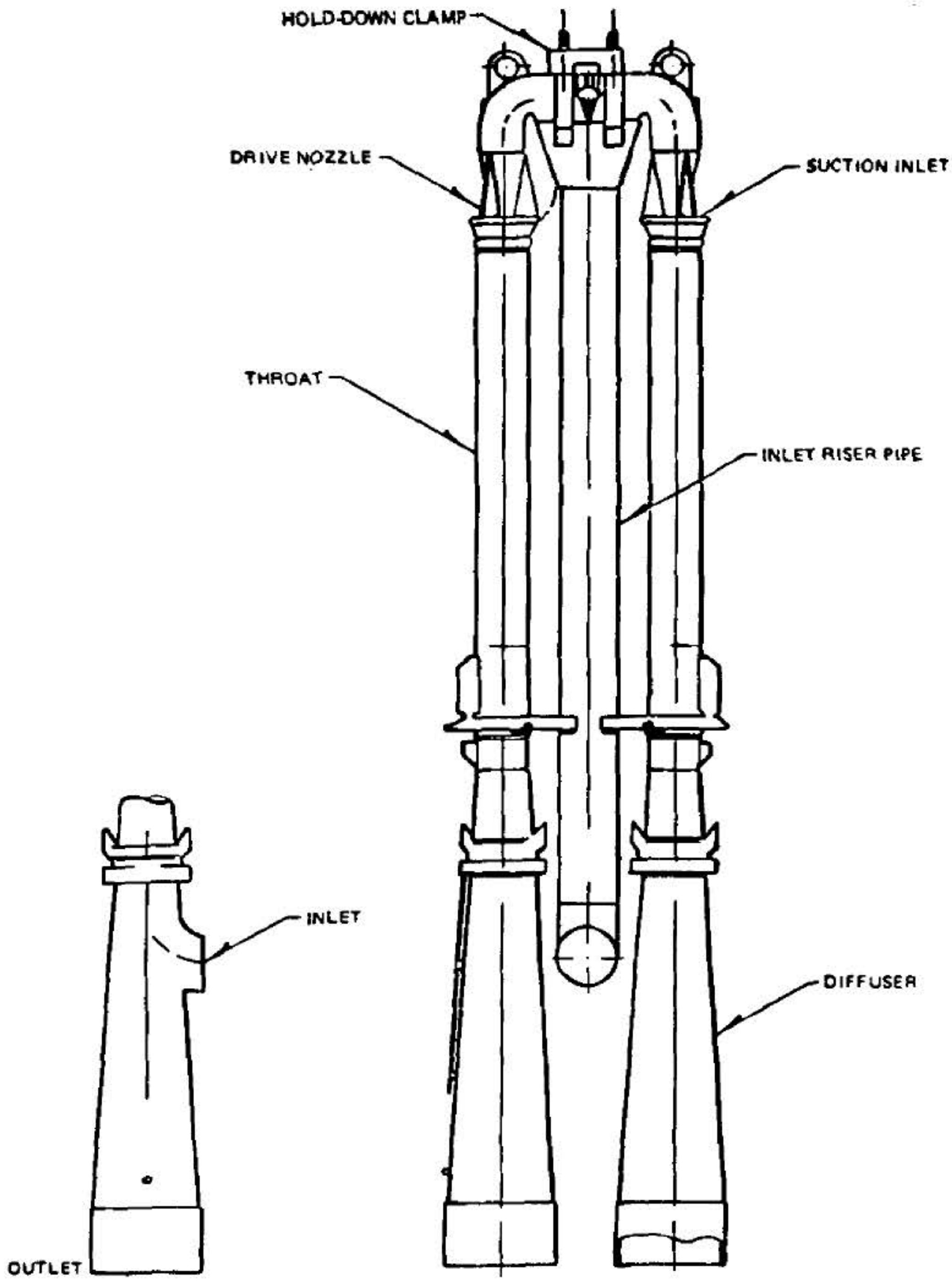

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

JET PUMP

FIGURE 3.9-6, Rev. 48

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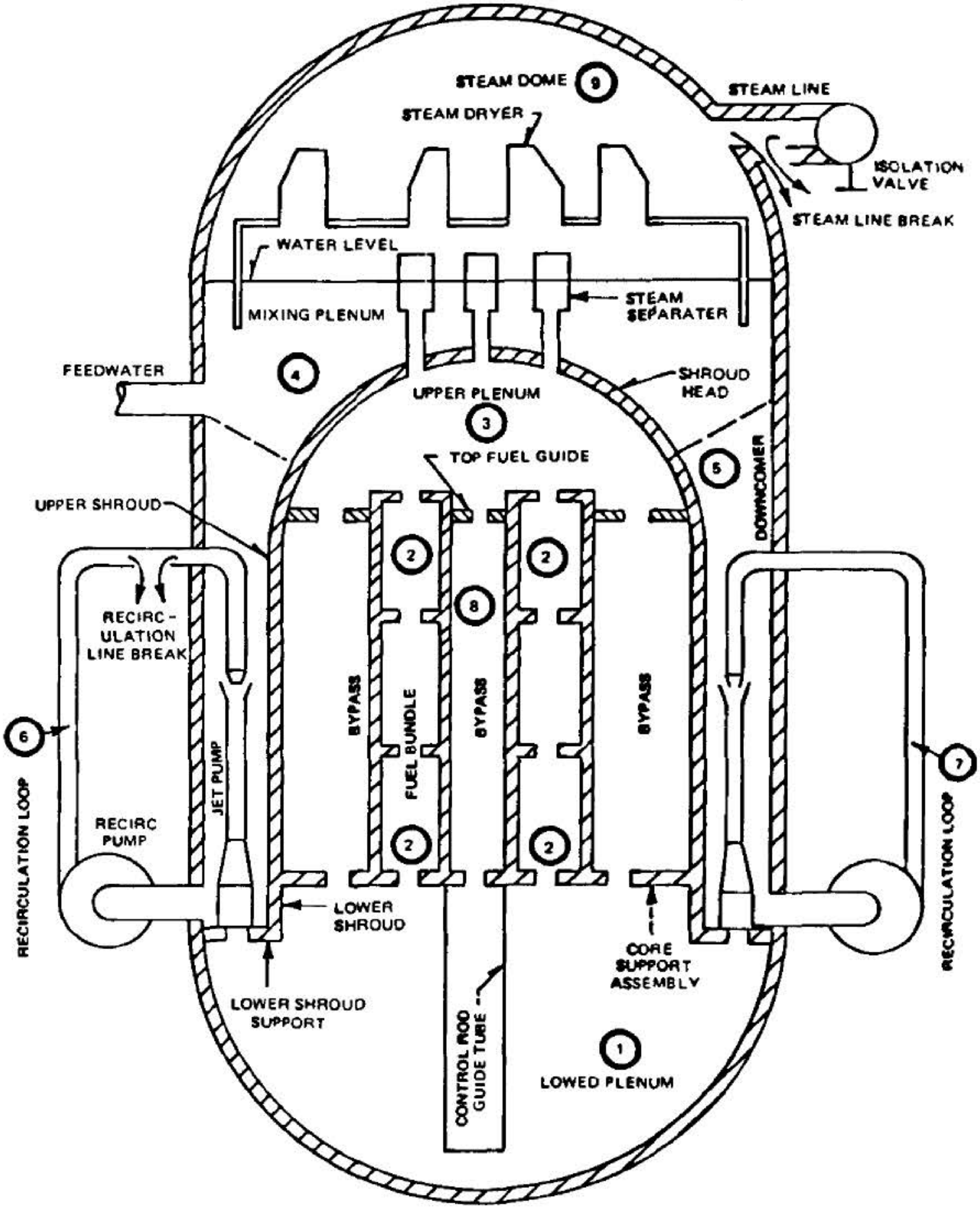

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UNITS 1 & 2
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PRESSURE NODES USED FOR
DEPRESSURIZATION ANALYSIS

FIGURE 3.9-7, Rev. 48

Auto-Cad Figure Fsar 3_9_7.dwg

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FIGURE 3.9-9, Rev. 48

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FIGURE 3.9-10, Rev. 48

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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

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FIGURE 3.9-11, Rev. 48

AutoCAD Figure 3_9_11.doc

APPENDIX 3.9A

COMPUTER PROGRAMS

START HISTORICAL

Introduction

The computer programs discussed in this section are those programs used by Bechtel for the original plant design. Changes to later versions of these programs or the addition of entirely new computer programs for safety related applications is controlled by procedures under our Operational Quality Assurance Program.

3.9A.1 ME101

Program Description

ME101 is a finite element computer program which performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program. In addition, modifications aimed at achieving an improved model are performed automatically.

The output may be used directly for piping design and for conformation to code and other regulatory requirements. Two piping codes, ASME BPV code 1974 and B31.1 Summer 1973 addenda are incorporated in the program to the extent of computing flexibility factors, stress intensification factors and stresses.

ME101 may be used for static and seismic load analysis of piping systems and also performs effective weight calculations.

Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly distributed loads, and externally applied forces, moments, displacements and rotations, or individual force loads.

Seismic analysis is based on standard normal mode techniques and uses response spectrum data. Two methods of eigen value solution are available. Determinant Search or Subspace Iteration considers all data points as mass points. Kinematic Reduction and Householder QR considers masses only at specified data points in designated directions. Differential seismic anchor movement analysis and static seismic analysis are also provided.

ME101 generates isometric plots of the piping configuration with optional node numbering. The plots are obtained by either ZETA or CALCOMP 1036 plotter. The program uses out-of-core solution techniques for both static and dynamic analysis, and

has no practical limitations to the number of equations or band width. However: very large systems may become prohibitive due to cost of computation. The maximum number of mode shapes allowable is currently 125.

Program Version and Computer

The current UNIVAC version (C3) of ME101 is being used by Bechtel Power Corporation.

Extent of Application

ME101 is a piping program developed by Bechtel Power Corporation (BPC). Its development began in July 1975 and is being continuously supported by BPC. It has been used by various projects in the BPC.

Test Problems

The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three-dimensional structure as described in Reference 3.9A-1.

The following table lists the natural frequencies from ME101 and Reference 3.9A-1:

Natural Frequency Comparisions, CPS		
Mode No.	*Reference 3.9A-1*	*ME 101*
1	*110*	*112*
2	*117*	*116*
3	*134*	*138*

Additional test problems can be found in Reference 3.9A-2.

3.9A.2 ME632

Program Description

ME632 performs stress analysis of 3-dimensional piping systems. The effects of thermal expansion, uniform load of the pipe, pipe contents and insulation, concentrated loads, movements of the piping system supports, and other external loads, such as wind and snow, may be considered. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program.

The output may be used directly for piping design and for conformation to code and other regulatory requirements. Piping codes, ASME BPV code, B31.1 code and B31.3

code have been incorporated in the program to the extent of computing flexibility factors, stress intensification factors and stresses.

A response spectrum analysis may be performed to analyze the effect of earthquake forces on the piping system, and transient effects of water hammer, steam hammer, or other impulsive type dynamic loading are also handled by the program. Also, a plot of piping geometry and/or response spectrum curves may be obtained to verify the accuracy of the model.

Program Version and Computer

The current UNIVAC version (B9) of ME632 is being used by Bechtel Power Corporation.

Extent of Application

ME632 is a piping program developed by Bechtel Power Corporation (BPC). Its development began in 1970 and is being continuously supported by BPC. It has been used by various projects in the BPC.

Test Problems

The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three-dimensional structure as described in Reference 3.9A-1.

The following table lists the natural frequencies for ME632 and Reference 3.9A-1:

Natural Frequency Comparisions, CPS		
Mode No.	*Reference 3.9A-1*	*ME632*
1	*110*	*111*
2	*117*	*116*
3	*134*	*137*

Additional test problems can be found in Reference 3.9A-3.

3.9A.3 ME912

Program Description

Finite-difference representation of the heat diffusion equation is used for the pipes or component wall section in contact with fluid of specified temperature and flow rate time histories. The program is quasi-two-dimensional, so that reduction of severity of a given transient with distance from inlet is accounted for.

Thermal properties of water, liquid sodium, stainless and carbon steel are built in the program. Film transfer coefficients for water or liquid sodium are computed by the program for each time step and pipe section. For other fluids such as steam, the program is used on a one-dimensional basis with user supplied film coefficients. Sequential computations are done for pipe lengths of different diameters or wall thicknesses. Fluid outlet temperature data from one pipe length are stored for use as inlet to the next length downstream. Average temperature differences T_a - T_b are thus calculated for structural discontinuity.

Program Version and Computer

The ME912 program has been used by Bechtel Power Corporation in Gaithersburg, and San Francisco offices on various BPC projects. A Univac 1110 computer is used to run the ME912 program.

Extent of Application

The ME912 program was developed from References 3.9A-4, 3.9A-5 and 3.9A-6 by the Stress Group of Gaithersburg and San Francisco offices of BAC. The ME912 program has been extensively used since 1975 for nuclear Class I component design on FFTF project.

Test Problem

For local gradients, the program has been compared with analytical flat plate data of Ref. 3.9A-5 and numerical results by in-house program ME643, Ref. 3.9A-7. The results were acceptable. For axial variations of fluid and wall temperatures, the program agrees closely with the analytical solution of Ref. 3.9A-6. Table 3.9A-2 shows the comparison of ME912 with ME 643 and analytical results.

The ME643 program was developed from References 3.9A-11 and 3.9A-12 by the Stress Group of Los Angeles Power Division of BPC. The results of ME 643 transient temperature responses on both inside and outside surfaces are compared with Chart 36 of Reference 3.9A-13 and plotted Figure 3.9A-1.

3.9A.4 ME 913

Program Description

ME913 can determine stress intensity levels for Class 1 nuclear power piping components for Equations 9 through 14 of subarticle NB-3650, ANALYSIS OF PIPING COMPONENTS of Section III, ASME Boiler and Pressure Vessel Code. Before attempting to exercise this program, the user should be familiar with the requirements and procedures set forth in subarticle NB-3650.

Prior to using this program, the user should have the following information external to the program.

1. *Piping configuration*

2. *Piping and piping component properties*

3. *Moment reactions due to*

   a. *Thermal expansion loads*
   b. *Weight loads and*
   c. *Earthquake loads*

4. *The thermal response of the piping system due to the specified transients:*

   ΔT_1, ΔT_2, *and the* $(T_a - T_b)$ *values for the key points during system life.*

Program Version and Computer

The current ME913 version is being used by Bechtel Power Corporation in its Gaithersburg, Los Angeles, Ann Arbor and San Francisco offices. A Univac 1100 computer is used to run the ME913 program.

Extent of Application

ME913 is the revised and expanded version of the LOTEMP program which was originally developed by the pipe stress group of the San Francisco Power Division of BPC and made available for use through the CDC 6600 computer. The LOTEMP program has been extensively used by the Bechtel Fast Flux Test Facility (FFTF) Systems Analysis Group since 1972 in the preliminary design of FFTF Class I piping. The ME913 program has been used to analyze nuclear Class I piping for Bechtel nuclear power plant projects.

Test Problems

The Grand Gulf Project feedwater line was selected as a test problem. Hand calculations of a selected component in the piping system were performed in accordance with the sample problem (Reference 3.9A-8). Their results were compared with the computer output for code equations 9 through 14 in ME 913 (Reference 3.9A-10).

Table 3.9A-1 shows the comparison between the ASME sample problem (Reference 3.9A-8) and ME913 results (Butt Welding Tee, Location 10).

3.9A.5 References

3.9A-1 Pressure Vessel and Piping 1972 Computer Programs Verification, the American Society of Mechanical Engineers.

3.9A-2 Verification Report on ME101, Linear Elastic Analysis of Piping Systems, Revision 1, February, 1977, Bechtel Power Corporation.

3.9A-3 Verification Report on ME632, Seismic Analysis of Piping Systems, October 1977, Bechtel Power Corporation.

3.9A-4 Tung, T.K. and Chem,, C.Y., "DELTAT, a Quasi-Two-Dimensional Program for Pipe Thermal Transients," ASME PVP-36, June 1979.

3.9A-5 McNeil, D. R. and Brock, J. E., "Charts for Transient Temperatures in Pipes," Heating/Piping/Air Conditioning, Nov. 1979, pp. 107-119.

3.9A-6 Carslaw, H.S. and Jaeger, J.C., "Conduction of Heat in Solids," Oxford University Press, 1959, pp. 392-394.

3.9A-7 ME643 Thermal and Stress Analyses Program Stress Group, Los Angeles Power Division, October 1977, Bechtel Power Corporation.

3.9A-8 "Sample Analysis of a Class I Piping System," prepared by the Working Group on Piping (SGD, ScIII) of the ASME Boiler and Pressure Vessel Code, December 1971.

3.9A-9 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1974 Edition.

3.9A-10 ME9I3 Verification by Bechtel Power Corporation, February 1975.

3.9A-11 Wilson, E. and Nickell, S.R., "Application for the Finite Element Method to Heat Conduction Analysis," Nuclear Engineering and Design 4, 1966.

3.9A-12 Wilson, E., "Structural Analysis of Axisymmetric Solids", AIAA Journal, Vol. 3, No. 1 12, December 1965.

3.9A-13 Schneider, P. J., "Temperature Response Charts," John Wiley and Sons, Inc., 1963.

END HISTORICAL

TABLE 3.9A-1

COMPARISON BETWEEN SAMPLE PROBLEM AND
COMPUTER PROGRAM ME 913 RESULTS

	ME 913	Sample Problem (Ref. 3.9A-8)
Eq. 9	20,810 psi	20,825 psi
Eq. 10	65,567 psi	65,596 psi
Eq. 11	128,950 psi	128,920 psi
Eq. 12	39,536 psi	39,564 psi
Eq. 13	23,152 psi	23,155 psi
Total Usage Factor	0.3439	0.3699

NOTE:

Comparison made for Butt Welding Tee, Location 10.

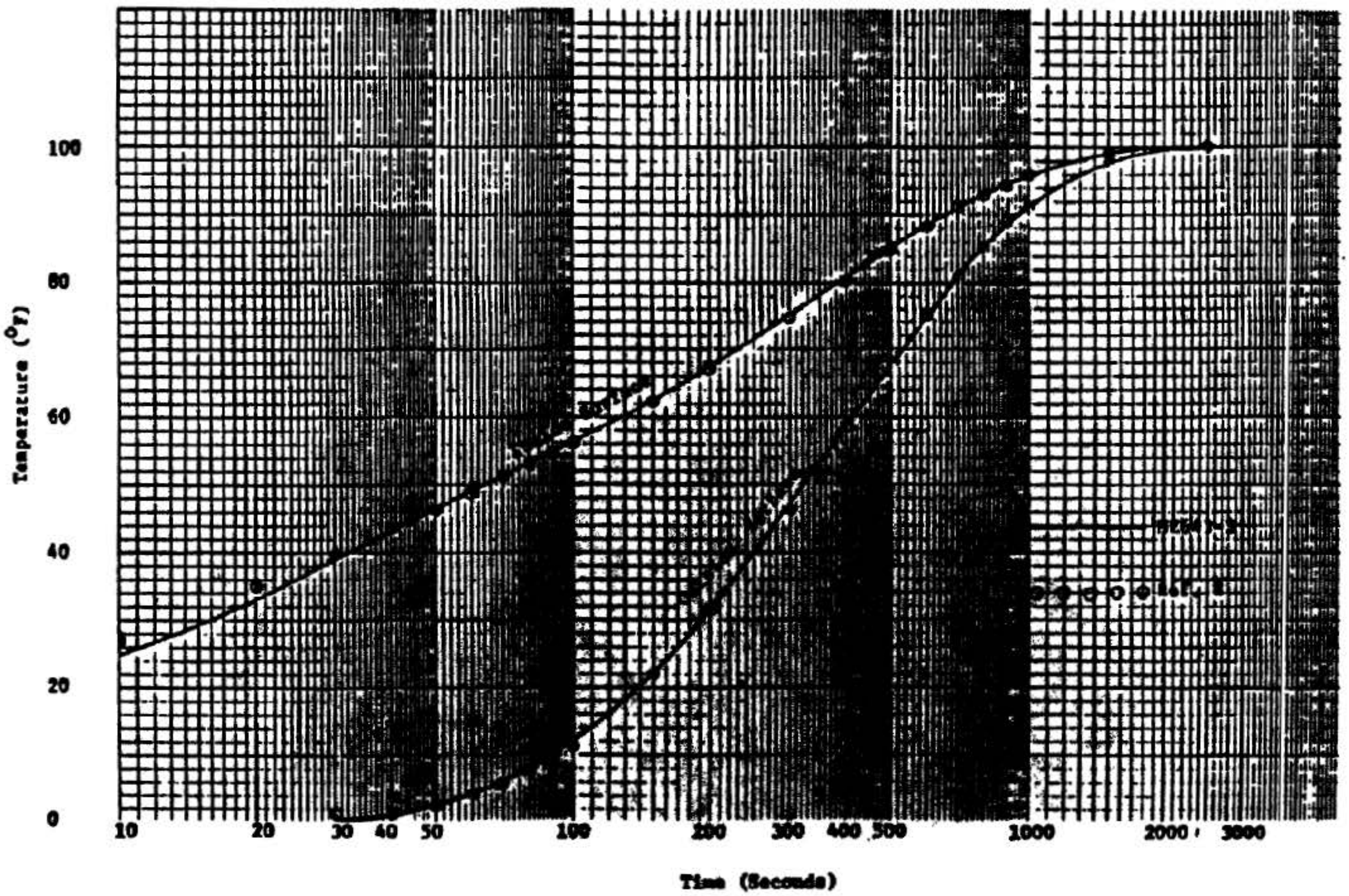
HISTORICAL INFORMATION

TABLE 3.9A-2

COMPARISON OF ME 912 WITH ME 643 AND ANALYTICAL RESULTS

Case	Gradients of Program	Pipe ΔT_1	Pipe ΔT_2	Ta - Tb
450° to 553°F Step	Me643	79.0	38.0	24.0
3" Sch. 160, Stainless	ME912	79.7	40.6	24.3
Thicknesses 1.50:1	B/M**	82.0	41.0	--
408° to 100°F Step 12" Sch. 80 Carbon Steel Thicknesses 1.69:1	ME643 ME912 B/M**	136.2 134.4 139.0	40.1 41.9 43.0	83.0 81.6 --

** Ref. 3.9A-5



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TRANSIENT TEMPERATURE RESPONSE

FIGURE 3.9A-1, Rev. 55

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3.10 SEISMIC QUALIFICATION* OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

The seismic qualification* of Seismic Category I instrumentation and electrical equipment is described in the following subsections:

3.10a - NSSS Instrumentation and Electrical Equipment

3.10b - Non-NSSS Instrumentation

3.10c - Non-NSSS Electrical Equipment

In addition to seismic qualification, all Seismic Category I instrumentation and electrical equipment located in the Containment and the Reactor and Control Buildings are qualified for the combined seismic and hydrodynamic vibratory loadings. Procedures for the assessment and requalification of Seismic Category I instrumentation and electrical equipment for the additional hydrodynamic loads are described in Sections 7.1.6 and 7.1.7 of the Design Assessment Report (DAR).

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* The term "Seismic Qualification" in this section is synonymous with "Seismic and Hydrodynamic Qualification."

3.10a SEISMIC QUALIFICATION OF SEISMIC CATEGORY I NSSS INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10a.1 SEISMIC QUALIFICATION CRITERIA

3.10a.1.1 Seismic Category I Equipment Identification

Seismic Category I instrumentation and electrical equipment, as well as other equipment, can be found in Table 3.2-1. Pumps and valves which are qualified as seismically "active" are listed in Table 3.9-3.

All NSSS Seismic Category I instrumentation and electrical equipment will be designed to resist and withstand the effects of the postulated earthquakes. Seismic Category I instrumentation and electrical equipment is designed to withstand the effects of the Safe Shutdown Earthquake (SSE) defined in Subsection 3.7a, and to withstand the effects of hydrodynamic loads without functional impairment.

From the basic input ground motion data, a series of response curves at various building elevations are developed after the building layout is completed. This information is included in the purchase specifications for Seismic Category I equipment. Suppliers of equipment such as batteries and racks, instrument racks, control consoles, etc., are required to submit test data, operating experience and/or calculations to substantiate that their components, systems, etc., will not suffer loss of function during or after seismic and hydrodynamic loadings. The magnitude and frequency of the dynamic loadings which each component will experience is determined by its specific location within the plant.

The Class 1E instrumentation and electrical equipment (excluding motors and valve-mounted equipment) supplied by GE requiring seismic qualification are identified in Table 3.10a-1.

3.10a.1.2 Dynamic Design Criteria

3.10a.1.2.1 NSSS Equipment

The seismic criteria used in the design and subsequent qualification of all Class 1E instrumentation and electrical equipment supplied by GE was as follows: The Class 1E equipment shall be capable of performing all safety-related functions during (1) normal plant operation, during (2) anticipated transients, during (3) design basis accidents, and during (4) post-accident operation, while being subjected to, and after the cessation of the accelerations resulting from the OBE and SSE at the point of attachment of the equipment to the building or supporting structure.

The criteria for each of the devices used in the Class 1E systems depend on the use in a given system; for example, a relay in one system may have as its safety function to deenergize and open its contacts within a certain time, while in another system it must energize and close its contacts. Since GE supplies devices for many applications, the approach taken was to test the device in all modes in which it might be used. In this way, the capability of protective action initiation and the proper operation of safety-failure circuits is ensured.

3.10a.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION (EXCLUDING MOTORS AND VALVE MOUNTED EQUIPMENT)

3.10a.2.1 Methods of Showing NSSS Equipment Compliance with IEEE 344-1971

(a) Scope - Compliance not applicable.

(b) Definition - Compliance not applicable.

(c) Procedures - GE supplied Class 1E equipment meets the requirement that the qualification should demonstrate the capability to perform the required function during and after the seismic and hydrodynamic load event. Both analysis and testing were used but most equipment was tested. Analysis was primarily used to determine the adequacy of mechanical strength (mounting bolts, etc.) after operating capability was established by testing.

1. Analysis - GE supplied Class 1E equipment performing primarily a mechanical safety function (pressure boundary devices, etc.) was analyzed since the passive nature of its critical safety role usually made testing impractical. Analytical methods sanctioned by IEEE 344-1971 were used in such cases (see Table 3.10a-1 for indication of which items were qualified by analysis).

2. Testing - GE supplied Class 1E equipment having primarily an active electrical safety function was tested in compliance with IEEE 344-1971, Section 3.2.

(d) Documentation - Available documentation verifies that the seismic qualification of GE supplied Class 1E equipment is in accordance with the requirements of IEEE 344-1971, Section 4.

3.10a.2.2 Testing Procedures for Qualifying Electrical Equipment and Instrumentation (Excluding Motors and Valve-Mounted Equipment)

(The following procedures are not applicable for the Diesel Generator 'E' facility where the seismic qualification conforms to project specification C-1041 or SD-140 and IEEE Standard 344-75.) In addition, replacement equipment may be seismically qualified to a version of the IEEE Standard 344 that is more recent than the 1971 version. Non-GE supplied replacement equipment is qualified to the provisions of FSAR Section 3.10b.

The test procedure required that the devices be mounted on the table of the vibration machine in a manner similar to which it was to be installed. The device was tested in the operating states that it is to be used in performing its Class 1E functions. These states were monitored before, during, and after the test to ensure proper function and absence of spurious function. In the case of a relay, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay is used in those configurations in its Class 1E functions.

The dynamic excitation was a single frequency continuous test in which the applied vibration was a sinusoidal table motion at a fixed peak acceleration and a discrete frequency at any given time. Each frequency and acceleration combination was maintained for about 30 seconds except when a resonance search was made (see IEEE 344-1971, paragraph 3.2).

The vibratory excitation was applied in three orthogonal axes individually with the axes chosen as those coincident with the most probable mounting configuration.

The first step was to search for resonances in each device. This was done since resonances cause amplification of the input vibration and are the most likely cause of malfunction. The resonance search was usually run at low acceleration levels (0.2G) to avoid destroying the test sample in case a severe resonance was encountered. The resonance search was performed in accordance with IEEE 344 in no less than 7 minutes; if the device was large enough, the vibrations were monitored by accelerometers placed at critical locations. Resonances were determined by comparing the acceleration level with that at the table of the vibration machine. Usually, the devices were either too small for an accelerometer, had their critical parts in an inaccessible location, or had critical parts that would be adversely affected by the mounting of an accelerometer. In these cases, the resonances were detected by visual (strobe light), audible observation, or performance.

Following the frequency scan and resonance determination, the devices were tested to determine their malfunction limit. This test was a necessary adjunct to the assembly test as will be shown later. The malfunction limit test was run at each resonant frequency as determined by the frequency scan. In this test, the acceleration level was gradually increased until either the device malfunctioned or the limit of the device (usually the case) was considered to be rigid (all parts move in unison) and the malfunction limit was therefore independent of frequency.

To achieve maximum acceleration from the vibration machine, rigid devices were malfunction tested at the upper test frequency since that allowed the maximum acceleration to be obtained from deflection- limited machines. The summary of the tests on the devices used in Class 1E applications given in Table 3.10a-1 includes the qualification limit for each device tested.

The above procedures were required of purchased devices as well as those manufactured by GE. Vendor test results were reviewed and if unacceptable, the tests were repeated either by GE or the vendor. If the vendor tests were adequate, the device was considered qualified to the limits of the test.

3.10a.2.3 Qualification of Valve-Mounted Equipment

The piping analyses establishes the response spectra, power spectral density function or time history characteristics, and develops a horizontal and a vertical acceleration for the pipe-mounted equipment. Class 1E motor-operated valves actuators were qualified per IEEE 382-1972, with the exception of DG-E motor-operated valve actuators which were qualified to IEEE 382-1980.

The safety/relief valve, including the electrical components mounted on the valve, are subjected to a dynamic test. This testing is described in Subsections 3.9.2.2a.2.15 and 3.9.3.2a.5.2.

3.10a.2.4 Qualification of NSSS Motors

Seismic qualification of the ECCS motors is discussed in Subsection 3.9.2.2a.2.7, in conjunction with the ECCS pump and motor assembly. Seismic qualification of the Standby Liquid Control (SLC) pump motor is discussed in Subsection 3.9.2.2a.2.10 in conjunction with the SLC pump motor assembly.

3.10a.3 METHODS AND PROCEDURE OF ANALYSIS OR TESTING OF SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

3.10a.3.1 Dynamic Analysis Testing Procedures and Restraint Measures

3.10a.3.1.1 NSSS Equipment (Other Than Motors and Valve Mounted Equipment)

The Class 1E equipment supplied by GE is used in many systems on many different plants under widely varying seismic requirements. The dynamic qualification was performed in accordance with IEEE-344.

Some GE supplied Class 1E devices were qualified by analysis only (as noted in Table 3.10a-2). One of the analysis methods is shown in Subsection 3.10a.5. Analysis was used for passive mechanical devices and was sometimes used in combination with testing for larger assemblies containing Class 1E devices. For instance, a test might have been run to determine if there were natural frequencies in the equipment within the critical seismic frequency range (see IEEE 344-1971, paragraph 3.2). If the equipment was determined to be free of natural frequencies, then it was assumed to be rigid and a static analysis was performed (see IEEE 344-1971, paragraph 3.2). If it had natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations were determined to see if Class 1E devices mounted in the assembly would operate without malfunctioning. In general, the testing of Class 1E equipment was accomplished using the following procedure.

Assemblies (i.e., control panels) containing devices which have had dynamic load malfunction-limits established were tested by mounting the assembly on the table of a vibration machine in the manner in which it was to be mounted in use. It was vibration tested by running a low level resonance search. As with the devices, the assemblies were tested in the three major orthogonal axes. The resonance search was run in the same manner as described for devices. If resonances were present, the transmissibility between the input and the location of each Class 1E device was determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response at any Class 1E device location for any given input. (It was assumed that the transmissibilities were linear as a function of acceleration even though they actually decrease as acceleration is increased-therefore a conservative assumption.).

Since control panels and racks constitute the majority of Class 1E electric assemblies supplied by GE, seismic qualification testing of these will be discussed in more detail. There are basically four generic panel types. One or more of each type was tested using the above procedures.

Figures 3.10a-1, 3.10a-2, 3.10a-3 and 3.10a-4 illustrate the four basic panel types referenced above and show typical accelerometer locations. The status of the dynamic tests on the Class 1E panels supplied by GE is summarized in Table 3.10a-2.

The full acceleration level tests described above disclosed that most of the panel types had more than adequate mechanical strength. A given panel design acceptability was shown to be only a function of its amplification factor and the malfunction levels of the devices mounted in it. Subsequent panels were, therefore, tested at lower acceleration levels and the transmissibilities measured to the various devices as described above. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input, the panel seismic qualification level could be determined. Several high level tests have been run on selected generic panel designs to ensure the conservatism in using the transmissibility analysis described.

3.10a.4 OPERATING LICENSE REVIEW

3.10a.4.1 NSSS Control and Electrical Equipment (Other Than Motors and Valve Mounted Equipment)

The dynamic test results for safety-related panels and control equipment within the NSSS scope are maintained in a permanent file by GE and can be readily audited in all cases. The equipment used in Class 1E applications passed the prescribed tests. Where equipment failed to pass the tests, it was rejected. In some cases, equipment which failed one test was modified to meet the performance requirements and retested. If the retested equipment passed the latter test, it could be used in a Class 1E application.

Table 3.10a-1 lists the NSSS control devices by item number and vendor. Also, a summary of the test conditions for the devices used in Class 1E applications is given in Table 3.10a-2.

The acceleration level shown in the right hand columns of Table 3.10a-1 is the acceleration at which either the device malfunctioned or the limit of the vibration machine was reached.

3.10a.4.2 NSSS Motors

Seismic qualification test results for the ECCS motors are discussed in Subsection 3.9.2.2a.2.7 in conjunction with the ECCS pump and motor assembly. Seismic qualification test results for the Standby Liquid Control (SLC) motor are discussed in Subsection 3.9.2.2a.2.10 in conjunction with the SLC pump motor assembly.

3.10a.4.3 Valve-Mounted Equipment

The safety relief valves (including the electrical components mounted on the valve) are subjected to dynamic tests. The results of these tests are discussed in Subsections 3.9.2.2a.2.15 and 3.9.3.2a.5.2.

3.10a.5 Dynamic Analysis By Response Spectrum Method

The system stiffness and mass matrices are generated using standard techniques. A dynamic analysis is performed using the following equations of motion and procedure to uncouple these equations

The equations of motion in matrix form are as follows:

$$M(\ddot{X} + \ddot{Y}) + C\dot{X} + KX = 0 \qquad \text{(Eq. 3.10a-1)}$$

where

M	= mass matrix, nxn (this includes the hydrodynamic mass)
X	= column vector of displacement relative to ground* (nxl)
C	= damping matrix (nxn)
K	= stiffness matrix (nxn)
Y	= column vector of ground accelerations (nxl)
.	= first derivative with respect to time
..	= second derivative with respect to time

It should be noted that for equipment containing fluid, a hydrodynamic mass coupling exists between real structural masses. This hydrodynamic mass appears as diagonal and off-diagonal terms in the mass matrix. The overall system stiffness matrix K is determined by either the matrix force method or the matrix displacement method. The resulting stiffness matrix is similar.

Removing the driving-point acceleration vector to the right side of Eq. 3.10a-1, the equation reduces to the classical form:

$$M\ddot{X} + C\ddot{X} + KX = M\ddot{Y} \qquad \text{(Eq. 3.10a-2)}$$

In order to decouple Eq. 3.10a-2, we set:

$$X = \phi q \qquad \text{(Eq. 3.10a-3)}$$

Eq. 3.10a-2 then becomes

$$M\phi q + C\phi q + K\phi q = -M\ddot{Y} \qquad \text{(Eq. 3.10a-4)}$$

Pre-multiplying by ϕT, the transpose of ϕ, and performing the coordinate transformation described in Eq. 3.10a-4 such that is defined by the following orthogonality conditions:

$$\phi\, TM\, \phi = I \qquad \text{(Eq. 3.10a-5)}$$

$$\phi\, TK\, \phi = w^2 \qquad \text{(Eq. 3.10a-6)}$$

where I is an identifying matrix (Nxn) and w^2 is a diagonal matrix of the eigenvalues. Then Eq. 3.10a-4 becomes

$$\phi^T M\phi\ddot{q} + \phi^T C\phi\dot{q} + \phi^T K\phi q = \phi^T M\ddot{Y} \qquad \text{(Eq. 3.10a-7)}$$

$$\ddot{q} + \phi^T O\phi\dot{q} = w^2\ q = \phi^T M\ddot{Y} \qquad \text{(Eq. 3.10a-8)}$$

The above procedure for decoupling the equation of motion by using the modal matrix of the undamped system assumes that damping in the system is small. It will further be assumed that the damping matrix C is such that $\phi^T C\phi$ is a diagonal matrix. The elements of this diagonal-matrix are the modal damping values.

With the above assumptions, Eq. 3.10a-8 may be written in the following uncoupled form:

$$\ddot{q}_i + \ + 2\beta_i w_i \dot{q}_i + w_i^2 q_i = S\ U \qquad \text{(Eq. 3.10a-9)}$$

i = 1, 2, --- n

where

$$x_i = \begin{matrix} \chi^1 i \\ \chi^2 i \\ \cdot \\ \cdot \\ \cdot \\ \cdot \\ \chi^n i \end{matrix} \qquad \phi_i = \begin{matrix} \phi 1_i \\ \phi 2_i \\ \cdot \\ \cdot \\ \cdot \\ \cdot \\ \phi n_i \end{matrix}$$

The maximum physical displacement for each mass is then taken to the square root of the sums of the squares of each of the maximum displacement responses for each mode, i.e.,

where:

$$X\max. = \left[\sum_{j=1}^{n} x_{ij}^2 \right]^{1/2}$$

(X) maximum is the column vector of maximum displacements. Similarly, the maximum load response for the i mode is found from

$$L_{ji} = \beta_j X_i$$

$$L_{ji} = \begin{matrix} L_{1i} \\ L_{2i} \\ \cdot \\ \cdot \\ \cdot \\ L_{mi} \end{matrix}$$

where

β_j is the stress matrix for element j, j=1, ...m

m	=	total number of elements.

where

β_i	=	damping ratio for the i^{th} mode expressed as percent of critical damping
w_i	=	i^{th} natural angular frequency of the system
S_i	=	modal participation factor the i^{th} mode = ϕ_i^t MD
U_g	=	ground or floor acceleration time history
ϕ_i^t	=	transpose of the i^{th} mode shape
D	=	earthquake direction vector

The response is calculated using the response spectra specified for the location of the input to the analytical model. The analytical procedure is described briefly in the following paragraphs.

The system of one degree-of-freedom equations represented by Eqs. 3.10a-8 or 3.10a-9 can be solved by the response spectrum method. With this method, the maximum modal response for each natural frequency of interest is found from the applicable response spectra. Response spectrum curves are essentially plots of the maximum responses of single degrees-of-freedom systems described by Eq. 3.10a-9 with S = 1.0 as a function of their natural frequencies.

Having found the maximum modal displacements q_i, i = I...m, the maximum physical displacement for the i[th] mode is given by:

$$X_i = \phi_i S_i q_i$$

The maximum load response is taken to be the square root of the sums of the squares of each of the maximum responses for each mode, i.e.,

$$Lj\,max. = \left[\sum_{i=1}^{n} L^2_{ji} \right]^{1/2} \quad : j = 1, 2, \dots m$$

where (L) max is the column vector of maximum loads

The accelerations for each mode are determined by multiplying the displacements vector for that mode (X_i) by the natural frequency of (w_{2_i}) that mode.

$$A_i = X_i w_{2_i}$$

The maximum accelerations are then determined by

$$A\,max. = \left[\sum_{i=1}^{n} A_i^2 \right]^{1/2}$$

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------					-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------					
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
SYSTEM TITLE - REACTOR										
B11-D193	Power Range Detector	GE	43	In Vessel						
SYSTEM TITLE – NUCLEAR BOILER										
B21-N002	Pressure Switch		1	Area II						
B21-N004	Temp Element	PYCO	16	Area II.7				Note 2		
B21-N006	Diff Press Switch	BARTON	18	Area II	N007	N008	N009,	5	10	10
					N021	A,B,C,D				
B21-N010	Temp Element	CALIF. ALLOY	13	Area II	N014,	N016,	N017	5	5	5
B21-N015	Press Switch	BARKSDALE	4	Area III.3				5	10	10
B21-N020	Press Switch	BARKSDALE	34	Area II.1			N023,	5	15	15
					N039	N044,	N022			
B21-N024	Level Ind Switch	BARTON	10	Area II.1	N031,	N042		15	15	15
B21-N025	Level Ind Switch	YARWAY	4	Area II.1				1.5	1.5	1.5
B21-N026	Level Ind Trans Switch	BARTON	6	Area II.1	N037			5	5	5
B21-N027	Level Trans	ROSEMOUNT	25	Area II.1	N033,	N034		Note 2		
B21-N043	Press Trans	ROSEMOUNT	1	Area II.1				3	3	3
B21-N055	Press Trans	ROSEMOUNT	2	Area II.1				3	3	3
B21-N056 A&C	Vacuum Switch (Unit 1)	STATIC-O-RING	2	Area III.5						
B21-N056 B&D	Vacuum Switch (Unit 1)	BARKSDALE	2	Area III.5						
B21-N056B	Vacuum Switch (Unit 2)	STATIC-O-RING	1	Area III.5						
B21-N056 A,C, & D	Vacuum Switch (Unit 2)	BARKSDALE	3	Area III.5						

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------					-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------					
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT[1]	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION[4]		
								X	Y	Z
B21-N064	Temp Element	CALIF. ALLOY	1	Area II				2	2	2
B21-N600	Temp Switch	RILEY	8	Area V	N603			Note 2		
B21-R004	Press Indicator	ROBERTSHAW	2	Area II.1				4	4	4
B21-R005	Diff Press Ind	BARTON	1	Area II.1				Note 2		
B31-N014	Flow Trans	ROSEMOUNT	8	Area II.1	N024			2	2	2
B31-N015	Diff Press Trans	ROSEMOUNT	1	Area II.1				Note 2		
B31-N016	Diff Press Switch	BARTON	13	Area II.1	N018A,	N019 thru		5	10	10
					N022					
B31-N018B	Press Switch	STATIC-O-RING	1	Area II.1				15	15	15
B31-N023	Temp Element	ROSEMOUNT	2	Area 1.4				Note 2		
B31-N035	Temp Element		2					Note 2		
SYSTEM TITLE – CRD HYDRAULIC CONTROL										
C12-N013	Level Switch	MAGNETROL	5	Area II.1				4.1	5	9.5
SYSTEM TITLE – FEEDWATER CONTROL										
C32-N003	Transmitter (Diff Press)	ROSEMOUNT	6	Area II.1	N004			Note 2		
C32-N005	Transmitter (Pressure)	ROSEMOUNT	2	Area II.1	N008			Note 2		
C32-N017	Diff Press Trans	STATHOM	2	Area II.1				Note 2		
SYSTEM TITLE – STAND BY LIQUID										
C41-N003	Temp Switch	CALIF. ALLOY	1	Area II.8				Note 2		
C41-N004	Press Trans	ROSEMOUNT	1	Area II.8				Note 2		
C41-N006	Temp Element		1	Area II.8				Note 2		
C41-R003	Press Indicator	ROBERTSHAW	1	Area II.8				Note 2		
SYSTEM TITLE – NEUTRON MONITORING										

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------					-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------					
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
C51-J004	Valve, Guide Tube	GE	5	Area II.10						
C51-J008	Guidetubes	GE	1	Area II.10				Note 2		
C51-K002	Volt Preamplifier	GE	8	Area V				4	5	5
C51-K601	Intermediate Range Mon	GE	8	Area V				6	6	6
C51-K605	Pwr Rnge Instr	GE	1	Area V						
C51-N002	Detector	GE	8	Area I.3						
SYSTEM TITLE – REACTOR PROTECTION										
C72-N002	Prim Cont Press Switch	STATIC-O-RING	4	Area II.1				15	15	15
C72-N003	Turbine 1st Stage Pr SW	BARKSDALE	4	Area III.3				15	15	15
C72-N005	Turbine EMC Press SW	BARKSDALE	4	Area III.3						
C72-N006	Turb Stop Vlv POS SW	ACME CLEVELAND	4	Area III.3						
C72-N008	Turbine Bypass Vlv POS SW	ACME CLEVELAND	4	Area III.3						
C72-S003(A-H)	Elec. Prot. Assy.	GE	8	Not Required				Later		
SYSTEM TITLE – PROCESS RADIATION MONITORING										
D12-K603	Rad Mon & Ind (Mn St Ln)	GE	4	Area V						
D12-K609	Ind & Trip Unit	GE	12	Area V	K615,	K616	K617	3	3	3
					K618					
D12-N006	Detector (Mn St Ln)	GE	4	Area II.7						
D12-N015	Detector	GE	8	Area II.9	N016	N017,	N018	15	15	15
SYSTEM TITLE – RESIDUAL HEAT REMOVAL										

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------				-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------						
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
E11-N001	Cond Element	BALSBAUGH	2	Area II.5				Note 2		
E11-N007	Diff Press Trans	ROSEMOUNT	5	Area II.5	N013,	N015		Note 2		
E11-N008	Diff Press Trans	BARTON	2	Area II.5				Note 2		
E11-N009	Temp Element	CALIF ALLOY	12	Area II.5	N029,	N030		2	2	2
E11-N010	Press Switch	STATIC-O-RING	8	Area II.5	N011			15	15	15
E11-N016	Press Switch	STATIC-O-RING	9	Area II.5	N018,	N020		15	15	15
E11-N019	Diff Press Switch	BARTON	2	Area II.5				5	10	10
E11-N021	Diff Press Ind Switch	BARTON	2	Area II.5				15	15	15
E11-N022	Press Switch	BARKSDALE	2	Area II.5				Note 3		
E11-N023	Level Switch	MAGNETROL	3	Area II.5		N024		Note 3		
E11-N026	Press Trans	ROSEMOUNT	3	Area II.5		N028		Note 3		
E11-N033	Flow Switch	FISHER & PORTER	2	Area II.5				Note 3		
E11-N600	Temp Switch		8	Area V	N601			4.5	4.5	4
E11-R002	Press Indicator	ROBERTSHAW OR CONTROL SPECIALTIES	8	Area II.5	R003			Note 3		
SYSTEM TITLE – CORE SPRAY										
E21-N001	Press Trans	ROSEMOUNT	2	Area II.5				Note 2		
E21-N003	Diff Press	ROSEMOUNT	2	Area II.5				Note 2		
E21-N004	Diff Press	BARTON	2	Area II.5				5	10	10
E21-N006	Flow Switch		2	Area II.5						
E21-N007	Press Switch		2	Area II.5				Note 2		
E21-N008	Press Switch		4	Area II.5						

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT[1]	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION[4] X	Y	Z
E21-R001	Pressure Indicator	ROBERTSHAW OR CONTROL SPECIALTIES	2	Area II.5				Note 2		
SYSTEM TITLE – MSIV LEAKAGE CONTROL										
E32-K601	Power Supply	GE	2	Area V				2.5	2.5	2.5
E32-N006	Flow Element	S & K INSTRUMENTS	4	Area II						
E32-N050	Press Trans	ROSEMOUNT	8	Area II	N055,	N058,	N060	3	3	3
					N061					
E32-N051	Press Trans	ROSEMOUNT	5	Area II	N056			3	3	3
E32-N053	Flow Trans	S & K INSTRUMENTS	4	Area II						
E32-N054	Diff Press Trans	ROSEMOUNT	2	Area II	N059			3	3	3
E32-N600	Timer	EAGLE SIGNAL	13	Area V	N601,	N602,	N604	2.5	2.5	2.5
E32-N650	Alarm	BAILEY METER	19	Area V	N651,	N653,	N654,	9	9.5	13
					N655,	N656,	N658,			
					N659,	N660,	N661			
E32-R601	MV/I	BAILEY METER	4	Area V				Note 2		
E32-R651	Meter	GE	18	Area V	R653 thru	R656,		Note 2		
					R658 thru	R661				
SYSTEM TITLE – HIGH PRESSURE COOLANT INJECTION										
E41-K600	Power Supply	GE	1	Area V				2.5	2.5	2.5
E41-K601	SO Root Converter	BAILEY METER	1	Area V				9	9	13

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------					-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------					
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
E41-K603	Inverter	TOPAZ	1	Area V				5	10	8.5
E41-N001	Press Switch	BARKSDALE	6	Area II	N027	N031		29	29	29
E41-N002	Level Switch	MAGNETROL	5	Area II.4	N003,	N015,	N018	1.2	6	9.5
E41-N604	Diff Press Switch	BARTON	2	Area II	N005			5	10	10
E41-N005	Diff Press Switch	BARTON	1	Area II				Note 3		
E41-N008	Diff Press Trans	ROSEMOUNT	1	Area II				3	3	3
E41-N009	Press Trans	ROSEMOUNT	4	Area II	N013,	N016,	N019	Note 2		
E41-N010	Press Switch	STATIC-O-RING	7	Area II	N012,	N017		15	15	15
E41-N014	Level Switch	MAGNETROL	1	Area II				Note 2		
E41-N024	Temp Element	CALIF. ALLOY	16	Area II	N025,	N028,	N029,	2	2	2
					N030					
E41-N600	Temp Switch	GE	6	Area V	N601,	N602				
E41-R601	Press Indicator	ROBERTSHAW	4	Area II	R003,	R004,	R005	Note 2		
E41-R002	Temp Indicator	MOELLER	1	Area II				Note 2		
E41-R600	Controller	BAILEY METER	1	Area V				9	9	8
SYSTEM TITLE – REACTOR CORE ISOLATION COOLING										
E51-K603	Inverter(DC to AC)	TOPAZ	1	Area V				5	10	8.5
E51-N602	Timer		4	Area V	N603					
E51-N003	Diff Press Switch	BARTON	1	Area II.4				15	15	15
E51-N003	Diff Press Transmitter	ROSEMOUNT	1	Area II.4				3	3	3
E51-N004	Press Transmitter	ROSEMOUNT	4	Area II.4	N005,	N007,	N008	Note 2		
E51-N006	Press Switch	STATIC-O-RING	5	Area II.4	N019			15	15	15
E51-N009	Press Switch	BARKSDALE	8	Area II.4	N012,	N020,	N030	29	29	29

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------				-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------						
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT(1)	OTHERS OF SAME TYPE IN SAME ENVIRONMENT			SEISMIC QUALIFICATION(4)		
								X	Y	Z
E51-N010	Level Switch	MAGNETROL	1	Area II.4				Note 2		
E51-N011	Temp Element		20	Area II	N021,	N022,	N023			
					N025,	N026,	N027			
E51-N017	Diff Press Switch	BARTON	2	Area II	N018			5	10	10
E51-N600	Temp Switch	GE	14	Area V	N601 thru	N604				8
E51-R001	Press Indicator	ROBERTSHAW	4	Area II.4				Note 2		
E51-R005	Temp Indicator	MOELLER	1	Area II.4				5	5	5
E51-R600	Flow Indicator Cont	BAILEY METER	1	Area V				9	9	8
SYSTEM TITLE – REACTOR WATER CLEANUP										
G33-K600	Power Supply	GE	1	Area V				2.5	2.3	2.5
G33-K602	SQ Root Conv	GE	3	Area V	K603,	K605				
G33-K604	Summer	BAILEY METER	1	Area V				4	9	13
G33-N011	Flow Element	VICKERY SIMS	2	Area II.2.b	N035,	N040		Note 2		
G33-N012	Diff Press Trans	ROSEMOUNT	3	Area II	N036,	N041		3	3	3
G33-N016	Temp Element	CALIF. ALLOY	18	Area II	N022,	N023		2	2	2
G33-N044	Diff Press Switch	BARTON	2	Area II				15	15	15
G33-N600	Temp Switch	GE	12	Area V	N602					
G33-N603	Alarm	BAILEY METER	12	Area V				9	9.5	13
NOTES:										

TABLE - 3.10a-1

ESSENTIAL ELECTRICAL COMPONENTS AND INSTRUMENTS

-------------------- DESCRIPTION --------------------				-------------------- SEISMIC AND ENVIRONMENTAL QUALIFICATIONS --------------------				
ITEM NO.	NAME	VENDOR	QUANTITY	ENVIRONMENT[1]	OTHERS OF SAME TYPE IN SAME ENVIRONMENT	SEISMIC QUALIFICATION[4]		
						X	Y	Z

1. Refer to Tables 3.11-1, 3.11-2 and 3.11-3.
2. Classified as Pressure Integrity Instrument; Seismic qualification not required.
3. Hydrostatic test only required for qualification.
4. This table is based on the original seismic qualification effort performed by GE. Replacement equipment may be seismically qualified to a version of the IEEE Standard 344 that is more recent than the 1971 version. Non-GE supplied replacement equipment is qualified to the revisions of FSAR Section 3.10b.

TABLE 3.10a-2

SEISMIC QUALIFICATION TEST SUMMARY
CLASS 1E CONTROL PANELS AND LOCAL PANELS & RACKS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H12-P601	Reactor Core Cooling System	Benchboard	SBM & CR 2940 switches GEMAC instruments	Too long for test table – not tested [1] qualified by analysis
H12-P680	Unit Operating Bd. Reactor Water Cleanup & Recirculation	Benchboard	SBM & CR 2940 switches BEMAC instruments	Seismic test on similar type panel[2]
H12-P680	Reactor Control	Benchboard	Mode switch, range switches	Seismic test completed[3]
H12-P606	Radiation Monitor	2 Bay instrument rack	Startup neutron monitoring electronics	Seismic test completed
H12-P609	Reactor Protection System Division 1 & 2 Logic	Vertical board	HFA & HMA Relays, CR 105 contactor	Identical to U13-P611 panel tested[4]
H12-P611	Reactor Protection System Division 3 & 4 Logic	Vertical board	HFA & HMA Relays, CR 105 contactor	Seismic test completed
H12-P612	FW & Recirc Instruments	2 Bay instrument rack	GEMAC Instruments	Seismic test completed
H12-P613	NSSS Process Instruments	2 Bay instrument rack	GEMAC Instruments	Seismic test completed
H12-P618	Division 2 RHR/RCIC Relay	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P621	Reactor Core Isolation Cooling Relays	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P622	Inboard Isolation Valve Relays	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P623	Outboard Isolation Valve Relays VB	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H12-P628	ADS Channel A Relay VB	Vertical board	HFA & HMA Relays	Seismic test on similar type panel
H22-P001	CS System Loc. Pnl. A	Local rack	Pressure Switch	Seismic test completed
H12-P631	ADS Channel B Relay VB	Vertical board	HFA & HMA Relays	Seismic test on similar type panel

TABLE 3.10a-2

SEISMIC QUALIFICATION TEST SUMMARY
CLASS 1E CONTROL PANELS AND LOCAL PANELS & RACKS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
			Mon., Timers	
H12-P633	Radiation Monitor Instrument Panel B	2 Bay instrument rack	Startup Neutron Monitoring Electronics	Identical to H13-P606; Panel tested
H22-P002	Reactor Water Cleanup	Local rack	Pressure transmitters	Seismic test completed
H22-P004	Reactor Vessel Level & Pressure – A	Local rack	Pressure switches, level indicator/transmitter	Seismic test on similar type panel
H22-P005	Reactor Vessel Level & Pressure – B	Local rack	Pressure switches, level indicator/transmitter	Seismic test on similar type panel
H22-P006	Recirc A/Main Steam Flow A	Local rack	Pressure transmitter	Seismic test on similar type panel
H22-P009	Jet Pump Division 1	Local rack	Pressure transmitter	Seismic test completed
H22-P010	Jet Pump Division 2	Local rack	Pressure transmitter	Identical to H22-P009 panel tested
H22-P015	Main Steam Flow A	Local rack	Pressure switch	Identical to H22-P025 panel tested
H22-P017	RCIC Division 1	Local rack	Pressure transmitter/switches	Seismic test on similar type panel
H22-P018	RHR Panel A	Local rack	Pressure switches	Seismic test on similar type panel
H22-P021	RHR Panel B	Local rack	Pressure transmitter/switches	Seismic test on similar type panel
H22-P025	Main Steam Flow D	Local rack	Pressure switches	Seismic test completed
H22-P030	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Seismic test completed
H22-P031	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Identical to H22-P030 enclosure tested
H22-P032	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Identical to H22-P039 enclosure tested

TABLE 3.10a-2

SEISMIC QUALIFICATION TEST SUMMARY
CLASS 1E CONTROL PANELS AND LOCAL PANELS & RACKS

PANEL	DESCRIPTION	TYPE	CLASS 1E EQUIPMENT DESCRIPTION	COMMENTS
H22-P033	SRM & IRM Preamp A-D	NEMA 12 – Enclosures	SRM-IRM Preamplifiers	Identical to H22-P030 enclosure tested
H12-P700	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P701	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P702	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P703	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P704	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P705	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P706	Termination Cabinet	4 Bay Cabinet	Cables	
H12-P732	Termination Cabinet	4 Bay Cabinet	Cables	

FOOTNOTES:

Seismic tests on essential C&I panels fall into the following categories:

1. Panels not tested — Due to size limitations, qualification completed by analysis.
2. Tests on similar panels — When panel size and configuration are very similar to but not necessarily identical, test results for a similar panel are used.
3. Seismic test completed — Tests run on essentially identical panels but possibly build for a different plant.
4. Tests on identical panels — When two panels are exact duplicates of one another, tests are run on only one panel (e.g., H12-P609 and H12-P611 are identical – only H12-P611 was tested).

TABLE 3.10a-3

SUMMARY OF SAMPLE SEISMIC STATIC ANALYSIS
FOR THREE TYPICAL CABINETS

Panel Description	Center of Gravity (in)	Number of Studs	Panel Forces (lb)/Ft of Panel FR to BK (F-B)	Panel Forces (lb)/Ft of Panel Side to Side (S-S)	Panel Forces (lb)/Ft of Panel Up	Panel Forces (lb)/Ft of Panel Down	Axial Load/Stud Tension (lb)	Axial Load/Stud Comp (lb)	Shear Load/ Stud (lb)	Combined Stress Shear (psi)	Combined Stress Normal (psi)	Margin of Safety* Yield	Stud Tensile Load (lb)**
NSSS Cabinet H-12-P608 Power Range Monitor	45	40	736	736	1656	2576	1748	174.8	349.6	6633	12,793	Tensile 0.95 Shear 1.07	1815
PGCC Computer Cabinet	40.5	4	561	561	1262	1963	1797	88	380	6878	13,206	Tensile 0.89 Shear 1.0	1874
Electro-Hydraulic Cabinet H-12-P863	43	24	637	637	1432	2228	2111	321	398	7953	15,398	Tensile 0.63 Shear 0.73	2185

* - A value for the margin of safety which is greater than zero (> 0) represents an adequate installation.

** - A value for the stud tensile load which is less than 4800 lbs. (< 4800) represents an adequate installation.

TABLE 3.10a-4

SEISMIC_DESIGN_VERIFICATION_DATA_SHEET

Cabinet Name: Area Radiation Monitor, H12-P605

Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	720 LBS
Number of Mounting Bolts	4
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	10,592 PSI
Combined Stress (Shear)	5,490 PSI
Margin of Safety (Tensile)	1.36
Margin of Safety (Shear)	1.50

Cabinet Name: TIP Control, H12-P607

Applied Horizontal Acceleration	1.5G
Applied Vertical Acceleration	2.6 G
Allowable Shear Stress in Weld	21,000 PSI
Weight of Cabinet (Approx.)	755 LBS
Number of Plug Welds Used for Mounting	8
Height of Center of Gravity	50 Inches
Total Normal Force per Plug Weld	858.8 LBS
Total Shear Force per Plug Weld	141.6 LBS
Stress in Weld	1,243 PSI
Margin of Safety (Shear)	15.9

Cabinet Name: Division A Radiation Monitor, H12-P606

Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	1,440 LBS
Number of Mounting Bolts	8
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	10,539 PSI
Combined Stress (Shear)	5,465 PSI
Margin of Safety (Tensile)	1.37
Margin of Safety (Shear)	1.52

Cabinet Name: Power Range Monitor, H12-P608

Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	5,750 LBS
Number of Mounting Bolts	40
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	12,793 PSI
Combined Stress (Shear)	6,633 PSI
Margin of Safety (Tensile)	0.95
Margin of Safety (Shear)	1.07

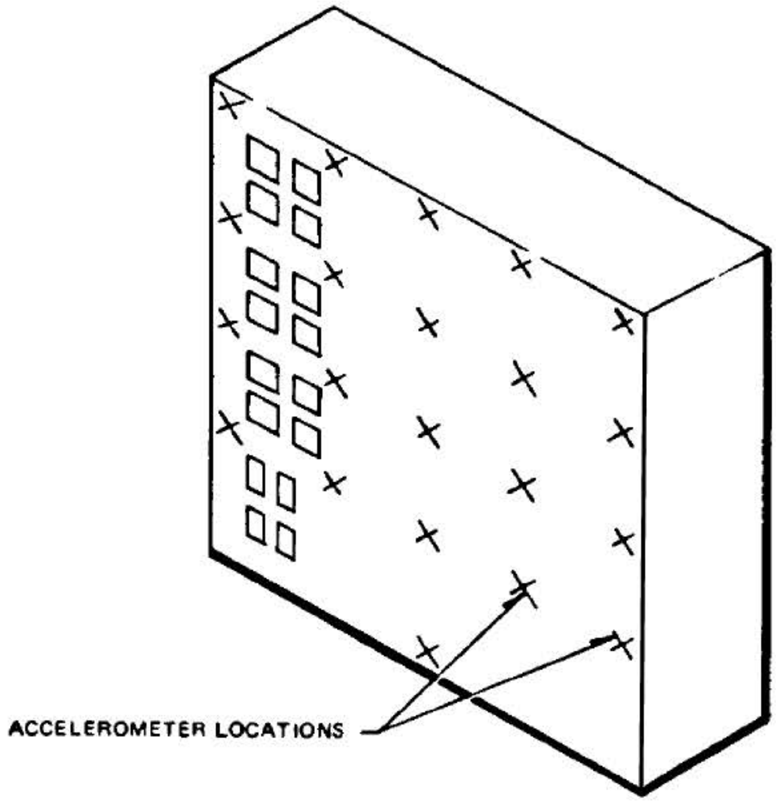
Table 3.10a-4

Cabinet Name: Rod Position Information System, H12-P615

Applied Horizontal Acceleration	1.6 G
Applied Vertical Acceleration	4.6 G
Tension Stress (Maximum-Yield)	25,000 PSI
Shear Stress (Maximum-Yield)	13,750 PSI
Weight of Cabinet (Approx.)	1,425 LBS
Number of Mounting Bolts	12
Height of Center of Gravity	45 Inches
Combined Stress (Tensile)	6,953 PSI
Combined Stress (Shear)	3,605 PSI
Margin of Safety (Tensile)	260
Margin of Safety (Shear)	2.81

IV. CONCLUSION

Review of the Margin of Safety for each standard cabinet indicates that the mounting bolts of each cabinet are capable of withstanding a seismic disturbance.



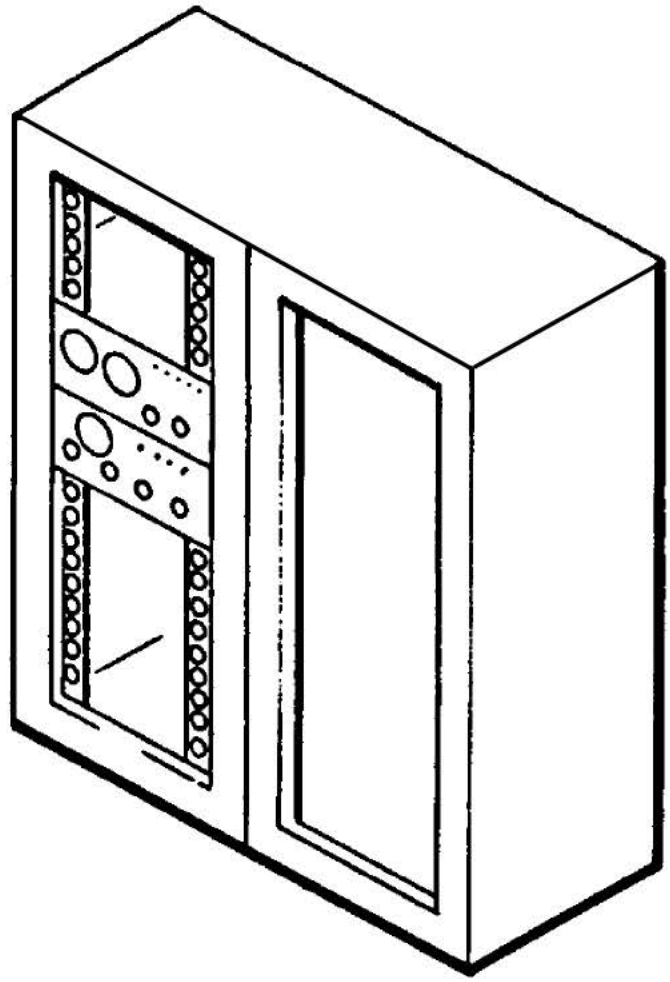
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL VERTICAL BOARD
(BENCHBOARD WOULD BE THE
SAME WITH A BENCH SECTION
PROTRUDING ABOUT
HALF-WAY DOWN)

FIGURE 3.10A-1, Rev. 47

Auto-Cad Figure Fsar 3_10A_1.dwg



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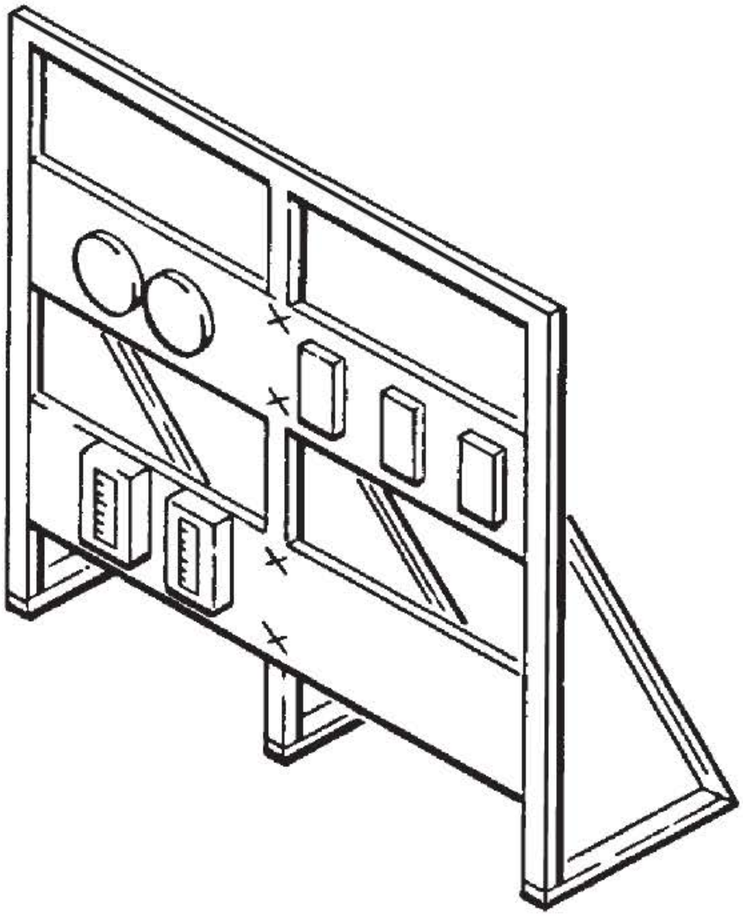
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

INSTRUMENT RACK

FIGURE 3.10A-2, Rev. 47

Auto-Cad Figure Fsar 3_10A_2.dwg



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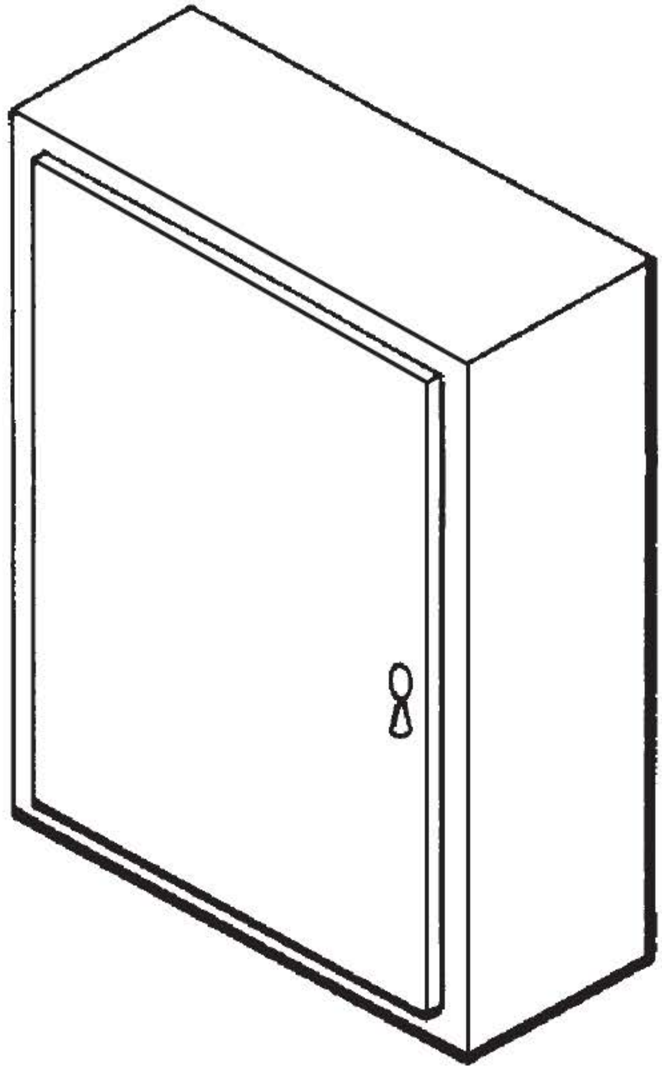
TYPICAL LOCAL RACK
(PIPING AND OTHER EXTERNAL
CONNECTIONS NOT SHOWN)

FIGURE 3.10A-3, Rev. 47

Auto-Cad Figure Fsar 3_10A_3.dwg



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NEMA TYPE-12 ENCLOSURE (INSTRUMENTS MOUNTED INSIDE ON INTERNAL MEMBRANE MOUNTED ON STANDOFFS ATTACHED TO BACK)
FIGURE 3.10A-4, Rev. 47

Auto-Cad Figure Fsar 3_10A_4.dwg

THIS FIGURE HAS BEEN
RENUMBERED TO 3.10a-5-1, 3.10a-5-2, 3.10a-5-3

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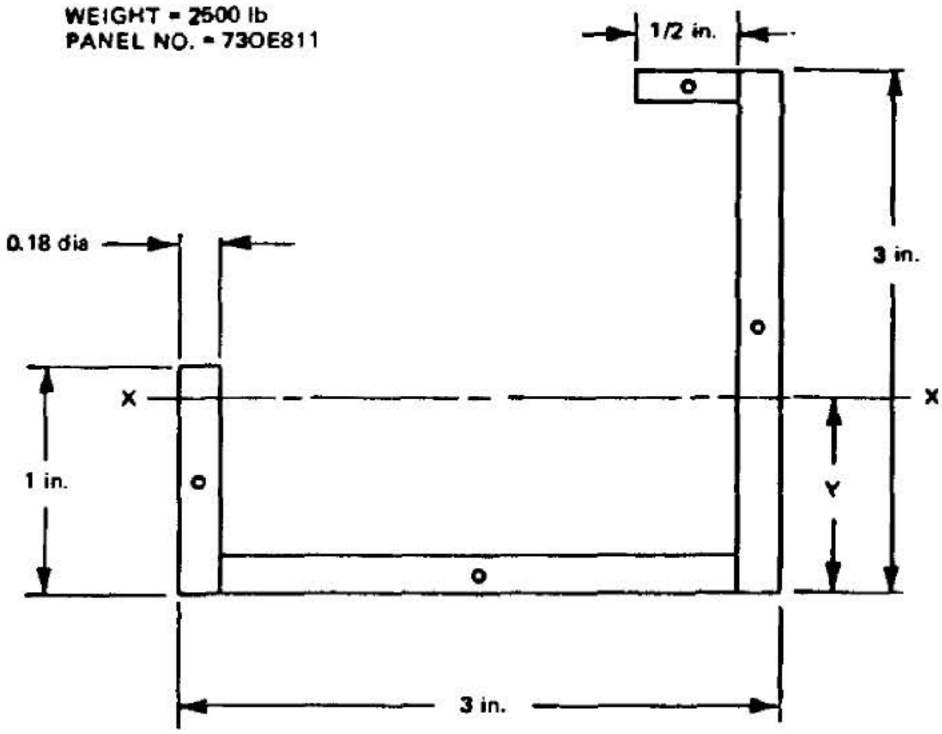
Figure renumbered from 3.10A-5 Sht. 1, Sht. 2, Sht. 3 to Figures 3.10a-5-1, 3.10a-5-2, 3.10a-5-3 FIGURE 3.10a-5, Rev. 47

AutoCAD Figure 3_10a_5.doc



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FINAL SAFETY ANALYSIS REPORT

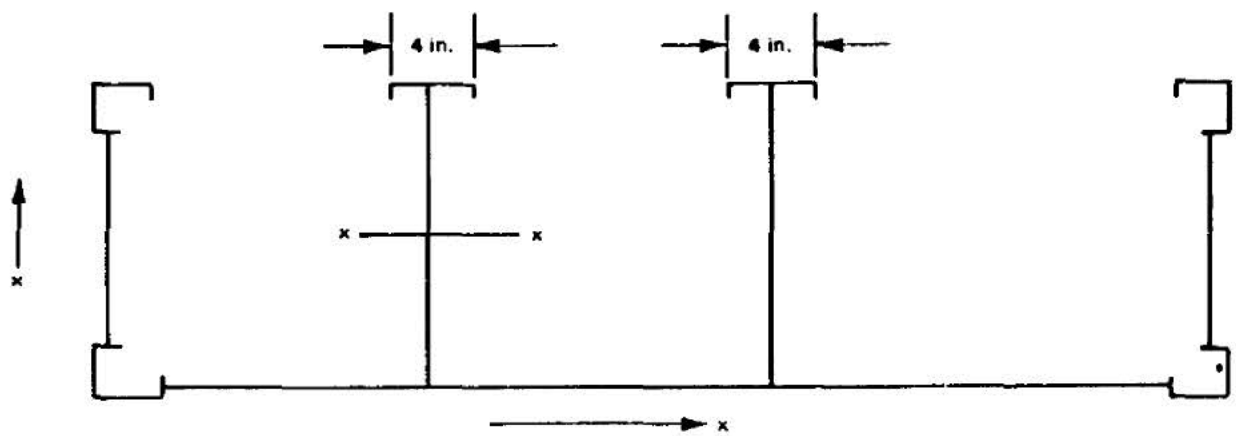
CORNER POST

FIGURE 3.10A-6, Rev. 47

Auto-Cad Figure Fsar 3_10A_6.dwg

SECOND APPROXIMATION

For a second approximation, consider two 0.18 in. x 20 in. barriers in addition to the corner posts. The plan view of the panel is shown in Figure 3.10C-2.



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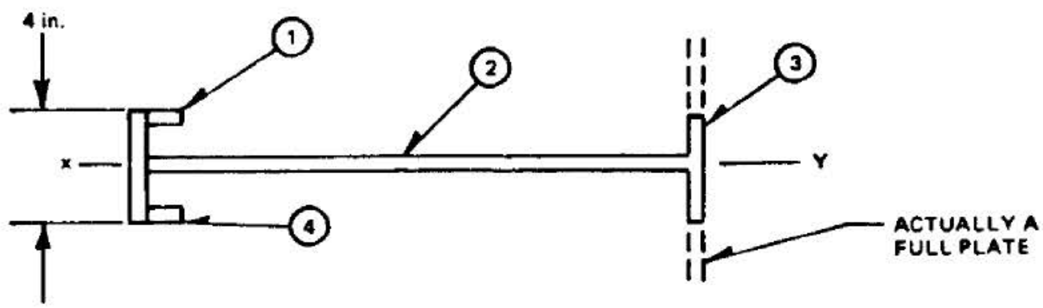
PLAN VIEW OF PANEL

FIGURE 3.10A-7, Rev. 47

Auto-Cad Figure Fsar 3_10A_7.dwg

In the X direction just one barrier will raise the frequency to 30HZ. Use 4 inches of the back panel for each of the two barriers (see Figure 3.10C-3) and the natural frequency in the Y direction becomes 4 HZ.



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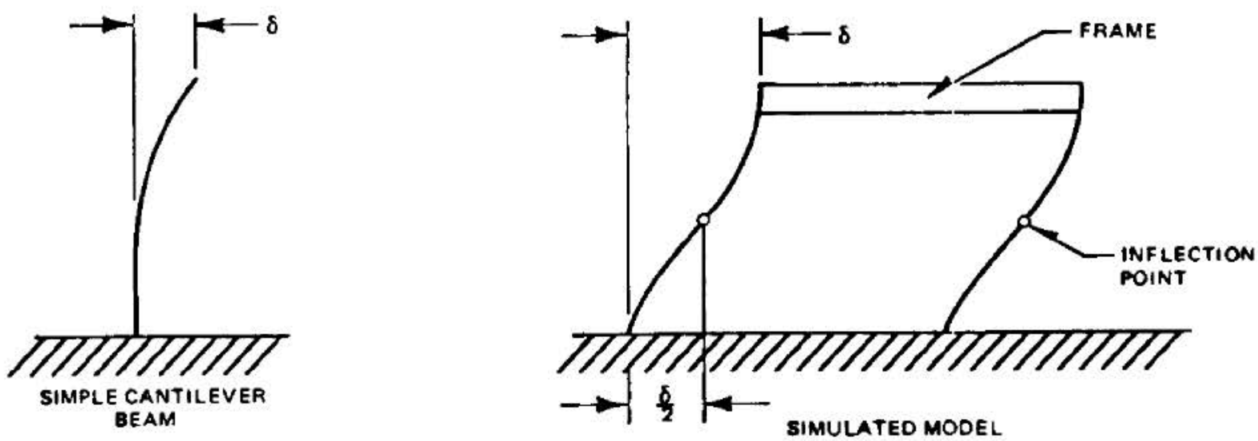
BARRIER WITH TWO END PLATES

FIGURE 3.10A-8, Rev. 47

Auto-Cad Figure Fsar 3_10A_8.dwg



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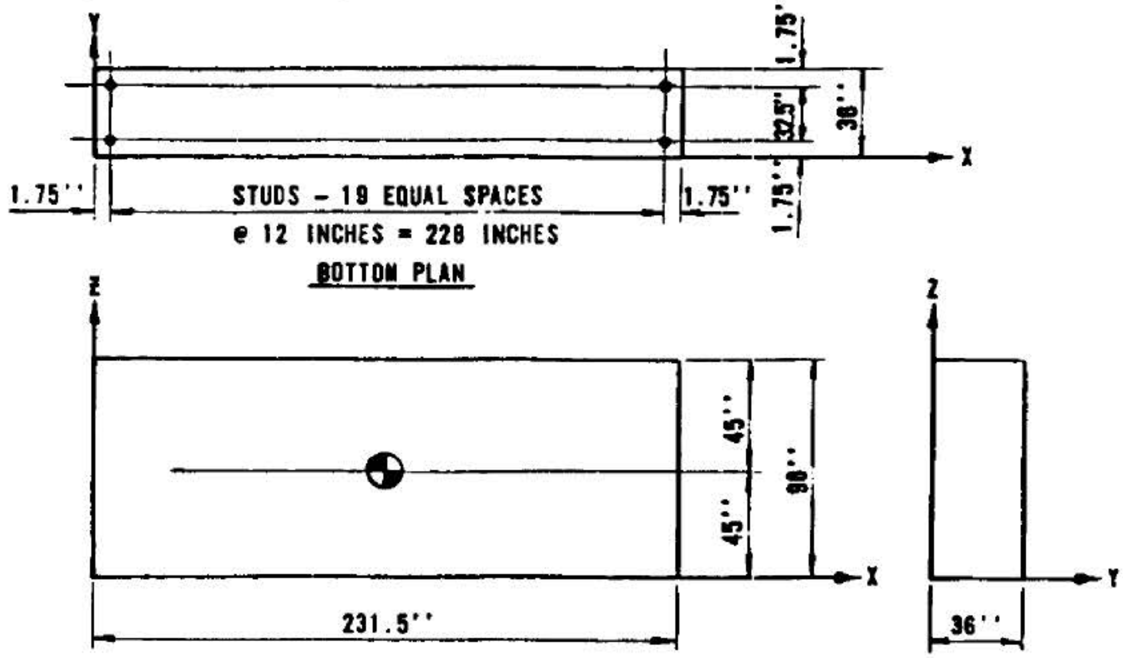
PANEL DEFLECTIONS

FIGURE 3.10A-9, Rev. 47

Auto-Cad Figure Fsar 3_10A_9.dwg

STEP (1): H12-P608, POWER RANGE MONITOR CABINET



FRONT ELEVATION SIDE ELEVATION

LEGEND OF TERMS

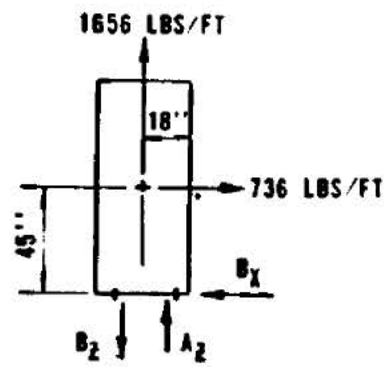
B_Z, A_Z = TENSION/COMPRESSION LOAD IN MOUNTING BOLT

B_X = SHEAR LOAD IN MOUNTING BOLT

B_X' = MAXIMUM COMBINED SHEAR LOAD AT A POINT IN BOLT DUE TO OVERTURNING AND UPLIFT

B_Y' = MAXIMUM COMBINED TENSION LOAD AT A POINT IN BOLT DUE TO OVERTURNING AND UPLIFT

S_t = MAXIMUM COMBINED TENSILE STRESS AT A POINT IN BOLT



FREE BODY DIAGRAM

STEP (2):

ABSOLUTE COMBINED LOADS (G's @ 3% DAMPING)

HORIZONTAL (3 TO 80HZ) 1.5G TO 1.6G [Gx, Gy]
VERTICAL (13 TO 18HZ) 4.6G [Gz]

PANEL FORCES PER FOOT OF PANEL

H12-P608

FRONT TO BACK Fx = 1.6G (460) = 736LB
SIDE TO SIDE Fy = 1.6G (460) = 736LB
UP +Fz = (4.6-1G) 460 = 1656LB
DOWN -Fz = (4.6+1G) 460 = 2576LB

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CABINET INSTALLATION FOR
SEISMIC AND HYDRODYNAMIC
LOADS - SAMPLE CALCULATION
(CABINET H12-P608)

FIGURE 3.10A-5-1, Rev. 47

Auto-Cad Figure Fsar 3_10A_5_1.dwg

STEPS (3) & (5): CASE 1

FRONT TO BACK

$$\overset{+}{\curvearrowleft}\Sigma M_A \quad \left[(3.75\ FT)(736\ LBS/FT)(19\ FT) + (1.5\ FT)(1656\ LBS/FT)(19\ FT) - (2.65\ FT)(B_Z)\right] \div 20\ STUDS = 0$$

$$B_Z = \frac{34,960\ LBS}{20\ STUDS}$$

$$B_Z = LBS\ PER\ STUD = \frac{34,960\ LBS}{20\ STUDS} = 1,748\ LBS/STUD\ TENSION$$

$$\Sigma F \updownarrow = 0 \quad \left[(1656\ LBS/FT)(19\ FT) + A_Z - 34,960\ LBS\right] \div 20 = 0$$

$$A_Z = \frac{3496\ LBS}{20\ STUDS}$$

$$A_Z = LBS\ PER\ STUD = \frac{3496\ LBS}{20\ STUDS} = 174.8\ LBS/STUD\ COMPRESSION$$

STEPS (4) & (5): SHEAR LOAD

$$B_X = \frac{(736\ LBS/FT)(19\ FT)}{40\ STUDS} = 349.6\ LBS/STUD$$

STEP (6): COMBINED STRESS

SHEAR

$$B'_X = \sqrt{B_X^2 + \left(\frac{B_Z}{2}\right)^2} = \sqrt{(349.6)^2 + \left(\frac{1748}{2}\right)^2}$$

$$B'_X = 941.33\ LBS/STUD$$

FORMULAS FOR COMBINED TENSION AND SHEAR AT A POINT MAY BE FOUND IN STRENGTH OF MATERIALS, 2ND ED. BY SINGER, HARPER & ROW PUBLISHERS, 1962.

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CABINET INSTALLATION FOR
SEISMIC AND HYDRODYNAMIC
LOADS - SAMPLE CALCULATION
(CABINET H12-P608)

FIGURE 3.10A-5-2, Rev. 47

Auto-Cad Figure Fsar 3_10A_5_2.dwg

NORMAL (TENSILE)

$$B'_Y = \frac{B_Z}{2} + \sqrt{B_X^2 + \left(\frac{B_Z}{2}\right)^2} = \frac{1748}{2} + \sqrt{349.6^2 + \left(\frac{1748}{2}\right)^2}$$

$$B'_Y = 874 + 941.33 = 1815.33 \text{ LBS/STUD}$$

$$S_T = \frac{B'_Y}{A_T} = \frac{1815.33}{.1419} = 12{,}793 \text{ PSI}$$

STEP (7): MARGIN OF SAFETY

TENSILE

$$\text{M.S. YIELD STRENGTH} = \frac{25{,}000}{12{,}793} - 1 = +.95$$

SHEAR

USE SHEAR YIELD = .55 TENSILE YIELD

$$\text{M.S. YIELD STRENGTH} = \frac{.55(25{,}000)}{\left(\frac{941.33}{.1419}\right)} - 1$$

$$= +1.07$$

ALL MARGINS OF SAFETY ARE POSITIVE, THERFORE, MOUNTING OF CABINET IS ADEQUATE TO RESTRAIN DESIGN LOADS.

STEP (8): STUD PRE-LOAD VS. STUD TENSILE LOAD

PRE-LOAD = 4800 LBS > B'_Y = 1815 LBS, THEREFORE, MOUNTING OF CABINET IS ADEQUATE TO RESTRAIN DESIGN LOADS.

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CABINET INSTALLATION FOR SEISMIC AND HYDRODYNAMIC LOADS - SAMPLE CALCULATION (CABINET H12-P608) FIGURE 3.10A-5-3, Rev. 47

Auto-Cad Figure Fsar 3_10A_5_3.dwg

3.10b SEISMIC QUALIFICATION OF NON-NSSS SUPPLIED SEISMIC CATEGORY I INSTRUMENTATION

3.10b.1 SEISMIC QUALIFICATION CRITERIA

3.10b.1.1 Seismic Category I Equipment Identification

Seismic Category I instrumentation devices and panels were designed to withstand the dynamic effects of the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) established for this power plant. In addition, when required due to equipment location, instrumentation devices and panels are designed to withstand vibration due to Safety Relief Valve (SRV) and LOCA conditions in combination with OBE and SSE loads.

Instrumentation devices are mounted in instrumentation panels, on equipment racks, on piping, and mounted on building walls or wall structural elements. All devices, panels, and racks that are classified Seismic Category I are mounted on similarly classified supporting members and located within Seismic Category I structures.

Seismic qualification requirements were imposed on equipment through the purchasing documents that were used to procure that equipment. Accordingly, seismic qualification was achieved through analysis and/or testing of items identified on each purchase order for each type of device.

Qualification of instrumentation devices and of assemblies through testing was not mandatory. Equipment suppliers were permitted to qualify their equipment by any of the methods allowable in IEEE 344. However, when practical, testing was the preferred method. Analysis methods were not used when equipment was required to perform an active function under seismic conditions.

Instrument racks, instrument tube tray and their supports, and instrument impulse sensing lines were qualified by analysis.

3.10b.1.2 Seismic and Hydrodynamic Design Criteria

The seismic and hydrodynamic (i.e., SRV and LOCA) design and test criteria for qualification of safety related instrumentation for balance-of-plant systems are described below.

3.10b.1.2.1 Functional Criterion

Every instrumentation device shall be capable of performing its safety related function during plant operating conditions of startup, constant power operation, and normal or emergency shutdown without impairment of its safety related function while undergoing seismic and hydrodynamic excitation. The safety related function of instrumentation devices can be either passive or active. Where one type of device is used in both types of applications, the device is qualified for the worst-case application.

3.10b.1.2.2 Qualification Levels

From the plant OBE, SSE, SRV, and LOCA conditions a family of acceleration required response spectra (RRS) were generated for each building elevation for north-south, east-west and vertical directions. The spectra for each elevation where instrumentation is located were examined to establish the worst-case response spectra.

Pipe-mounted devices are procured for certain generic acceleration values (such as 3g or 6g) applied in the vertical and the weakest lateral axis simultaneously. These values are checked against the piping analysis to ensure that the piping response does not exceed the qualification level. Where equipment was not capable of meeting this generic value, the actual "g" value for that equipment was used for qualification.

For devices mounted in panels, the RRS used was derived from the panel analysis or from the panel shaker table test data.

3.10b.1.2.3 Instrumentation Supports

Instrumentation devices, assemblies, and control panels shall be seismically qualified using the supports that will be used during in-plant installation. These items of equipment are required to maintain their functional capability while undergoing earthquake excitation at the equipment supports.

3.10b.1.3 Device Qualification Test Criteria

Devices that were qualified by test were tested in accordance with IEEE Standard 344-1975. In general, test requirements and acceptance criteria are summarized as follows:

a) Devices under test are mounted in a manner that simulates intended use.

b) Devices are tested while in their normal operating condition (e.g., energized) to determine that vibratory conditions do not produce a malfunction or failure. Seismic Category I devices shall not malfunction during or after a safe shutdown earthquake.

c) Devices are tested in all three axes. Simultaneous excitation in all three axes is preferred; however, tests may be run one axis at a time and then be repeated for the other two axes as an acceptable alternative.

d) Where appropriate a frequency sweep (varying the frequency of excitation with time) is conducted at a low "g" value, e.g., 0.2g as noted in IEEE Standard 344. This test was performed to identify resonant frequencies in the range of interest.

e) Devices that are floor- or panel-mounted are subjected to five OBEs and one SSE in each axis tested. Each OBE and SSE consists of random input motion that envelopes the RRS for that device.

f) Devices that are pipe-mounted are subjected to sine-beat tests over the frequency range of 1 to 100 Hz where required. Each sine-beat test is performed at certain generic peak acceleration values (such as 3g or 6g) or to the peak acceleration for the specific mounting location. If used, the generic acceleration values are checked against the piping analysis to insure that the piping response does not exceed the qualification level.

g) The criteria for malfunction or failure include as many of the following characteristics as are applicable to the safety related function of the device during and after testing:

   1) Loss of output signal; e.g., open or short circuit
   2) Output variations greater than ±10 percent of full range
   3) Spurious or unwanted output; e.g., relay contact bounce
   4) Major calibration shift; e.g., greater than ±10 percent of range
   5) Structural failure; e.g., broken or loosened parts.

3.10b.2 SEISMIC CATEGORY I EQUIPMENT QUALIFICATION

Detailed information about seismic qualification of Non-NSSS Supplied Seismic Category I Instrumentation is maintained in a central file within PP&L. A synopsis of this information was by SQRT forms previously submitted to the NRC.

3.10b.3 Methods and Procedures of Analysis or Testing of Supports of Instrumentation

Instrumentation equipment was qualified by test. The instrument support design was considered during the qualification process.

3.10b.4 Operating License Review

Results of tests and analyses were provided in individual SQRT Forms.

TABLE 3.10b-1

THIS TABLE HAS BEEN DELETED

3.10c SEISMIC QUALIFICATION OF NON-NSSS SEISMIC CATEGORY I ELECTRICAL EQUIPMENT

3.10c.1 SEISMIC QUALIFICATION CRITERIA

Seismic qualification of Seismic Category I electrical equipment and supports was demonstrated by the suppliers test laboratories, or consultants by analysis and/or by tests.

Seismic qualification of electrical equipment and supports was performed by analysis when the equipment could be modeled and the structural and functional integrity was adequately represented.

The analysis were performed by an equivalent static analysis or by a dynamic analysis. See Subsection 3.7b.3.5 for details of equivalent static load method of analysis.

The dynamic analysis was performed by the response spectrum method. See Subsection 3.7.3.1 for details of dynamic analysis. When analysis was not sufficient to determine seismic integrity, then tests or a combination of tests and analysis was performed to qualify the electrical equipment and supports.

3.1Oc.1.1 Equipment Location

Electrical equipment and supports are located within the several buildings on the Susquehanna Steam Electric Stations Units 1 and 2.

3.10c.1.2 Response Spectrum Curves for the Electrical Equipment and Supports

Response spectrum curves are based upon the seismic analysis of the supporting structure and represent the maximum seismic response, as a function of oscillator frequency, of an array of single degree of freedom damped oscillators at a particular location within the structure (See Section 3.7).

3.10c.1.3 Seismic Category I Electrical Equipment Loads

Seismic Category I electrical equipment will withstand simultaneously the horizontal and vertical accelerations caused by the OBE and the design SSE as defined herein, in conjunction with applicable electrical, mechanical, and thermal loads. The functions of electrical equipment or components, which are necessary for the functional requirements of the equipment, shall not be impaired when the equipment is subjected to the OBE or the SSE in conjunction with applicable electrical, mechanical, and thermal loads.

3.10c.1.4 Safe Shutdown Earthquake (SSE) Conditions

SSE is defined as an earthquake that produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are necessary to ensure the following:

a) Integrity of reactor coolant pressure boundary

b) Capability to shut down the reactor and maintain it in safe shutdown condition

c) Capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures to the radioactive material released to the environment.

The load combinations include gravity loads and operating loads. Allowable stresses in the structural portions may be increased to 150 percent of allowable working stress limits. The resulting deflections, misalignment or binding of parts, or effects on electrical performance (microphonics, contact bounce, etc) do not prevent the operation of the equipment during or after the seismic disturbance.

3.10c.1.5 Operating Basis Earthquake (OBE) Conditions

The load combinations include gravity loads and operation loads.

Allowable stresses in the structural steel portions may be increased to 125 percent of the allowable working stress limits as set forth in the appropriate design standards, that is, AISC Manual of Steel Construction, ANSI and other applicable industrial codes. The customary increase in normal allowable working stress due to earthquake is used if, according to the appropriate code, it is less than 25 percent. The resulting deflections, misalignment or binding of parts, or effects on electrical performance (microphonics, contact bounce, etc); does not prevent continuous normal operation of the equipment during and after the seismic disturbance.

For the Diesel Generator 'E' facility, the above 25% increase in allowable working stress limit is not allowed.

3.10c.1.6 Prevention of Overturning and Sliding

Stationary electrical equipment is designed to prevent overturning or sliding by the use of anchor bolts, welding, or other suitable mechanical anchorage devices.

3.10c.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT

3.10c.2.1 Seismic Analysis Method

For the purpose of analysis, the equipment has been idealized as a mathematical model consisting of lumped masses connected by massless elastic structural members. For dynamic analysis, the frequencies and mode shapes have been determined for vibration in the vertical and two orthogonal horizontal directions, termed global directions. The effects of coupling between vibrations in all three global directions have been considered. The spectral acceleration per mode has been obtained from the appropriate response spectrum curve, which has been provided for the appropriate damping value. For determining the spectral acceleration from the response spectrum curves, the value chosen is the largest value on the curve when the

frequency in question varies by ±10 percent. Seismic response in terms of inertia forces, shears, moments, stresses, and deflections are determined for response to seismic excitation in each of the global directions for each mode. (See Subsection 3.7b.3.7)

For the consideration of stress or deflection at any point, the total seismic load consists of the most severe seismic load in one of the horizontal global directions combined by the sum of the absolute values method with the vertical seismic load. (See Subsection 3.7b.3.6)

For the Diesel Generator 'E' facility, responses at a point are obtained by taking the square root sum of the squares of corresponding responses due to three orthogonal components of earthquake acting simultaneously.

3.10c.2.2 Seismic Qualification for Electrical Equipment Operability

The seismic qualification of Category I electrical equipment, equipment supports, and material except in the Diesel Generator 'E' Building meets as a minimum the requirements of IEEE 344-1971 and project specification G-10, "General Project Requirements for A Seismic Design and Analysis of Class I Equipment and Equipment Supports" and complemented by Project Specification G-22, "Design Assessment and Qualification of Seismic Category I Equipment & Equipment Supports for Seismic & Hydrodynamic Loads." Project Specification G-10 is summarized in comparison to IEEE-344-1975 and Regulatory Guide 1.100 in Table 3.9-18.

Electrical equipment is qualified for functional operability during and after an earthquake of magnitude up to and including the SSE according to at least one of the following input excitation tests:

a) Single frequency sinusoidal motion or sine beat motions were continuously inputted during the test at specified frequencies to cover the frequency range up to 33 Hz.

b) Random waveform, multifrequency tests.

For the Diesel Generator 'E' facility, the seismic qualification of Category I electrical equipment, equipment supports and material meets as a minimum the requirements of IEEE 344-1975 and project specification C-1041, "General Specification for Seismic Criteria for Design and Qualification of Seismic Category I Equipment and Equipment Supports Located in the Diesel Generator 'E' Building".

3.10c.2.3 Seismic Test Report Analysis and Methods

The analysis and test reports furnished by the supplier demonstrate the ability of electrical equipment to perform its required function during and after the time it is subjected to the forces resulting from one SSE and a required number of OBE.

Four categories of reports are provided by the supplier of electrical equipment and material applicable to Seismic
Category I qualification:

a) Electrical equipment qualified by testing method

b) Electrical equipment support and material qualified by analysis and calculation method

c) Electrical equipment qualified by supplier's certification of Seismic Category I requirements.

d) Combination of analysis and testing.

3.10c.2.3.1 Electrical Equipment Qualified by Testing and Combination of Testing and Analysis Method

Qualification of the electrical equipment listed below is based
on testing performed by the suppliers or test laboratories. (Qualification may be based on tested equipment which is similar in design and assembly).

a) Indoor secondary unit substation and indoor power transformers (see Table 3.10c-1)
b) 480 V ac motor control centers (see Table 3.10c-2)
c) Soother monitors and fuse boxes (see Table 3.10c-3)
d) DC distribution panels (see Table 3.10c-4)
e) Battery racks (see Table 3.10c-5)
f) Electrical cable penetrations (see Table 3.10c-6)
g) Battery charger racks and cabinets (see Table 3.10c-8)
h) Panels and termination cabinets (see Table 3.10c-10)
i) Battery chargers (see Table 3.10c-11)
j) 4.16 kV switchgear (see Table 3.10c-12)
k) DC control and load centers (see Table 3. 10c-13)
l) Instrument ac transformers (see Table 3.10c-14)
m) Automatic transfer switches (see Table 3.10c-15)
n) Load isolation motor generator sets (see Table 3.10c-16)
o) Inverters and 120V AC instrument panels (see Table 3-10c-18)
p) Battery Shunt Box (See Table 3.10c-3)

3.10c.2.3.2 Electrical Equipment and Supports Qualified by Analysis Method

Cable trays were qualified by analysis method based on similarity in design and assembly, and representing the type of equipment shown in Table 3.10c-7.

3.10c.2.3.3 Electrical Equipment Qualified by Suppliers' Certification

Large induction motors (see Table 3.10c-9) were certified by the suppliers the motors had been previously qualified by tests equivalent to those described in Subsections 3.10c.1, or were analyzed and calculated.

3.10c.2.3.4 Minimum Operating Voltage of Voltage Relays

All non-NSSS and non-ACR voltage relays which must be energized or must remain energized to perform safety functions during a seismic event are tabulated in Table 3.10c.17.

3.10c.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment

Electrical equipment supports were qualified or tested with their associated equipment. See Subsection 3.10c.2 for a description of the applicable method or procedure.

3.10c.4 Operating License Review

A summary of tests and analyses is identified in Tables 3.10c-1 to 3.10c-16.

TABLE 3.10c-1 SECONDARY UNIT SUBSTATIONS AND POWER TRANSFORMERS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "EI" SIGNED BY:
8856-E-117-57&58	Single Ended Secondary Unit Substation Consisting of: a. Terminal Chamber, b. 750 kVA Transformer, c. L.W. Switchgear	1B-210 1B-220 1B-230 1B-240 2B-210 2B-220 2B-230 2B-240	Reactor	749 749 719 719 749 749 719 719	1 1 1 1 2 2 2 2	I.T.E. Imperial Corporation	Wyle Laboratories Norco, California	Project Spec G-10* & IEEE-344-1975	Report # 26340-2 26340-3 26340-4 By: G. Shipway
Spec E-1023	1000 kVA Transformer 4.16KV-480V	0X565	D. Gen. 'E' Bldg.	675	Comm.	B.B.C.	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Report # 37-55778-STA By: C. E. Kunkel
Spec E-1023	5kV Switch	0S569	D. Gen. 'E' Bldg.	675	Comm.	B.B.C.	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Report # 37-55778-STA By: C. E. Kunkel

* NOTE: Specification G-10 is complemented by Specification G-22.
For G-10 Specification Summary, See Table 3.9-31.

TABLE 3.10c-2 MOTOR CONTROL CENTERS (Page 1 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-118	Motor Control	0B-136	Control	783	Cmmn	Cutler-Hammer	Wyle	Project Spec	Report #42966-1
	Center	0B-146	Control	783	Cmmn		Laboratories	G-10* & IEEE-	
		0B-516	D.Gen.	677	Cmmn		Huntsville,	344-1975	By: J. Foreman
			'A-D'				Alabama		
		0B-517		677					Wyle
		0B-526		677	Cmmn				Report #45590-1
		0B-527		677	Cmmn				#45590-2
		0B-536		677	Cmmn				
		0B-546		677	Cmmn				By:
									Vincent F. Kearns III
		1B-216	Reactor	683	1				
		1B-217		749	1				C. H. Eaton
		1B-219		670	1				Report #DA57-3251 By:
		1B-226		683	1				Vincent F. Kearns III
		1B-227		749	1				
		1B-229		719	1				
		1B-236		719	1				
		1B-237		670	1				
		1B-246		719	1				
		1B-247		670	1				
		2B-216	Reactor	683	2				
		2B-217		749	2				
		2B-226		683	2				
		2B-227		749	2				
		2B-236		719	2				
		2B-237		670	2				
		2B-246		719	2				
		2B-247		670	2				

* Note: Specification G is complemented by Specification G-22

TABLE 3.10c-2 MOTOR CONTROL CENTERS (Page 2 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
		1Y-216	Reactor	683	1				
		1Y-218		719	1				
		1Y-226		683	1				
		1Y-236		719	1				
		1Y-246		719	1				
		2Y-216		683	2				
		2Y-218		719	2				
		2Y-226		683	2				
		2Y-236		719	2				
		2Y-246		719	2				
Spec E-1024	Motor Control Center	OB-565	D. Gen. 'E'	675		Comm. Telemecanique	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Telemecanique Report No. SC-655 By: Paul W. Higgins

TABLE 3.10c-3
BATTERY MONITORS AND FUSE BOXES

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT IDENTIFICATION EQUIPMENT NO.	LOCATION BLDG	LOCATION ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-0119AC	Battery Monitor 24V	1D-675	Control	771	1	Power Conversion Products, Inc.	Wyle. Laboratories, Huntsville, Alabama	Project Spec G-10* & IEEE 344-1975	Vincent F. Kearns Test Report #45463-1 Rev. A
		1D-676			1				
		1D-685			1				
		1D-686			1				
		2D-675			2				
		2D-676			2				
		2D-685			2				
		2D-686			2				
	Battery Monitor 125V	1D-691			1				
		1D-692			1				
		1D-693			1				
		1D-694			1				
		2D-691			2				
		2D-692			2				
		2D-693			2				
		2D-694			2				
	Battery Monitor 250V	1D-695			1				
		1D-696			1				
		2D-695			2				
		2D-696			2				
	125V Fuse Box 2-1000A	1D-611			1				
		1D-621			1				
		1D-631			1				
		1D-641			1				
		2D-611			2				

*NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-3
BATTERY MONITORS AND FUSE BOXES

ITEM NO.	EQUIPMENT IDENTIFICATION		LOCATION		UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.					
		2D-621			2				
		2D-631			2				
		2D-641			2				
	250V Fuse Box 2-1600A	1D-651			1				
		1D-661			1				
		2D-651			2				
		2D-661			2				
	Fuse Box, 24V, 2-100A	1D-671			1				
		1D-681			1				
		2D-671			2				
		2D-681			2				
Spec E-1025	Battery Monitor 125V	0D-601	D. Gen. 'E'	656	Comm.	Vitro Corp.	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	C&D Power Systems Test Report No. QR2-13201-1 By: Paul Wagner
	Battery Float Current Shunt Box 125V	0D595A			0				
		1D610A, 2D610A			1 2				
		1D620A, 2D620A			1 2				
		1D630A, 2D630A			1 2				
		1D640A, 2D640A			1 2				
	Battery Float Current Shunt Box 250V	1D650A, 2D650A			1 2				
		1D660A, 2D660A			1 2				

TABLE 3.10c-4 DC DISTRIBUTION PANELS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-120	DC Distribution	1D-614	Control	771'	1	I.T.E.	Wyle	Project Spec	Report #26340-5
	Panels	1D-615			1	Imperial	Laboratories	G-10* & IEEE-	By: G. Shipway
	125V 225A	1D-624			1	Corporation	Novco,	344-1975	
	Main Bus	1D-625			1		California		Report #26340-3
		1D-634			1				26340-6
		1D-635			1				By: G. Shipway
		1D-644			1				
		1D-645			1				
	24V 100A	1D-672			1		*NOTE: Specification G-10 is complemented by		
	Main Bus	1D-682			1		Specification G-22.		
	125V 225A	2D-614			2				
	Main Bus	2D-615			2				
		2D-624			2				
		2D-625			2				
		2D-634			2				
		2D-635			2				
		2D-644			2				
		2D-645			2				
	24V 100A	2D-672			2				
	Main Bus	2D-682			2				
Spec E-1027	DC Switchboard 125V	OD-597	D. Gen 'E'	656	Comm	Square 'D' Company	(Qualified by Test) Farwell & Hendricks, Inc. Milford, Ohio	Spec C-1041 & IEEE 344-1975	Square 'D' Report No. 8998-10.09-L74 8998-10.09-L84 By: R. A. Diersing
	DC Distribution Panel 125V	OD-599	D Gen. 'E'	660	Comm	Square 'D' Company	(Qualified by Test) Farwell & Hendricks, Inc. Milford, Ohio	Spec C-1041 & IEEE 344-1975	Square 'D' Report No. 8998-10-09-L74 By: R. A. Diersing

TABLE 3.10c-5
BATTERY RACKS

ITEM NO.	EQUIPMENT IDENTIFICATION: DESCRIPTION	EQUIPMENT IDENTIFICATION: EQUIPMENT NO.	LOCATION: BLDG	LOCATION: ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-119B	Stationary Batteries 24V 75AH	1D-670	Control	771	1	C&D Batteries Co.	Structural Dynamic Research Corporation, Milford, Ohio for Corporate Consulting Development Co.	Project Spec G-10* & IEEE-344-1975	Report #A379-81-01 Stephen A. Lehrman Dr. John Roland Yow
		1D-680			1				
		2D-670			2				
		2D-680			2				
	Stationary Batteries 125V 720AH	1D-610			1				
		1D-620			1				
		1D-630			1				
		1D-640			1				
		2D-610			2				
		2D-620			2				
		2D-630			2				
		2D-640			2				
	Stationary Batteries 250V 1800AH	1D-650			1				
		1D-660			1				
		2D-650			2				
		2D-650			2				
Spec E-1025	Stationary Batteries 125V 825AH	OD-595	D Gen. 'E'	656	Comm	C&D Power Systems	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	C&D Power Systems Report No. QR2-54035-1 By: Paul Wagner

NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-6 ELECTRICAL CABLE PENETRATION (Page 1)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-135	Electrical Cable Penetrations:					Westinghouse	Action Labs	Project Spec G-10* & IEEE-344-1975	Report #PEN-TR-16404-81N By: A. Lebourdais
	Neutron Monitor	1W-100A	Reactor	707'	1				
		1W-100B		707'	1				
		1W-100C		707'	1				
		1W-100D		707'	1				
		2W-100A		707'	2				
		2W-100B		707'	2				
		2W-100C		707'	2				
		2W-100D		707'	2				
	Medium Voltage	1W-101A		735'-9"	1				
		1W-101B		733'	1				
		1W-101C		733'	1				
		1W-101D		700'	1				
		1W-101E		730'-2"	1				
		1W-101F		727'-0"	1				
		2W-101A		735'-9"	2				
		2W-101B		733'	2				
		2W-101C		733'	2				
		2W-101D		700'	2				
		2W-101E		730'-2"	2				
		2W-101F		727'-0"	2				
	Low Level Signal	1W-102A		729'-1"	1				
		1W-102B		729'-1"	1				
		2W-102A		729'-1"	2				
		2W-102B		729'-1"	2				
		1W-103A		707'	1				
		1W-103B		712'	1				
		2W-103A		707'	2				
		2W-103B		712'	2				
	Control Rod Drive	1W-104A		707'	1				
		1W-104B		712'	1				
		1W-104C		712'	1				
		1W-104D		712'	1				
		2W-104A		707'	2				
		2W-104B		712'	2				
		2W-104C		712'	2				
		2W-104D		712'	2				

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-6 ELECTRICAL CABLE PENETRATION (Page 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "EI" SIGNED BY:
8856-E-135	Electrical Cable Penetrations					Westinghouse	Action Labs	Project Spec G-10* & IEEE-344-1975	Report #16404-81" By: A. Labouriais
	Power	1W-105A	Reactor	729'-1"	1				
		1W-105B		729'-1"	1				
		1W-105C		729'-1"	1				
		1W-105D		741'	1		Seismic Test for Modular Penetration 5" Dia.		
		2W-105A		729'-1"	2				
		2W-105B		729'-1"	2				
		2W-105C		729'-1"	2				
		2W-105D		741'	2				
	Low Voltage	1W-106A		729'-1"	1				
		1W-106B		729'-1"	1				
		1W-106C		729'-1"	1				
		1W-106D		741'	1				
		1W-107		741'	1				
		1W-108		729'-1"	1				
		2W-106A		729'-1"	2				
		2W-106B		729'-1"	2				
		2W-106C		729'-1"	2				
		2W-106D		741'	2				
		2W-107		741'	2				
		2W-108		729'-1"	2				
	Suppression Pool Low Voltage, Control and Power	1W-300		688'-6"	1				
		1W-301		688'-1"	1				
		2W-300		688'-6"	2				
		2W-301		688'-1"	2				

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-7 CABLE TRAYS "SAFEGUARD" (Page 1 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-132	Cable Trays: 3"D x 24"W 3"D x 18"W 3"D x 12"W 5"D x 24"W 5"D x 18"W 5"D x 12"W	 S9N1-24-144 S9N1-18-144 S9N1-12-144 S9N1-24-144 S9N1-18-144 S9N1-12-144	Control Reactor	670' to 770'	1&2	Husky Product Inc.	Husky Products, Inc. 7405 Industrial Rd Florence, Kentucky	Project Spec G-10 & IEEE-344-1975	1-29-76 a. Test No. 977-978 Load Test- (Trays) By: T. O'Hara B. Heinz b. (Hold Down Test 4/12/76 Test No. 1127-L,H,V, 5/14/76 1151& 1152 7/21/76 1188 8/10/76 1196-H,V c. Electric Test 12/12/72 Harper-Morrez B. Schuster d. Seismic Calculation 8/11/76 By: B. Schuster

TABLE 3.10c-7 CABLE TRAYS "SAFEGUARD" (Page 2 of 2)

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
Spec E-1032	Cable Trays: 3"D x 12"W 3"D x 18"W 3"D x 24"W 5"D x 12"W 5"D x 18"W 5"D x 24"W 5"D x 36"W		D. Gen 'A-D' and 'E'			T. J. Cope	(By Analysis)	Spec C-1041 & IEEE 344-1975	PP&L Calc. No. JI-CMHR-102 By: T. A. Gorman

TABLE 3.10c-8
BATTERY CHARGER RACKS AND CABINETS

ITEM NO.	EQUIPMENT ID: DESCRIPTION	EQUIPMENT ID: EQUIPMENT NO.	LOCATION: BLDG	LOCATION: ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-119-AC	Battery Chargers 125V 100A	1D-613	Control	771'	1	Power Conversion Products Inc. 42 East Street, Crystal Lake, Illinois 66014	Wyle Laboratories Huntsville, Alabama	Project Spec G-10* & IEEE-344-1975	Test Report #45463-1 Rev. A Vincent F. Kearns
		1D-623			1				
		1D-633			1				
		1D-643			1				
		2D-613			2				
		2D-623			2				
		2D-633			2				
		2D-643			2				
		0D-673			Cmmn				
	Battery Chargers 250V 300A	1D-653A			1				
		1D-653B			1				
		1D-663			1				
		2D-653A			2				
		2D-653B			2				
		2D-663			2				
		0D-683			Cmmn				
	Battery Chargers 24V 25A	1D-673			1				
		1D-674			1				
		1D-683			1				
		1D-684			1				
		2D-673			2				
		2D-674			2				
		2D-683			2				
		2D-684			2				
		0D-685			Cmmn				
Spec E-1025	Battery Charger 125 V 200A	0D-596	D. Gen. 'E'	656	Comm	C&D Power Systems	Wyle Laboratories Huntsville, Alabama	Spec. C-1041 & IEEE 344-1975	C&D Test Report QR2-52666-1 By: Paul Wagner

*NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-9 LARGE INDUCTION MOTORS 4000V

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "EI" SIGNED BY:
8856-E-112	Large Induction Motors 4000V 3ø 450 HP 1800 RPM Emergency Service Water Pump	0P-504-A 0P-504-B 0P-504-C 0P-504-D		685	Cmmn Cmmn Cmmn Cmmn	General Electric	The motors are qualifieid by seismic analysis by McDonald Engineering Analysis, Inc., Birmingham, Alabama	Project Spec. G-10* IEEE 344-1975	C. K. McDonald Report # ME-573 ME-574
	600 HP 4000V 1200 RPM RHR Service Water Pump	1P-506-A 1P-506-B 2P-506-A 2P-506-B		685	1 1 2 2				

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-10
PANELS AND TERMINATION CABINETS

ITEM NO.	EQUIPMENT IDENTIFICATION		LOCATION		UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.					
	Transfer Panels	0C512E-A	D. Gen. 'E'	656'-6"	1	York Electro-Panel	(By Analysis)	Project Spec C-1041* & IEEE 344-1975	N&S Reports 1290-1 and 1290-2 By: M. Randall
		0C512E-B			2				
		0C512E-C			1				
		0C512E-D			2				
	Termination Cabinets	0TC512-A/C			1				
		0TC512-B/D			2				
	Transfer Panels	OC512-A	D. Gen. 'A-D'	710'-9"	1				
		0C512-B			2				
		0C512-C			1				
		0C512-D			2				
	Synchronizing Panel	0C619	D. Gen. 'E'	675'-6"	Comm	Golden Gate Switchboard Co.	Wyle Labs Norco Calif.	Project Spec. C-1041# IEEE 344-1975	Wyle Labs Report No. 53444 By: C. C. Lee

* NOTE: Spec C-1041 is complemented by Spec. E-1026.
# NOTE: Spec C-1041 is complemented by Spec. E-1022.

TABLE 3.10c-11
BATTERY CHARGERS

EQUIPMENT IDENTIFICATION									
ITEM NO.	DESCRIPTION	EQUIPMENT NO.	BLDG	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-119-AC	Battery Chargers 125V 100A	1D-613	Control	771	1	Power Conversion Products, Inc.	Wyle Laboratories, Huntsville, Alabama	Project Spec G-10* & IEEE-344-1975	Test Report #45463-1 Rev. A Vincent F. Kearns
		1D-623			1				
		1D-633			1				
		1D-643			1				
		2D-613			2				
		2D-623			2				
		2D-633			2				
		2D-643			2				
		2D-673			Comm				
	Battery Chargers 250V 300A	1D-653A			1				
		1D-653B			1				
		1D-663			1				
		2D-653A			1				
		2D-653B			2				
		2D-663			2				
		2D-683			2				
		2D-684			2				
		0D-685			Comm				
	Battery Chargers 24V 25A	1D-673 1D-674 1D-683 1D-684 2D-673 2D-674 2D-683 2D-684 0D-685			1 1 1 1 2 2 2 2 Comm				
Spec E-1025	Battery Charger 125V 200A	0D596	D. Gen. 'E'	656	Comm	C&D Power Systems	Wyle Laboratories Huntsville, Alabama	Spec. C-1041 & IEEE 344-1975	C&D Test Report QR2-52666-1 QR2-52666-1 By: Paul Wagner

*NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-12 4.16 KV SWITCHGEAR

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-109-34	4.16 kV Switchgear	1A-201	Reactor	749	1	Westinghouse	Wyle	Project Spec	Report #'s
		1A-202		749	1		Laboratory,	G-10* & IEEE-	57577-1
		1A-203		719	1		Huntsville,	344-1975	57588
		1A-204		719	1		Alabama		
		2A-201		749	2		and		58642
		2A-202		749	2		Wyle		58664
		2A-203		719	2		Laboratory		
		2A-204		719	2		Novco, CA		G. Shipway
Spec E-1022	4.16 kV Switchgear	OA510	D. Gen. 'E'	657	Comm	B.B.C.	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	C&D Report No. 37-55736-SSA By: C. E. Kunkel
		OA510A	D. Gen. 'A-D'	710	Comm	B.B.C.			
		OA510B							
		OA510C							
		OA510D							

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-13 DC CONTROL AND LOAD CENTERS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-121-22-1	DC Control Centers 250V	1D-254	Reactor	670	1	General Electric Co.	Wyle Laboratory, Novco, CA	Project Spec C-10* & IEEE-344-1971	Report# 26340-8 By: G. Shipway
		1D-264	Reactor	683	1				
		1D-274	Reactor	683	1				
		2D-254	Reactor	670	2				
		2D-264	Reactor	683	1				
		2D-274	Turbine	729	2				
8856-E-121-22-3	DC Load Centers 250V	1D-652	Control	771	1			Project Spec G-22 & IEEE-344-1975	Report #'s 26340-2 26340-3 26340-7 By: G. Shipway
		1D-662	Control	771	1				
		2D-652	Control	771	2				
		2D-662	Control	771	2				
	DC Load Centers 125V	1D-612	Control	771	1				
		1D-622	Control	771	1				
		1D-632	Control	771	1				
		1D-642	Control	771	1				
		2D-612	Control	771	2				
		2D-622	Control	771	2				
		2D-632	Control	771	2				
		2D-642	Control	771	2				
Spec E-1024	DC Motor Control Center 125V	0D598	D Gen 'E'	657	Comm	Telemecanique	Wyle Laboratories Huntsville, Alabama	Spec. C-1041 & IEEE 344-1975	Telemecanique Report No. SC-655 By: Paul Wiggins

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-14 INSTRUMENT AC TRANSFORMERS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E1" SIGNED BY:
8856-E-136	Instrument AC	1X-216	Reactor	683	1	Federal Pac.	Wyle	Proj Spec	Report #
	Transformers	1X-226		683	1	Electric Co.	Laboratory,	G-10* &	45455-1
	37.5 kVA, 3Ø	1X-236		719	1		Huntsville,	IEEE 344-	By:
	4 Wire, 480-	1X-246		719	1		Alabama	1975	Vincent F. Kearns III
	208y/120V	2X-216		683	2				
		2X-226		683	2				
		2X-236		719	2				
		2X-246		670	2				
	25 kVa, 1Ø	1X-201A	Reactor	761	1				
	480-120/240 V	1X-201B		761	1				
		2X-201A		761	2				
		2X-201B		761	2				
	15 kVa, 1Ø	0X-507	D. Gen. 'A-D'	710	Comm				
	480-120/240V	0X-508		710	Comm				
		0X-509		710	Comm		Qualified by Analysis		
		0X-510		710	Comm				
		0X-512	ESSW	685	Comm				
		0X-513	Pump-house	685	Comm				
Spec E-1024	30 kVa, 3Ø 480-480/277	0LX-5B	D. Gen. 'E'	675	Comm	Telemecanique	Wyle Laboratories Huntsville, Alabama	Spec C-1041 & IEEE 344-1975	Telemecanique Report No. SC-665 By: J. D. Owens

* NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-15 AUTOMATIC TRANSFER SWITCHES

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "EI" SIGNED BY:
8856-E-152	Automatic Transfer Switch	OATS-516	D Gen. 'A-D'	677'	Comm		Russelectric, Inc. Alabama for C.C & D Company Ltd.	Wyle Laboratory, 344-1975	ProjectSpec G-10* & IEEE- By: James W. Foreman Report # 44434-1
		OATS-526	Gen.	677'	Comm				
		OATS-536		677'	Comm				
		OATS-546		677'	Comm				
		1ATS-219	Reactor	670'	1				
		1ATS-229		719'	1				
		2ATS-219	Reactor	670'	2				
		2ATS-229		719'	2				
		OATS-556	D Gen. 'E'	656'-6"	Comm	Gould/ Telemecanique (TE)	Wyle Laboratory Huntsville, Alabama	Project Spec. C-1041# & IEEE 344-1975	TE Report SC-657, Rev. 1 By: P. Higgins

* NOTE: Specification G-10 is complemented by Specification G-22.
\# NOTE: Spec. C 1041 is complemented by Spec. E-1024.

TABLE 3.10c-16

LOAD ISOLATION MOTOR-GENERATOR SETS

ITEM NO.	EQUIPMENT IDENTIFICATION DESCRIPTION	EQUIPMENT NO.	LOCATION BLDG.	ELEV.	UNIT NO.	SUPPLIER	TESTING FACILITIES	QUALIFICATION CRITERIA	QUALIFICATION "E" SIGNED BY:
8856-E-151	Load Isolation Motor Generator Sets:								
	Motor 150 HP. 3ϕ, 480 V	1S-246	Reactor	670'	1	Engine Power Co./ Kato Engineering	Wyle Labs and R. W. Siegfried Associates	Project Spec G-10* & IEEE-344-1975	Wyle Report # 58393 58411 By: W. M. West Report #4058
	Generator	1G-202		670'	1				
	100 kv, 3ϕ 480 V	1S-247		719'	1				
		1G-203		719'	1				R. W. Siegfried Report #4058
		2S-246		670'	2				
		2G-202		670'	2				
		2S-247		719'	2				
		2G-203		719'	2				
	Control Panels for Motor-Generator Sets	1C-246		670'	1				
		1C-247		719'	1				
		2C-246		670'	2				
		2C-247		719'	2				

NOTE: Specification G-10 is complemented by Specification G-22.

TABLE 3.10c-17

NON-NSSS AND NON-ACR RELAYS REQUIRED TO BE ENERGIZED
(Units 1 & 2 Devices Are Identical)

Device No.	Relay Function	Location	Manufacturer Type	Operating (Volt) Normal	Minimum	Seismically Type Tested	Remarks
27	Supervise 480 V Auto Transfer Switches	0ATS219, 0ATS229 0ATS516, 0ATS526 0ATS536, 0ATS546	Russel-Electric UV-100/42 480/500	480 ac	432 ac	432 Vac	
27A	Initiates 4 kV Bus Auto Transfer	1A201, 1A202 1A203, 1A204	ABB/West. – SVF31	119	107	24 Vac	
27AI	Permissive to Close 4 kV Incom Breakers	1A201, 1A202 1A203, 1A204	ITE-270	119	107	110 Vac	
43	Transfer Relay for 480 V Auto Transfer Switches 0ATS536, 0ATS546	1ATS219, 1ATS229 0ATS516, 0ATS526	Ward-Leon ARD Bul-130	480 ac	432 ac	432 Vac	
44	Initiation of 4 kV ESF Loads	1A201, 1A202 1A203, 1A204	ABB/West – SSV-T	120 ac	90 ac	90 ac	
59N	Trip ±24 vdc Battery Charger on Overvoltage	1D672, 1D682	GE – NSV	24 dc	28 dc	30 dc	
51V	480 V Swing Bus M-G Set Protection	1C246, 1C247	ABB/West – Cov-9	120 ac	108 ac	80 ac	
62	Time Delay Relay	Various inplant Locations	Agastat 7000 Series	125 dc	105 dc	120 dc	
62	Time Delay Relay	Various inplant Locations	Agastat 7000 Series	120 ac	108 ac	120 ac	
62	Time Delay Relay (480 V Auto Transfer Switches	1AT219, 1AT229 0ATS516, 0ATS526 0ATS536, 0ATS546	Ind. Timer CSF-30M	120 ac	108 ac	120 Vac	
X	Auxiliary Control Relays	1ATS219, 1ATS229 0ATS516, 0ATS526 0ATS536, 0ATS536	Ward – Leon ARD 130-6429	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays	1C246, 1C247	GE-HFA	125 dc	105 dc	125 Vdc	

TABLE 3.10c-17

NON-NSSS AND NON-ACR RELAYS REQUIRED TO BE ENERGIZED
(Units 1 & 2 Devices Are Identical)

Device No.	Relay Function	Location	Manufacturer Type	Operating (Volt) Normal	Operating (Volt) Minimum	Operating (Volt) Seismically Type Tested	Remarks
X	Auxiliary Control Relays	1C661A, 1C661B 0C877A, 0C877B 0C876A, 0C876B	GE-HFA	120 ac	108 ac	125 Vdc	
X	Auxiliary Control Relays	1C661A, 1C661B	GE-HMA	120 ac	108 ac	96 ac	
X	Auxiliary Control Relays	1C661A, 1C661B	GE-HMA	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays	1C661A, 1C661B 0C578, 0C681 1C681, 0C877A 0C877B, 0C883A 0C883B, 0C876A 0C876B	Agastat-GPI	120 ac	108 dc	96 Vac	
X	Auxiliary Control Relays	0C519A, 0C519B 0C519C, 0C519D 0C519E, 0C521A 0C521B, 0C521C 0C521D, 0C521E	Agastat-GPD	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays (prevent cycling of 4 kV Bkr)	1A201, 1A202 1A203, 1A204	ABB/West – AR	125 dc	105 dc	125 Vdc	
X	Auxiliary Control Relays	1A201, 1A202 1A203, 1A204	ABB/West – MG6	125 dc	105 dc	125 Vdc	
X	Isolation Relays	1A201, 1A202 1A203, 1A204	P.B. MDR – 5062/ 5151	125 dc	105 dc	105 dc	(1)
X	Isolation Relays	1C661A, 1C661B 0C877A, 0C877B 0C876A, 0C876B 0C529A, 0C529B	P.B. MDR – 4094/ 4094-1/4165	120 ac	92 ac	90 ac	(1)

Remarks

(1) Test made by Arkansas Unit 1.

TABLE 3.10C-18 - INVERTERS AND 120 VAC INSTRUMENT PANELS

Item No.	Equipment Identification		Location		Unit No.	Supplier	Testing Facilities	Qualification Criteria	Qualification "E1" Signed By:
	Description	Equipment No.	Bldg.	Elev.					
	Inverters 2KVA, 120VAC	1D115 1D125 2D115 2D125	Control	754' 714' 754' 714'	1 1 2 2	General Electric	Wyle Laboratories Novco, California	Project spec G10* and IEEE 344-1975	ADO-01294-1 and -9 signed by D.A. Kneerea
	Instrument Distr. Panels 120VAC, 100A Main Bus	1Y115 1Y125 2Y115 2Y125	Control	754' 714' 754' 714'	1 1 2 2	Eaton Corp.	Wyle Laboratories Novco, California	Project Spec G10* and IEEE 344-1975	45590-1 signed by V.F. Kearns, III

*NOTE: Specification G-10 is complemented by Specification G-22. For G-10 specification summary, see Table 3.9-31.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Environmental design criteria for the facilities conform to 10CFR50, Appendix A, General Design Criteria 1, 4, and 23. Compatibility of mechanical and electrical equipment with environmental conditions is provided to fulfill the following design criteria:

a) For normal operation, systems and components required to mitigate the consequences of a design basis accident (DBA) or required for a safe shutdown, are designed to remain functional after exposure to the following environmental conditions:

   1) Design temperatures, pressures, and relative humidity values maintained at the equipment location during normal operating by the heating, ventilating, and cooling systems described in Section 9.4.

   2) Maximum expected integrated radiation exposures, for 40 years at the equipment location during normal operation. For service beyond 40 years, the SSES EQ Program is used to manage the aging of equipment to ensure the components continue to perform their intended function.

The environmental conditions expected during normal operation are given in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12. The environmental conditions identified in Table 3.11-1 are for the turbine building and for all elevations in the control building except elevation 806'. The environmental conditions in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 are for primary containment, the reactor building and elevation 806' of the control structure.

b) In addition to the normal operation environmental requirements listed in a) above, the systems and components required to mitigate the consequences of a DBA, or to effect a safe shutdown of the reactor are designed to remain functional after exposure to the applicable accident environmental conditions. The applicable environmental conditions are those anticipated to follow the DBA that the systems or components are intended to mitigate and are listed below:

   1) Components Inside Containment

      The temperature, pressure, and humidity inside containment after a design basis Loss of Coolant Accident (LOCA) conditions are indicated in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

Containment zones are shown on Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

The post-LOCA radiation environment is calculated assuming that 100 percent of the core noble gas inventory, 50 percent of the core halogen inventory, and 1 percent of the core solid fission product inventory are released. The calculational method is in accordance with NUREG 0588, Rev. 1, Section 1.4 and appendix D. The total calculated post-accident dose is the integrated dose from the time of the LOCA to 180 days.

2) Components Outside Containment

The expected temperature, pressure and humidity conditions are specified in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

In computing the expected integrated accident doses for equipment in contact with or in proximity to Emergency Core Cooling System (ECCS) - water, it is assumed that 50 percent of the core halogen inventory and 1 percent of the core solid fission product inventory are diluted by the Reactor Coolant System water plus the suppression pool water after a design basis LOCA. For equipment located remotely from ECCS water, the appropriate accidental release is assumed.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Class 1E safety-related equipment is installed in accordance with mechanical and electrical separation requirements and designed and qualified in accordance with the provisions of IEEE 323-1971 and 1974, with appropriate margins, to function properly in the environments listed.

An Environmental Qualification Master Equipment List (EQMEL) is maintained for the SSES equipment which requires environmental qualification through the current Procedure.

1) required to detect a steam or water line accident condition;
2) required to perform a steamline isolation function;
3) required to perform a water line isolation function and could be subjected to the steam environment such as electrical cable or valve operator;
4) required for safety system operation and is located so a steamline break in some other system exposes the safety system equipment to the local accident environment; and,
5) required to track the post-accident environment condition such as pressure, temperature and radiation monitors.

Electrical switchgear and motor control centers required for safety system operation are located outside of the drywell accident environment to ensure operation.

Harsh environments may arise in primary and secondary containments as a result of a Loss of Coolant Accident (LOCA) inside primary containment. In addition, harsh environments may arise in localized areas outside primary containment as a result of a High Energy Line Break (HELB). The environmental conditions (both normal and maximum) to which Class 1E equipment is exposed are given in Table 3.11-1 and Dwgs.
C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4,
C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8,
C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

The nonseismic vibration of safety-related equipment conforms to requirements of the following standards:

Equipment	Standard
Diesel Fuel Oil Transfer Pumps	Hydraulics Institute Standards
RHR Service Water Pumps	Hydraulics Institute Standards
Emergency Service Water Pumps	Hydraulics Institute Standards
Control Structure Chilled Water Pumps	Hydraulics Institute Standards
All other Safety-Related Pumps	API 610, Section IV or better
HVAC Fans	ASHRAE Systems Handbook 1973 Edition, Chapter 35, Page 24
Diesel Generator Engines	DEMA Standard Practices for low and medium speed stationary diesel and gas engines
Electric Motors	NEMA MGI

The absence of any significant nonseismic vibration caused by pipe vibration interaction with above equipment is verified in Subsection 3.9.2.1.

A containment spray system may be utilized following the LOCA; therefore, exposed safety-related systems located in the containment are designed to withstand the effects of the containment spray.

3.11.2 QUAIIFICATION TEST AND ANAIYSIS

The qualification tests conform to the requirements of Appendix B, Section XI of 10CFR50.

For the Class 1E equipment, qualification tests and analysis performed on electrical equipment, including motors, are maintained in an auditable manner as discussed in Section 3.11.3. Class 1E equipment installed at SSES is subject at a minimum to the requirements of NUREG-0588, Category II as detailed in Section 3.11.2a.1. Certain new or replacement Class 1E equipment installed at SSES is subject at a minimum to the requirements of IEEE Standard 323-1974 as detailed in Section 3.11.2a.2.

3.11.2.1 CLASS 1E EQUIPMENT QUALIFIED TO IEEE STANDARD 323-1971

The original Class 1E equipment at SSES have been qualified at a minimum to NUREG-0588, Category II and IEEE Standard 323-1971. This is because SSES received its construction permit before July 1, 1974. Nuclear power plants which received their construction permits after

July 1, 1974 are required to qualify the Class 1E equipment to NUREG-0588, Category I and IEEE Standard 323-1974. New or replacement equipment purchased after May 23, 1980 is qualified to the requirements of 10CFR50.49 (NUREG-0588, Category I; IEEE Standard 323-1974) except in cases where sound reasons to the contrary have been established per the guidelines of Regulatory Guide 1.89.

3.11.2.2 CLASS 1E EQUIPMENT QUALIFIED TO IEEE STANDARD 323-1974

A large number of pieces of Class 1E equipment at SSES are qualified to NUREG-0588, Category I and to IEEE Standard 323-1974. Qualification to IEEE Standard 323-1974 requires type testing of a prototype to demonstrate that the equipment will perform its safety function in the combined temperature, pressure, humidity, chemical and radiation environment.

Equipment qualified to IEEE 323-1974 is qualified to the test sequence and margins specified in IEEE 323-1974 unless justification for using another sequence or other margins is provided in the SSES EQ documentation.

3.11.2.3 Class 1E Component Environment Design and Qualification for Normal Operation

Class 1E equipment is designed for 40 years of continuous operation in the most severe temperature, pressure, humidity, and radiation environments that exist at the equipment location during normal operation, assuming that proper routine preventive maintenance is performed, such as periodic replacement of seals, packing, and consumable materials. For service beyond 40 years, the SSES EQ Program is used to manage the aging of equipment to ensure the components continue to perform their intended function.

Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 provide the maximum normal and maximum DBE values for the environmental parameters of temperature, pressure, and humidity for each area in which Class 1E safety-related equipment is located, as well as the exposures to radiation.

For most equipment, special qualification tests to verify operability at normal operating temperature, pressure, and humidity conditions are not required. Verification for this equipment is based on proven operating capability in similar environments in industrial and previous nuclear power plant applications. The pre-operational and post-operational test programs for safety-related components further ensure that safety-related components will be available when required. Since the normal and accident integrated radiation doses have cumulative effects, the integrated radiation dose during normal operation is discussed in Subsection 3.11.5.3.

3.11.2.4 Class 1E Component Environmental Design and Qualification for Operation After a Design Basis Accident

Class 1E safety-related equipment is designed to remain functional in the most severe combination of temperature, pressure and humidity conditions that exist at the equipment location after a design basis LOCA. This equipment is also designed for the maximum calculated integrated radiation exposure of the LOCA or of the accident, as discussed in Subsection 3.11.5. The temperature, pressure, and humidity environment inside the primary containment after a LOCA is presented and discussed in detail in Subsection 6.2.1. The integrated post accident radiation dose for plant locations in which the equipment is located is given in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12. In addition, possible steam and feedwater line breaks outside the containment are analytically checked to ensure that no additional qualifications need be applied to components that could be affected by these breaks.

The requirements of the general design criteria, 1, 4, and 23 of Appendix A to 10CFR50, are met as discussed in Section 3.1.

The recommendations contained in the regulatory guides listed below (listings a) through g)) have been utilized as described in Section 3.13. Additional discussion is included in listings f) through j).

a) Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment.

b) Regulatory Guide 1.40, Qualification Test of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants.

   Continuous duty motors used inside the containment are type tested under simulated LOCA conditions. IEEE Standard 334-1971 is used. Insofar as practicable, auxiliary equipment that is part of the installed motor assembly is likewise qualified in accordance with IEEE Standard 334 under simulated design bases event conditions.

c) Regulatory Guide 1.63, Electrical Penetrations Assemblies in Containment Structures for Water Cooled Nuclear Plants.

   Electrical containment penetrations are tested in accordance with IEEE Standard 317-1972. Refer to Section 8.1 for discussion on this guide.

d) Regulatory Guide 1.73, Qualification Test of Electric Valve Operator, Installed Inside the Containment of Nuclear Power Plants.

   Motor operated valves used inside the containment are type tested as a minimum in accordance with IEEE 382-1972 (ANSI N41.6).

e) Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants.

The qualification methods and documentation requirements of IEEE 323-1971 are discussed in Section 3.11.

f) Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident, Rev. 2, December 1980.

g) Regulatory Guide 1.131, Qualification Tests of Electric Cables, Filed Splices, and Connections for Light-Water-Cooled Nuclear Power Plants, August 1977.

h) Type tests to ensure acceptability for use in the containment post accident environment are performed for each type of cable in accordance with IEEE Standard 383-1974.

i) Pressure boundary components inside the containment are designed for the temperature, pressure, and humidity environment in accordance with the applicable codes to which the component is constructed. Qualification testing is not considered necessary for such components.

j) A total (normal plus accident) integrated dose of less than 10^4 rad will not affect the strength or properties of materials used; hence, further qualification analyses and tests for components which will be exposed to less than 10^4 rad are not necessary. However, certain electronic equipment such as metal oxide semi-conductive devices are sensitive to radiation levels of less than 10^4. Therefore, radiation qualification is evaluated on a case by case basis even when the postulated accident dose is less than 10^4. For higher integrated doses, components are qualified either by qualification testing or by evaluation of materials used. Reliable accumulated data on radiation effects, such as contained in Reference 2, is used to analyze the dose effects of particular materials.

k) The sources used in calculating radiation levels following LOCA are consistent with those set forth in NUREG-0588, Rev. 1. All the active non-NSSS safety-related equipment located inside the primary containment is designed to withstand the maximum integrated doses during the life of the plant. Suitability of materials used is verified by test for all electrical penetration assembly materials, for an integrated dose rate of at least 5 x 10^7 rad, in accordance with requirements of IEEE 317-1972.

l) The materials used in the fabrication of the reactor coolant system pressure boundary and other mechanical and structural components are selected to minimize corrosion and hydrogen generation.

3.11.3 QUALIFICATION TEST RESULTS

Environmental qualification documentation for Class 1E electrical equipment is contained in EQ binders maintained by Susquehanna Nuclear Records in File R34. In some cases, additional details of test results for GE safety-related equipment are maintained in a permanent file by GE and can be readily audited. In all cases, the equipment used in Class 1E applications passed the prescribed tests. Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12, identify the temperature pressure, humidity, and radiation environments to which the Class 1E equipment has been qualified.

3.11.4 LOSS OF VENTILATION

The maximum temperatures considered in the sizing of air conditioning systems serving safety-related systems are determined by additive analysis of the following factors:

a) Maximum outdoor design temperature for the geographical area of the plant (both wet bulb and dry bulb readings)

b) Maximum internal piping thermal loads, if applicable, for the room, using maximum normal operating temperatures for the pipe contents and maximum footage of active pipe for each mode of operation.

c) Maximum internal electrical load assuming full lighting for the room, and using, if applicable, the maximum control and equipment resistance losses for each mode of operation.

d) Maximum heat transfer from miscellaneous equipment surfaces, if applicable (e.g., outer surface of the diesel generator).

e) Maximum heat transfer from the surface of open pools and tanks, if applicable, using the maximum operating temperature of the contents.

f) Maximum heat transfer from the room envelope including walls, floor, and ceiling, or roof (this value may be negative).

Seismic Category I air conditioning and air cooling systems, described in Section 9.4 are powered from the Class 1E electrical power supplies and are provided for the locations listed in Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

These Category I systems are designed so that the single failure of an active mechanical component, or an active or passive electrical component, after a DBA, cannot impair the ability of the systems served by the air conditioning equipment to fulfill their safety functions. Should a train in a Seismic Category I air conditioning system become inoperative during normal operation, sufficient equipment is still available to mitigate the consequences of a DBA.

Two redundant Seismic Category I emergency air conditioning trains are provided for the control room.

Power cable is rated for a conductor temperature of 194°F (90°C). Class 1E cables are qualified for the plant specific worst case temperatures expected during normal operation and post accident conditions by testing and analysis. The allowable current carrying capacity of the cable is based on not exceeding this insulation design temperature while the surrounding air is at an ambient temperature of 150°F (65.5°C) inside the containment and for the rest of the plant area a temperature of 122°F (50°C) or 104°F (40°C) depending on location. The cable ampacity is determined as discussed in Section 8.3.3.1.

Instrumentation cable is rated for a conductor temperature of 194°F (90°C). Class 1E cables are environmentally qualified for the plant specific worst case temperatures expected during normal operation and post accident conditions by testing and analysis. Operating currents of these cables are low and will not cause this temperature to be exceeded at maximum design ambient temperature. Instruments required to operate following a DBA are not located in pipe tunnels, but are mounted outside the tunnels.

3.11.5 ESTIMATED CHEMICAL, PHYSICAL, AND RADIATION ENVIRONMENT

3.11.5.1 Suppression Pool, Residual Heat Removal System, and Emergency Core Cooling System Water Quality

The water in these systems shall not be chemically inhibited. The maximum limits for the suppression pool have been established to be compatible with the primary coolant limits for the shutdown condition and are listed in Table 3.11-7 for comparison.

Observations made of suppression pool water quality over a period of several years in suppression pool with and without coatings, indicate that the feed and bleed to radwaste that occurs during normal system testing and level adjustments maintain the water quality well within the above limits.

During reactor shutdown cooling, the RHR system water mixes with reactor water. Therefore, to insure reactor water quality, as much as practicable of the shutdown cooling piping and equipment shall be flushed with either reactor water or water of the quality specified above for maximum limit.

3.11.5.2 Physical Environment

Engineered safety feature (ESF) systems are designed to perform their safety-related functions in the temperature, pressure, and humidity conditions described in Subsection 3.11.2, and Sections 6.2 and 6.3.

The containment atmosphere is maintained below 4 percent by volume hydrogen consistent with the recommendations of Regulatory Guide 1.7 as discussed in Subsection 6.2.5.

3.11.5.3 Radiation Environment

ESF systems and components are designed to perform their safety-related functions after the normal operational exposure plus an accident exposure. The normal operational exposure is based on the design source terms presented in Chapter 11 and Subsection 12.2.1 and the equipment and shielding configurations presented in Section 12.3.

Post-accident ESF system and component radiation exposures are dependent on equipment location. In the containment and control room area, exposures are due to a hypothesized LOCA. Source terms and other accident parameters are consistent with the recommendations of NUREG 0588, Rev. 1.

In the reactor building, post-accident exposures to ECCS systems recirculating depressurized reactor fluids are based on source terms assuming an inventory of 50 per cent of the core halogens and one percent of the core solid fission products. Post Accident exposures to the High-Pressure Coolant Injection and Reactor Core Isolating Cooling Systems are based on steam source terms of control rod drop accident as described in Section 15.4.9. The minimum exposure within the Reactor Building is based on airborne sources originating from one per cent per day drywell leakage of post-accident airborne activity.

Normal, accident, and design (normal plus accident) radiation exposures for plant areas, based on the above assumptions, are presented in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12.

Organic materials that exist within the containment are identified in Subsection 6.1.2. The design radiation exposures identified in Table 3.11-1 and Dwgs. C-1815, Sh. 1, C-1815, Sh. 2, C-1815, Sh. 3, C-1815, Sh. 4, C-1815, Sh. 5, C-1815, Sh. 6, C-1815, Sh. 7, C-1815, Sh. 8, C-1815, Sh. 9, C-1815, Sh. 10, C-1815, Sh. 11, and C-1815, Sh. 12 are based on gamma radiation exposure only except as noted. The attenuation of beta radiation by small amounts of shielding, such as conduits and jackets for cable and casings for equipment, is evaluated. Where such shielding is not completely effective, the equipment is qualified for radiation exposures which include the appropriate portion of the postulated beta TID.

3.11.6 REFERENCES

3.11-1. J.J. DiNunno, R.E. Baker, F.D. Anderson, and R.L. Waterfield, "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, Division of Licensing and Regulation, AEC, Washington, D.C. (1962).

3.11-2. J.F. Kircher and R.E. Bowman, "Effects of Radiation on Materials and Components", Van Nostrand Reinhold, New York, 1964.

Table Rev. 55

TABLE 3.11-1

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

ABEA	KEY (3)	PRESSURE	NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS WITH NUREG-0588 SOURCE TERM				
			TEMP °F MAX/MIN	RELATIVE HUMIDITY MAX/MIN %	DOSE RATE (R/HR) (4)	INTEGRATED DOSE (RAD)	PRESSURE (2)	TEMP °F (2)	RELATIVE HUMIDITY % (2)	LOCA DOSE RATE (RAD/HR) (6) (7)	TOTAL INTEGR. DOSE (1) (RAD) (6) (7)
Control Room	CS1	+.125" wg	80 / 70	55 / 45	.0005	$1.8X10^2$	+.125" wg	80	55	<1.0	$1.8X10^2$
Cable Spreading Rooms & HVAC Equipment Room, Relay Rooms Elect. Equip. Rooms	CS2	+.125" wg	80 / 60	60 / 10	.0005	$1.8X10^2$	+.125" wg	90	60	<1.0	$1.8X10^2$
Battery Room	CS5	+.125" wg	80 / 60	60 / 10	.0005	$1.8X10^2$	+.125" wg	80	60	<1.0	$1.8X10^2$
Computer Room	CS3	+.125" wg	85 / 65	60 / 40	.0005	$1.8X10^2$	+.125" wg	85	60	<1.0	$1.8X10^2$
Diesel Generator 'A-D' Rooms (5)	G	Atmos	104 / 72	90 / 5	.0005	$1.8X10^2$	Atmos	120	50	<1.0	$1.8X10^2$
ESW Pumphouse	SW	Atmos	104 / 40	100 / 5	.0005	$1.8X10^2$	Atmos	104	100	<1.0	$1.8X10^2$
UPS Rooms	CS3	+.125" wg	104 / 65	60 / 40	.0005	$1.8X10^2$	+.125" wg	104	60	<1.0	$1.8X10^2$
Turbine Building Operating Floor	T2a	Atmos	104	90 / 10	.0025	$8.8X10^2$	-.125" wg	104	90	≤1.0	$8.8X10^2$
Diesel Generator E Building (5)	G	Atmos	104 / 72	90 / 5	.0005	$1.8X10^2$	Atmos	120	50	≤1.0	$1.8X10^2$
Turbine Building General Areas (Shielded)	T1	-.125" wg	104	90 / 10	.0025	$8.8X10^2$	-.125" wg	104	100	≤1.0	$8.8X10^2$
HP Turbine	T2b	-.125" wg	-	-	.5	$1.8X10^4$	-.125" wg	-	-	≤1.0	$1.8X10^3$

TABLE 3.11-1

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

AREA	KEY (3)	PRESSURE	NORMAL OPERATING CONDITIONS					MAXIMUM CONDITIONS WITH NUREG-0588 SOURCE TERM			
			TEMP °F MAX/MIN	RELATIVE HUMIDITY MAX/MIN %	DOSE RATE (R/HR) (4)	INTEGRATED DOSE (RAD)	PRESSURE (2)	TEMP °F (2)	RELATIVE HUMIDITY % (2)	LOCA DOSE RATE (RAD/HR) (6) (7)	TOTAL INTEGR. DOSE (1) (RAD) (6) (7)
LP Turbine	T2c	-.125" wg	-	-	.1	$3.5X10^4$	-.125" wg	-	-	≤1.0	$3.5X10^4$
Feedwater Heaters Condensers	T3	-.125" wg	120	90 / 10	5	$1.8X10^6$	-.125" wg	120	100	≤1.0	$1.8X10^6$
Steam Jet Air Ejectors	T4	-.125" wg	120	90 / 10	15	$5.3X10^6$	-.125" wg	120	100	≤1.0	$5.3X10^6$
Condensate Treatment	T5	-.125" wg	120	90 / 10	10	$3.5X10^6$	-.125" wg	120	100	≤1.0	$3.5X10^6$

(1) Includes integrated accident and normal dose TID for 180 days. After 180 days, the TID is essentially saturated and will not increase significantly.

(2) Pressure, temperature, and humidity maximum are not simultaneous. Above normal pressure, temperature and humidity are considered to persist for 100 days. After 100 days, the thermal environment will be equal to or less than the "maximum" given for normal operation.

(3) Key letter and number identifies a particular group of environmental parameters.

(4) If not otherwise noted, dose is gamma.

(5) For DG rooms: Normal operation means DG in Standby, maximum condition means DG operating.

(6) Maximum Condition Dose rates and TIDs are maximum contact doses in each room, and specific equipment may be subject to a reduced dose based upon the appropriate attenuation factors.

(7) For Beta Sensitive equipment only, the post-accident airborne Beta doses shown in this note must be corrected with appropriate attenuation factors and then added to the tabulated gamma doses to determine total TID.

Area	Max Beta Dose Rate (R/HR)	Beta TID (RAD)
Control Building	2.0	$1.0X10^2$
Turbine Building	20.0	$1.0X10^3$

TABLE 3.11-2

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TABLE 3.11-7

WATER QUALITY

PARAMETER	REACTOR WATER LIMITS SHUTDOWN CONDITION	PRESSURE SUPPRESSION POOL WATER QUALITY EXPECTED	SUPPRESSION POOL WATER MAXIMUM LIMIT
Conductivity	≤ 10 μs/cm at 25°C	≤ 5 μs/cm at 25°C	≤ 10 μs/cm at 25°C
Chlorides (as C1)	≤ 0.5 ppm	≤ 0.2 ppm	≤ 0.5 ppm
pH	5.3 to 8.6 at 25°C	6.0 to 7.5 at 25°C	5.3 to 8.6 at 25°C

TABLE 3.11-8

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Figure 3.11-1 replaced by dwg.
C-1815, Sh. 1

FIGURE 3.11-1, Rev. 56

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Figure 3.11-2 replaced by dwg.
C-1815, Sh. 2

FIGURE 3.11-2, Rev. 51

AutoCAD Figure 3_11_2.doc

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C-1815, Sh. 3

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Figure 3.11-3 replaced by dwg.
C-1815, Sh. 3

FIGURE 3.11-3, Rev. 51

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Figure 3.11-4 replaced by dwg.
C-1815, Sh. 4

FIGURE 3.11-4, Rev. 56

AutoCAD Figure 3_11_4.doc

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Figure 3.11-5 replaced by dwg.
C-1815, Sh. 5

FIGURE 3.11-5, Rev. 56

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Figure 3.11-6 replaced by dwg.
C-1815, Sh. 6

FIGURE 3.11-6, Rev. 56

AutoCAD Figure 3_11_6.doc

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Figure 3.11-7 replaced by dwg.
C-1815, Sh. 7

FIGURE 3.11-7, Rev. 56

AutoCAD Figure 3_11_7.doc

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UNITS 1 & 2
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Figure 3.11-8 replaced by dwg.
C-1815, Sh. 8

FIGURE 3.11-8, Rev. 56

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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Figure 3.11-9 replaced by dwg.
C-1815, Sh. 9

FIGURE 3.11-9, Rev. 56

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C-1815, Sh. 10

Figure 3.11-10 replaced by dwg.
C-1815, Sh. 10

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C-1815, Sh. 11

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Figure 3.11-11 replaced by dwg.
C-1815, Sh. 11

FIGURE 3.11-11, Rev. 56

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C-1815, Sh. 12

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Figure 3.11-12 replaced by dwg.
C-1815, Sh. 12

FIGURE 3.11-12, Rev. 56

AutoCAD Figure 3_11_12.doc

Appendix 3.11A has been deleted.

3.12 SEPARATION CRITERIA FOR SAFETY RELATED MECHANICAL AND ELECTRICAL POWER EQUIPMENT

3.12.1 INTRODUCTION

This section describes the various separation criteria utilized in the design of mechanical and electrical safety related systems and auxiliary support systems; outside of the NSSS scope delineated in Section 7.1

3.12.2 MECHANICAL SYSTEMS

The mechanical safety related auxiliary support and safety related systems to which the separation criteria apply are identified in Table 3.12-1 and Table 3.12-2. Mechanical descriptions of the systems covered by this section are given in Chapters 6 and 9.

3.12.2.1 Criteria

3.12.2.1.1 General Criteria

Redundant systems are separated from each other so that single failure of a component or channel will not interfere with the proper operation of its redundant/diverse counterpart.

The affected mechanical systems and equipment are separated so that systems important to safety are protected from the following hazards:

a) The pipe break dynamic effects outlined in Section 3.6.

b) Environmental effects as a result of pipe breaks and as outlined in Section 3.11

c) Flooding effects as a result of pipe breaks and as outlined in Section 3.4.

d) Missiles as defined in Section 3.5.

e) Fires capable of damaging redundant mechanical safety equipment.

The need for adequacy of separation to protect the safety equipment from the above hazards are determined in conjunction with the criteria specified in Sections: 3.4 (flood protection), 3.5 (missile protection), 3.6 (pipe rupture), 3.11 (environmental design), and the Fire Protection Review Report.

3.12.2.1.2 Svstem Separation Criteria

Piping from a redundant safety system is run independently of its counterparts, unless it can be shown that no single credible event, e.g., LOCA, is capable of causing piping failure that could prevent reactor shutdown. Supports and restraints of redundant mechanical components and piping are not shared, unless such sharing does not significantly impair their ability to perform their safety function.

Penetrations to the primary containment are separated or other adequate provisions are made so that the initial break of one piping branch of a system does not render its redundant counterpart(s) inoperable.

3.12.2.1.3 Phvsical Separation Criteria

Mechanical equipment and piping are separated from each other so that single failure of a device or component will not interfere with the proper operation of its redundant counterpart.

3.12.2.2 Separation Techniques

The methods used to protect redundant Auxiliary Support Systems from the above hazards (Subsection 3.12.2.1.1) fall into four categories of separation techniques: plant arrangement, barriers, spatial separation, and alternatives.

a) Plant Arrangement

A basic design consideration of plant layout is that redundant divisions of a safety system should not share common equipment areas. However, equipment common to a particular safety system division can share a common area if that equipment does not constitute a hazard within itself to another safety system of the same division.

Failure of any non-safety related structure, system, or component shall not result in failure of any safety related structures, system, or component.

To accomplish Auxiliary Support Systems separations through plant arrangement, redundant division of a safety system may be placed in different compartments or even on different elevations. Non-safety equipment, components, or piping should not be run above safety equipment unless they are adequately restrained or it can be demonstrated that failure will not impair function of the safety equipment.

b) Barriers

Barriers are most often used in restricted areas where a particular hazard (e.g., small turbine missiles) is more easily identified or where other techniques are inappropriate (e.g., separation between control boards). Separation by barriers is an extension of separation by the use of compartments in plant arrangement. Separation was also accomplished through the use of suitably designed equipment that in itself acts as a barrier. In many cases, the barrier may enclose the hazard (e.g., a compartment around a high speed turbine driven pump) in lieu of effecting a direct separation between redundant systems.

c) Spatial Separation

Spatial separation is another method of separating redundant safety systems and protecting them from the hazards described in Subsection 3.12.2.1.1.

For example, in areas where a barrier would be impractical, piping has been rerouted so that jet impingement resulting from a break would be dissipated by the distance traveled. In this example, partial barriers or restraints could also be used, as well as by hardening design (e.g., heavier housing construction) of system components within the hazard area. When it can be shown that a hazard would have only a certain sphere of effectiveness (e.g., for pipe whip, a rotation about a plastic hinge at the next restraint), spatial separation was considered adequate.

d) Alternatives

When one of the above techniques is impractical, a suitable alternative was used, some of which were additional restraints, hardening design, or temporary system isolation under accident conditions. When the redundant safety component cannot be held safe from common hazards by the alternatives outlined above, more resistant components were selected. An example would be the use of high pressure piping in a low pressure safety system to ensure its ability to withstand the effect of a break in adjacent high pressure lines.

3.12.3 ELECTRICAL SYSTEMS AND EQUIPMENT SEPARATION CRITERIA

Electrical and actuation systems are described in detail in Chapters 8 and 7, respectively.

3.12.3.1 Affected Svstems

The electrical portions of the systems identified in Tables 3.12-1 and 3.12-2 are designed to the criteria of Subsection 3.12.3.2. Equipment covered by the requirements in this Subsection include: instrument channels, trip systems, and trip actuators.

3.12.3.2 General Criteria

The resulting installations satisfy the criteria of IEEE 279-1971, 10CFR50 Appendix A, General Design Criteria 3, 17, and 21, as further clarified and limited below. The affected electrical systems and equipment are separated such that systems important to safety are protected from the following hazards:

a) Fires in cable raceways due to an electrical fault that could cause failure of insulation on other cables.

b) Mechanical damage of electrical equipment in a single location.

c) Single Design Base Event (DBE) should not disable essential automatic or manual protective function, i.e., reactor scram, primary containment isolation, core cooling, etc.

Identification

Identification and division/channels conform to the following:

a) Panels and racks, not part of the PGCC, are labeled with distinctive marker plates. The marker plates include identification of the proper division/channel, as listed in Table 3.12-1.

b) Junction and/or pull boxes, not part of the PGCC, have identification similar to and compatible with the panels and racks considered above.

c) Cables external to cabinets and/or panels, not part of the PGCC, are marked to distinguish them in color from other cables and to identify their separation division/channel as applicable.

d) Raceways, not part of the PGCC, identified as described in Subsection 3.12.3.4.2.1.b.

e) For PGCC panels and racks refer to Section 7.1.2a.3.

f) For cables external to panels and racks but within the PGCC, refer to Section 7.1.2a.3.

3.12.3.3 Svstem Separation Criteria

See Section 7.1.2a.3.

3.12.3.4 Electrical Phvsical Separation Criteria

3.12.3.4.1 General Separation Criteria

Methods of Separation

The separation of circuits and equipment is achieved by separate safety class structures, distance, or barriers, or any combination thereof.

Compatibilitv with Mechanical Svstems

Class 1E circuits are routed and/or protected such that failure of related mechanical equipment of one redundant system cannot disable Class 1E circuits or equipment essential to the operation of the other redundant system(s).

Racewav Sharing of Class 1E and non-Class 1E Circuits

See Subsection 8.1.6.1.

3.12.3.4.2 Specific Separation Criteria

3.12.3.4.2.1 Cables and Racewavs

a) General

The minimum separation distances specified in paragraphs d) and e) are based on open ventilated trays. Where these distances are used to provide adequate physical separation:

1) Cable splices in raceways are prohibited
2) Cables and raceways involved are flame retardant
3) The design basis is that the cable trays will not be filled above the side rails
4) Hazards will be limited to failures or faults internal to the electric equipment or cables.

In areas where the raceway separation criteria cannot be met due to physical limitation, each circuit in the raceway is to be analyzed to assure that the Class 1E function is not degraded to an unacceptable level. The specific analysis of each case is documented by a controlled document.

b) Identification of Non-PGCC Cables and Racewavs

Exposed Class 1E raceways are identified in a distinct and permanent manner at intervals not to exceed 15 ft. In addition, these raceways are also identified where

they pass through walls and/or floors, and enclosed areas. Class 1E raceways are identified prior to the installation of their cables.

Cables installed in these raceways are identified with the separation group color at intervals not exceeding 5 ft. to facilitate initial verification that the installation conforms to the separation criteria. These cable identifications are applied prior to or during their installation.

Class 1E cables are identified by a permanent marker at each end in accordance with the design drawings or cable schedule.

Color coding is used to meet the above requirements and to distinguish between redundant Class 1E cables and non-Class 1E cables.

c) Identification of PGCC Cables and Racewavs

Refer to Subsection 7.1.2a.3.2.

d) Cable Spreading Areas/Control Structure Complex

The control structure complex consists of two elevations of relay rooms, two cable spreading areas, and the main control room. Below the main control room is the lower cable spreading room, which facilitates cable convergence from the computer room and the lower relay rooms (which are below the lower cable spreading room) to the general plant areas, and to the cable entrance areas at the bottom of the control room panels. The lower relay rooms consist mainly of control and instrument panels of non-Class 1E systems and one division (i.e., Division II) of redundant systems as listed in Subsection 3.12.3.1. The main control room panels are mounted on a raised floor assembly with cable trays and wireways that enter the bottom of the main control room panels. Above the main control room are the upper relay rooms and the upper cable spreading area. The upper cable spreading area facilitates cable convergence from the upper relay room to the general plant areas, to the top of the main control room panels and to the control room raised floor. The upper relay room consists mainly of control and instrument panels of non-Class 1E systems and the other division (i.e., Division I) of the redundant systems listed in Subsection 3.12.3.1. The relay room panels and cabinets are integrated with a module type floor assembly with lateral and longitudinal ducts that act as raceways and barriers. The cabling interface between the PGCC and the spreading area is made at termination cabinets on the periphery of the relay room floor assemblies.

The relay rooms and spreading room areas do not contain high energy equipment (such as switchgear, transformers) or potential sources of missiles or pipe whip and are not used for storing flammable materials.

Circuits in the relay room and main control room are limited to control functions, instrument functions, and those power supply circuits and facilities serving the main control room and instrument systems.

Where for operational reasons redundant channel/division Class 1E cables are not separated by different safety class structures (e.g., two relay rooms and spreading areas), the minimum separation distance between the redundant Class 1E cable trays is 1 ft. horizontally and 3 ft. vertically. Where 1 ft. horizontal separation is not possible, one of the two following requirements is met: a fire barrier is placed between the redundant cable trays 1 ft. above the trays or to the ceiling; or cables of each channel/division are installed in rigid steel conduit or totally enclosed raceway up to a point where the 1 ft. spacing requirement is met. Where cables of redundant channel/divisions must be stacked one above the other with less than 3 ft. vertical spacing, one of the following requirement is met: a) a fire barrier is placed between the trays and extended to 6 in. of each side of the tray system or to the wall, or b) a solid steel tray cover is installed on the lower cable tray and the upper tray has a solid bottom up to a point where 3 ft. vertical separation is met; or c) the cables of each redundant channel/division are installed in rigid steel conduit or totally enclosed raceway to a point where the 3 ft. vertical separation exists. The minimum separation distance between these rigid steel conduit and totally enclosed raceway is 1 inch, except as noted in Section 8.1.6.1 (Regulatory Guide 1.75 (1/75), Part 7).

Separation requirements between Class 1E trays and non-Class 1E trays are the same as separation of redundant channel/division, except that the minimum separation distance between Class 1E tray and non-Class 1E conduit or totally enclosed raceway is 1 inch.

Free air temporary cables can be installed with no separation distance from totally enclosed Class 1E raceways. Temporary cables are non-Class 1E and have a specified removal date or removal event. Tests have demonstrated the acceptability of a single solid metal cable cover as a barrier when the worst case electrical fault occurs to a cable resting on the metal cable tray cover. The cables inside the cable tray maintained their functional capability during the testing.

Permanent free air telephone (PABX) cables can be installed with a separation distance of one-foot horizontal and three-foot vertical from Class 1E ventilated cable tray. Physical separation between the permanent free air telephone (PABX) cables and Class 1E enclosed raceway shall be 6 inches. Tests have demonstrated the acceptability of a single solid metal conduit as a barrier when the worst case electrical fault occurs to a cable resting on the barrier. The cables inside the conduit maintained their functional capability during the test[jcb1]

In confined spaces, where fire barriers cannot be installed, lesser separation distance between open trays than those specified above shall be allowed in the following areas that are protected by ionization type detectors with total flooding or

manual spurt CO_2 suppression systems. (Refer to Fire Protection Review Report, Table 6.1-1, "SSES Fire Areas" for room numbers and fire zones.) Examples of confined spaces are:

1) Cable chases/soffits

2) Raised floor section

e) General Plant Areas

In plant areas from which potential hazards such as missiles, external fires, and pipe whip are excluded, the minimum separation distance between redundant Class 1E cable trays is 3 ft. between trays separated horizontally if no physical barrier exists between trays. If a horizontal separation of less than 3 ft. exists, alternate methods as stated in paragraph d) above are required. Vertical stacking of trays is avoided wherever possible; however, where cable trays of redundant channel/divisions are stacked, a minimum vertical separation distance of 5 ft. is required, or alternate methods as stated in paragraph d) above are required. Where a cross-over of one tray over another carrying redundant channel/division is made, and minimum vertical separation distance cannot be maintained, one of the following requirements is met; a) a solid cover is installed on the lower tray to extend 1 ft. 0 in. minimum either side of the upper tray, b) fire barriers are installed minimum 1 in. from the upper tray and extend 1 ft. 0 in. minimum beyond the crossing tray.

Separation requirements between Class 1E and non-Class 1E trays are the same as separation of redundant channel/division, except that the minimum separation distance between Class 1E tray and non-Class 1E conduit or totally enclosed raceway is 1 inch.

Free air temporary cables can be installed with no separation distance from totally enclosed Class 1E raceways. Temporary cables are non-Class 1E and have a specified removal date or removal event. Tests have demonstrated the acceptability of a single solid metal cable cover as a barrier when the worst case electrical fault occurs to a cable resting on the metal cable tray cover. The cables inside the cable tray maintained their functional capability during the testing.

Permanent free air telephone (PABX) cables can be installed with a separation distance of one-foot horizontal and three-foot vertical from Class 1E ventilated cable tray. Physical separation between the permanent free air telephone (PABX) cables and Class 1E enclosed raceway shall be 6 inches. Tests have demonstrated the acceptability of one-foot horizontal and three-foot vertical distances to prevent migration of electrical faults from the low energy free air telephone cables to the Class 1E cables. Tests also have demonstrated the acceptability of a single solid metal conduit as a barrier when the worst case

electrical fault occurs to a cable resting on the barrier. The cables inside the conduit maintained their functional capability during the test[jcb2].

f) Power Generation Control Complex - (PGCC)

Refer to Subsection 7.1.2a.3.3.6.

g) The Lighting Fixture Cords

The non-Class 1E lighting fixture cord connects a lighting fixture to a single phase, 277V power supply. The cord is a #14 AWG SO insulated cable (with grounding conductor) installed in free air carrying a maximum load current of 1.0 ampere. The minimum separation between the free air lighting fixture cord and a Class 1E open tray is 6 inches. If the above minimum separation cannot be satisfied, tray covers or conduits will be provided for the Class 1E raceway in the vicinity of the free air lighting fixture cord. The lighting fixture cord does not have sufficient combustible material or energy which could cause failure of the nearby Class 1E cables.

Non-Class 1E lighting conduits, containing a single circuit rated less than 300 VAC and 10 amp or 20 amp for receptacles, are treated as control conduits for separation purposes. The separation criteria is as stated in Section 8.1.6.1 (Regulatory Guide 1.75(1/75), Part 7). The lighting fixture cords, which are exposed in free air containing #12 AWG wires and about 3 to 5 feet in length, shall be maintained at 6" minimum separation from a Class 1E raceway.

h) An exception to the above subsections d) and e) is the 450 MHz radio antenna cable network.

The 450 MHz radio antenna is the plant security communication radio system (non-Class 1E). This system utilizes an antenna cable network installed exposed (not enclosed in raceway) on the cable raceway supports throughout the plant. The jacketing material of the antenna cable is flame retardant. The cable has been tested and passed IEEE 383 and ASTM Proc. D2633 part 30.

Separation between this antenna cable and other class 1E raceways is not required because:

1) The antenna cable is a low energy circuit. A short circuit of the antenna cable would not produce enough energy to cause degradation of any other circuits.

2) The antenna cable is not routed with any other cables.

3) The antenna cable jacket is made of flame retardant material.

4) The antenna cable does not terminate in close proximity or routed through any equipment with voltage level higher than 120V AC.

5) The maximum radio frequency (rf) power output level of the antenna cable is 37.5 watts.

6) Where redundant safe shutdown raceways are separated by less than 20 feet, fire barriers have been provided to protect one division per Fire Protection Review Report Section 4.11.

3.12.3.4.2.2 Standbv Power Supply

a) Emergencv Diesel Generators

Redundant Class 1E diesel generator units are located in separate safety class structures and have independent air and fuel supplies.

b) Auxiliaries and Local Controls

The auxiliaries and local controls for diesel generators are in the same safety class structure as the unit they serve, except for the Diesel Generator A, B, C and D fuel oil transfer pumps that are located in separate safety class structures at the fuel oil storage tanks (see Subsection 9.5.4).

3.12.3.4.2.3 DC Svstem

a) Batteries

Redundant Class 1E batteries are placed in separate safety class structures. The structures are served by redundant ventilation equipment.

b) Battery Chargers

Battery chargers and their respective switchgears are placed in separate safety class structures from their respective redundant Class 1E batteries.

3.12.3.4.2.4 Distribution Svstem

a) Switchgear

Redundant Class 1E distribution switchgear groups are placed in separate safety class structures.

b) Motor Control Centers

Redundant Class 1E motor control centers are physically separated in accordance with the requirements of Subsection 3.12.3.4.1.

c) Distribution Panels

Redundant Class 1E distribution panels are physically separated in accordance with the requirements of Subsection 3.12.3.4.1.

3.12.3.4.2.5 Primary Containment Electrical Penetrations

Redundant Class 1E primary containment electrical penetrations are physically separated in accordance with the requirements of Subsection 3.12.3.4.1. The minimum physical separation for redundant penetrations meets the requirements for cables and raceways given in Subsections 3.12.3.4.2.1 through 3.12.3.4.2.6.

3.12.3.4.2.6 Main Control Room and Relay Room Panels

a) For NSSS panels see Subsection 7.1.2a.3.1.1.

b) All non-NSSS panels containing safety-related equipment and circuits are provided as follows:

1) Generally, panels are divisionalized (i.e., are devoted to one (1) division only) and are physically separated from the redundant division's panels.

2) In cases where redundant channel/division Class 1E circuits, or RPS and other Class 1E and non-Class 1E circuits are located in the same enclosure, physical separation is achieved by minimum of 6" spatial separation, steel barriers, metallic enclosure, or metallic flexible conduit.

   Where the above separation methods are not feasible, one of the separation group circuits are to be covered with a qualified non-metallic barrier material. A description of the material and analysis to regulatory requirements is provided in Subsection 8.1.6.t.14 (Conformance to Reg. Guide 1.75).

3) All requirements for connection of control circuits between separated divisions are accomplished with MDR relays to provide positive isolation of the circuits.

4) All the annunciator and computer digital inputs are classified as non-Class 1E circuits. These circuits are not separated from the Class 1E circuits within the Class 1E panels in which the non-Class 1E input is

derived. The interface devices used in the Class 1E circuits to develop the annunciator and computer digital inputs are listed in Table 3.12-3. An analysis for each circuit in which these devices are used has shown that a failure mode which prevents the Class 1E circuits from meeting their minimum performance requirements does not exist. This is based upon the application/function of the interface devices in each individual circuit.

TABLE 3.12-1

ESP DIVISION SEPARATION

Division I	Division II
Core Spray Loop A	Core Spray Loop B
Automatic Depressurization System A	Automatic Depressurization System B
Residual Heat Removal Loop A	Residual Heat Removal Loop B
High Pressure Coolant Injection System (Inboard Valve)	High Pressure Coolant Injection System except the inboard steam line isolation valve
Reactor Core Isolation Cooling System except the inboard steam line isolation valve	Reactor Core Isolation Cooling System (Inboard Valves)
Nuclear Steam Supply Shutoff System (Inboard Valves)	Nuclear System Supply Shutoff System (Outboard Valves)
Recirculation Pump Trip Loop A	Recirculation Pump Trip Loop B
Emergency Service Water Loop A	Emergency Service Water Loop B
RHR Service Water Loop A	RHR Service Water Loop B
Containment Instrument Gas Loop A	Containment Instrument Gas Loop B
Containment Atmospheric Control System A	Containment Atmospheric Control System B
Standby Gas Treatment System Train A	Standby Gas Treatment System Train B
Reactor Building HVAC Isolation and Recirculation System A	Reactor Building HVAC Isolation and Recirculation System B

TABLE 3.12-1

ESP DIVISION SEPARATION

Division I	Division II
Drywell HVAC System A	Drywell HVAC System B
Control Structure HVAC System Train A	Control Structure HVAC System Train B
Control Structure Chilled Water System Loop A	Control Structure Chilled Water System Loop B
Battery Room Ventilation System A	Battery Room Ventilation System B
HVAC Coolers for Div I	HVAC Coolers for Div II
Standby liquid Control System Pumps A(1) and B(1) and Explosive Valves A(1) and B(1)	
Class 1E 250V DC Supply System I	Class 1E 250V DC Supply System II
480V Swing Bus and Associated Motor-Generator Set Div I	480V Swing Bus and Associated Motor-Generator Set Div II
Class 1E 480V AC MCCs	Class 1E 480V AC MCCs
Class 1E 120V AC Distribution Panels	Class 1E 120V AC Distribution Panels
Class 1E 125V DC Distribution Panels	Class 1E 125V DC Distribution Panels

(1)The redundant standby liquid control pumps and explosive valves are powered from different electrical buses.

TABlE 3.12-2

CHANNEl SEPARATION

Channel A	Channel B	Channel C	Channel D
Standby Diesel Generator & Auxiliaries A (Common to Units 1 and 2)	Standby Diesel Generator & Auxiliaries B (Common to Units 1 and 2)	Standby Diesel Generator & Auxiliaries C (Common to Units 1 and 2)	Standby Diesel Generator & Auxiliaries D (Common to Units 1 and 2)
Standby Diesel A Ventilation System (Common to Units 1 and 2)	Standby Diesel B Ventilation System (Common to Units 1 and 2)	Standby Diesel C Ventilation System (Common to Units 1 and 2)	Standby Diesel D Ventilation System (Common to Units 1 and 2)
Class 1E 4160 V Switchgear	Class 1E 4160 V Switchgear	Class 1E 4160 V Switchgear	Class 1E 4160 V Switchgear
Class 1E 480 V Load Center	Class 1E 480 V Load Center	Class 1E 480 V Load Center	Class 1E 480 V Load Center
Class 1E 480 V MCC (Common to Units 1 and 2)	Class 1E 480 V MCC (Common to Units 1 and 2)	Class 1E 480 V MCC (Common to Units 1 and 2)	Class 1E 480 V MCC (Common to Units 1 and 2)
Class 1E 125 V Distribution Panel	Class 1E 125 V Distribution Panel	Class 1E 125 V Distribution Panel	Class 1E 125 V Distribution Panel

Note:

Additionally, a fifth diesel generator is provided which can be manually realigned as a replacement for any one of the other four diesel generators. This fifth diesel generator has its own ventilation and electrical support systems.

When this fifth diesel generator is substituted for any one of the other four diesel generators, the fifth diesel generator and it's auxiliaries assimilate the separation channel of the diesel generator which was substituted.

TABLE 3.12-3

Main Control Room and Relay Panel Annunciator and Computer Interface Device

ANNUNCIATOR INTERFACE DEVICES	
Agastat Type EGP	GE Type HFA
Riley 86 T/C Monitor	P&B Type KH-4690
Westronics Recorder	Agastat Type E7000
GE Type CR2940 SW	GE Type 2820
Bailey 745 Alarm	P&B Type MDR
GE Type CR105	Agastat Type TR
C-H Type 10250T PB	GE Type HMA
COMPUTER INTERFACE DEVICES	
GE Type CR 105	Agastat Type E7024
GE Type HFA	P&B Type KH-4690
GE Type HMA	GE IRM Switch 216X494G19
Agastat Type EGP	

3.13 COMPLIANCE WITH NRC REGULATORY GUIDES

This section discusses the compliance of the plant design with the guidelines presented in the NRC Regulatory Guides. Where applicable, reference is made to the Final Safety Analysis Report (FSAR) section(s) in which the appropriate design feature is described.

Since the application for the construction permit for this station was docketed in March 1971; therefore, it should be noted that the implementation paragraphs of many of the Regulatory Guides render the provisions contained therein inapplicable to Susquehanna SES by virtue of their effective dates. Nonetheless, the Applicant has evaluated the design and construction against versions of the Regulatory Guides which were current when the application for an Operating License was tendered and has complied with the listed revisions to the extent practicable.

Where compliance to the regulatory guide has been qualified by an interpretation of the regulatory guide, these variances are discussed in either this section or in an appropriate section referenced in a particular response.

The use of an Alternative Source Term (AST) requires changes to source term assumptions and dose acceptance criteria. Regulatory Guides have not been reviewed in detail to determine if exceptions are required for the below listed items. This note captures that the following criteria may be applicable exception(s) to a specific Regulatory Guide.

a. New AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the following accidents: Loss of Coolant Accident, Main Steam Line Break, the Refueling Accident and the Control Rod Drop Accident.

b. Dose acceptance criteria is based on the Total Effective Dose Equivalent (TEDE) versus thyroid, whole body and beta dose.

c. Changed from 10CFR100.11 to 10CFR50.67 for dose acceptance criteria.

d. Changed from 10CFR50, Appendix A, General Design Criteria 19 to 10CFR50.67 for control room personnel dose acceptance criteria.

e. No longer committed to Regulatory Guides 1.3, 1.5 and 1.25.

3.13.1 DIVISION 1, REGULATORY GUIDES - POWER REACTORS

Regulatory Guide 1.1 - NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS (November 2, 1970)

As discussed in Subsections 6.2.2 and 6.3.2, Susquehanna SES has been designed to comply with this regulatory guide.

Regulatory Guide 1.2 - THERMAL SHOCK TO REACTOR PRESSURE VESSELS (November 2, 1970)

With respect to the regulatory positions of this regulatory guide, Susquehanna SES is in compliance as follows:

The reactor pressure vessel utilized for Susquehanna SES employs no significant core or vessel design changes from previously approved BWR pressure vessels such as Browns Ferry.

NOTE: Although this regulatory guide has been withdrawn, any prior or existing commitments based on its use are not altered.

An investigation of the structural integrity of boiling water reactor pressure vessels during a design basis accident (DBA) has been conducted (refer to NEDO-10029). It has been determined, based upon methods of fracture mechanics, that no failure of the vessel by brittle fracture as a result of a DBA will occur.

The investigation included:

a. a comprehensive thermal analysis considering the effect of blowdown and the low-pressure coolant injection (LPCI) system reflooding;
b. a stress analysis considering the effects of pressure, temperature, seismic load, jet load, dead weight, and residual stresses;
c. the radiation effect on material toughness (NDTT shift and critical stress intensity); and
d. methods for calculating crack tip stress intensity associated with a non-uniform stress field following the design basis accident.

This analysis incorporated very conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis, effects of radiation on material toughness, and crack tip stress intensity). Therefore, the results reported in NEDO-10029 provide an upper bound limit on brittle fracture failure mode studies. Because of the upper bound approach, it is concluded that catastrophic failure of the pressure vessel due to the DBA is shown to be impossible from a fracture mechanics point of view. In the case studied, even if an acute flaw does form on the vessel inner wall, it will not propagate as the result of the DBA.

For further discussion of fracture toughness of the Reactor Pressure Vessel refer to Subsection 5.2.3.3.1.

Regulatory Guide 1.3 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS (Revision 2, June 1974)

Not Applicable.

Regulatory Guide 1.4 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS (Revision 2 June, 1974)

Not Applicable.

Regulatory Guide 1.5 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS (March 10, 1971)

Not Applicable.

Regulatory Guide 1.6 - INDEPENDENCE BETWEEN REDUNDANT - STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS (March 10, 1971)

As discussed in Subsections 7.1.2.6, 8.1-6.1.a, and 8.3.1.4, independence between redundant standby (onsite) power sources and between their distribution systems is provided with the exception of Position D.4.c. Swing buses Supply power to the LPCI injection valves and the recirculation piping isolation and bypass valves. Motor generator sets are used to protect redundant power sources from faults that might develop on the swing bus, thus ensuring the requisite degree of independence between the redundant power sources.

Regulatory Guide 1.7 - CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT (Revision 3, March 2007)

The design guidance and assumptions of this regulatory guide are followed as discussed in Subsection 6.2.5.

Regulatory Guide 1.8 - PERSONNEL SELECTION AND TRAINING (Revision 2, April 1987)

Commitment to this regulatory guide is described in FSAR Table 17.2-1 and Technical Specification 5.3.1. Additional information relating to personnel selection and training can be found in FSAR Sections 12.5, 13.1 and 13.2.

Regulatory Guide 1.9 - SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES (March 10, 1971, For Diesel Generators ‘A-D’ and December 1979 for Diesel Generator ‘E’)

The standby power system is discussed fully in Subsections 8.1.6.1.b and 8.3.1, AC Power Systems. Standby diesel generator power supplies comply with Regulatory Guide 1.9.

Except as indicated below,

(1) Reference: Position C.4 Power quality is in accordance with IEEE 387-1972, Section 5.1.2(4) for Diesel Generators A-D and IEEE 387-1977, Section 5.1.2(5) for Diesel Generator E. At no time during the loading sequence will the frequency or voltage drop to a level which will degrade the performance of any of the loads below their minimum requirements.

(2) Reference: Position C.5. The suitability of each Susquehanna SES Diesel Generator is confirmed by factory qualification testing and preoperational test. Discussion of the factory test results is in Section 8.3.

Regulatory Guide 1.10 - MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURES (Revision 1, January 2, 1973)

The testing and inspection program for all mechanical (Cadweld) splices in reinforcing bars of Category I structures are in compliance with this regulatory guide.

Regulatory Guide 1.11 - INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT (March 10, 1971)

The design of the instrument lines penetrating the primary reactor containments of the Susquehanna SES complies with the provisions of this Regulatory Guide. Instrument lines which directly communicate with containment atmosphere, or do not communicate with reactor coolant pressure boundary are treated as extensions of the containment.

Regulatory Guide 1.12 - INSTRUMENTATION FOR EARTHQUAKES (Revision 1 April, 1974)

As described in Subsection 3.7b.4, seismic instrumentation is provided. The instrumentation meets the regulatory position set forth in this guide except for Section C.1.a. since no triaxial peak accelerographs are provided.

Regulatory Guide 1.13 - SPENT FUEL STORAGE FACILITY DESIGN BASIS (Revision 1, December 1975)

The fuel storage facility design basis is described in Section 9.1 and Appendix 9A. Regulatory positions are complied with subject to the following exceptions and clarifications:

(1) Reference: Position C.2. The fuel pool is designed to prevent significant loss of watertight integrity caused by tornadic winds and missiles generated by these winds. The reactor building above the refueling floor consists of steel framing with metal siding.

(2) Reference: Positions C.3 and C.5.a. Interlocks are provided to prevent the 125 ton hook of the reactor building crane from passing over or near stored fuel. These interlocks preclude any load suspended from this crane from tipping over on the stored fuel in the event of a crane failure. The 5 ton auxiliary hook suspended from the same crane trolley is prevented from passing over stored fuel when fuel handling is not in progress by administrative controls. There are no planned transfers of loads heavier than a new fuel element over the stored fuel.

(3) Reference: Position C.8. A Seismic Category I makeup water supply from each emergency service water loop is permanently connected to each spent fuel pool by two independent Seismic Category I piping routes. The make-up is provided for filling the Spent Fuel Pool to the proper level to support operation of the RHR Fuel Pool Cooling mode, and to provide for make-up from evaporative losses during cooling by RHR. The make-up rate is sized based on boiling so as to be conservative. The normal makeup system to the fuel pool is not Seismic Category I.

Regulatory Guide 1.14 - REACTOR COOLANT PUMP FLYWHEEL INTEGRITY (Revision 1, August 1975)

Not Applicable.

Regulatory Guide 1.15 - TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES (Revision 1, December 28, 1972)

Testing of reinforcing bars for Category I concrete structures is in compliance with this regulatory guide.

Regulatory Guide 1.16 - REPORTING OF OPERATING INFORMATION-APPENDIX A TECHNICAL SPECIFICATIONS (Revision 4, August 1975)

In lieu of the positions stated in this Regulatory Guide, the reporting of operating information for the Susquehanna SES complies with Technical Specifications, 10CFR50.73 and G.L. 97-02.

Regulatory Guide 1.17 - PROTECTION OF NUCLEAR POWER PLANTS AGAINST INDUSTRIAL SABOTAGE (June 1973)

In lieu of the positions stated in this regulatory guide, the protection of Susquehanna SES against industrial sabotage complies with 10CFR73.

Regulatory Guide 1.18 - STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS (Revision 1, December 28, 1972)

The compliance with this regulatory guide is achieved subject to certain test modifications as discussed in Subsection 3.8.1.7.1.1.

Regulatory Guide 1.19 - NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS (Revision 1, August 11, 1972)

Nondestructive examination of the primary containment liner welds is conducted as discussed in Subsection 3.8.1.

Regulatory Guide 1.20 - COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING (Revision 2, May 1976)

The vibration assessment program for reactor internals as discussed in Subsections 1.5.1, 3.9.2.4 and NEDE 24057P complies with this regulatory guide.

Regulatory Guide 1.21 - MEASURING, EVALUATING AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (Revision 1, June 1974)

Operation of the radwaste systems will be conducted in accordance with this regulatory guide as permitted by the design. Operation and design of the systems are discussed in Section 11.5.

Regulatory Guide 1.22 - PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTION (February 17, 1972)

As discussed in Subsections 7.1.2.6.4, 7.2.2.1.2.1.2, 7.2.4.1.1.2.2.1, 7.3.2a.1.2.1.3, 7.3.2a.2.2.1.2, 7.3.2a.5.2.1, 7.4.2.1.2.1.3, 7.4.2.2.2.1.3, 7.6.2a.3.2.3.1, 7.6.2a.5.4.2, 8.1.6.1d and 8.3.1.3.1.5 periodic testing of protection system actuation functions complies with this regulatory guide.

Regulatory Guide 1.23 - METEOROLOGICAL MEASUREMENT PROGRAM FOR NUCLEAR POWER PLANTS Second Proposed Revision 1, (April, 1986)

The onsite meteorological system and program comply with this regulatory guide.

The commitment to accuracy criteria for delta temperature is Atomic Energy Commission Safety Guide 23, February 1972 (Regulatory Guide 1.23 Rev. 0).

Regulatory Guide 1.24 - ASSUMPTIONS USED FOR EVALUATION THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER GAS STORAGE TANK FAILURE (March 23, 1972)

Not Applicable.

Regulatory Guide 1.25 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL-HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS (March 23, 1972)

Not Applicable.

Regulatory Guide 1.26 - QUALITY GROUP CLASSIFICATION AND STANDARDS (Revision 3, February 1976)

In general the requirements of Regulatory Guide 1.26 are met for the Susquehanna Plant. Some exceptions exist due to the purchase date of the NSSS equipment and design changes in a few systems. These exceptions have been documented in correspondence with NRC. Quality Group classifications are detailed in Tables 3.2-1, 3.2-2, and SSES-FSAR 3.2-3.

Regulatory Guide 1.27 - ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS (Revision 2, January 1976)

Subject to the exception indicated below, the design of the ultimate heat sink satisfies the requirements of this regulatory guide.

(1) Reference: Position C.2. Position C.2 states that the ultimate heat sink features which are not required to be designed to withstand the Safe Shutdown Earthquake (SSE) should, nonetheless, be designed and constructed to withstand the Operating Basis Earthquake and waterflow based on severe historical events in the region. The requirements of this regulatory guide without the necessity for any makeup operations following the occurrence of an SSE. Therefore, the potential makeup water sources, such as the cooling tower basins and the makeup system from the Susquehanna River, are not required to perform any safety-related function following the occurrence of any seismic event, and design criteria specified in Position C.2 have not been employed in their design.

Compliance is discussed in Subsection 9.2.7. A description of the analysis performed to demonstrate the ability of the ultimate heat sink to meet the requirements of this Regulatory Guide is presented in Subsection 9.2.7.3.

Regulatory Guide 1.28 - QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION) (Revision 1, March 1978)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is discussed in Section 17.2.

Regulatory Guide 1.29 - SEISMIC DESIGN CLASSIFICATION (Revision 2, February 1976) (Revision 3, September 1978 for the Diesel Generator 'E' Facility)

Subject to the clarifications and/or exceptions indicated below, Susquehanna SES complies with this regulatory guide.

(1) Reference: Position C.1.b. For the NSSS, application of this guide is limited to the reactor core and reactor internals which are engineered safety features.

(2) Reference: Position C.1.d and C.1.g. The normal spent fuel pool cooling system is non-seismic Category I. If a seismic event would occur, cooling of the spent fuel is achieved by use of the RHR Fuel Pool Cooling (RHRFPC) mode as described in Sections 5.4.7.1.1.6, 5.4.7.2.6c, 9.1.3.1, and 9.1.3.3. Either or both of two Seismic Category I ESW makeup water supplies to each pool can provide make-up in support of the RHRFPC mode. Additionally, ESW is capable of supplying make-up for the boiling Spent Fuel Pool (SFP) analysis as described in Appendix 9A.

(3) Reference: Position C.1.e. The Main Steam System (MSS) beyond the outer isolation valves up to and including the turbine stop valves and all branch lines 2 1/2 in. in diameter and larger, up to and including the first valve (including their restraints) are not classified Seismic Category I; because portions of the pipe are routed in a non-Seismic Category I building (the Turbine Building). However, the turbine building has been designed to withstand an SSE as stated in Subsection 3.7b.2.8. Further description of the turbine building is given in Subsection 3.8.4.1; applicable load combinations are given in Table 3.8-10. The subject piping is designed in accordance with ASME Section III, Class 2 requirements for the OBE and SSE as described in Subsection 10.3.3.

(4) Reference: Position C.1.h. The component cooling water portions of the reactor recirculation pumps are not Seismic Class I since they do not involve a safety function.

(5) Reference: Paragraph C.2 of the Regulatory Guide. Items which would otherwise be classified non-Seismic Category I, "but whose failure could reduce the functioning" of items important to safety "to an unacceptable safety level" are to be "designed and constructed so that the SSE would not cause such failure." In addition, Paragraph C.4 of the guide requires that the "pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both of these positions are considered to be adequately met by applying the following practices to such items:

    (a) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

(b) Field work is performed under the direction of experienced field construction superintendents and is inspected by the staff of field engineers stationed at the site. The field engineers are responsible for verifying that construction is performed in accordance with the design drawings and specifications and with applicable standard codes and specifications.

(6) Reference: Paragraph C.3 of the Regulatory Guide. Seismic Category I design requirements are required to be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system requires division of the systems into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system which can be treated as an anchor to the plant structure. In addition, Paragraph C.4 of the Regulatory Guide takes the position that "the pertinent quality assurance requirement of Appendix B to 10CFR Part 50 should be applied to the safety requirements" of such items. Both these requirements are considered to be met adequately by applying the following practices to such items:

(a) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

(b) Field audits are performed by representatives of the originating design group to assure that the final installation of such items is in accordance with documents that formed the basis for the seismic analysis of the items.

Regulatory Guide 1.30 - QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT (August 11, 1972)

The Susquehanna SES quality program for construction of safety related items was conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Program with this guide is described in Table 17.2-1.

Regulatory Guide 1.31 - CONTROL OF STAINLESS STEEL WELDING

Control of stainless steel welding (except NSSS scope of supply) complies with Interim Position on Regulatory Guide 1.31 (Branch Technical Position MTEB 5-1 dated November 24, 1975) except as discussed below.

(1) Reference: Paragraph B.1.b of Interim Position. Austenitic stainless steel welding filler materials used in the fabrication and installation of ASME Section III, Class 1, 2, and 3 components are controlled to deposit from 8 to 25 percent delta ferrite. Welding filler materials 309 and 309L are controlled to deposit from 5 to 15 percent delta ferrite and are used only for welding carbon or low alloy steel to austenitic stainless steel. The use of 309L welding filler material is further limited to the overlay deposit on the carbon or low alloy steel component nozzles or connecting pipe when postweld heat treatment is required.

These limits for delta ferrite in austenitic stainless steel welding materials comply with Interim Regulatory Guide 1.31 because the upper limit of 20 percent delta ferrite does not apply to welds that are not heat treated after welding (Paragraph 3b). Solution heat treatment, although not required after welding, is permitted to avoid sensitization.

The procedure for determining the amount of delta ferrite in each heat or lot of austenitic stainless steel welding material does not comply with the Interim Position of the Regulatory Guide. Determination of delta ferrite is in accordance with ASME Section III, Division 1, 1974 Edition,

Paragraph NB02433, except that an undiluted weld deposit is required for each heat of bare wire used with the Gas Metal-Arc process.

(2) Reference: Paragraph B.2 of the Interim Position. This paragraph is complied with for all tests and examinations required by ASME Section III, Division 1, 1974 Edition.

(3) Reference: Paragraph B.3.a of the Interim Position. This paragraph is not complied with. Magnetic measurement of production welds for delta ferrite is unnecessary when austenitic stainless steel welding materials are controlled to deposit 8 to 25 percent delta ferrite based on chemistry, except for 309 and 309L welding materials which are controlled to deposit 5 to 15 percent delta ferrite based on chemistry.

Three Bechtel projects are committed to measuring production welds for delta ferrite in order to collect data and demonstrate that the welding material controls described above are more than adequate for the purpose of avoiding microfissuring. Since this represents a sufficient number of welds for the purpose of collecting data, measurement of production welds for delta ferrite on this project is not planned.

(4) Reference: Paragraph B.3.b of the Interim Position. This paragraph is complied with for welding material certification.

(5) Reference: Paragraphs B.4.a, b, and c of the Interim Position. These paragraphs are not complied with since measurement of production welds for delta ferrite is not performed.

The NSSS scope of supply control of welding is described in Subsection 5.2.3.4.2.

Regulatory Guide 1.31 - CONTROL OF FERRITE CONTENT IN (DIESEL GENERATOR E ONLY) STAINLESS STEEL WELD METAL (Revision 3, April 1978)

The design of the Diesel Generator "E" meets the intent of this Regulatory Guide. Methods of control of welding, in fabricating and joining safety-related austenitic stainless steel components and systems, were implemented which led to meeting the intent of 10CFR50, Appendix A, GDC 1.

Regulatory Guide 1.32 - CRITERIA FOR SAFETY RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS (Revision 1, March 1976) for Diesel Generators "A-D" and (Revision 2, February 1977 for the Diesel Generator 'E' Facility)

Diesel Generators "A-D" are designed in accordance with Regulatory Guide 1.32, Revision 1 (March 1976). Diesel Generator "E" and the transfer points in the Diesel Generator "A-D" rooms are designed to Regulatory Guide 1.32, Revision 2 (February 1977), as discussed in FSAR Sections 8.1.6.1 and 8.3.2.2. Positions C.1.e and C.1.f of this regulatory guide are addressed in the responses to Regulatory Guides 1.75 and 1.9 respectively.

Regulatory Guide 1.33 - QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION) (Revision 2, February 1978)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.34 - CONTROL OF ELECTROSLAG WELD PROPERTIES (December 28, 1972)

The electroslag weld method was not used for the fabrication of any core support structures or any ASME B&PV Section III, Class 1 or 2 vessels and components. Therefore, this regulatory guide is not applicable to Susquehanna SES.

Regulatory Guide 1.35 - IN-SERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES (Revision 2, January 1976)

Not Applicable.

Regulatory Guide 1.36 - NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL (February 1973)

As discussed in Subsections 4.5.2.4 and 6.1.1.1, the use of nonmetallic thermal insulation for austenitic stainless steel complies with this regulatory guide.

Regulatory Guide 1.37 - QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS (March 16, 1973)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments.

Compliance of the operational Quality Assurance program with this guide is discussed in Section 17.2.

Regulatory Guide 1.38 - QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, May 1977)

The Susquehanna SES construction quality program is being conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program is described in Section 17.2.

Regulatory Guide 1.39 - HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, September 1977)

Compliance of the Operational Quality Assurance Program is described in Section 17.2.

Regulatory Guide 1.40 - QUALIFICATION TESTS OF CONTINUOUS DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER COOLED NUCLEAR POWER PLANTS (March 16, 1973)

As described in Subsection 3.11.2.2, the present design of Susquehanna SES complies with the provisions of this regulatory guide.

Regulatory Guide 1.41 - PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEM TO VERIFY PROPER LOAD DESIGN ASSIGNMENTS (March 16, 1973)

The requirements of this regulatory guide were met. The testing procedures are outlined in Section 14.2.

Regulatory Guide 1.42 - INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHTWATER-COOLED NUCLEAR POWER REACTORS

Withdrawn March 18, 1976.

Regulatory Guide 1.43 - CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS (May 1973)

This regulatory guide prescribes qualification and production cladding controls for ASME SA 508-2 material made to coarse grain practice. This material is not used for any of the safety class components within the NSSS. ASME SA 508-2 composition material employed on the reactor pressure vessel for Susquehanna SES is produced to fine grain practice.

Regulatory Guide 1.44 - CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL (May 1973)

Subject to the following clarifications and exceptions, the use of unstabilized austenitic stainless steel for components that are part of (a) the reactor coolant pressure boundary, (b) systems required for reactor shutdown, (c) systems required for emergency core cooling, (d) reactor vessel internals required for emergency core cooling, and (e) reactor vessel internals which are relied upon to permit adequate core cooling during any mode of normal operation or postulated accident conditions complies with Regulatory Guide 1.44.

(1) Reference: Position C.1. Contamination of austenitic stainless steel (Type 300 series) by compounds that could cause stress corrosion cracking is avoided during all stages of fabrication and installation. Cleaning is limited to solutions that contain not more than 100 ppm of chlorides. Rinsing or flushing is with water containing less than 100 ppm of chlorides. Foreign substances in contact with austenitic stainless steel (die lubricants, penetrant materials, marking materials, masking tape, etc.) either are controlled to contain the following amounts of contaminates, or are removed immediately following the operation in which they are used.

1. The inorganic halogen content shall be less than 200 ppm by weight.

2. The halogen (inorganic and organic) content shall be less than 1 percent by weight measured in accordance with ASTM D808-63.

3. Sulfur content shall be less than 1 percent by weight as measured in accordance with ASTM D129-64.

4. Total low melting point metal (lead, bismuth, zinc, mercury, antimony, and tin) content shall be less than 200 ppm by weight and no individual metal content shall be greater than 50 ppm by weight.

Completed components are packaged so that they are protected from the weather, dirt, wind, water spray, and other deleterious environmental conditions that may be encountered during shipment and subsequent site storage.

In the field, austenitic stainless steel components are stored clean and dry to prevent contamination. System hydrostatic tests are performed with demineralized water containing less than 10 ppm of chlorides. The influent water quality during final flushing or preoperational testing of the completed system is at least equivalent to the quality of demineralized water as defined in ANSI N45.2.1-1973.

Leachable chlorides and fluorides in nonmetallic insulation materials, which come in contact with austenitic stainless steel, are held to the lowest practical level by the inclusion of the requirements of Regulatory Guide 1.36 in the insulation purchase specifications.

(2) Reference: Position C.2. All grades of austenitic stainless steels (Type 300 series) are required to be furnished in the solution heat treated condition before fabrication or assembly into components or systems. The solution heat treatment varies according to the applicable ASME or ASTM material specification.

(3) Reference: Position C.3. All austenitic stainless steels are furnished in the solution heat treated condition in accordance with the material specification. For material that has been solution heat treated by the material manufacturer, testing to determine susceptibility to intergranular corrosion is performed only when required by the material specification. During fabrication and installation austenitic stainless steels are not permitted to be exposed to temperatures in the range of 800° to 1500°F, except for welding and hot forming. Welding practices are controlled to avoid severe sensitization, as described in (6), and solution heat treatment in accordance with the material specification is required following hot forming in the temperature range 800° to 1500°F. Unless otherwise required by the material specification, the maximum time for cooling from the solution heat treat temperature to below 800°F is 3 minutes. Corrosion testing in accordance with ASTM A 262-70, Practice A or E, or ASTM A 393 may be required if the maximum length of time for cooling to below 800°F is exceeded or the solution heat treat condition is in doubt.

(4) Reference: Position C.4. Use of low carbon (0.03 percent maximum) unstabilized austenitic stainless steel is not required since the reactor coolant meets the conductivity and chloride limits of Table 2 of Regulatory Guide 1.56. However, it is used as described in Section 6(c) below.

(5) Reference: Position C.5. Heat treating austenitic stainless steel in the temperature range 800 to 1500°F is not permitted and solution heat treatment is required following hot forming. Since sensitization is avoided, testing to determine susceptibility to intergranular attack is not performed.

(6) Reference: Position C.6. Welding practices are controlled to avoid severe sensitization in the heat-affected zone of unstabilized austenitic stainless steel as described below. Unless otherwise stated, the position applies to both Bechtel and Bechtel suppliers and subcontractors.

a) Weld Heat Input

Bechtel controls weld heat input during field installation by using shielded metal-arc welding and gas tungsten-arc welding processes only, and by limiting the size of electrodes for each process to 5/32 in. and 1/8 in. diameter maximum, respectively. In addition to these two processes, Bechtel suppliers and subcontractors are permitted to use automatic submerged-arc welding and gas metal-arc welding. Hardsurfacing operations are not included.

b) Interpass Temperature

The interpass temperature is controlled so as not to exceed 350°F.

c) Carbon Content

Susceptibility to sensitization is reduced significantly by selecting materials with the lowest reported carbon content. Specifically Type 304 stainless steel with carbon content limited to .030 maximum or 304L Stainless steel with carbon content limited to .030 maximum was used as follows:

Pipe Description	Size	Material
Head Spray	6"	304SS
Core Spray Influent	12"	304SS
Recirculation System	4"	304SS
Standby Liquid Control	1 1/2"	304LSS
Reactor Water Cleanup (Effluent from Reactor)	4"	304LSS
Instrument Piping	1" & 2"	304LSS
Bottom Drain	4"	304LSS
Vent, Drain, and Test Connections	1"	304LSS

d) Solution Heat Treatment

All austenitic stainless steels are provided in the solution annealed condition. This is accomplished by following the manufacturer solution annealing by water quenching from the solution annealing temperature to below 800°F in 3 minutes. Solution heat treatment is not required after welding.

Severe sensitization is avoided by not permitting heat treatment in the temperature range 800 to 1500°F following welding. This requires a special technique when welding stainless steel safe ends (transition pieces) to carbon or low-alloy steel component nozzles or piping. Specifically, a 309L stainless steel overlay or an Inconel weld overlay is deposited on the component and the component is postweld heat treated. Following final postweld heat treatment of the component, the stainless steel safe end is welded to the weld overlay using 308 or 308L austenitic stainless steel or Inconel type welding materials.

Intergranular corrosion testing is not performed on a routine basis. Performing intergranular corrosion tests for each welding procedure serves no useful purpose when welding practices and reactor coolant water chemistry are controlled as described above.

Regulatory Guide 1.45 - REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS (May 1973)

The design of the leakage detection systems is described in Subsection 5.2.5.1.

Regulatory Guide 1.46 - PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT (May 1973)

The criteria given in NRC Branch Technical Position MEB3-1, dated 11-24-75, is used in lieu of the criteria prescribed in Regulatory Guide 1.46 dated May 1973 for the non-NSSS scope of supply. Section 3.6 describes how the present design meets these protection requirements.

This regulatory guide is applicable to the main steam, HPCI, RCIC, RWCU, SLCS and recirculation pipelines within the NSSS scope of supply.

The design of the containment structure, component arrangement, Class 1 pipe runs, pipe restraints and compartmentalization was done in consonance with the acknowledgement of protection against dynamic effects associated with postulated rupture of piping. Analytically sized and positioned pipe restraints were engineered to preclude damage based on the pipe break evaluation.

Pipe whip requirements for fluid system piping within the primary containment that under normal operation, has service temperatures 200º F or pressures greater than 275 psig comply with ANS 58.2 - "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture" and Regulatory Guide 1.46 except as delineated in the following criteria for no breaks in Class 1 piping:

1. If Equation 10 of NB-3653-1, ASME Code III results in S_n less than or equal to 2.4 S_m for ferritic or austenitic steels, no other requirement need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and an OBE event transient.

2. If Equation 10 results in S_n between 2.4 S_m and 3,0 S_m for ferritic or austenitic steels, the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6, must be less than 0.1.

3. If Equation 10 results in S_n greater than or equal to 3.0 S_m for ferritic or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653.6 must not exceed 2.4 S_m and the cumulative usage factor, U, must be less than 0.1.

Regulatory Guide 1.47 - BYPASSED AND INOPERATIONAL STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS (May 1973)

The design, as discussed in Subsections 7.1.2, 7.1.2.6.10, 7.2.2.1.2.1.5), 7.3.2a.1.2.1.7, 7.3.2a.2.2.1.5, 7.3.2a.5.2.5, 7.4.2.1.2.1.7, 7.4.2.2.2.1.71, 7.5.1b.7, 7.5.2a.5.4, 7.5.2b.5, and 8.1.6.1.m (Regulatory Guide 1.47), complies with the provisions set forth in this regulatory guide.

Regulatory Guide 1.48 - DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS (May 1973)

The design loading combinations for non-NSSS systems for Positions C.1 to C.12 are described in Table 3.9-6. Operability of active pumps and valves is assured as described in Subsection 3.9.3.2.

The design limits and loading combinations for seismic category I fluid system components for the Diesel Generator 'E' facility are in compliance with this regulatory guide.

GE practice is representative of industry practice and is in general agreement with the requirements of Regulatory Guide 1.48 with the following clarifications:

a. The probability of an OBE of the magnitude postulated for the Susquehanna SES is consistent with its classification as an Emergency Event. However, for design conservatism, loads due to the OBE vibratory motion have been included under upset conditions. Loads due to the OBE vibratory motion plus associated transients, such as turbine trip, have been considered in the equipment design under emergency conditions consistent with the probability of the OBE occurrence.
b. The use of increased stress levels for Class 2 components is consistent with industry practice as specified in ASME B&PV Code Section III.

For a comparison of NSSS compliance with Regulatory Guide 1.48 see Table 3.13-1. This comparison reflects a GE practice on BWR 4's and 5's and therefore, is applicable to the Susquehanna SES (see Subsections 3.9.2 and 3.9.3).

Regulatory Guide 1.49 - POWER LEVELS OF NUCLEAR POWER PLANTS (Revision 1, December 1973)

Regulatory Guide 1.49 states, in part,

> Anaylses and evaluation in support of the application should be made at an assumed core power level equal to 1.02 times the proposed licensed power level. . . for (a) normal operating conditions, (b) transient conditions anticipated during the life of the facility. . . and (c) accident conditions necessary to evaluate the adequacy of structures, systems and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

Fuel dependent analyses include the effects of a 2 percent power uncertainty factor discussed in Regulatory Guide 1.49. Most of the analyses were performed at 100% power level and the impact of the two percent power uncertainty factor is accounted for either statistically or through the inherent conservatism of the methodology. For three of the analyses, ASME over-pressurization (Section 5.2), loss of feedwater flow (Section 15.2.7), and LOCA-ECCS analyses (Section 6.3.3), the effects of the 2 percent power uncertainty factor are not directly included in methodology used for the analyses. Therefore, these analyses were performed at 102% of CPPU rated power to account for the 2 percent power uncertainty factor. Non-fuel dependent analysis power levels include a two percent power uncertainty factor unless a smaller value is specifically justified or the uncertainty is accounted for in the analysis methods. Additionally, Regulatory Guide 1.49 does not apply to some events that have been historically analyzed from nominal initial conditions, which include the Anticipated Transient Without Scram (section 15.8) and Station Blackout (section 15.9) events.

Thus, the SSES units continue to meet the intent of the Guide, which is to assure that all design calculations are performed at the highest possible power level that the plant can be operating.

Regulatory Guide 1.50 - CONTROL OF PREHEAT TEMPERATURE FOR WELDING OF LOW-ALLOY STEEL (May 1973)

The control of preheat temperature for welding of low-alloy steel is described in Subsection 5.2.3.3.2.1.

Regulatory Guide 1.51 - INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS

Withdrawn July 15, 1975.

Regulatory Guide 1.52 - DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ENGINEERED SAFETY FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT WATER COOLED NUCLEAR POWER PLANTS (Revision 1, July 1976 and Revision 2, March 1978)

The filter adsorber systems are designed to mitigate exposures resulting from a design basis accident. The Control Structure Emergency Outside Air Supply System (CSEOASS) and the Standby Gas Treatment System (SGTS) are the only systems that are subject to the requirements of this regulatory guide.

Subject to the clarifications and/or exceptions indicated below, the general intent of this regulatory guide has been met by the current design of the plant. Items (1) through (10) and (13) apply to Revision 1 and items (11) and (12) apply to Revision 2 of the Regulatory Guide 1.52.

(1) Reference: Position C.2.a. Moisture separators are used only where moisture impingement may be a problem. The SGTS is the only system with moisture separators. Heaters are used on both systems (SGTS and CSEOASS) to control humidity before filtration.

(2) Reference: Position C.2.d. Devices, such as pressure relief valves, are not used on either the CSEOASS or SGTS. Neither filtration system is subject to containment pressures (internal or external) or hazardous pressure surges.

(3) Reference: Position C.2.g. The pertinent pressure drop which is instrumented to signal, alarm, and record in the control room is the pressure drop across the first HEPA filter. The SGTS also alarms on high differential pressure across the entire filter system. The flow rate and low flow alarm also indicate proper functioning of the fan.

(4) Reference: Position C.2.i. Overall design considerations include reduction of radiation exposures during routine maintenance and testing insofar as effectually possible. It is envisioned, however, that workers will not handle filter units after a design basis accident and will thereby avoid exposures associated with immediate post-accident filter handling. Accordingly, no efforts were made toward a unitized atmosphere cleanup train design in the interest of accident exposure reduction.

(5) Reference: Position C.3.d. Since none of the HEPA filters separators are exposed to potential iodine removal spray, the units are not designed for contact with the spray. The referenced military standards (MIL-F-51068C and MIL-F-51079A) have been deleted, but represent acceptable standards for installed (or previously purchased) HEPA filters. New HEPA filters will meet the standards presented in ASME AG-1-1997.

(6) Reference: Position C.3.e through C.3.h. In these sections and all others where reference is made to ORNL-NSIC-65, the reference is understood to be to ERDA 76-21 or ANSI N509 where appropriate.

(7) Reference: Position C.3.i. The adsorber beds are designed for 2.5 mg of iodine (both stable and radioactive) per gram of activated carbon averaged over the bed depth. This is consistent with the background information. Each replacement batch of impregnated, activated carbon shall meet the qualification and batch test results summarized in Table 5-1 of ANSI N509-1980, rather than those of Table 2 in Regulatory Guide 1.52, Revision 1, except that the 350 ft^3 batch size limit specified in Table 2 shall be retained.

(8) Reference: Position C.3.k. All systems are designed for low flow in order to control temperature rise. Oxidation effects are considered. A water spray system, provided only to minimize property loss in the event of fire, is designed for the control structure OA supply units, to extinguish a fire by flooding the adsorber units. The fire extinguishing system in the SGTS filter, sprays large quantities of water over the charcoal adsorber, until the charcoal temperature drops below its ignition temperature. The water is removed from the SGTS housing through automatic drain valves. The sprays or quenches are not provided for prevention of fire inception.

(9) Reference: Positions C.4.c and C.4.d. The spacing requirement is applicable to systems requiring operator access to remove filters and adsorber trays. Where unnecessary, the space is not provided, e.g., gasketless carbon absorbers which are filled and emptied externally.

(10) Reference: Position C.4.d. The length of pipe associated with manifolding would promote plate-out of the constituents of the sampled gas stream, thereby resulting in erroneous test results. The test probes are located in readily accessible locations; a minimum run of piping is used and manifolding is not employed.

(11) Reference: Positions C.5.a, C.5.c and C.5.d - In-place testing criteria identified in Paragraph C.5.a, C.5.c and C.5.d of Rev. 2 dated March, 1978 of the Regulatory Guide 1.52 are implemented, however, the testing frequency for C.5.c is 24 months rather than 18 months.

(12) Reference: Positions C.6.a and C.6.b - Laboratory Testing Criteria for activated carbon identified in Paragraph C.6.a and C.6.b of Rev. 2 dated March, 1978 of the Regulatory Guide 1.52 are implemented with the following exceptions:

a. Representative samples of used activated carbon will be tested at $\leq 30°C$ and $\geq 70\%$ relative humidity and in accordance with ASTM D3803-89.

b. New activated carbon will meet the performance requirements and physical property specifications given in Table 5-1 of ANSI N509-1980.

Note: Table 6.5-2 provides details of how various positions (C1 to C4) of Revision 0 of Regulatory Guide 1.52 are met in the design of SGTS and CSEAOSS.

(13) Reference: Position C.4.e. The frequency and duration of operating the cleanup train with the heater in operation is in accordance with the Plant Technical Specifications and Surveillance Requirements.

Regulatory Guide 1.53 - APPLICATION OF THE SINGLE-FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS (June 1973)

The Susquehanna SES complies with this guide in the design of protection, safeguards actuation, and Class 1E electrical systems. Related considerations pertaining to cable separation and associated circuits are considered in the discussion of Regulatory Guide 1.75 and environmental considerations in the discussion of Regulatory Guide 1.89.

Regulatory Guide 1.54 - QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER COOLED NUCLEAR POWER PLANTS (June 1973)

For the non-NSSS scope of supply, a quality assurance program for coatings was in compliance with this regulatory guide.

For the NSSS scope of supply, the quality assurance records requirements in this regulatory guide were not imposed on painting material and paint application for Susquehanna SES since these coatings cover a relatively small exposed surface area.

Regulatory Guide 1.55 - CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (June 1973)

Concrete placement in Seismic Category I structures is in accordance with the regulatory positions of this guide as described in Appendix 3.8B.

Concrete placement for the Diesel Generator 'El Building is in accordance with ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures" and ANSI N45.2.5 "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

Regulatory Guide 1.56 - MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (June 1973)

GE Report, NEDO-10899, "Chloride Control in BWR Coolants," establishes General Electric's position on water purity.

Regulatory Guide 1.57 - DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (June 1973)

The primary containments for Susquehanna SES are reinforced concrete structures. Nonetheless, the provisions of this regulatory guide are applicable to the following components of each containment:

1) Equipment hatch with personnel lock

2) Equipment hatch

3) Drywell head assembly

4) CRD removal hatch

5) Suppression chamber access hatches

6) Pipe and electrical penetrations

These items were designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, for Class MC components, the 1971 Edition with addenda through Summer 1972. Allowable stress limits used for the design are in conformance with Regulatory Guide 1.57, Paragraph C.1.d and the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE-3131.2 as specified in the Winter 1973 Addenda. A detailed discussion of these design features is contained in Subsection 3.8.2.

Regulatory Guide 1.58 - QUALIFICATIONS OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL (Revision 1, September 1980)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is discussed in Section 17.2.

Regulatory Guide 1.59 - DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS (Revision 2 August 1977)

The design basis flood, discussed in Section 2.4, is determined in accordance with the regulatory positions of this guide.

Regulatory Guide 1.60 - DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (Revision 1, December 1973)

The design response spectra used in the analysis of Susquehanna SES, except the Diesel Generator 'E' facility, are different from those of the regulatory guide. A detailed discussion of the design response spectra is presented in Subsection 3.7b.1.

Regulatory Guide 1.61 - DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (October 1973)

The damping values used in the original seismic design of Susquehanna SES, except the Diesel Generator 'E' facility, are different from the regulatory guide. A detailed discussion of the damping values is presented in Subsection 3.7b.1. For snubber elimination or other piping modifications, damping values from this Regulatory Guide may be used.

Regulatory Guide 1.62 - MANUAL INITIATION OF PROTECTIVE ACTIONS (October 1973)

The provisions for manual initiation of protective actions are described in Subsections 7.2.2.1.2.1.7, 7.2.4.1.1.2.2.4, 7.3.2a.1.2.1.9, 7.3.2a.2.2.1.7, 7.3.2a.5.2.7, 7.4.2.1.2.1.9, 7.4.2.2.2.1.9, 7.6.1b.3.1 and 8.1.6.1.0.

Regulatory Guide 1.63 - ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR WATER COOLED NUCLEAR POWER PLANTS (Revision 1, May 1977)

Since the construction permit for Susquehanna SES was issued in November 1973, the provisions of Revision 1 to this regulatory guide (which supplements IEEE 317-1976) were not specifically considered in the design of Susquehanna SES. The design of the electric penetration assemblies is therefore in compliance with Regulatory Guide 1.63 dated October 1973 (which supplements IEEE 317-1972). Specifically, Sections 4.2.3, 4.2.4, 5.1.6, 5.2.2, 6.2, 6.3.3, and 6.4 of IEEE 317-1976 have not been incorporated.

The penetration assemblies are type tested. There are no provisions for periodic testing under simulated fault conditions.

Electrical penetration circuits are summarized as follows:

1. 480 Volt Circuits

   Loads powered from 480 volt motor control centers are supplied via electrical penetrations equipped with #4 awg, #10 awg, or 4/0 awg copper conductors. Typical single line diagrams for each penetration conductor size are shown on Figure 3.13-1.

   Overcurrent protection for penetrations considers both connected load characteristics and penetration time-current capabilities in accordance with the following guidelines:

   a. 480V motor, less than 1.5 hp (Figure 3.13-2A)

      1. Penetration conductor #10 awg

      2. Overcurrent protection redundant, adjustable, magnetic only circuit breakers with a maximum setpoint of 45 amperes and a minimum setpoint which exceeds 200% of the motor locked rotor current.

   b. 480V motor, 10 hp or less (Figure 3.13-2B)

      1. Penetration conductor #10 awg

      2. Overcurrent protection redundant, fixed, thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 40 amperes and a minimum thermal trip rating which exceeds 200% of the motor full load current.

c. 480V motor, 10.1 to 20 hp (Figure 3.13-3)

1. Penetration conductor #4 awg

2. Overcurrent protection redundant, fixed, thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 100 amperes and a minimum thermal trip rating which exceeds 200% of the motor full load current.

d. 480V non-motor loads (presently only non-motor loads are hydrogen recombiners.)

1. Penetration conductor 4/0 awg (Figure 3.13-8)

2. Overcurrent protection-redundant, fixed thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 150 amperes and a minimum thermal trip rating which exceeds 150% of the connected load full load current.

Credit is not taken for penetration protection provided by other overload detectors, such as, motor overloads shown in Figure 3.13-1. Penetration seal withstand curves (Figures 3.13-2, 3.13-3, and 3.13-8) apply for the condition when mechanical seal integrity is maintained, but electrical integrity may be comprised.

Adequacy of the subject molded case circuit breaker selections are demonstrated on Figures 3.13-2, 3.13-3 and 3.13-8 by showing time-current characteristic curves with the protective device total clearing time below, and to the left of the penetration seal withstand curve.

2. <u>120 Volt AC Control Circuits</u>

There are two types of 120 volt AC control circuits to be considered: (1) circuit powered by a control transformer located in an MCC cubicle and (2) circuit powered by a 120 volt AC instrument distribution panel (see Figure 3.13-4). #14 AWG is the minimum size used for control circuits.

a. The motor control circuits are powered by control transformers located in the respective MCC cubicles. Control transformers are sized to meet the requirement of the control circuit (one for each starter). Typically a 120 VA control transformer is used with a NEMA Size 1 starter and a 200 VA is used with a NEMA Size 2 starter. The largest control transformer used in connection with a penetration is 350 VA. The maximum short circuit current that can be delivered by a 350 VA transformer in a control circuit is approximately 100 amps. A fuse, rated 3.2 amp or less, located in the respective MCC provides the circuit protection. Since sustained short circuit current will destroy the control transformer before the integrity of the penetration assembly is compromised, backup fuses are not utilized (refer to Figure 3.13-4).

b. Control circuits that emanate from the fuse control panel will have a 10 amp or lower rated fuse at the panel and as a back-up breaker either an identical fuse in series or a breaker in the 120 V instrument AC distribution panel. 20 amp or smaller breaker or fuse will be used for backup protection (refer Figure 3.13-4). All breakers shown in Figure 3.13-4, Case 2, are molded case and self-actuated (short circuit current trip with manual closing).

3. <u>125 VDC Control Circuits</u>

Each 125 VDC control circuit is protected by a 20 amp or smaller fuse located in a control panel with back-up protection provided by a 20 amp breaker (ITE type E) in the dc distribution panel. The mechanical integrity of the penetration assembly is maintained under overload or faults conditions (refer to Fig. 3.13-5).

4. <u>120 Volt Lighting and Space Heater Circuits</u>

Mechanical integrity of all 120 V lighting and space heater circuits is maintained under overload or fault condition. Each type of circuit is discussed below:

a. Each 120 V lighting circuit is provided with a 20 amp breaker with back-up protection of a 50 amp breaker as shown on Fig. 3.13-6.

b. Each 120 V motor (except reactor recirc pump motor) or motor operated valve (MOV) space heater is provided with a 20 amp or smaller fuse protection. Backup protection is provided by a 20 amp or smaller breaker located in a lighting panel (see 4(a) of above).

c. Each of the 120 V space heater circuits for the reactor recirc pump motors is provided with a 40 amp breaker as the primary protection. Backup protection is provided by a 50 amp breaker located in a 280/120 V lighting panel.

5. <u>Medium Voltage Circuits (above 480 volt)</u>

The only medium voltage equipment located inside the containment are two variable frequency reactor recirc. pump motors. These two recirc. pump motors are fed by two independent 13.8 kV M-G (motor-generator) sets with the generator output rated at 3,920 volts. The M-G sets are located in the turbine building. A calculated M-G set generator decrement curve together with the penetration cable thermal curves is shown on Figure 3.13-7. The maximum short circuit current that can be produced by the generator is 7000 amp asymmetrical. Figure 3.13-7 shows that a line-to-line short circuit (across lines which do not furnish input power to the voltage regulator) of about 6,000 amps can be sustained if the generator field breaker and the main feeder to the M-G set drive motor are not tripped. However, the existing protection schemes have redundant detection devices and redundant protection devices for the penetration circuits. In addition, redundant Class 1E overcurrent protection is provided (one overcurrent relay for each Recirculation Pump Trip (RPT) breaker). This protection scheme ensures that the fault current would be cleared before any damage. Therefore, as shown by Figure 3.13-7, the mechanical integrity of the penetration assembly is maintained under the most severe fault condition.

6. <u>Instrumentation Circuits</u>

The instrumentation circuits are low level signal circuits (ma or mv range) such as thermocouples, RTD, etc. These circuits are current limiting. In addition, instrument cables are not routed with any other types of circuits in the same raceway. Therefore, backup protection is not needed.

The requirement for periodic testing and inspecting of fuses, breakers, and containment circuit protection schemes are stated in SSES Technical Requirements Manual. Requirements for back-up penetration circuit protection are met.

Regulatory Guide 1.64 - QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS (Revision 2, June 1976)

The quality program for the design for the Susquehanna SES is described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this regulatory guide is described in Section 17.2.

Regulatory Guide 1.65 - MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS (October 1973)

Subsections 5.2.4 and 5.3.1.7 describe the materials and inspections for reactor vessel closure studs.

Operation will be conducted in accordance with this regulatory guide with the following exception:

All studs will not be removed from the vessel prior to flooding the reactor cavity. Other means will be employed to prevent corrosion.

Regulatory Guide 1.66 - NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS (October 1973)

Withdrawn September 28, 1977.

Regulatory Guide 1.67 - INSTALLATION OF OVERPRESSURE PROTECTION DEVICES (October 1973)

This regulatory guide is not applicable to Susquehanna SES since the main steamline safety/relief valves discharge into closed systems.

Regulatory Guide 1.68 - INITIAL TEST PROGRAM FOR WATER-COOLED REACTOR POWER PLANTS (Revision 1, January 1977)

Subject to the clarifications and/or exceptions indicated in Subsection 14.2.7, the provisions of this regulatory guide were met by the test programs instituted during the startup of each unit.

The design of the A-D Diesel Generators meets the intent of this Regulatory Guide. The applicable Appendix A of Regulatory Guide 1.68 provides acceptable preoperational testing criteria for emergency/standby AC power supplies. These testing requirements were implemented and led to meeting the intent of 10CFR50, Appendix B.

Regulatory Guide 1.68 - (TASK SC 704-5) INITIAL TEST (DIESEL GENERATOR E ONLY) PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, August 1978)

The design of the Diesel Generator "E" meets the intent of this Regulatory Guide. This revision added "Emergency loads supplied should be confirmed to be in agreement with design sizing assumptions used for power supplies" to the applicable Appendix A, Section g(3) of Regulatory Guide 1.68. Methods of control of welding, in fabricating and joining safety-related austenitic stainless steel components and systems, were implemented which led to meeting the intent of 10 CFR 50, Appendix B.

Regulatory Guide 1.68.1 - PREOPERATIONAL AND INITIAL STARTUP TESTING OF FEEDWATER AND CONDENSATE SYSTEMS FOR BOILING WATER REACTOR POWER PLANTS (Revision 1, January 1977)

The preoperational testing and initial startup testing of the feedwater and condensate systems associated with Susquehanna SES were conducted in accordance with the provisions of this regulatory guide.

Regulatory Guide 1.68.2 - INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS (January 1977)

The portion of the initial startup test program used to demonstrate the remote shutdown capability of each unit was conducted in accordance with the provisions of this regulatory guide.

Regulatory Guide 1.69 - CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS (December 1973)

The design and placement of the concrete used for radiation shielding differs from the provisions of this Regulatory Guide. The practices employed for the Susquehanna SES are described in Appendix 3.8B.

Regulatory Guide 1.70 - STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS-LWR EDITION (Revision 2, September 1975)

The format of this FSAR complies with this regulatory guide except that replacement pages will contain a change indicator (vertical bar in the margin of text and table pages) and a page change identification consisting of a revision number. The date of the change will not be shown on replacement pages. This is consistent with the requirements of 10 CFR 50.71(e)(5).

With respect to physical specifications, material submitted may be submitted on CD-ROM in accordance with the guidance of NRC Regulatory Issue Summary 2001-05, dated January 25, 2001.

Regulatory Guide 1.71 - WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY (December 1973)

Exceptions are taken to this regulatory guide as specified below:

(1) Reference: Position C.I. Performance qualifications for field personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX.

For the welder qualifications for the reactor coolant pressure boundary, see Subsection 5.2.3.3.2.3.

Regulatory Guide 1.72 - SPRAY POND PLASTIC PIPING December, 1973

Plastic piping is not used in safety related applications.

Regulatory Guide 1.73 - QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS (January 1974)

The requirements of this regulatory guide are met as described in Section 3.11.

Regulatory Guide 1.74 - QUALITY ASSURANCE TERMS AND DEFINITIONS (February 1974)

The Susquehanna SES construction quality program is described in PSAR Appendix D and amendments. Compliance of the operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.75 - PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS (Revision 1, January 1975)

A. This Regulatory Guide endorses IEEE 384-1974 subject to the additions and clarifications delineated in Section C of the guide. Although there is no requirement for Susquehanna SES to comply with Regulatory Guide 1.75 and IEEE 384, Susquehanna SES follows this separation criteria, subject to the clarifications and exceptions below for the NSSS scope of supply. The paragraphs below reference sections of the IEEE standard in all cases and the specific paragraphs of the regulatory position statement where applicable.

(1) Reference: Regulatory Guide 1.75 and Section 4.5 of IEEE 384-1974

Certain power cables are subject to the same requirements as a Class 1E circuit. This applies to derating, environmental qualification, flame retardance, splicing restrictions, and raceway fill.

(2) Reference: Sections 4.5 and 4.6 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

(3) Reference: Sections 5.1.3, 5.1.4, and 5.6.2 of IEEE 384-1974

All annunciator and computer input circuits are classified as non-Class 1E circuits. These non-Class 1E circuits are not separated from Class 1E control circuits within Class 1E panels in which the non-Class 1E circuit derives its input, i.e., circuit breaker auxiliary contact used for computer input, etc., nor are they separated in PGCC cable ducts. These non-Class 1E instrument circuits are considered to be low energy and the probability of these non-Class 1E circuits providing a mechanism of failure to the Class 1E circuits is extremely low.

(4) Reference: Position C.15 of Regulatory Guide 1.75 and Section 5.3.1 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

(5) Reference: Sections 5.6.2 and 5.6.3 of IEEE 384-1974

In general, the circuits for redundant Class 1E systems and the circuits for non-Class 1E systems are located in separate panelboards, boxes, racks, and enclosures. Panels, racks, and boxes that contain wiring and devices for Class 1E circuits are labeled distinctly to externally identify the separation system and grouping. Internal to the enclosures, devices such as relays, switches, and instruments, are uniquely identified. In addition, external cables are color coded and marked to be readily identifiable. These methods of identification are described in Subsection 3.12.3.2.

Where required, physical separation is achieved either by a minimum of 6" horizontal and vertical separation, steel barriers, metallic enclosures, or metallic flexible conduit.

Where the above separation methods are not feasible, one of the separation groups are to be covered with one of the following qualified non-flammable materials:

i. Haveg Industries, Siltemp sleeving type S or woven tape type WT65.

ii. Carborundum, Fiberfrax sleeving type HP144T or woven tape type 3L144T.

These materials have been qualified to be used as separation barriers (Wyle Lab. Test Report No. 56669 dated May, 1980). Applications of these materials are controlled and documented.

(6) Reference: Section 5.5 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

Additional information is found in Sections 7.1, 7.2, 7.3, 7.4 and 7.6

B. Compliance to this Regulatory Guide for the non-NSSS scope of supply is discussed in Section 8.1.6.1 (Regulatory Guide 1.75), and for D/G E in Subsection 8.1.6.1.r.

Regulatory Guide 1.76 - DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (April 1974)

For all tornado-resistant structures except the Diesel Generator 'E' Building, the following parameters were used in lieu of those presented in Table I of this regulatory guide.

Tangential speed:	300 mph
Translational speed:	60 mph
Rate of pressure drop:	1 psi/sec (for 3 seconds)

A detailed discussion of these parameters is contained in Subsection 3.3.2.

The design basis tornado used for the Diesel Generator 'E' Building is in accordance with this regulatory guide.

Regulatory Guide 1.77 - ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS (May 1974)

Not Applicable.

Regulatory Guide 1.78 - ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (Revision 1 December 2001)

As described in Subsection 6.4.4.2, the steps taken to protect control room habitability conform to requirements of this regulatory guide.

Regulatory Guide 1.79 - PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS (Revision 1, September 1975)

Not Applicable.

Regulatory Guide 1.80 - PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS (June 1974)

The primary containment instrument gas system will be tested in accordance with the requirements of Regulatory Guide 1.80, Sections C1 through C.6. The portions of the instrument air system which supply safety-related equipment will also be tested in accordance with Section C.1 through C.6 of the Regulatory Guide 1.80 (June, 1974). Loss of air testing will be done in the various system preoperational tests. Systems/components which have separate accumulators will be tested with a loss of air/gas to ensure that the accumulators function in accordance with design. Testing described in Regulatory Guide 1.80, Sections C7 through C10 will not be done in the instrument air system or primary containment gas system tests.

Regulatory Guide 1.81 - SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS (Revision 1, January 1975)

The design of the standby electric power systems which use shared diesels complies with this Regulatory Guide by invoking the provisions of Position C.2. See Section 8.1.6.1.

Regulatory Guide 1.82 - SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS (June 1974)

Not Applicable.

Regulatory Guide 1.83 - IN-SERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES (Revision 1, July 1975)

Not Applicable.

Regulatory Guide 1.84 - CODE CASE ACCEPTABILITY, ASME SECTION III DESIGN AND FABRICATION

The revision date to this Regulatory Guide is intentionally not stated. Regulatory Guide 1.84 is frequently revised to append new Code Cases. No previous compliance requirements to earlier revisions are negated by updates to this Regulatory Guide.

The Susquehanna SES for the non-NSSS scope of supply will use only those code cases listed in this regulatory guide. In accordance with Section D of the Regulatory Guide, Code Cases approved by earlier revisions of the Regulatory Guide may have been invoked. In addition, ASME Code Case N-316 and N-640 have been approved by the NRC for use at Susquehanna Steam Electric Station. Should it be deemed necessary or beneficial to apply other code cases, a specific request shall be made to the NRC.

ASME Code Case N-516-2 has been approved by the NRC for use during the second 10-year inspection interval at Susquehanna SES in a letter dated February 21, 2002 (Relief Request for Authorization to use Code Case N-516-2 as an Alternative to the ASME Code). The use of Code Case N-516-2 is subject to the following three conditions and limitations:

1. Performance qualifications shall be in accordance with Paragraph 3.2 in Code Case N-516-2, except that immediate retest following a failed mechanical bend test shall be in accordance with ASME, Section IX, QW-320.

2. Procedure qualification shall be in accordance with Paragraph 3.1 in Code Case N-516-2. The Alternative Procedure Qualification Requirements of Paragraph 5.0 shall not be used except as noted in Paragraph 4.(b)(4) for the additional requirements for qualification of filler metal.

3. When welding is to be performed on high neutron fluence Class 1 material, then a mockup, using material with similar fluence levels, should be welded to verify that adequate crack prevention measures were used.

The NSSS scope of supply procedure for meeting the regulatory requirements is to obtain NRC approval for code cases applicable to Class I components only. Approval of code cases for Class 2 and 3 equipment was not required at the time of the design of the Susquehanna SES, and is not currently required by 10CFR50.55a. Therefore, GE believes that this procedure in conjunction with 10CFR50APPB and other regulatory requirements provide adequate assurance of quality in the design and fabrication of safety-related equipment (see Subsection 5.2.1.2).

Regulatory Guide 1.85 - CODE CASE ACCEPTABILITY, ASME SECTION III MATERIALS

The revision date to this Regulatory Guide is intentionally not stated. Regulatory Guide 1.85 is frequently revised to append new code cases, no previous compliance requirement to earlier revisions are negated by updates to this Regulatory Guide. The Susquehanna SES for non-NSSS scope of supply will use only those code cases listed in this regulatory guide. In accordance with Section D of the Regulatory Guide, Code Cases approved by earlier revisions of the Regulatory Guide may have been invoked. In addition, ASME Code Case 1481-1 has been approved by the NRC for use at Susquehanna Steam Electric Station. Should it be deemed necessary or beneficial to apply other code cases a specific request shall be made to the NRC.

The NSSS scope of supply procedure for meeting the regulatory requirements is to obtain NRC approval for code cases applicable to Class 1 components only. Approval of code cases for Class 2 and 3 equipment was not required at the time of the design of the Susquehanna SES, and is not currently required by 10CFR50.55a. Therefore, GE believes that this procedure in conjunction with 10CFR50APPB and other regulatory requirements provide adequate assurance of quality in the materials of safety-related equipment (see Subsection 5.2.1.2).

Regulatory Guide 1.86 - TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS (June 1974)

The Susquehanna SES will comply with this regulatory guide.

Regulatory Guide 1.87 - CONSTRUCTION CRITERIA FOR CLASS 1 COMPONENTS IN ELEVATED TEMPERATURE REACTORS (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, 1596) (Revision 1, June 1975)

Not Applicable.

Regulatory Guide 1.88 - COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS (Revision 2, October 1976)

The quality assurance program for the construction of the Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.89 - QUALIFICATION OF CLASS 1E EQUIPMENT FOR NUCLEAR POWER PLANTS (November 1974)

For the non-NSSS scope of supply, the degree of compliance with this regulatory guide and justification for any exceptions to this guide are provided in Section 3.11.

For the NSSS scope of supply, see Subsections 3.9.2.2a.2.7, 3.9.2.2a.2.9, 3.11.2, 7.2.2.1.2.1.11, and 7.3.2a.2.2.1.10.

Regulatory Guide 1.90 - IN-SERVICE INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS (Revision 1, August 1977)

Not Applicable.

<u>Regulatory Guide 1.91</u> - <u>EVALUATION OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANT SITES (January 1975)</u>

An examination of the historical data for the surface transportation of explosive material near the Susquehanna SES indicates that the commercial truck and rail transportation routes are farther from the station's vital structures than the distances delineated in Figure 2 of this regulatory guide for plants situated in Tornado Region 1, as defined in Regulatory Guide 1.76. Although the closest point of approach from the Susquehanna River to the vital structures of the Susquehanna SES is less than the distance specified in this regulatory guide for the largest probable quantity of explosive material transported by ship (i.e., 5,000 tons of TNT), the Susquehanna River is not commercially navigable for purposes of transporting large quantities of explosives. Therefore, this type of accident is not considered to be probable enough to evaluate.

<u>Regulatory Guide 1.92</u> - <u>COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS (Revision 1, February 1976)</u>

Since the construction permit for the Susquehanna SES was issued in November 1973, the methods of combining modal responses and spatial components in seismic response analysis, as described in this regulatory guide, were not specifically considered in the original design, except for the Diesel Generator 'E' facility. The methods of design and analysis for structures, components, and piping systems that have been employed are described in Sections 3.7a, 3.7b, and 3.9.

Regulatory Guide 1.92 shall be invoked for the analysis of snubber elimination or other piping modifications whenever Code Case N-411 or Regulatory Guide 1.61 damping values are used.

<u>Regulatory Guide 1.93</u> - <u>AVAILABILITY OF ELECTRIC POWER SOURCES (December 1974)</u>

Compliance with this guide is discussed in Subsection 8.1.6.1.u.

<u>Regulatory Guide 1.94</u> - <u>QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS (Revision 1, April 1976)</u>

The quality assurance program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

<u>Regulatory Guide 1.95</u> - <u>PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE (February 1975)</u>

This regulatory guide was superceded by Revision 1 of Regulatory Guide 178.

Regulatory Guide 1.96 - DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS (Revision 1, June 1976)

Subject to the clarification indicated below, the provisions of this regulatory guide are met by the current plant design.

(1) Reference: Appendix A, Paragraph 6. The design and inspection of this portion of the leakage control system is in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code. The 100% volumetric inspection of this portion of the system is specifically exempted from the requirement for volumetric inspection by paragraph IWB-1220(a) of Section XI of the ASME Boiler and Pressure Vessel Code (Summer 1975 Addenda).

Note: MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7), approved by the NRC as an alternative.

Regulatory Guide 1.97 - INSTRUMENTATION FOR LIGHT WATER COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT (Revision 2, December 1980)

The accident monitoring instrumentation was designed prior to this Regulatory Guide being issued. The instrumentation for accident monitoring has not been evaluated against Revision 1 to the Regulatory Guide. With the exception of the item discussed below, our position on Revision 2 to the Regulatory Guide is provided in PLA-965, dated November 13, 1981. Compliance is described in more detail in the applicable sections of Chapter 7.

The instrumentation for accident monitoring is not specifically identified on the control panels. The control panel layouts and instrument identification at Susquehanna SES are based on good human factors engineering for the presentation of information to the control room operator. The Susquehanna SES Detailed Control Room Design Review validated our choice of instrument identification scheme. That review indicated that the use of redundant labeling as would have been required to specifically identify accident monitoring equipment would serve to confuse and distract operator performance rather than enhance it.

Subject to the clarifications delineated in PLA-2222, the exception above and the exceptions discussed in FSAR section 7.1.2.6.21, the provisions of this regulatory guide are met.

<u>Regulatory Guide 1.98</u> - <u>ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFFGAS SYSTEM FAILURE IN A BOILING WATER REACTOR (March 1976)</u>

Subject to the clarifications or exceptions indicated below, the assumptions of Regulatory Guide 1.98 are followed in the analyses of the offgas system failure in Subsection 15.7.1.

(1) Reference: Position C.4.a. Dose consequences are expressed in terms of REM TEDE. Dose conversion factors are given in Appendix 15B.

<u>Regulatory Guide 1.99</u> - <u>RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS (Revision 2, May 1988)</u>

The methods of Regulatory Guide 1.99, Revision 2 are followed in the analyses of reactor pressure vessel fracture toughness in Subsection 5.3.

<u>Regulatory Guide 1.100</u> - <u>SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS (March 1976)</u>

The implementation paragraph of this regulatory guide states that the requirements of the position statements will only be applied to plants that received construction permits after November 16, 1976. The Construction Permit for Susquehanna SES was issued in November 1973 and therefore the guidelines of this regulatory guide were not utilized in the design of this nuclear power station. However, PP&L conducted a reassessment of the original equipment qualification using the criteria contained in Regulatory Guide 1.100, Rev. 1.

Seismic qualification of the safety related electric equipment (non-NSSS scope of supply) has been conducted in accordance with the IEEE Standard 344-1971. Section 3.10 describes the complete qualification methods and procedures that have been utilized.

The safety-related electric equipment (NSSS scope of supply) meets IEEE 323-1971 and IEEE 344-1971.

The Diesel Generator 'E' facility is in accordance with Regulatory Guide 1.100, Rev. 1 and IEEE Standard 344-1975.

<u>Regulatory Guide 1.101</u> - <u>EMERGENCY PLANNING FOR NUCLEAR POWER PLANTS (Revision 3, August 1992)</u>

The Susquehanna Emergency Plan complies with the provisions of this regulatory guide.

<u>Regulatory Guide 1.102</u> - <u>FLOOD PROTECTION FOR NUCLEAR POWER PLANTS (Revision 1, September 1976)</u>

The present design of the Susquehanna SES complies with the provisions of this regulatory guide.

Regulatory Guide 1.103 - POST-TENSIONED PRESTRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS (Revision 1, October 1976)

Not Applicable.

Regulatory Guide 1.104 - OVERHEAD CRANE HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS (February 1976)

Subject to the clarifications and exceptions indicated below, the safety related overhead crane handling systems of this station comply with the provisions of this regulatory guide.

(1) Reference: Position C.1.b(2). The nil-ductility transition temperature for the structural steel associated with the cranes was not determined as suggested by this position. Position C.1.b(3) states that a cold proof test represents an acceptable alternative to the requirements of Position C.1.b(2). Accordingly, a cold proof test will be performed in accordance with the general procedures and testing frequency suggested in Position C.4.d except as modified in Item 7 below.

(2) Reference: Position C.3.f. The paragraph states that "The fleet angles between individual sheaves for rope should not exceed 1 1/2 degrees.," however, the fleet angles for the cranes that have been purchased for the Susquehanna SES are 3 degrees 7 minutes.

This position also states that "the pitch diameter of the lead sheave should be 30 times the rope diameter for the 180-degree reverse bend, 26 times the rope diameter for running sheaves and drum,...". The pitch diameter of the running sheaves is 24 times that of the wire rope diameter.

Consistent with established industry standards and in light of the limited number of loading cycles, the present design is acceptable.

(3) Reference: Position C.3.g. This position states that the head block, rope reeving system and load block should be subjected to a load test of 200 percent of the design rated load. No such test has been specified for these components. ANSI Standard B30.2 allows for a 125 percent load test of these components and the purchase orders for these cranes have so specified this type of test.

(4) Reference: Position C.3.j. This position suggests that the designer should provide means within the reeving system located on the head or on the load block combinations to absorb or control the kinetic energy of rotating machinery prior to the incident of two blocking or load hangup. As an alternative to the regulatory position, each crane is provided with dual upper limit switches to preclude the possibility of a "two block" occurrence and an overload switch combined with overcurrent and rate of current rise cutouts to prevent a load hangup.

(5) Reference: Position C.3.p. This paragraph states that provisions should be made for manual operation of the brakes.

This regulatory position has not been incorporated in that there has been no provision made for manual bridge and trolley holding brake operation.

The position also recommends that the trolley and bridge speeds be limited to a maximum speed of 30 fpm for the trolley and 40 fpm for the bridge. The maximum speed for the bridge is actually 50 fpm, but the potential effects of this difference in maximum speed is compensated for by the substantial runway length (approximately 320 ft.) and a stepless type bridge speed control. Administrative controls will be instituted to ensure that a maximum trolley speed of 30 fpm is used when the main hoist is loaded.

(6) Reference: Position C.4.b. This position states that the complete hoisting machinery should be allowed to "two block" during the hoisting test and should also be tested for ability to sustain a load hangup condition. The testing of these conditions was not specified in the purchasing documents for these cranes since the reeving system is not designed for two blocking or load hangup as indicated in Item 4 above, which describes the alternate design that has been incorporated in lieu of the regulatory position C.3.j.

(7) Reference: Position C.4.d. This position states that the cold proof test should be performed at or below the minimum operating temperature of the structural members essential to the structural integrity of the crane and should use a dummy load equal to 1.25 times the maximum working load. It further states that "If it is not feasible to achieve the minimum operating temperature during the test, the dummy load should be increased beyond the design rated load 1.5 percent per °F temperature difference." The minimum operating temperature for the cranes is 60°F and by appropriate scheduling of the cold proof test and adjustment of the HVAC systems, it will be attempted to maintain this temperature. However, in no case will the dummy load be increased above an amount of 125 percent of the rated load. Crane testing in excess of 125 percent of the rated load will adversely affect the safety of the cranes, since any such tests may propagate undetectable material defects and thus increase the probability of crane component failures. Furthermore, such testing would violate the ANSI Standard B30.2 and Title 29CFR Part/1910.179(k). This position also states that cold-proof "test frequency should be approximately 40 months or less ...." The cold proof test will be conducted only once because the manufacturer recommends against repeatedly overloading the crane. Thereafter, all accessible welds whose failure could cause a critical load drop will be nondestructively examined every 4 years or less. This exception is consistent with Section 2.4 of NUREG 0554.

<u>Regulatory Guide 1.105</u> - <u>INSTRUMENT SPANS AND SET POINTS (Revision 1, November 1976)</u>

Subject to the clarifications and/or exceptions indicated below, the provisions of this regulatory guide are met by the current plant design.

(1) Reference: Position C.4. The guide requires that instrumentation not exhibit certain mechanical characteristics during the environmental qualification tests performed in accordance with Regulatory Guide 1.89. These tests have not been performed for the Susquehanna SES as discussed in the response to Regulatory Guide 1.89.

(2) Reference: Position C.5. The guide requires that a securing device or equivalent be installed on all instrument set point mechanisms. All set points are provided with multiturn, "screwdriver" adjustments that have "built-in" friction to obviate the effect of vibratory inputs.

Regulatory Guide 1.106 - THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES (Revision 0, November 1975; Revision 1, March 1977)

The requirements of Revision 0, November 1975 of this regulatory guide are met for Susquehanna SES except for the 'E' diesel generator. The requirements of Revision 1, March 1977 of this regulatory guide are met for the 'E' diesel generator. Compliance is discussed in Subsection 8.1.6.1.

Regulatory Guide 1.107 - QUALIFICATIONS FOR CEMENT GROUTING FOR PRESTRESSING TENDONS IN CONTAINMENT STRUCTURES (Revision 1, February 1977

Not Applicable.

Regulatory Guide 1.108 - PERIODIC TESTING OF DIESEL GENERATORS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANT (Revision 1, August 1977) (INCLUDING ERRATA, SEPTEMBER 1977)

The design of the diesel generators are in compliance with this regulatory guide (including errata) except position C.1.b(5). Test according to requirements in Sections 14.2 and SSES Technical Specifications.

Regulatory Guide 1.109 - CALCULATION OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10CFR50, APPENDIX I (Revision 1,October, 1977)

The assumptions of Regulatory Guide 1.109 are followed in the analysis of annual doses to man from routine releases presented in Sections 11.2 and 11.3.

Regulatory Guide 1.110 - COSTS-BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT WATER COOLED NUCLEAR POWER REACTOR (March 1976)

The requirements of this regulatory guide are met.

Regulatory Guide 1.111 - METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASE FROM LIGHT WATER COOLED REACTORS (Revision 1, July 1977)

The assumptions of Regulatory Guide 1.111 are followed in the analyses of atmospheric dispersion factors presented in Section 2.3 and of deposition rates presented in Section 11.3.

Regulatory Guide 1.112 - CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT WATER COOLED POWER REACTORS (April 1976)

The requirements of Regulatory Guide 1.112 are met.

Regulatory Guide 1.113 - ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I (Revision 1, April 1977)

The requirements of Regulatory Guide 1.113 are met.

Regulatory Guide 1.114 - GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT (Revision 1, November 1976)

PP&L will be in compliance with this regulatory guide.

Regulatory Guide 1.115 - PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES (Revision 1, July 1977)

Since the construction permit for the Susquehanna station was issued in November 1973, the methods of turbine missile protection, as described in this Regulatory Guide, were not specifically considered in the design. The design methods and demonstrative analyses employed for the tangential turbine orientation of Susquehanna SES are described in Section 3.5.

Regulatory Guide 1.116 - QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS (Revision 0-R, May 1977)

The Susquehanna SES quality program for construction of safety related items was conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.117 - TORNADO DESIGN CLASSIFICATION (June 1976)

All the structures, systems, and components listed in the appendix to this regulatory guide are protected against the effects of tornadoes except that the spent fuel pool is only designed to prevent significant loss of watertight integrity caused by tornadic winds and missiles generated by these winds.

Regulatory Guide 1.118 - PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS (June 1976) and (June 1978 for the 'E' Diesel Generator only)

For compliance with this regulatory guide refer to Section 8.1.6.1.

Regulatory Guide 1.119 - SURVEILLANCE PROGRAM FOR NEW FUEL ASSEMBLY DESIGNS (June 1976)

Withdrawn June 23,1977.

Regulatory Guide 1.120 - FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS (June 1976)

Regulatory Guide 1.120 was withdrawn on August 15, 2001. Susquehanna SES is not committed to this Regulatory Guide. Refer to the Fire Protection Review Report for the Susquehanna SES Fire Protection Program Commitments.

Regulatory Guide 1.121 - BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES (August 1976)

Not Applicable.

Regulatory Guide 1.122 - DEVELOPMENT OF FLOOR DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF FLOOR-SUPPORTED EQUIPMENT OR COMPONENTS (September 1976)

The methods used for developing the floor design response spectra for Susquehanna SES are in compliance with the positions of this regulatory guide except as follows:

1. The frequencies used for the calculation of the response spectra are different and are described in Subsection 3.7b.2.5.

2. The procedure for smoothing the spectra (broadening of peaks) is different and is discussed in Subsection 3.7b.2.9.

The above exceptions are Not Applicable for the Diesel Generator 'E' Building where the methods used for developing floor response spectra are in compliance with Regulatory Guide 1.122, Rev. 1 (February 1978).

Regulatory Guide 1.123 - QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR PLANTS (Revision 1, July 1977)

The Susquehanna SES quality assurance program for the construction phase is detailed in PSAR Appendix D and amendments. Compliance of the operational Quality Assurance Program with this regulatory guide is discussed in Section 17.2.

Regulatory Guide 1.124 - DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS 1 LINEAR-TYPE COMPONENT SUPPORTS (November 1976)

Since the construction permit for Susquehanna SES was issued in November 1973, this regulatory guide was not specifically considered in the design. The methods used to determine design loading combinations for Susquehanna SES are described in Section 3.9.

Regulatory Guide 1.125 - PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR POWER PLANTS (March 1977)

No physical models were used during the design of Susquehanna SES.

Regulatory Guide 1.126 - AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION (Revision 1, March 1978)

This position will be supplied later.

Regulatory Guide 1.127 - INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS (April 1977)

Susquehanna SES will comply with this regulatory guide.

Regulatory Guide 1.128 - INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (April 1977 and October 1978 For the Diesel Generator 'E' Facility)

Installation design and installation of Class 1E batteries are in compliance with this regulatory guide except the design ambient temperature of the battery rooms is 80°F - 5°F and 104°F (Max.) to 65°F (Min.) for the Diesel Generator 'E' facility.

Regulatory Guide 1.129 - MAINTENANCE, TESTING AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (April 1977 and February 1978 for the Diesel Generator 'E' Facility)

The Susquehanna SES complies with Regulatory Guide 1.129 dated April, 1977 and February, 1978 which invoke IEEE Std. 450-1975. As a result of the conversion from the Current Technical Specification to the Improved Technical Specification, IEEE Std. 450-1995 has been established as the applicable IEEE standard for the Class 1E station battery maintenance, testing and replacement. Compliance with Regulatory Guide 1.129 dated April, 1977 and February, 1978 remains unchanged because the intent of Regulatory Guide 1.129 is not changed as a result of the commitment to IEEE Std. 450-1995.

Regulatory Guide 1.130 - DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS I PLATE AND-SHELL-TYPE COMPONENT SUPPORTS (July 1977)

Since the construction permit for Susquehanna SES was issued in November 1973, this regulatory guide was not specifically considered in the design. The methods used to determine design loading combinations for Susquehanna SES are described in Subsection 5.4.14.

Regulatory Guide 1.131 - QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (August 1977)

The electric cables, field splices, and connections for the non-NSSS scope supply are qualified in accordance with IEEE 383-1974. Exceptions to the regulatory positions are as follows:

(1) Paragraph C.2 - The design basis event conditions meet the most severe postulated conditions for Susquehanna SES. Factors for margin given in Section 6.3.1.5 of IEEE 323-1974 were not used.

(2) Paragraph C.4 - Only one aging data point (121 C) has been applied to the cables used on Susquehanna SES.

(3) Paragraph C.6 - Flame tests were done in accordance with IEEE 383-1974. No tests were performed on aged specimen.

(4) Paragraph C.10 - Gas burner position is in accordance with IEEE 383-1974.

(5) Panel internal wires are not qualified to Regulatory Guide 1.131.

The electric cables, field splices, and connections for the NSSS scope, of supply have not been evaluated against this regulatory guide.

The electric cables, field splice, and connections for the Diesel Generator 'E' facility and ties to the transfer points in each of the Diesel Generator A-D bays are qualified in accordance with IEEE 383-1974. Exceptions, to the regulatory positions are as follows:

(1) Paragraph C.2 - The design basis event conditions meet the most severe postulated conditions for the Diesel Generator 'E' facility and Diesel Generator A/D bays. Factors for margin given in Section 6.3.1.5 of IEEE 323-1974, were not used.

(2) Paragraph C.6 - Flame tests were done in accordance with IEEE 383-1974. No tests were performed on aged specimen.

(3) Paragraph C.10 - Gas burner position is in accordance with IEEE 383-1974.

(4) Paragraph C. - The gas and air pressures of Section 2.5.4.4.3 were utilized.

(5) Panel internal wires are not qualified to Regulatory Guide 1.131.

Regulatory Guide 1.137 - FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS (Revision 0, January 1978 for Diesels A-D, Revision 1, October 1979 for Diesel E)

The design of diesel generators “A-D” meet the intent of the fuel oil quality and testing requirements in Regulatory Guide 1.137, Revision 0. The design of the “E” diesel generator meets the intent of Regulatory Guide 1.137, Revision 1. The diesel generator fuel oil storage and transfer system is discussed/clarified in FSAR Section 9.5.4 and the Fuel Oil Monitoring Program.

Regulatory Guide 1.144 - AUDITING OF QA PROGRAMS (January 1979)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.145 - ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENT AT NUCLEAR POWER PLANTS (NOVEMBER 1982)

Compliance with the Regulatory Guide and Methodology used for the Atmospheric Diffusion Models is described in Section 2.3.

Regulatory Guide 1.146 - QUALIFICATION OF QA PROGRAM AUDIT PERSONNEL (August 1980)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.148 - FUNCTIONAL SPECIFICATION OF ACTIVE (DIESEL GENERATOR E ONLY) VALVE ASSEMBLIES IN SYSTEMS IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS (March 1981)

The design of the "E" Diesel Generator System meets the intent of this Regulatory Guide, which defines operating requirements for safety related valve assemblies. This regulatory guide endorses the use of ANSI Standard N278.1-1975.

Regulatory Guide 1.163 - PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM (Revision 0, July 1995)

Leakage rate testing of the primary containment for compliance with 10CFR50, Appendix J, Option B is performed in accordance with the guidance provided in this regulatory guide. This regulatory guide also endorses the methodology for testing presented in NEI 94-01 and ANSI/ANS-56.8-1994.

Regulatory Guide 1.183 - ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (JULY 2000)

Regulatory positions in this guide are complied with subject to the following exceptions and clarifications:

Reference: Section 6.0. An Alternative Source Term (AST) assessment was not performed for equipment qualification. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification and radiation zone maps/shielding calculations.

Reference: Section 3.3, Appendix A. No credit is conservatively taken in the AST analyses for fission product reduction due to the initiation of the drywell sprays.

Reference: Section 3.4, Appendix A. No credit is taken in the AST analyses for reduction of airborne radioactivity in the containment by in-containment recirculation filter systems.

Reference: Section 3.6, Appendix A. No credit is taken in the AST analyses for reduction of airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above.

Reference: Section 3.8, Appendix A. The primary containment is not routinely purged during power operations.

Reference: Section 5.4, Appendix A. The temperature of the leakage does not exceed 212°F.

Reference: Section 7.0, Appendix A. Primary containment purging as a combustible gas or pressure control measure was analyzed and not deemed to be required. Containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis.

Reference: Sections 5.1 to 5.5, Appendix B. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.

Reference: Section 2.0, Appendix C. Postulated accident assumes fuel clad failure and fuel melt.

Reference: Section 3.5, Appendix C. The MSIVs and main steam drain lines do not automatically trip closed following a CRDA. The operators manually scram the reactor and close all MSIVs and drain line valves in the event of a main steam line radiation monitor (MSLRM) high-high radiation alarm in accordance with station procedures.

<u>Regulatory Guide 1.194</u> - ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL ASSESSMENTS AT NUCLEAR POWER PLANTS (JUNE 2003)

Regulatory positions in this guide are complied with subject to the following exceptions and clarifications:

Reference: Sections 3.2.1 – 3.2.4, 3.2.4.1 – 3.2.4.8. The diffusion models used are based on point-source formulations and ground level releases.

Reference: Sections 3.3.2 – 3.3.4. The Control Room ventilation system has only a single outside air intake.

Reference: Sections 4.1 – 4.4. Source-receptor geometry, type and locations.
All ground level releases were determined per the ARCON96 methodology utilizing standard time intervals; no χ/Q correction per wind speed averaging.

Reference: Section 6.0. Plume rise was not considered in the χ/Q determinations.

Reference: Section 7.0. No experimental data was utilized to calculate the χ/Qs.

<u>Regulatory Guide 1.197</u> - DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS (May 2003)

The following regulatory portions are compiled with:

Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Section C.1 and C.2 and

(ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2.

TABLE 3.13-1

COMPARISON WITH REGULATORY GUIDE 1.48

COMPONENT	PLANT CONDITION	NRC REGULATORY GUIDE 1.48 LOADING COMBINATION 1/	NRC REGULATORY GUIDE 1.48 DESIGN LIMIT		REGULATORY GUIDE PARAGRAPH	SSES LOADING COMBINATION (F)	SSES CODE ALLOWABLE STRESSES	SSES ASME SECTION 111 REFERENCE	HOW SSES COMPARES WITH NRC REGULATORY GUIDE 1.48
Class 1 Vessels	Upset (U) Emergency (E) Faulted (F)	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	NB-3223 NB-3224 NB-3225	2/	1.a 1.b 1.c	(NPC OR UPC), 0.5 SSE EPC,.5 SSE + TRANSIENT NPC + SSE + DSL	3.0 Sm (INCLUDES SECONDARY STRESSES) 1.8 Sm APP F - SECT 111	NB-3223 NB-3224 NB-3225	GE REFLECTS INDUSTRY POSITION
Class 1 PIPING	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	NB-3654 NB-3655 NB-3656	2/	1.a 1.b 1.c	(NPC or UPC), 0.5 SSE EPC,.5 SSE + TRANSIENT NPC + SSE + DSL	3.0 Sm (INCLUDES SECONDARY STRESSES) 2.25 Sm 3.0 Sm	NB-3654 NB-3655 NB-3656	GE REFLECTS INDUSTRY POSITION
Class 1 Pumps (Inactive)	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	NB-3223 5/ NB-3224 NB-3225	1/	2.a 2.b 2.c	(NPC or UPC), 0.5 SSE EPC, .5 SSE + TRANSIENT NPC + SSE + DSL	1.65 Sm 1.8 Sm APP F - SECTION 111	NB-3223 NB-3224 NB-3225	GE REFLECTS INDUSTRY POSITION
Class 1 Pumps (Active)	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	NB-3222 NB-3222 NB-3222	5/ 6/ 7/ 8/	4.a 4.a 4.a	(NPC or UPC), 0.5 SSE EPC NPC + SSE + DSL	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
Class 1 Valves (Inactive) By Analysis	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	NB-3223 5/ NB-3224 2/ NB-3225 2/	4/	2.a 2.b 2.c	(NPC or UPC), 0.5 SSE EPC NPC + SSE + DSL	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
Class 1 Valves (Inactive) Designed by Either Std or Alternative Design Rules	U E F	(NPC or UPC) + 0.5 SSE EPC NPC + SSE + DSL	1.1 Pr 1.2 Pr 1.5 Pr		3.a 3.b 3.c	(NPC or UPC), 0.5 SSE EPC, .5 SSE + TRANSIENT NPC + SSE + DSL	1.1 Pr 1.2 Pr 1.5 Pr	NB-3525 NB-3526 NB-3527	GE REFLECTS INDUSTRY POSITION
Class 1 Valves (Active) By Analysis	U E F	(NPC or UPC) 0.5 SSE EPC NPC + SSE + DSL	NB-3222 NB-3222 NB-3222	5/ 6/ 7/ 8/	4a.1 4a.2 4a.3	(NPC or UPC), 0.5 SSE EPC NPC + SSE + DSL	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
Class 1 Valves (Active) Designed by Std or Alternative Design Rules	U E F	(NPC or UPC) 0.5 SSE EPC NPC • SSE + DSL	1.0 Pr 1.0 Pr 1.0 Pr	6/	5a.1 5a.2 5a.3	(NPC or UPC), 0.5 SSE EPC, .5 SSE + TRANSIENT NPC + SSE + DSL	1.0 Pr 1.0 Pr 1.0 Pr (a)	NB-3525 NB-3526 NB-3527	GE REFLECTS INDUSTRY POSITION

TABLE 3.13-1 (Continued)

COMPARISON WITH REGULATORY GUIDE 1.48

COMPONENT	PLANT CONDITION	NRC REGULATORY GUIDE 1.48: LOADING COMBINATION 1/	NRC REGULATORY GUIDE 1.48: DESIGN LIMIT	REGULATORY GUIDE PARAGRAPH	SSES: LOADING COMBINATION (F)	SSES: CODE ALLOWABLE STRESSES	SSES: ASME SECTION 111 REFERENCE	HOW SSES COMPARES WITH NRC REGULATORY GUIDE 1.48
Class 2 & 3 Vessels (Division 1) of Section VIII of the ASME Code	U	(NPC or UPC) + 0.5 SSE	1.1S 9/	6a	(NPC or UPC), 0.5 SSE	σm 1.15 (c)	CODE CASE 1607	FAULTED CONDITION NRC MORE CONSERVATIVE GE REFLECTS INDUSTRY POSITION
	E	EPC	1.1S	6b	EPC, .5 SSE + TRANSIENT		NC/NB3321 1 0b)	
	F	NPC + SSE + DSL	1.5S	6c	NPC + SSE + DSL	σm 2.08		
Class 2 Vessels (Divison 2) of Section VIII of the ASME Code	U	(NPC or UPC) + 0.5 SSE	NB-3223 2/	7a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3224	7b	EPC			
	F	NPC + SSE + DSL	NB-3225	7c	NPC + SSE + DSL			
Class 2 & 3 Piping	U	(NPC or UPC) + 0.5 SSE	NC3611.1(b) (4) (c) (b) (1)	8a	(NPC or UPC), 0.5 SSE	1.2 Sh	NC/ND3611 3(b) NC/ND3611 3(c) (4) (b) (b)	NRC MORE CONSERVATIVE
	E	EPC	NC3611.1(b) (4) 10/ (c) (b) (1)	8a	EPC,.5 SSE + TRANSIENT	1.8 Sh	CODE CASE 1606	GE REFLECTS INDUSTRY POSITION
	F	NPC + SSE + DSL	NC3611.1(b) (4) (c) (b) (2)	8b	NPC + SSE + DSL	2.4 Sh	NC/ND3611.3(d) (SEE NOTE (b))	
Class 2 & 3 Pumps (Inactive)	U	(NPC or UPC) + 0.5 SSE	$\sigma m < 1.1S > \frac{\sigma m + \sigma b}{1.5}$	9.a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	$\sigma m < 1.1S > \frac{\sigma m + \sigma b}{1.5}$	9.a	EPC			
	F	NPC + SSE + DSL	$\sigma m < 1.2S > \frac{\sigma m + \sigma b}{1.5}$	9.b	NPC + SSE + DSL			
Class 2 & 3 Pumps '(Active)	U	(NPC or UPC) + 0.5 SSE	$\sigma m < 1.0S > \frac{\sigma m + \sigma b}{1.5}$	10.a	(NPC or UPC), 0.5 SSE	σm 1.1S	CODE CASE 1636	GE RELECTS INDUSTRY POSITION
	E	EPC	$\sigma m < 1.0S > \frac{\sigma m + \sigma b}{1.5}$ 11/	10.a	EPC,.5 SSE + TRANSIENT	(e) (c)	NC/ND 3423	
	F	NPC + SSE + DSL	$\sigma m < 1.0S > \frac{\sigma m + \sigma b}{1.5}$	10.a	NPC + SSE + DSL	σm 1.2S	(See Note (b))	
Class 2 & 3 Valves (Inactive)	U	(NPC or UPC) + 0.5 SSE	.1 Pr	11.a	(NPC or UPC), 0.5 SSE	σm 1.1S (c)	CODE CASE 1636	EQUALLY CONSERVATIVE
	E	EPC	1.1 Pr	11.a	EPC,.5 SSE + TRANSIENT	(c)	NC/ND 3521	
	F	NPC • SSE + DSL	1.2 Pr	11.b	NPC + SSE + DSL	σm 2.0m	(See Note (b))	
Class 2 & 3 Valves (Active)	U	(NPC or UPC) + 0.5 SSE	1.0 Pr	12.a	(NPC or UPC), 0.5 SSE	σm 1.1S	CODE CASE 1636	EQUALLY CONSERVATIVE (a)
	E	EPC	1.0 Pr. 11/	12.a	EPC,.5 SSE + TRANSIENT	(a)	NC/ND 3621	
	F	NPC + SSE + DSL	1.0 Pr	12.a	NPC + SSE + DSL	σm 1.2S (c)	(See Note (b))	

TABLE 3.13-1 (CONTINUED)

NOTES FOR COMPARISON TABLE

1. Numerical indicators in the regulatory guide portion of the table (1/, 2/, etc.) correspond to footnotes of the Regulatory Guide 1.48.

2. Alphabetical indicators in Susquehanna SES portion of table (or comparative column) correspond to the following:

   a. In addition to compliance with the design limits specified, assurance of operability under all design loading combinations shall be in accordance with Subsection 3.9.3.2.

   b. Referenced paragraphs of code currently in course of preparation.

   c. The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150% of that allowed for general membrane (except as limited to 2.4S for inactive components under faulted condition). Refer to Subsection 3.9.5.2.

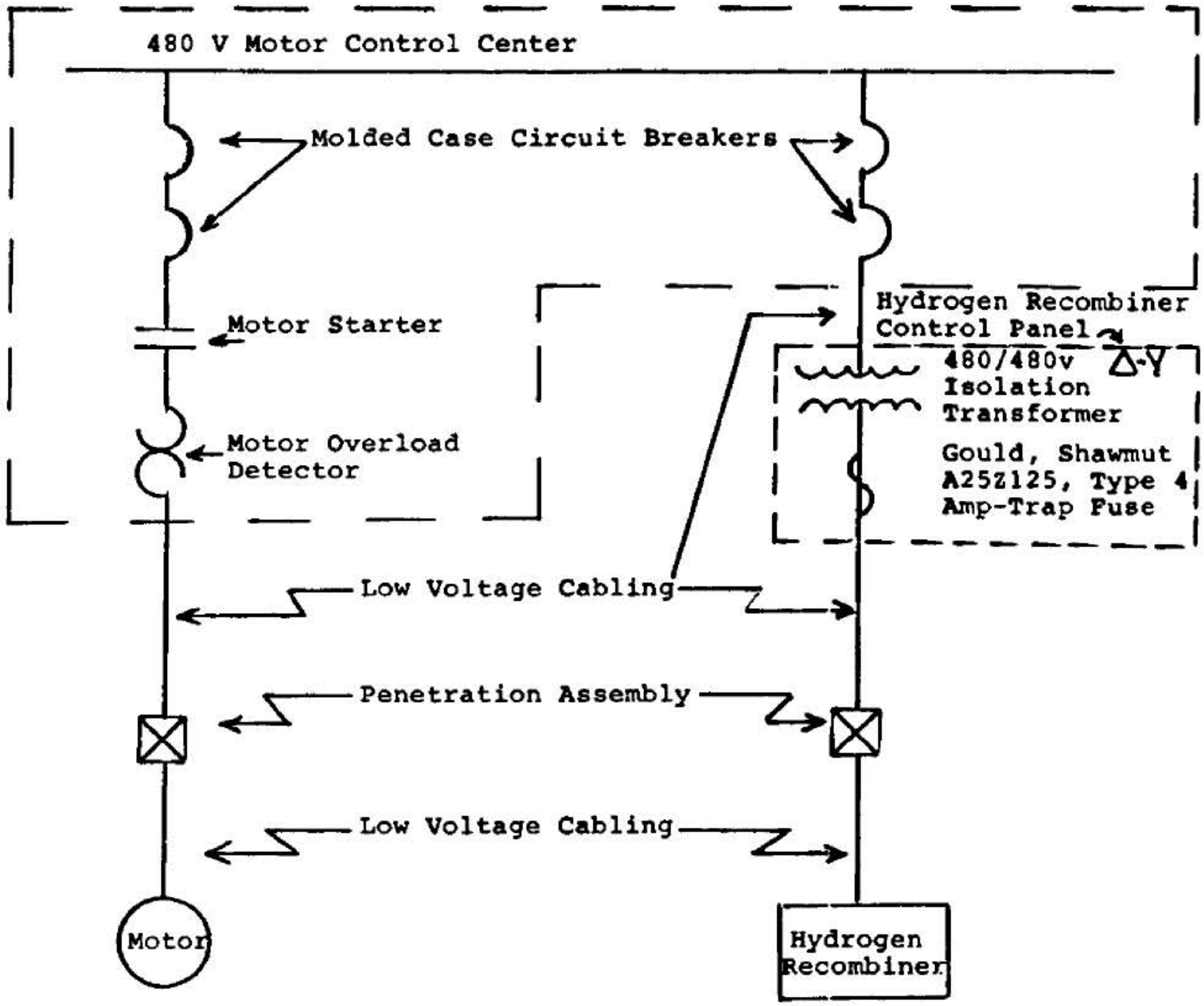
   d. Not used.

   e. Inactive limits may be used since operability will be demonstrated in accordance with Subsection 3.9.3.2.

   f. When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in Table 3.9.2(a).

3. Acronyms

   UPC Upset plant conditions
   NPC Normal plant conditions
   EPC Emergency plant conditions
   DSL Dynamic System Loadings
   SSE Safe Shutdown Earthquake



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RELATIONSHIP OF LOADS,
PENETRATION ASSEMBLIES &
PROTECTIVE DEVICES FOR CURVES
PRESENTED IN FIGURES
3.13-2, 3.13-3 & 3.18-8

FIGURE 3.13-1, Rev. 47

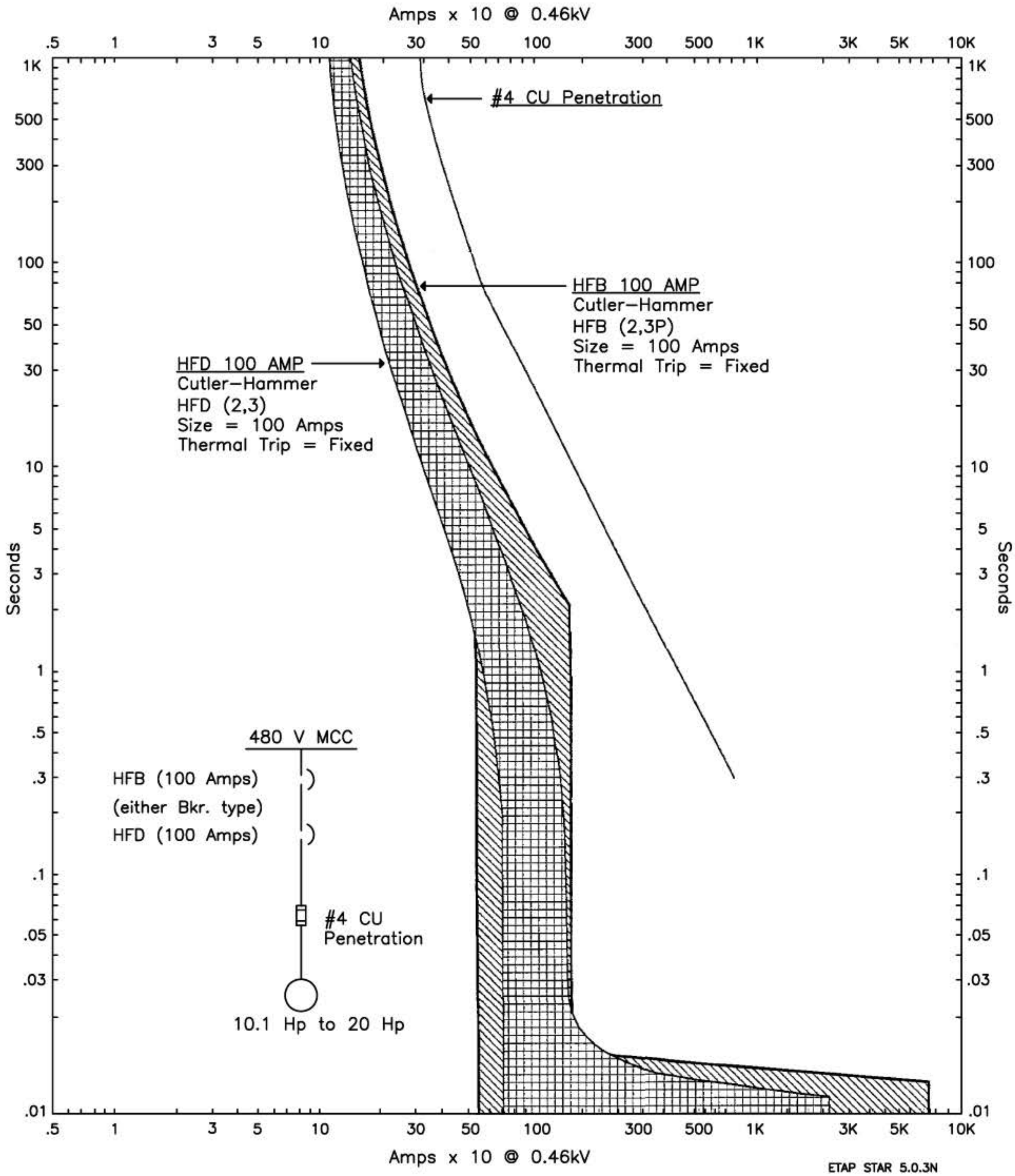
Auto-Cad Figure Fsar 3_13_1.dwg

THIS FIGURE HAS BEEN
REPLACED BY FIGURE 3.13-2A & FIGURE 3.13-2B

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Replaced by FIGURE 3.13-2A & FIGURE 3.13-2B FIGURE 3.13-2, Rev. 48

AutoCAD; Figure Fsar 3_13_2.doc



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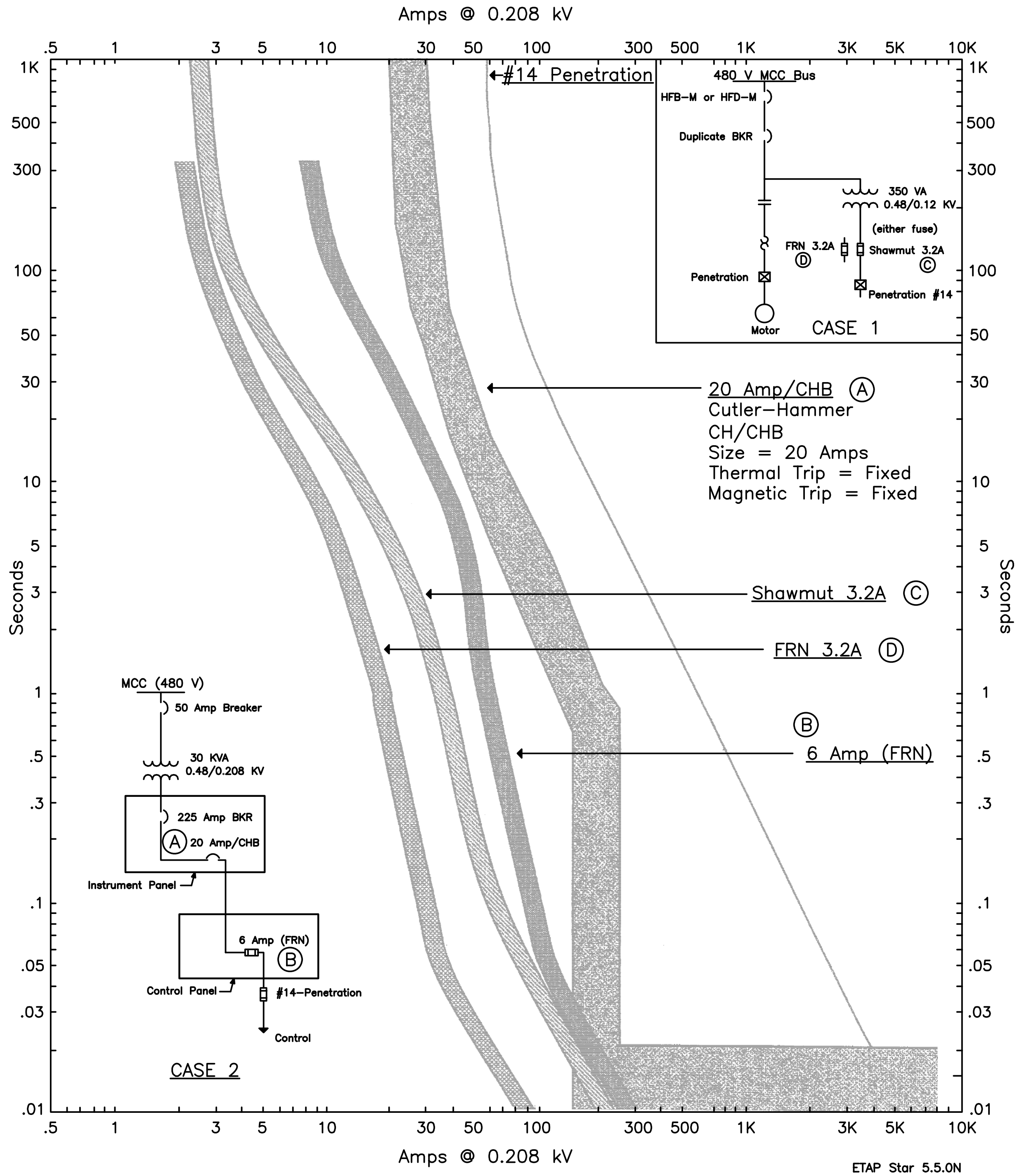
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#4 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-3, Rev. 48

Auto-Cad Figure Fsar 3_13_3.dwg



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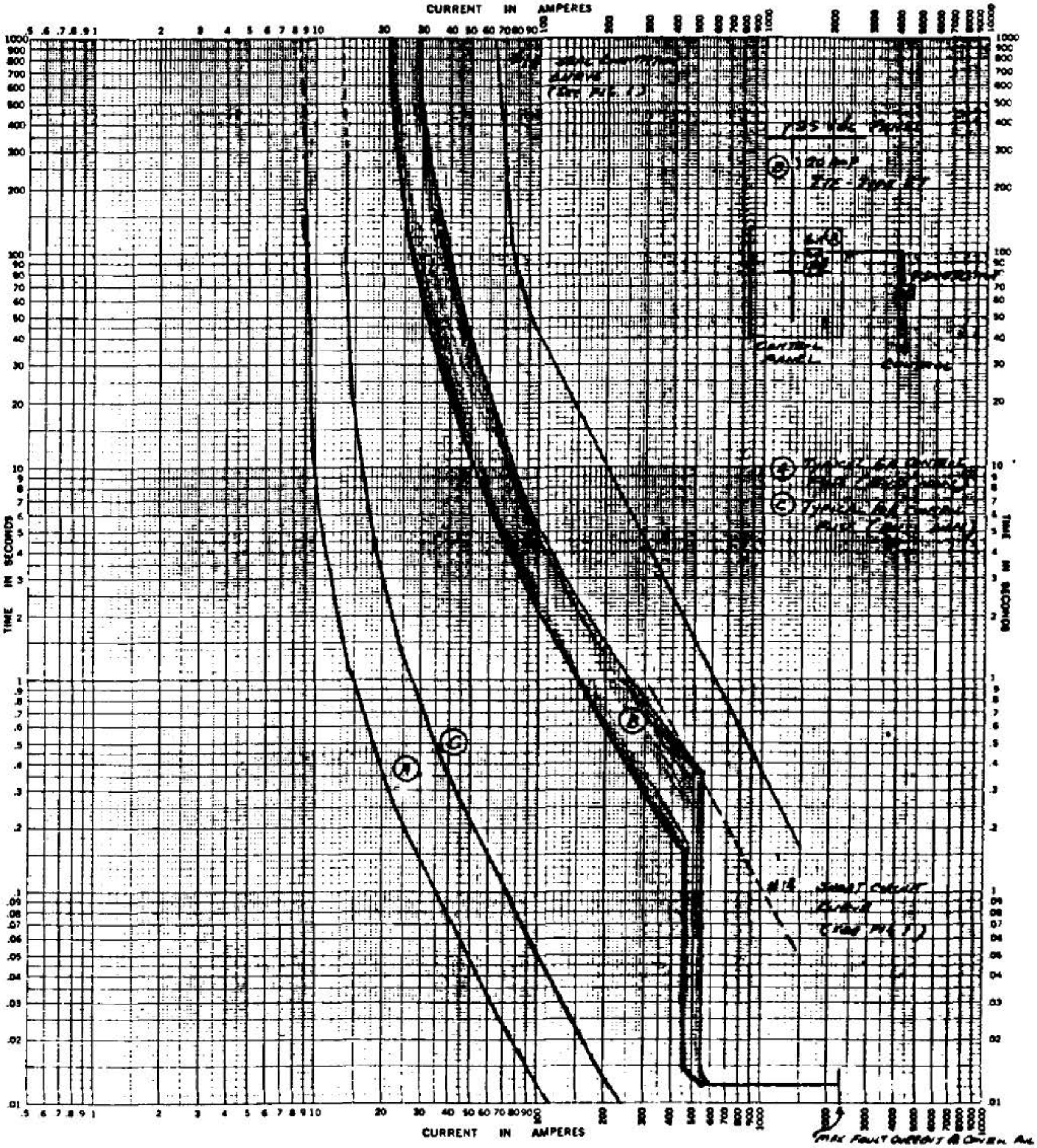
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

120V AC
CONTROL CIRCUITS

FIGURE 3.13-4, Rev. 48

Auto-Cad Figure Fsar 3_13_4.dwg



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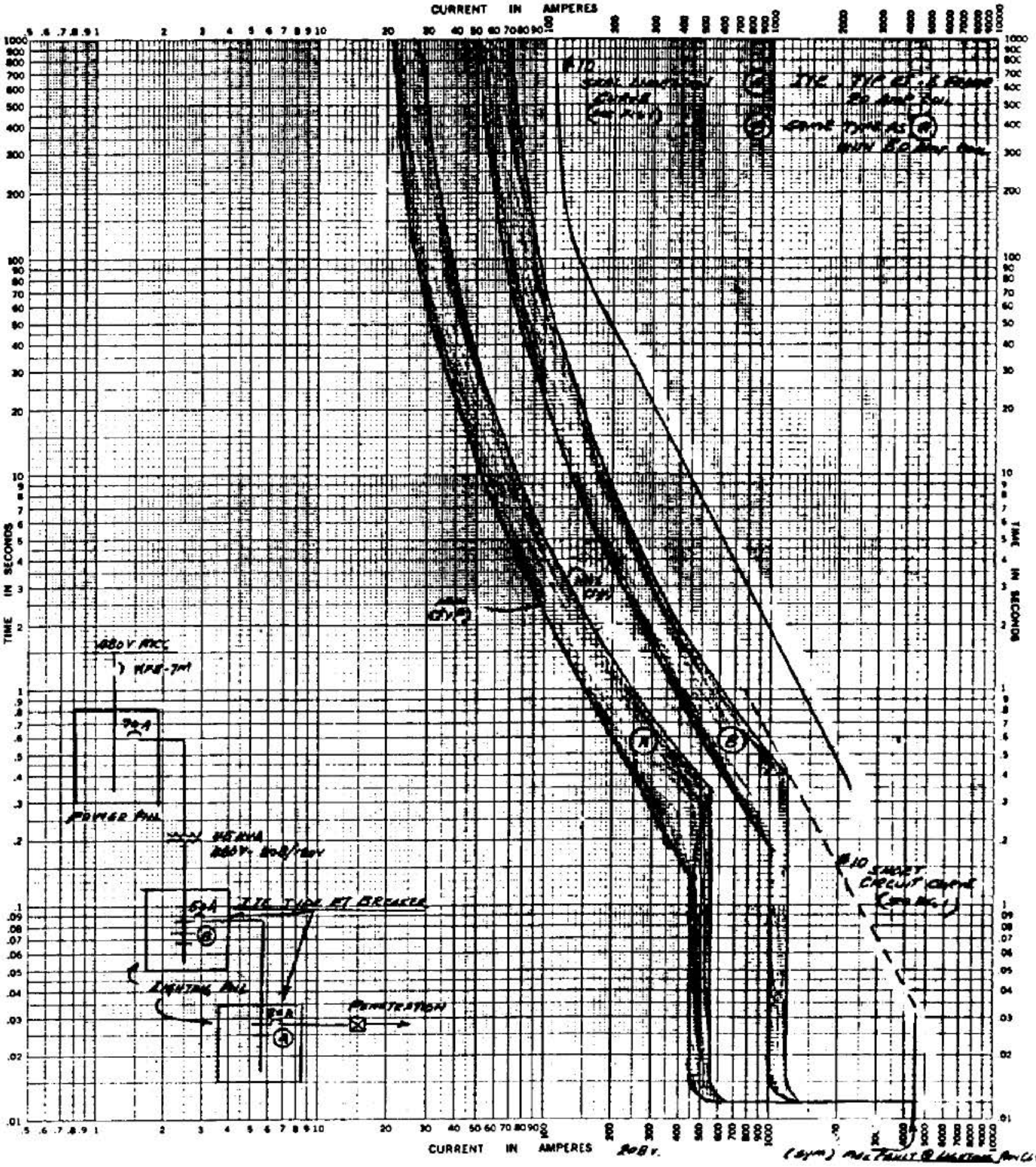
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

125 VOLT DC CONTROL CIRCUITS

FIGURE 3.13-5, Rev. 47

Auto-Cad Figure Fsar 3_13_5.dwg



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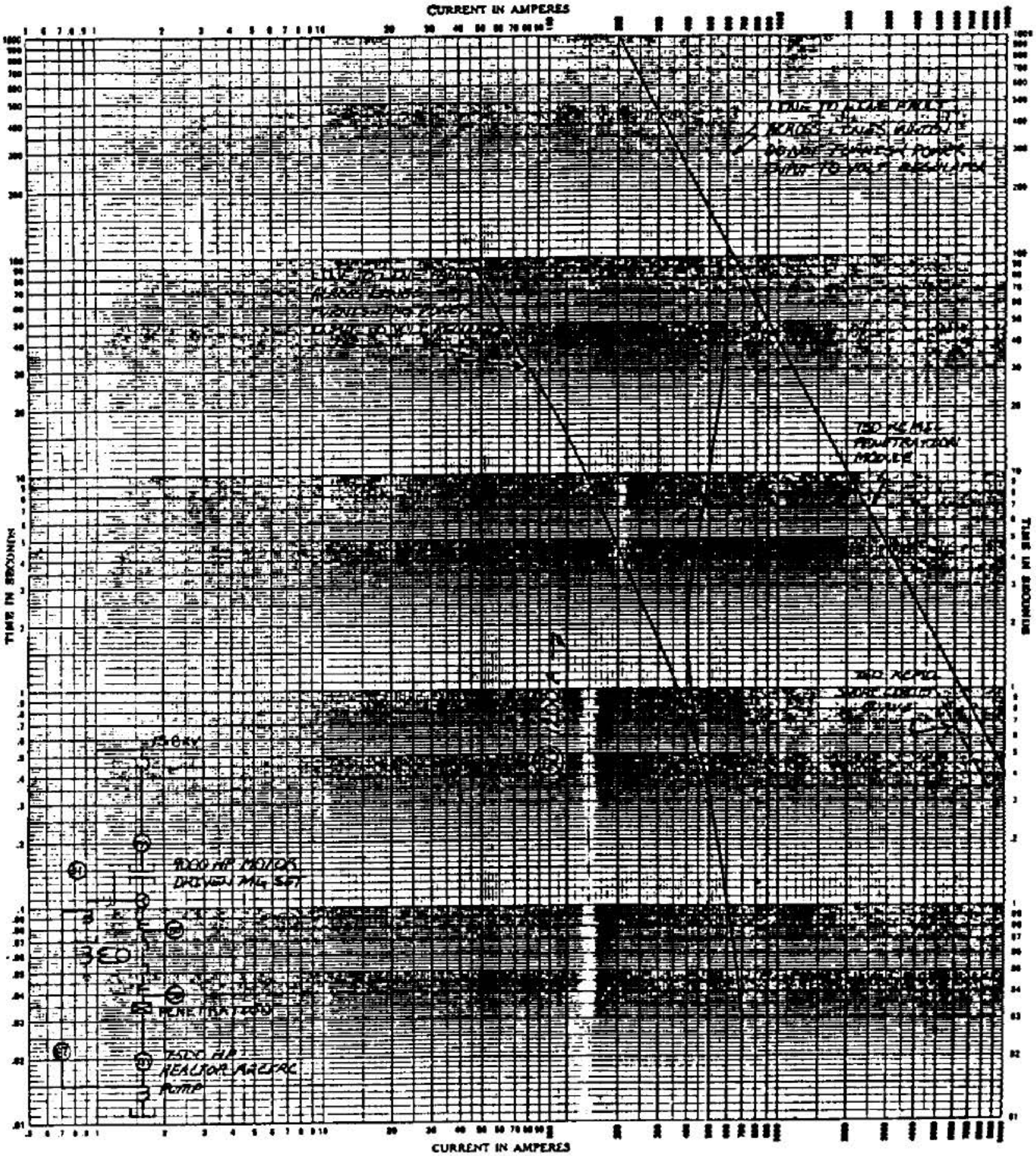
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

208/120 LIGHTING CIRCUITS

FIGURE 3.13-6, Rev. 47

Auto-Cad Figure Fsar 3_13_6.dwg



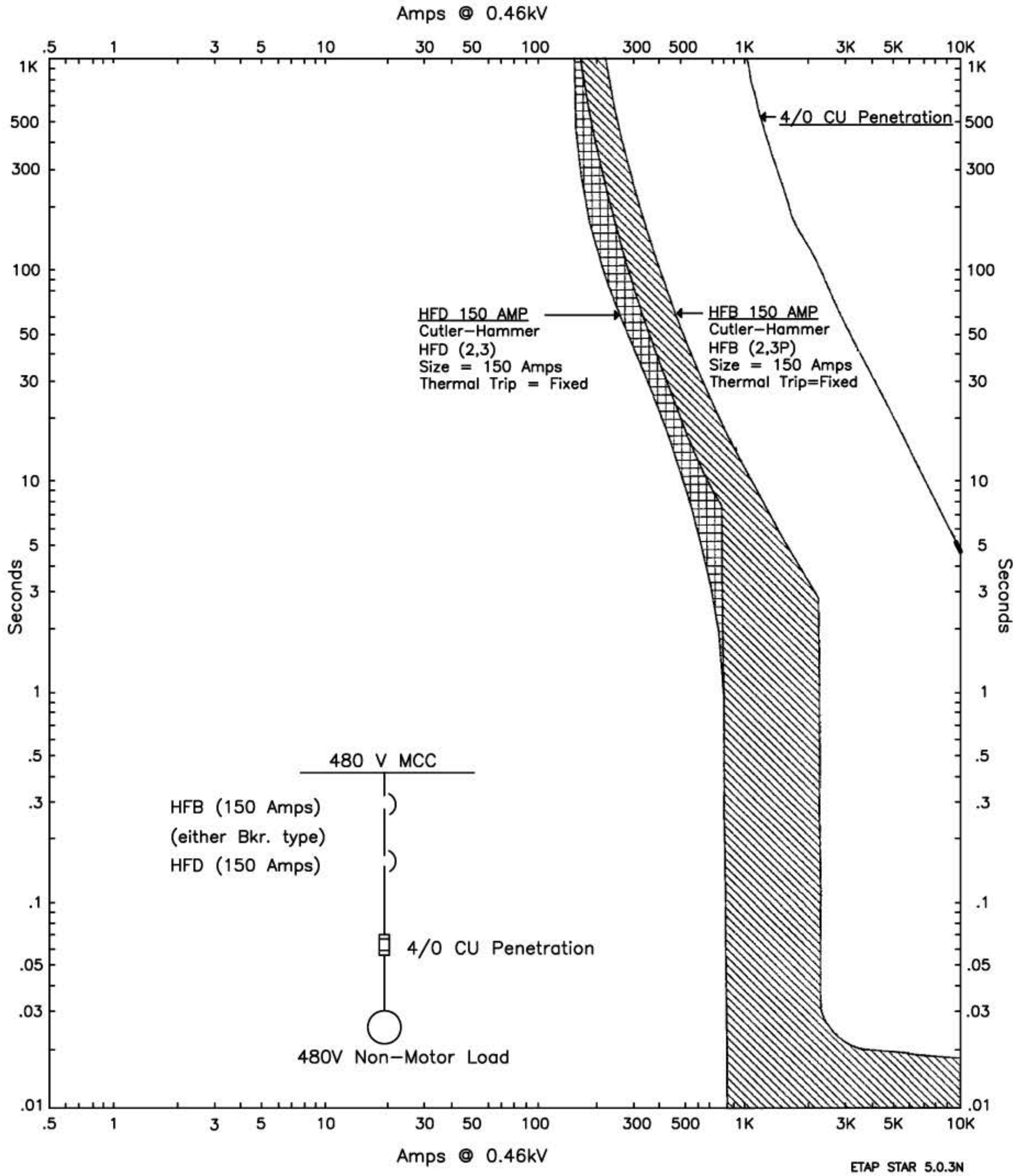
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR RECIRCULATION PUMP-
MAXIMUM SHORT
CIRCUIT CAPABILITY

FIGURE 3.13-7, Rev. 47

Auto-Cad Figure Fsar 3_13_47.dwg



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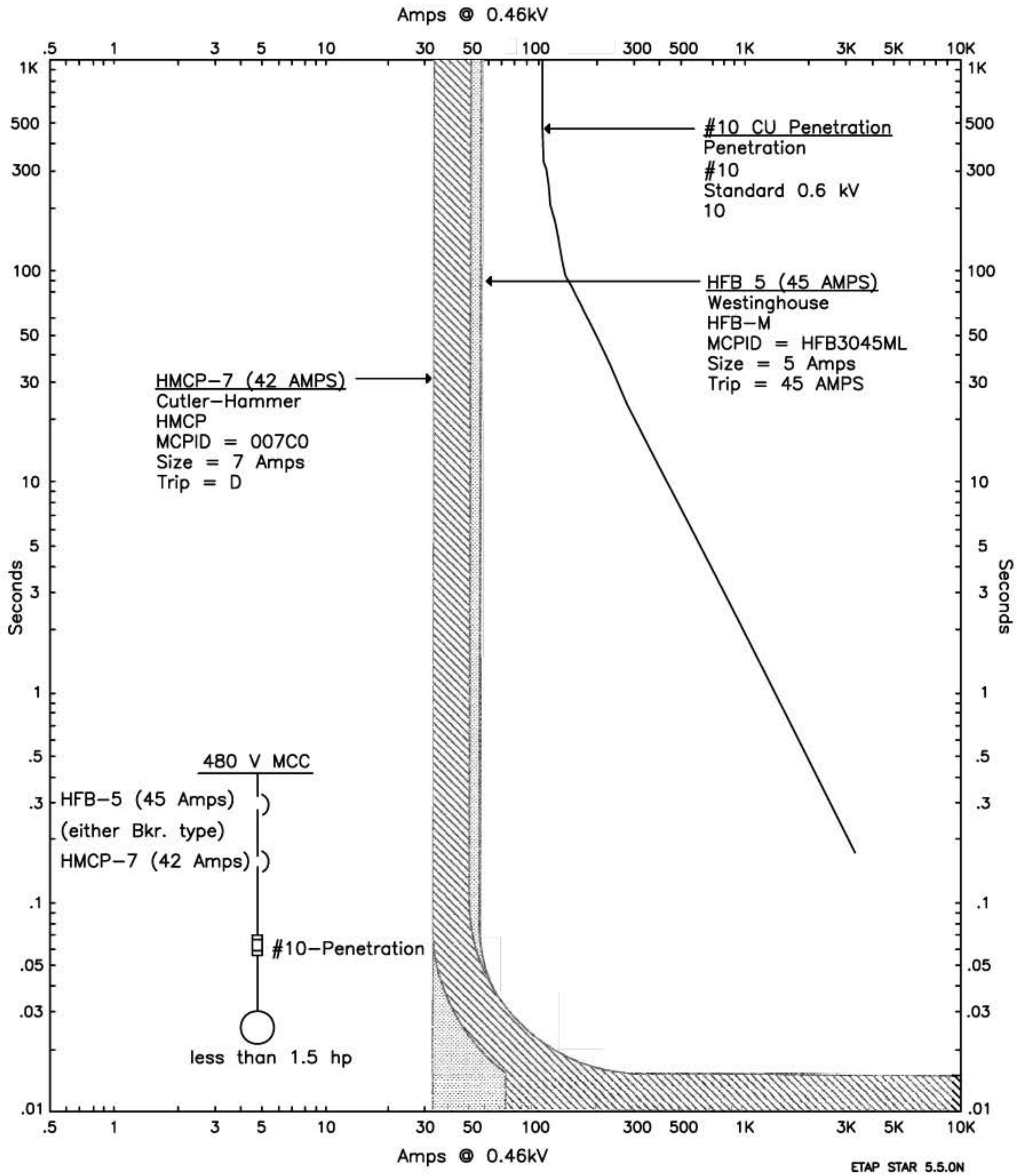
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT CHARACTERISTIC
CURVES FOR OVERCURRENT
PROTECTION OF 4/0 COPPER,
CONTAINMENT PENETRATIONS
USED WITH HYDROGEN RECOMBINERS

FIGURE 3.13-8, Rev. 48

Auto-Cad Figure Fsar 3_13_8.dwg



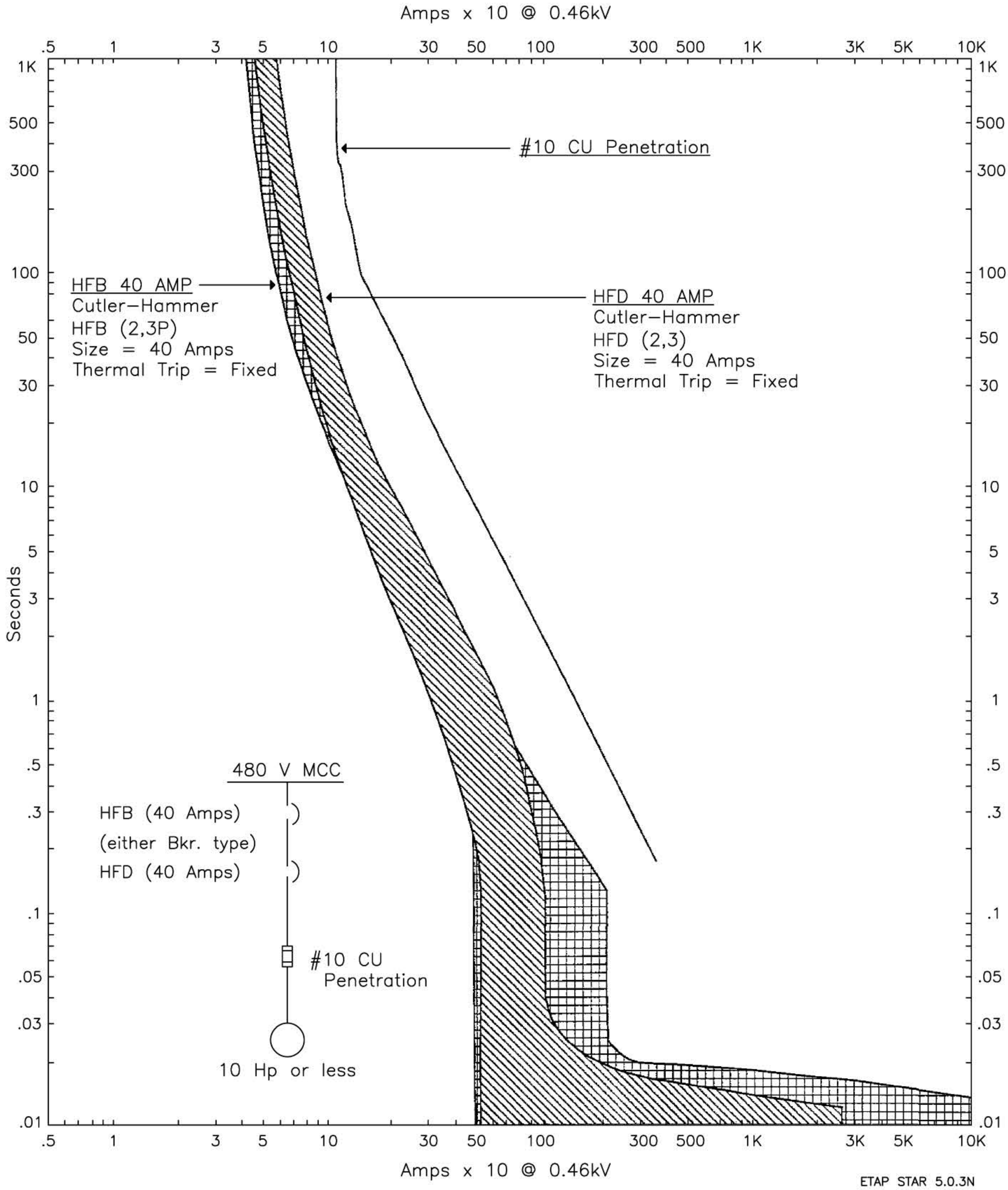
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#10 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-2A, Rev. 1

Auto-Cad Figure Fsar 3_13_2A.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#10 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-2B, Rev. 1

Auto-Cad Figure Fsar 3_13_2B.dwg

3.14 LICENSE RENEWAL PROGRAMS, TLAA, AND COMMITMENTS

3.14.1 INTRODUCTION

The License Renewal Rule, 10 CFR Part 54, governs the issuance of renewed operating licenses for nuclear power plants. The renewed SSES operating licenses were granted on November, 24, 2009 to be effective until July 17, 2042 for Unit 1 and March 23, 2044 for Unit 2.

This section contains the license renewal information to be included in the Final Safety Analysis Report as required by 10 CFR 54.21(d). The Safety Evaluation Report (SER) NUREG-1931, for renewed operating licenses for SSES Units 1 & 2 contains the results of the NRC review of technical information required by 10 CFR 54.21(a) and (c) as submitted in the SSES License Renewal Application (LRA) and supplemental RAI responses. The programs and activities that will be implemented to manage the effects of aging for the period of extended operation are listed throughout NUREG-1931. In addition, this section contains a listing of commitments associated with license renewal as identified in Appendix A of NUREG-1931.

3.14.2 AGING MANAGEMENT PROGRAMS (AMP)

The license renewal integrated plant assessment identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the required aging management programs identified during the integrated plant assessment. Except for one-time inspections, these programs will be implemented during the period of extended operation. One-time inspections will be conducted within the 10-year period prior to beginning the period of extended operation.

Three elements of an effective aging management program that are common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the SSES Operational Quality Assurance (OQA) Program, which implements the requirements of 10 CFR 50, Appendix B. Prior to the period of extended operation, the elements of corrective action, confirmation process, and administrative controls in the SSES OQA Program will be applied to required aging management programs for both safety-related and nonsafety-related structures and components determined to require aging management during the period of extended operation.

3.14.2.1 Area-Based NSAS Inspection (NUREG 1931, Section 3.0.3.3.1)

The Area-Based NSAS Inspection detects and characterizes the conditions on the internal surfaces of nonsafety-related components exposed to non-radioactive equipment/area drainage water or potable water environments. The Area-Based NSAS Inspection also detects and characterizes specific conditions on the internal surface of copper alloys exposed to raw water from the spray pond/cooling tower. Components identified as non-safety affecting safety (NSAS) are those nonsafety-related components with the potential to prevent a safety-related system or component from performing its safety function. The conditions in these environments are not expected to result in sufficient degradation to cause spatial interaction with safety-related components or are expected to result in effects that progress very slowly. To ensure

that spatial interactions do not occur that impair or prevent a safety-related function, a focused characterization of conditions is performed to provide confirmation of a lack of degradation or to serve as the basis for recurring actions during the period of extended operation, if required.

The Area-Based NSAS Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.2 Bolting Integrity Program (NUREG 1931, Section 3.0.3.2.5)

The Bolting Integrity Program is a combination of existing SSES activities that, in conjunction with other credited programs, addresses the management of aging for the bolting of subject mechanical components within the scope of license renewal. The Bolting Integrity Program relies on manufacturer/vendor information and industry recommendations for the proper selection, assembly and maintenance of bolting for pressure-retaining closures. The Bolting Integrity Program includes, through the Inservice Inspection (ISI) Program and System Walkdown Program, the periodic inspection of bolting for indication of degradation such as leakage, loss of material due corrosion, or cracking.

Prior to the period of extended operation, the Bolting Integrity Program will be enhanced to include a specific precaution against the use of sulfur (sulfide) containing compounds as a lubricant for bolted connections.

3.14.2.3 Buried Piping and Tanks Inspection Program (NUREG 1931, Section 3.0.3.2.13)

The Buried Piping and Tanks Inspection Program manages the effects of corrosion on the external surfaces of piping and tanks exposed to a buried environment. The Buried Piping and Tanks Inspection Program will be a combination of a prevention program (consisting of protective coatings and wrappings, where appropriate) and a condition monitoring program (consisting of visual inspections).

The Buried Piping and Tanks Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.4 Buried Piping Surveillance Program (NUREG 1931, Section 3.0.3.2.10)

The Buried Piping Surveillance Program manages the effects of corrosion on the external surfaces of piping with damaged coatings exposed to a buried environment. The Buried Piping Surveillance Program will be a combination of a prevention program (consisting of cathodic protection) and a condition monitoring program (consisting of periodic testing).

The Buried Piping Surveillance Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.5 BWR CRD Return Line Nozzle Program (NUREG 1931, Section 3.0.3.2.2)

The BWR CRD Return Line Nozzle Program is an existing program that manages cracking of the control rod drive return line (CRDRL) nozzle, cap, and connecting weld. The program was developed in response to industry events involving the CRD return line nozzle.

SSES modified the CRD System by cutting the CRDRL and capping the nozzle prior to initial plant startup. The CRDRL was not rerouted. SSES has completed all of the requirements specified in NUREG-0619 for the CRD System modifications performed at SSES, including the final liquid penetrant testing (PT) inspection and system performance testing. The SSES BWR CRD Return Line Nozzle Program monitors the effects of cracking on the intended function of the CRDRL nozzle by performing inservice inspections in conformance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB, Table IWB 2500-1 (edition and addenda described in 3.14.2.23). Any cracks that are detected will be dispositioned in accordance with the requirements of ASME Section XI.

3.14.2.6 BWR Feedwater Nozzle Program (NUREG 1931, Section 3.0.3.1.4)

The BWR Feedwater Nozzle Program is an existing program that manages cracking of the feedwater nozzles.

The program includes (a) enhanced inservice inspection in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB, Table IWB 2500-1 (edition and addenda described in 3.14.2.23) and the recommendations of report GE-NE-523-A71-0594, and (b) system modifications (completed on the spargers prior to initial startup) to mitigate cracking. The program specifies periodic ultrasonic inspection of critical regions of the feedwater nozzles.

3.14.2.7 BWR Penetrations Program (NUREG 1931, Section 3.0.3.2.3)

The BWR Penetrations Program is an existing program that manages cracking of selected reactor vessel penetrations. The BWR Penetrations Program is implemented via the Inservice Inspection (ISI) Program in compliance with ASME Section XI and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines.

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of NRC-approved BWRVIP reports and BWRVIP-27-A, BWRVIP-47-A, BWRVIP-49-A, and BWRVIP-74-A and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 to ensure the long-term integrity and safe operation of reactor vessel internal components. The BWRVIP-27-A report addresses the standby liquid control system nozzle or housing, the BWRVIP-47-A report addresses the control rod drive and flux monitor penetrations in the lower plenum, the BWRVIP-49-A report provides guidelines for instrument penetrations, and the BWRVIP-74-A report addresses the reactor vessel flange leak off penetrations and the reactor vessel drain penetrations.

3.14.2.8 BWR Stress Corrosion Cracking (SCC) Program (NUREG 1931, Section 3.0.3.1.5)

The BWR Stress Corrosion Cracking (SCC) Program is an existing program that manages stress corrosion cracking for stainless steel piping, valves, flow instruments, and pump casings. The program to manage stress corrosion cracking in pressure boundary piping made of stainless steel is delineated in NUREG-0313, Revision 2, and NRC Generic Letter 88-01 and its Supplement 1.

The program includes (a) preventive measures to mitigate intergranular stress corrosion cracking (IGSCC), and (b) inspection and flaw evaluation to monitor IGSCC and its effects. The

NRC-approved report BWRVIP-75-A allows for modifications of inspection scope in the Generic Letter 88-01 program.

3.14.2.9 BWR Vessel ID Attachment Welds Program (NUREG 1931, Section 3.0.3.1.3)

The BWR Vessel ID Attachment Welds Program is an existing program that manages cracking of the welds for internal attachments to the reactor pressure vessel.

The program includes (a) inspection and flaw evaluation in accordance with the guidelines of the NRC-approved report BWRVIP-48-A, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 to ensure the long-term integrity and safe operation of the vessel inside diameter (ID) attachment welds. The BWR Vessel ID Attachment Welds Program is based on the inspection and flaw evaluation guidelines of the BWRVIP, and is implemented by the Inservice Inspection (ISI) Program in accordance with the ASME Code, Section XI, Table IWB 2500-1.

3.14.2.10 BWR Vessel Internals Program (NUREG 1931, Section 3.0.3.2.4)

The BWR Vessel Internals Program is an existing program that manages aging of the reactor vessel internals in accordance with the requirements of ASME Section XI and the BWRVIP documents. The purpose of the BWR Vessel Internals Program is to manage cracking, loss of material, and reduction of fracture toughness for various subcomponents of the reactor vessel internals.

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and NRC-approved BWRVIP reports, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 to ensure the long-term integrity and safe operation of reactor vessel internal components.

3.14.2.11 BWR Water Chemistry Program (NUREG 1931, Section 3.0.3.1.1)

The BWR Water Chemistry Program is an existing program that mitigates damage due to loss of material and cracking of plant components that are within the scope of license renewal and contain treated water. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material or cracking through proper monitoring and control consistent with pertinent EPRI water chemistry guidelines. The SSES BWR Water Chemistry Program currently implements BWRVIP-29 and is in the process of incorporating the recommendations of BWRVIP-130. The relevant conditions are specific parameters such as sulfates, halogens, dissolved oxygen, and conductivity that could lead to or are indicative of conditions for, corrosion, erosion, or stress corrosion cracking (SCC) of susceptible materials. The BWR Water Chemistry Program is a mitigation program.

The BWR Water Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the BWR Water Chemistry Program in mitigating the effects of aging.

3.14.2.12 Chemistry Program Effectiveness Inspection (NUREG 1931, Section 3.0.3.1.10)

The Chemistry Program Effectiveness Inspection detects and characterizes the condition of materials in representative low flow and stagnant areas of plant systems influenced by the BWR

Water Chemistry Program, the Closed Cooling Water Chemistry Program, and the Fuel Oil Chemistry Program. The inspection provides direct evidence as to whether, and to what extent, cracking or a loss of material has occurred. The Chemistry Program Effectiveness Inspection will provide confirmation of the effectiveness of the BWR Water, Closed Cooling Water, and Fuel Oil Chemistry Programs in managing the effects of aging.

The Chemistry Program Effectiveness Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.13 Closed Cooling Water Chemistry Program (NUREG 1931, Section 3.0.3.2.7)

The Closed Cooling Water Chemistry Program is an existing program that mitigates damage due to loss of material and cracking of plant components that are within the scope of license renewal and that contain treated water in a closed cooling water system or component (e.g., a heat exchanger) served by a closed cooling water system. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material or cracking through proper monitoring and control of corrosion inhibitor concentrations consistent with the pertinent EPRI water chemistry guideline. The Closed Cooling Water Chemistry Program is a mitigation program.

The Closed Cooling Water Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the Closed Cooling Water Chemistry Program in mitigating the effects of aging.

3.14.2.14 Condensate and Refueling Water Storage Tanks Inspection (NUREG 1931, Section 3.0.3.1.9)

The Condensate and Refueling Water Storage Tanks Inspection detects and characterizes the conditions on the bottom surfaces of the Condensate Storage Tanks and the Refueling Water Storage Tank. The inspection provides direct evidence through volumetric and/or visual examination as to whether, and to what extent, a loss of material due to crevice, general or pitting corrosion has occurred or is likely to occur in inaccessible areas (i.e., tank base/bottom) that could result in a loss of intended function.

The Condensate and Refueling Water Storage Tanks Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.15 Containment Leakage Rate Test Program (NUREG 1931, Section 3.0.3.1.22)

The Containment Leakage Rate Test Program is an existing program that manages aging effects for the Primary Containment and systems penetrating the Primary Containment, which are the containment liner and Primary Containment penetrations including personnel airlock, equipment hatches, and control rod drive hatch. The Containment Leakage Rate Test Program provides assurance that leakage from the Primary Containment will not exceed maximum values for containment leakage.

3.14.2.16 Cooling Units Inspection (NUREG 1931, Section 3.0.3.1.11)

The Cooling Units Inspection detects and characterizes the condition of aluminum, carbon steel, copper alloy, and stainless steel cooling unit components that are exposed to a ventilation environment or to an uncontrolled raw water environment from cooling unit drain pans, and of certain heat exchanger components exposed to treated water or ventilation environments in the Control Structure Chilled Water, the Primary Containment Atmosphere Circulation, and the Control Structure and Reactor Building HVAC systems. The inspection provides direct evidence as to whether, and to what extent, a loss of material due to crevice, galvanic, general, or pitting corrosion, or reduction in heat transfer due to fouling of heat exchanger tubes and fins, has occurred or is likely to occur in these systems that could result in a loss of intended function.

The Cooling Units Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.17 Crane Inspection Program (NUREG 1931, Section 3.0.3.1.8)

The Crane Inspection Program is an existing program that manages loss of material for cranes (including bridge, trolley, rails, and girders), monorails, and hoists within the scope of license renewal. The Crane Inspection Program is based on guidance contained in ANSI B30.2 for overhead and gantry cranes, ANSI B30.11 for monorail systems and underhung cranes, and ANSI B30.16 for overhead hoists. The inspections monitor structural members for signs of corrosion other than minor surface corrosion.

3.14.2.18 Fire Protection Program (NUREG 1931, Section 3.0.3.2.8)

The Fire Protection Program is an existing program that is described in the Fire Protection Review Report (FPRR) and which is credited with aging management of components with fire barrier functions in the scope of license renewal. Periodic visual inspections and functional tests are performed, as appropriate, of fire dampers, fire barrier walls, ceilings and floors, fire rated penetration seals (fire stops), fire wraps, fireproofing, and fire doors to ensure that functionality and operability are maintained. The Fire Protection Program is a condition monitoring program, comprised of tests and inspections generally in accordance with the applicable National Fire Protection Association (NFPA) recommendations.

3.14.2.19 Fire Water System Program (NUREG 1931, Section 3.0.3.2.9)

The Fire Water System Program (sub-program of the overall Fire Protection Program) is an existing program that is described in the Fire Protection Review Report (FPRR) and which is credited with aging management of the water suppression components in the scope of license renewal. Periodic inspection and testing of the water-based fire suppression systems provides reasonable assurance that the systems will remain capable of performing their intended function. Periodic inspection and testing activities include hydrant and hose station inspections, fire main flushing, flow tests, and sprinkler inspections. The Fire Water System Program is a condition monitoring program, comprised of tests and inspections generally in accordance with the applicable National Fire Protection Association (NFPA) recommendations.

Prior to the period of extended operation, the Fire Water System Program will be enhanced to incorporate:

a) Sprinkler head sampling/replacements, in accordance with NFPA 25;

b) Ultrasonic testing of representative above ground portions of water suppression piping that are exposed to water but which do not normally experience flow, are associated with a dry-pipe sprinkler system and may contain stagnant water, or is pre-action or deluge piping that is normally dry but may have been wetted and not completely dried;

c) At least one visual inspection (opportunistic or focused) of the internal surface of buried fire water piping within the 10 year period prior to the period of extended operation; and

d) At least one inspection per year of 'wet' fire protection piping for wall thickness and pipe blockage, if no opportunistic inspection is completed.

3.14.2.20 Flow-Accelerated Corrosion (FAC) Program (NUREG 1931, Section 3.0.3.1.7)

The Flow-Accelerated Corrosion (FAC) Program is an existing program that manages loss of material for carbon steel components located in systems that are susceptible to flow-accelerated corrosion, also called erosion/corrosion. The Flow-Accelerated Corrosion (FAC) Program is a condition monitoring program which ensures that the integrity of piping systems susceptible to flow-accelerated corrosion is maintained. The program was developed in response to NRC Bulletin 87-01 and NRC Generic Letter 89-08. The Flow-Accelerated Corrosion (FAC) Program follows the guidance and recommendations of EPRI NSAC-202L and combines the elements of predictive analysis, inspections (to baseline and monitor wall-thinning), industry experience, station information gathering and communication, and engineering judgment to monitor and predict flow-accelerated corrosion wear rates.

3.14.2.21 Fuel Oil Chemistry Program (NUREG 1931, Section 3.0.3.2.11)

The Fuel Oil Chemistry Program is an existing program that maintains fuel oil quality in order to mitigate damage due to loss of material and cracking of susceptible materials for plant components that are within the scope of license renewal and that contain fuel oil. The program manages the relevant conditions that could lead to the onset and propagation of loss of material or cracking through proper monitoring and control of fuel oil contamination consistent with pertinent plant technical specifications/requirements and American Society for Testing of Materials (ASTM) standards for fuel oil. The relevant conditions are specific contaminants such as water or microbiological organisms in the fuel oil that could lead to corrosion or stress corrosion cracking (SCC) of susceptible materials. Exposure to these contaminants is minimized by verifying the quality of new fuel oil before it enters the storage tanks and by periodic sampling to ensure that the tanks are free of water and particulates. The Fuel Oil Chemistry Program is a mitigation program.

The Fuel Oil Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the Fuel Oil Chemistry Program in mitigating the effects of aging.

3.14.2.22 Heat Exchanger Inspection (NUREG 1931, Section 3.0.3.1.12)

The Heat Exchanger Inspection detects and characterizes the condition of heat exchanger tubes in the Control Structure Chilled Water (CSCW), Fire Protection (FP) High Pressure Coolant Injection (HPCI), and Reactor Core Isolation Cooling (RCIC) systems.

The scope of the Heat Exchanger Inspection includes the CSCW chiller oil cooler and chiller evaporator, the FP diesel engine driven fire pump heat exchanger and lube oil cooler, and the HPCI and RCIC lube oil coolers. The inspection provides direct evidence as to whether, and to what extent, cracking due to SCC or reduction in heat transfer due to fouling has occurred or is likely to occur that could result in a loss of intended function.

The Heat Exchanger Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.23 Inservice Inspection (ISI) Program (NUREG 1931, Section 3.0.3.2.1)

The Inservice Inspection (ISI) Program is an existing program that manages cracking and loss of material of multiple reactor coolant system pressure boundary components, including the reactor vessel, a limited number of internals components, and the reactor coolant system pressure boundary. The Inservice Inspection (ISI) Program was developed as required by 10 CFR 50.55a. The program is in accordance with the requirements detailed in the 1998 Edition through the 2000 Addenda, of ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWA, IWB, IWC, IWD, IWE, IWF, IWL, Mandatory Appendices, Inspection Program B of IWA-2432, and approved ASME Code Cases.

The inservice inspections conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of the ASME Code Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC) for 10 CFR 54 associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.24 Inservice Inspection (ISI) Program – IWE (NUREG 1931, Section 3.0.3.1.19)

The Inservice Inspection (ISI) Program – IWE is an existing program that establishes responsibilities and requirements for conducting IWE inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWE includes visual examination of all accessible surface areas of the steel liner for the reinforced concrete Primary Containment and its integral attachments, containment seals and gaskets, and containment pressure-retaining bolting in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWE.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of the ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC), for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.25 Inservice Inspection (ISI) Program – IWF (NUREG 1931, Section 3.0.3.1.21)

The Inservice Inspection (ISI) Program – IWF is an existing program that establishes responsibilities and requirements for conducting IWF Inspections for ASME Class 1, 2, and 3 component supports as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program –

IWF visual examination for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). The primary inspection method employed is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. Supports requiring corrective actions are re-examined during the next inspection period in accordance with the requirements of ASME Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWF.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC), for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.26 Inservice Inspection (ISI) Program - IWL (NUREG 1931, Section 3.0.3.1.20)

The Inservice Inspection (ISI) Program – IWL is an existing program that establishes responsibilities and requirements for conducting IWL Inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWL includes visual examination of all accessible surface areas of the reinforced concrete Primary Containment in accordance with the requirements of ASME Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWL.

No applicable aging effects have been identified for the Primary Containment concrete. However, the Inservice Inspection (ISI) Program – IWL will be used to confirm the absence of significant aging effects for the extended period of operation.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration, for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.27 Leak Chase Channel Monitoring Activities (NUREG 1931, Section 3.0.3.3.2)

The Leak Chase Channel Monitoring Activities is an existing program that consists of observation and surveillance activities to detect leakage from the spent fuel pool and the fuel shipping cask storage pool liners due to aging and age-related degradation. The Leak Chase Channel Monitoring Activities is a condition monitoring program.

3.14.2.28 Lubricating Oil Analysis Program (NUREG 1931, Section 3.0.3.2.15)

The Lubricating Oil Analysis Program is an existing program that mitigates damage due to loss of material and reduction of heat transfer due to fouling for plant components that are within the scope of license renewal and exposed to lubricating oil. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material, or reduction in heat transfer for heat exchanger tubes, through proper monitoring consistent with manufacturer's

recommendations, the equipment's importance to safe plant operation, equipment accessibility and American Society for Testing of Materials (ASTM) standards for lubricating oil. The relevant conditions are specific parameters including particulate and water concentrations, viscosity, neutralization number, and flash point that could lead to, or are indicative of, conditions for age-related degradation of susceptible materials. The Lubricating Oil Analysis Program is a mitigation program.

Prior to the period of extended operation, the Lubricating Oil Analysis Program will be enhanced to include sampling of the lubricating oil from the Control Structure Chiller, Reactor Building Chiller, and Diesel Engine Driven Fire Pump when the oil is changed. The oil will be tested for water and for particle count.

The Lubricating Oil Analysis Program is supplemented by the Lubricating Oil Inspection which provides verification of the effectiveness of the Lubricating Oil Analysis Program in mitigating the effects of aging.

3.14.2.29 Lubricating Oil Inspection (NUREG 1931, Section 3.0.3.1.13)

The Lubricating Oil Inspection detects and characterizes the condition of materials in systems and components for which the Lubricating Oil Analysis Program is credited with aging management. The inspection provides direct evidence as to whether, and to what extent, a loss of material or a reduction in heat transfer due to fouling has occurred.

The Lubricating Oil Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.30 Not Used (NUREG 1931, Section 3.0.3.1.14)

3.14.2.31 Masonry Wall Program (NUREG 1931, Section 3.0.3.2.16)

The Masonry Wall Program is an existing program that consists of inspection activities to detect cracking of masonry walls within the scope of license renewal. Masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program. The Masonry Wall Program is implemented as part of the Structures Monitoring Program. The Masonry Wall Program performs visual inspection of external surfaces of masonry walls.

Prior to the period of extended operation, the Masonry Wall Program will be enhanced to specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing be evaluated to ensure that the current evaluation basis is still valid.

3.14.2.32 Metal-Enclosed Bus Inspection Program (NUREG 1931, Section 3.0.3.1.26)

The Metal-Enclosed Bus Inspection Program manages the aging of the metal-enclosed bus within the scope of license renewal. The program provides for inspection of the applicable metal-enclosed bus on a 10-year interval, in order to determine if age-related degradation is occurring.

The Metal-Enclosed Bus Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.33 Monitoring and Collection System Inspection (NUREG 1931, Section 3.0.3.1.15)

The Monitoring and Collection System Inspection detects and characterizes the conditions on the internal surfaces of subject components that are exposed to equipment/area drainage water and other potential contaminants/fluids. The inspection provides direct evidence as to whether, and to what extent, a loss of material has occurred or is likely to occur in the Liquid Waste Management System that could result in a loss of intended function.

The Monitoring and Collection System Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.34 Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program (NUREG 1931, Section 3.0.3.1.24)

The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program manages the age-related degradation associated with non-EQ, low-current instrumentation cables and connections within the scope of license renewal.

The program applies to in-scope, non-EQ electrical cables and connections used in neutron monitoring and radiation monitoring circuits with sensitive, low-current signals. The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program will perform testing of the applicable cable systems to identify reduction in insulation resistance. The tests will be performed at least every ten years, with the frequency to be determined by engineering evaluation.

The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.35 Non-EQ Electrical Cables and Connections Visual Inspection Program (NUREG 1931, Section 3.0.3.1.23)

The Non-EQ Electrical Cables and Connections Visual Inspection Program manages the aging of non-EQ electrical cables and connections within the scope of license renewal. The program provides for visual inspection on a 10-year interval of accessible, non-EQ electrical cables and connections, in order to determine if age-related degradation is occurring, particularly in plant areas with high temperatures and/or high radiation levels.

The Non-EQ Electrical Cables and Connections Visual Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.36 Non-EQ Inaccessible Medium-Voltage Cables Program (NUREG 1931, Section 3.0.3.1.25)

The Non-EQ Inaccessible Medium-Voltage Cables Program manages the aging of non-EQ inaccessible medium-voltage electrical cables subject to wetting within the scope of license renewal. The program provides for the periodic testing of non-EQ inaccessible medium-voltage electrical cables, in order to determine if age-related degradation is occurring, and includes provision for the inspection of associated manholes to identify any collection of water. The

cable testing frequency will be based on plant operating experience, but will be performed at least once every ten years. The electrical manhole inspection frequency will be based on plant operating experience, but will be performed at least once every two years.

The Non-EQ Inaccessible Medium-Voltage Cables Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.37 Non-EQ Electrical Cable Connections Program (NUREG 1931, Section 3.0.3.1.27)

The Non-EQ Electrical Cable Connections Program manages the aging for the metallic parts of non-EQ electrical cable connections within the scope of license renewal. The program addresses cable connections that are used to connect cable conductors to other cables or electrical devices. Aging management for the metallic parts of the non-EQ electrical cable connections that are subject to aging stressors will be provided by testing. A representative sample of non-EQ electrical cable connections will be selected for testing, considering the effects of their application (high, medium, and low voltage), circuit loading, and location with respect to electrical connection stressors. Thermography will be used to test a representative sample of cable connections to provide an indication of the integrity of the connections. The tests will be performed at least every ten years, with the frequency to be determined by engineering evaluation.

The Non-EQ Electrical Cable Connections Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.38 Piping Corrosion Program (NUREG 1931, Section 3.0.3.2.6)

The Piping Corrosion Program is an existing program that manages fouling due to particulates (e.g., corrosion products) and biological material (micro- and/or macro-organisms), and loss of material due to corrosion and erosion for components located in systems within the scope of the program that are exposed to a raw water environment. The program also manages the applicable aging effects for the internal environments of heat exchanger components within the scope of the program.

The Piping Corrosion Program is a combination of condition monitoring program (consisting of inspections, surveillances, and testing to detect the presence of, and to assess the extent of, damaged coatings, fouling and loss of material) and a mitigation program (consisting of chemical treatments and cleaning activities to minimize fouling and loss of material, and use of protective coatings in areas vulnerable to erosion).

Prior to the period of extended operation, the Piping Corrosion Program will be enhanced to include the Standby Gas Treatment System loop seals, and to also incorporate performance, documentation and trending of opportunistic visual inspections (during normal maintenance/repair activities).

3.14.2.39 Preventive Maintenance Activities – RCIC/HPCI Turbine Casings (NUREG 1931, Section 3.0.3.3.3)

The Preventive Maintenance Activities – RCIC/HPCI Turbine Casings is an existing program that manages loss of material on the internal surfaces of the Reactor Core Isolation Cooling

(RCIC) and High Pressure Coolant Injection (HPCI) pump turbine casings, and on the internal surfaces of associated piping and piping components (rupture disks and valve bodies), that are constructed of carbon steel or cast iron.

The Preventive Maintenance Activities – RCIC/HPCI Turbine Casings is a condition monitoring program, consisting of inspections and surveillance activities to detect aging and age-related degradation.

Prior to the period of extended operation, the Preventive Maintenance Activities – RCIC / HPCI Turbine Casings will be enhanced to incorporate:

a) A specific step to perform a visual inspection of the RCIC turbine casing.

b) Performance of inspections by qualified personnel using VT-3 or equivalent inspection methods, and reporting and trending of inspection results.

c) Specific acceptance criteria for inspections.

3.14.2.40 Reactor Head Closure Studs Program (NUREG 1931, Section 3.0.3.1.2)

The Reactor Head Closure Studs Program is an existing program that manages cracking for the reactor head closure studs. The Reactor Head Closure Studs Program includes (a) inservice inspection in conformance with the requirements of ASME Code, Section XI, Subsection IWB (edition and addenda described in 3.14.2.23), Table IWB 2500-1, and (b) preventive measures in accordance with Regulatory Guide 1.65 to mitigate cracking. The Reactor Head Closure Studs Program is implemented by the design of the plant and the Inservice Inspection (ISI) Program.

3.14.2.41 Reactor Vessel Surveillance Program (NUREG 1931, Section 3.0.3.2.12)

The Reactor Vessel Surveillance Program is an existing program that manages reduction of fracture toughness for the low alloy steel reactor vessel shell and welds in the beltline region. The Reactor Vessel Surveillance Program is a condition monitoring program developed in response to 10 CFR 50 Appendix H.

The SSES Reactor Vessel Surveillance Program is part of the Integrated Surveillance Program (ISP) described in BWRVIP-78, BWRVIP-86-A and BWRVIP-116, and approved by the NRC staff. BWRVIP-116 extends the ISP to cover the period of extended operation. SSES will follow the requirements of the BWRVIP ISP and will apply the ISP data to SSES Units 1 and 2. The NRC approved the use of the BWRVIP ISP in place of a unique plant program at SSES.

The SSES Reactor Vessel Surveillance Program will be enhanced, as necessary, to ensure that additional requirements that are specified in the NRC safety evaluation dated March 1, 2006, for BWRVIP-116 will be addressed before the period of extended operation. The program will include a requirement that, if a standby capsule is removed from either of the SSES Unit 1 or Unit 2 reactor vessels, without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation, if necessary,

3.14.2.42 RG 1.127 Water-Control Structures Inspection (NUREG 1931, Section 3.0.3.2.18)

The RG 1.127 Water-Control Structures Inspection is an existing program that consists of inspection and surveillance activities to detect aging and age-related degradation. The RG 1.127 Water-Control Structures Inspection ensures the structural integrity and operational adequacy of the Spray Pond (including concrete liner, emergency spillway, and riser encasements), ESSW Pumphouse (including pump intake chambers, overflow weir and chamber, and structural components within the ESSW Pumphouse), and the earthen embankments along the Spray Pond.

Prior to the period of extended operation, the RG 1.127 Water-Control Structures Inspection will be enhanced to add the Spray Pond (including concrete liner, emergency spillway, riser encasements, and earthen embankments) to its scope for inspection. The program will be enhanced to include RG 1.127 inspection elements and degradation mechanisms for water-control structure inspection and to include acceptance criteria for water-control structures.

3.14.2.43 Selective Leaching Inspection (NUREG 1931, Section 3.0.3.1.17)

The Selective Leaching Inspection detects and characterizes the conditions on internal and external surfaces of subject components. The inspection provides direct evidence through a combination of visual examination and hardness testing of whether, and to what extent, a loss of material due to selective leaching has occurred or is likely to occur that could result in a loss of intended function.

The Selective Leaching Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.44 Small Bore Class 1 Piping Inspection (NUREG 1931, Section 3.0.3.1.18)

The Small Bore Class 1 Piping Inspection is a one-time inspection to detect cracking resulting from thermal and mechanical loading or intergranular stress corrosion. The inspection will provide assurance that cracking of small bore Class 1 piping is not occurring or an evaluation of any detected crack indications will be performed to justify continued operation with no further monitoring, such that an aging management program (AMP) is not warranted. The inspection will also confirm the effectiveness of the BWR Water Chemistry Program mitigating cracking due to intergranular stress corrosion. The Small Bore Class 1 Piping Inspection is applicable to small bore ASME Code Class 1 piping and systems less than 4 inches nominal pipe size (NPS 4), which includes pipes, fittings, and branch connections.

The Small Bore Class 1 Piping Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.45 Structures Monitoring Program (NUREG 1931, Section 3.0.3.2.17)

The Structures Monitoring Program is an existing program that manages age-related degradation of plant structures and structural components within its scope to ensure that each structure or structural component retains the ability to perform its intended function. Aging

effects are detected by visual inspection of external surfaces prior to the loss of the structure's or component's intended function.

This program implements provisions of the Maintenance Rule, 10 CFR 50.65, that relate to structures, masonry walls, and water-control structures. Concrete and masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program.

Prior to the period of extended operation, the Structures Monitoring Program will be enhanced to include structures within the scope of license renewal. The program will be enhanced to specify inspections of a below grade structural wall or structural component that becomes accessible through excavation. The program will be enhanced to clarify the component types included as "structural components". The program will be enhanced to specify degradation mechanisms for elastomer and earthen embankment inspection. The program will be enhanced to include RG 1.127 inspection elements for water-control structures. The program will be enhanced to include requirements for review of site groundwater and raw water parameters. The program will also be enhanced to specify inspection requirements for masonry walls. The program will be enhanced to include direction for quantifying, monitoring and trending of inspection results, guidance for inspection reporting, data collection and documentation, acceptance criteria and critical parameters to monitor degradation and to trigger level of inspection and initiation of corrective action; and better alignment with referenced Industry codes, standards and guidelines. The program will be enhanced to include specific qualification requirements for the inspector.

3.14.2.46 Supplemental Piping/Tank Inspection (NUREG 1931, Section 3.0.3.1.16)

The Supplemental Piping/Tank Inspection detects and characterizes the condition of carbon steel, stainless steel and cast iron components that are exposed to moist air environments, particularly the aggressive alternate wet/dry environment that exists at air-water interfaces, and for internal surfaces of diesel exhaust components due to periodic exposure to exhaust gases containing moisture and contaminants. The inspection provides direct evidence as to whether, and to what extent, a loss of material or cracking (of stainless steel exposed to diesel exhaust) has occurred or is likely to occur that could result in a loss of intended function.

The Supplemental Piping/Tank Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.47 System Walkdown Program (NUREG 1931, Section 3.0.3.2.14)

The System Walkdown Program is an existing program that manages the following aging effects for the external surfaces, and in some cases the internal surfaces, of mechanical components within the scope of license renewal:

a) Loss of material for metals that are exposed to indoor air, outdoor air, or ventilation environments, including both the HVAC-type internal environments and ambient air internal environments, such as that found in the upper portion of a vented tank.

b) Cracking and/or change in material properties for elastomers (neoprene and rubber) and polymers (Teflon) that are exposed to indoor air or ventilation environments.

The System Walkdown Program is a condition monitoring program, consisting of observation and surveillance activities to detect aging and age-related degradation.

Prior to the period of extended operation, the System Walkdown Program will be enhanced to include the license renewal systems that contain mechanical components whose external surfaces require aging management during the period of extended operation. The program will also be enhanced to address opportunistic inspections of normally inaccessible components (e.g., those that are insulated), and those that are accessible only during refueling outages. The program will also be enhanced by addition of a routine activity to inspect elastomers and polymers for cracking and/or change in material properties and to include inspection of other metals, copper alloy and stainless steel. The program will also be enhanced to sample normally inaccessible components in underground vaults, pits, and manholes. In addition, the program will be enhanced to include a visual and ultrasonic inspection of the external surfaces of piping passing into structures through penetrations (underground piping) for those penetrations with a history of leakage.

3.14.2.48 Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program (NUREG 1931, Section 3.0.3.1.6)

The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program augments the visual inspection of the reactor vessel internals done in accordance with the ASME Code, Section XI, Subsection IWB, Category B-N-2. The inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internal components. The aging management program includes (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and (b) for each potentially susceptible component, aging management is accomplished through either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the SSES 10-year Inservice Inspection (ISI) Program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness.

The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.49 Fatigue Monitoring Program (NUREG 1931, Section 3.0.3.2.19)

The Fatigue Monitoring Program manages fatigue for all Class 1 components, including the reactor pressure vessel. In order not to exceed the design basis limit on fatigue usage, the aging management program monitors and tracks the number and severity of critical thermal and pressure transients and calculates the fatigue usage for the limiting locations of the reactor coolant pressure boundary.

Prior to the period of extended operation the program will be enhanced by adding the following required actions:

a) The program will verify that components which have satisfied ASME Section III, Paragraph N-415.1 requirements (i.e., RPV nozzles N6A, N6B, and N7) continue to satisfy these

requirements prior to and during the period of extended operation, thereby allowing fatigue to be continued to be addressed under N-415.1.

b) The program will review Class 1 valve fatigue analyses and other fatigue-related TLAA, such as flued head analyses and high energy line break evaluations, when sufficient fatigue accumulation has occurred, to determine if additional actions are required to address fatigue-related concerns.

c) The program will define specific fatigue usage values for all monitored locations, including those locations that account for the effect of the reactor water environment that, if reached, will require further action. These fatigue usage values shall be conservatively set to values that will allow for not less than 4 years of additional plant operation before the actual fatigue usage at any location would reach the design basis limit. Upon reaching the defined usage at a location, the program will require an action request to be generated. The action request will require further engineering evaluation to resolve the issue.

d) The program will implement one or more of the following actions, if fatigue usage at a monitored location, including any location that accounts for the effect of the reactor water environment, is projected to reach the design basis limit prior to the end of the period of extended operation:

1) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable;

2) Repair of the affected components;

3) Replacement of the affected components;

4) Management by an inspection program that has been reviewed and approved by the NRC.

3.14.2.50 Preventive Maintenance Activities – Main Turbine Casing (NUREG 1931, Section 3.0.3.3.4)

The Preventive Maintenance Activities – Main Turbine Casing is an existing program that manages loss of material due to flow-accelerated corrosion on the internal surfaces of the high pressure casing for the main turbine.

The Preventive Maintenance Activities – Main Turbine Casing is a condition monitoring program consisting of inspections performed on a nominal 10-year (12-year maximum) frequency to detect aging and age-related degradation.

Prior to the period of extended operation, the Preventative Maintenance Activities – Main Turbine Casing will be enhanced to specify that the inspection of the high pressure turbine shell will consist of a VT-3 or equivalent visual inspection of accessible surfaces and an ultrasonic examination of selected locations for wall thickness.

3.14.2.51 Fuse Holders Program (NUREG 1931, Section 3.0.3.2.20)

The Fuse Holders Program is a new aging management program that manages increased

connection resistance due to fatigue of fuse holder clamps. The program provides for periodic inspection of fuse holder clamps within the scope of license renewal that are not in enclosures containing active components and whose fuses are scheduled for removal once every 12 months, or more frequently.

The Fuse Holders Program is a condition monitoring program consisting of inspections performed on a 10-year frequency to detect aging and age-related degradation.

The Fuse Holders Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.3 EVALUATION OF TIME-LIMITED AGING ANALYSES (TLAA)

In accordance with 10 CFR 54.21(c), an application for a renewed operating license requires an evaluation of time-limited aging analyses (TLAA) for the period of extended operation. The license renewal evaluation of time-limited aging analyses (TLAA) identified existing aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section identifies the TLAA, summarizes the evaluation of each, and, where necessary, describes the aging management programs that will be required. These aging management programs will be implemented during the period of extended operation. One-time inspections will be conducted within the 10-year period prior to beginning the period of extended operation.

Three elements of an effective aging management program that are common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the SSES Operational Quality Assurance (OQA) Program, which implements the requirements of 10 CFR 50, Appendix B. Prior to the period of extended operation, the elements of corrective action, confirmation process, and administrative controls in the SSES OQA Program will be applied to required aging management programs for both safety-related and nonsafety-related structures and components determined to require aging management during the period of extended operation.

3.14.3.1 Reactor Vessel Neutron Embrittlement

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. Reactor vessel neutron embrittlement is a TLAA. Neutron fluence, upper shelf energy, adjusted reference temperature, and vessel pressure-temperature limits are time-dependent items that must be investigated to evaluate vessel embrittlement.

3.13.3.1.1 Neutron Fluence

High energy (>1 MeV) neutron fluence for the welds and shells of the reactor pressure vessel beltline region was calculated using the RAMA fluence methodology. The RAMA methodology was developed for the Electric Power Research Institute and the Boiling Water Reactor Vessel and Internals Project and is approved by the NRC for use at SSES Units 1 and 2. Use of this methodology for evaluations of fluence for the SSES units was performed in accordance with guidelines presented in Regulatory Guide 1.190. The evaluations determined values for neutron fluence for extended power uprate (EPU) conditions and for extended operation out to

54 effective full power years (EFPY), i.e., at the end of 60 years of operation. Using actual reactor core power histories to-date and conservative estimates of future core designs for each unit, extended operation to 60 years will be bounded by 54 EFPY.

Neutron fluence is not a TLAA, it is a time-limited assumption used in various neutron embrittlement TLAA.

3.14.3.1.2 Upper Shelf Energy Evaluation

10 CFR 50, Appendix G requires that upper shelf energy (USE) values for reactor pressure vessel materials include the effects of neutron radiation. It states that USE for the beltline materials including plates and welds be maintained at no less than 50 ft-lb for the life of the reactor vessel. Calculated fluence values for extended power uprate (EPU) and extended operation to 54 EFPY exceed previously determined fluence values based on materials surveillance program information for Units 1 and 2. Therefore, projections of changes in USE for the period of extended operation are required in accordance with 10 CFR 50, Appendix G.

The projections of changes in USE for the period of extended operation for the RPV beltline plates and welds for Units 1 and 2 were determined in accordance with Regulatory Guide (RG) 1.99, Revision 2. For the plates and welds with projected USE values of 50 ft-lb or greater at 54 EFPY, the criterion of 10 CFR 50, Appendix G, has been met and no further evaluation is required.

For plates and welds that do not meet the 50 ft-lb criterion, the equivalent margin analyses (EMA) documented in BWRVIP-74-A were used to demonstrate that the 54 EFPY USE values remain in compliance with 10 CFR 50, Appendix G. As prescribed in BWRVIP-74-A, the predicted decrease in USE from RG 1.99, Figure 2 was compared to the decrease assumed in the EMA for each vessel beltline plate and weld that fails to meet the 50 ft-lb criterion. The results demonstrate that all evaluated plates and welds are bounded by the BWRVIP-74-A equivalent margin analyses.

Therefore, the effects of neutron radiation have been evaluated, and all RPV beltline materials for Units 1 and 2 have been demonstrated to remain in compliance with Appendix G of 10 CFR 50 for the period of extended operation.

3.14.3.1.3 Adjusted Reference Temperature (ART) Analysis

In addition to USE, the other key parameter that characterizes the fracture toughness of a material is the reference temperature for nil-ductility transition (RT_{NDT}). This reference temperature will change as its exposure to neutron radiation increases. The effects of neutron fluence on RT_{NDT} are reflected in the change in this reference temperature, ΔRT_{NDT}, and the resulting adjusted reference temperature, ART, is calculated by adding ΔRT_{NDT} to RT_{NDT} along with appropriate margin to account for uncertainties.

The methodology used to calculate ART for the vessel beltline plates and welds is provided in Regulatory Guide 1.99. The ART values projected to 54 EFPY are used to develop Pressure-Temperature (P-T) limit curves. There are no limits or specific acceptance criteria for the projected ART values.

3.14.3.1.4 Pressure-Temperature (P-T) Limits

To assure that adequate margins of safety are maintained for various modes of reactor operation, 10 CFR 50, Appendix G specifies pressure and temperature requirements for affected materials for the service life of the reactor vessel. The basis for these fracture toughness requirements is ASME Section XI, Appendix G. The ASME Code requires P-T limits be established for hydrostatic pressure tests and leak tests, for operation with the core not critical during heatup and cooldown, and for core critical operation.

Calculations were performed to develop P-T limit curves for SSES Units 1 and 2 for the period of extended operation (54 EFPY). The calculations were performed for the bounding regions of the reactor vessel to account for 54 EFPY fluence projections, which include the effects of EPU conditions. The P-T curves were developed in accordance with 10 CFR 50, Appendix G and the methodology in ASME Section XI, Appendix G. PPL will submit future P-T curve data to the NRC as necessary to comply with 10 CFR 50 Appendix G,

3.14.3.1.5 Reactor Vessel Circumferential Weld Examination Relief

BWRVIP-74-A reiterated the recommendation of BWRVIP-05 that reactor pressure vessel circumferential welds could be exempted from examination. The NRC safety evaluation report for BWRVIP-74 agreed, but required that plants apply for this relief request individually. The relief request should demonstrate that at the expiration of the current license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the (BWRVIP-05) evaluation. This evaluation of circumferential weld parameters is a TLAA.

PPL requested and received relief from circumferential vessel shell weld volumetric examinations. The SSES submittal included an analysis that showed that the reactor vessel parameters at 32 EFPY were within the bounding parameters for Chicago Bridge & Iron (CBI) vessels from the BWRVIP-05 safety evaluation report. As such, there is a lower conditional probability of failure for circumferential welds at SSES than that stated in the NRC's Final Safety Evaluation Report of BWRVIP-05.

The SSES reactor pressure vessel circumferential weld parameters at 54 EFPY will remain within the bounding parameters for CBI vessels at 64 EFPY from the BWRVIP-05 safety evaluation report. As such, the conditional probability of failure for circumferential welds remains below that stated in the safety evaluation report for BWRVIP-05.

PPL will process a relief request for circumferential vessel shell weld volumetric examinations for the period of extended operation in the same manner that has been the practice during the original licensing period.

3.14.3.1.6 Reactor Vessel Axial Weld Failure Probability

The NRC safety evaluation report for BWRVIP-74-A evaluated the failure frequency of axially oriented welds in BWR reactor pressure vessels, and determined failure frequency acceptance criteria for 40 years of reactor operation. Applicants for license renewal must evaluate axially oriented RPV welds to show that their failure frequency remains below the acceptance criteria calculated in the safety evaluation report for BWRVIP-74. An acceptable way to do this is to show that the mean RT_{NDT} of the limiting axial beltline weld at the end of the period of extended operation is less than the values specified in the safety evaluation report for BWRVIP-74.

The SSES axial weld mean RT_{NDT} at 54 EFPY is projected to be well below that in the SER, and thus the SSES axial weld failure frequency is well within the acceptable criteria.

3.14.3.1.7 Reflood Thermal Shock Analysis

FSAR Section 3.13.1, Regulatory Guide 1.2 “Thermal Shock to Reactor Pressure Vessels” addresses the possibility of brittle fracture of the reactor vessel resulting from reflooding of the vessel following a postulated loss of coolant accident. This concern is addressed in NEDO-10029, “An Analytical Study on Brittle Fracture of GE-BWR Vessels Subject to the Design Basis Accident,” in which a conservative analysis is documented. That document provides an upper bound limit on brittle fracture failure for the materials and concludes that catastrophic failure is not possible. However, the analysis performed in NEDO-10029 is only valid for 40 years of operation.

A more recent analysis provides a technical basis for addressing brittle fracture of BWR vessels due to vessel reflood following a design basis Loss of Coolant Accident (LOCA) during the period of extended operation. The analysis is documented in a paper by S. Ranganath of General Electric, entitled “Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident,” which was presented as Paper G1/5 at the Fifth International Conference on Structural Mechanics in Reactor Technology in Berlin, Germany, in August 1979. While the analysis was performed for BWR-6 vessels, it is applicable to the SSES (BWR-4) vessels, based on the following:

a) It evaluated the main steam line break LOCA event, which is bounding for the evaluation of thermal stresses and brittle fracture in the vessel beltline region.

b) It applied to 251-inch, 238-inch, and 201-inch diameter BWR-6 vessels, since the structural details and operating conditions are similar. The SSES vessels are 251-inch diameter and the SSES response to a main steam line break LOCA is equivalent to that of the BWR-6 vessel (i.e., rapid depressurization and blowdown, immediately followed by ECCS injections to reflood the vessel).

c) It analyzed a 6-inch thick wall of a BWR-6 vessel. The SSES vessels are 6.1875 inches thick. A critical parameter of the fracture mechanics analysis is the wall temperature at a depth of 1/4 of the total thickness, 1/4T. The 1/4T location of the thinner BWR-6 vessel will cool faster than the 1/4T location of the SSES vessels. Thus, the BWR-6 analysis is conservative for the SSES vessels, since a lower temperature at the 1/4T location is worse for brittle fracture concerns.

The analysis presented in the Ranganath paper assumes end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature (ART). The analysis determined that the peak stress intensity for the LOCA event at the 1/4T location on the BWR-6 vessel would be 100 ksi√inches and that the available fracture toughness at the 1/4T location that coincides with the peak stress intensity would be a minimum of 200 ksi√inches. Thus, since the available toughness exceeds the applied stress intensity, an existing 1/4T flaw in the vessel wall would not propagate following a LOCA.

The BWR-6 analysis conclusion applies to the SSES vessels because the end-of-life (54 EFPY) ART values for the SSES vessels are such that the available material toughness at the 1/4T location of the SSES vessels would remain on the upper shelf of 200 ksi√inches, which exceeds

the peak stress intensity of 100 ksi√inches for the analyzed event. Therefore, brittle fracture of the SSES vessels due to reflood thermal shock following a design basis LOCA is not possible during the period of extended operation.

3.14.3.2 Metal Fatigue

Fatigue evaluations for mechanical components are identified as TLAA; therefore, the effects of fatigue must be addressed for license renewal.

PPL monitors fatigue of the ASME Class 1 reactor coolant pressure boundary via the Fatigue Monitoring Program, which uses a computer program, FatiguePro, to count transient cycles and calculate fatigue usage.

Calculation of fatigue usage values is not required for non-Class 1 SSCs. Instead, stress intensification factors and lower stress allowables are used to ensure components are adequately designed for fatigue.

Certain components enveloped by the Primary Containment are also required to be evaluated for fatigue. These include penetrations, hatches, the drywell head, downcomer vents, safety relief valve (SRV) discharge piping, and SRV quenchers.

3.14.3.2.1 Reactor Pressure Vessel Fatigue Analysis

The design transients for the reactor pressure vessel (RPV) assembly are reported in FSAR Table 3.9-1. Design cumulative usage factors (CUFs) for the limiting reactor pressure vessel assembly locations are obtained from applicable design reports. These CUFs were calculated based on applicable design transients.

Metal fatigue for all reactor pressure vessel assembly components is managed by the existing SSES Fatigue Monitoring Program. This program includes requirements for continued monitoring and periodic updates to current and projected CUFs for the limiting reactor pressure vessel locations. The program will be enhanced to include an approach to address CUFs that will exceed the allowable before the end of the period of extended operation. The aging management approach will include one or more of the following, which is similar to the approach documented in ASME Code Section III Non-mandatory Appendix L:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

The original RPV design report was not required to provide an explicit fatigue analysis for nozzles N6A, N6B, and N7, since the nozzles satisfied all requirements of ASME Section III, Paragraph N-415.1. As such, design CUFs were not calculated for these nozzles. The SSES Fatigue Monitoring Program will be enhanced to include a requirement to periodically determine

if the requirements of N-415.1 remain satisfied, such that fatigue evaluations are not required for these nozzles prior to entering and during the period of extended operation.

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with reactor pressure vessel fatigue are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.2.2 Reactor Vessel Internals Fatigue Analyses

The Reactor Internals and Core Support Structures at SSES were designed in accordance with ASME Section III, Subsection NG. The fatigue evaluations performed to demonstrate the design adequacy of the internals for 40 years are TLAA.

Most recently, structural evaluations were performed to address the effects of operation under extended power uprate conditions and the extended period of plant operation to 60 years. The evaluations determined that the fatigue usage factors for all reactor pressure vessel internals remain within the ASME Section III Subsection NG allowable limits.

PPL also monitors the design transients using FatiguePro, as described above under Reactor Pressure Vessel Fatigue Analysis. This monitoring allows PPL to continually assess the potential for plant operating anomalies that could impact the assumptions made in the fatigue evaluations of plant components. In addition to plant transient monitoring, PPL has effectively implemented the inspection requirements of the BWRVIP program at SSES, as described in Section 3.14.2.10 above. These inspections provide further assurance that the effects of aging due to fatigue of the RPV internals will be managed during the period of extended operation.

Structural evaluations have demonstrated that fatigue usage will remain within design limits to the end of the period of extended operation. Also, the BWR Vessel Internals Program is credited for managing the aging effects of the reactor vessel internals during the period of extended operation. Therefore, the TLAA associated with fatigue of the reactor vessel internals are dispositioned in accordance with 10CFR54.21 (c)(1)(ii) and 10CFR54.21(c)(1)(iii).

3.14.3.2.3 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Applicants for license renewal are required to address the reactor coolant environmental effects on fatigue of plant components. The minimum set of components is suggested to be the six (6) components defined in NUREG/CR-6260, as follows:

a) Reactor vessel shell and lower head

b) Reactor vessel feedwater nozzle

c) Reactor recirculation piping (including inlet and outlet nozzles)

d) Core spray line reactor vessel nozzle and associated Class 1 piping

e) Residual heat removal return line Class 1 piping

f) Feedwater line Class 1 piping

Calculation of a fatigue life adjustment factor, F_{en}, is determined for each fatigue-sensitive component. The environmental fatigue life adjustment factors are applied to the appropriate component CUFs to verify acceptability of the components for the period of extended operation.

Using fatigue data projected by the SSES Fatigue Monitoring Program and methodology accepted by the NRC, as noted above, PPL evaluated the limiting locations (a total of eleven component locations corresponding to the six NUREG/CR-6260 components), as appropriate for the material for each component location. Seven of the eleven locations evaluated have an environmentally adjusted CUF of greater than 1.0.

Prior to entering the period of extended operation, for each location that may exceed a CUF of 1.0 when considering environmental effects, SSES will implement one or more of the following:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

Should PPL select the option to manage environmentally-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be provided to the NRC prior to the period of extended operation.

The effects of environmentally-assisted fatigue for the limiting locations identified in NUREG/CR-6260 have been evaluated. The effects of environmentally-assisted fatigue for these locations is addressed using one of the four approaches identified above.

The Fatigue Monitoring Program is credited for managing the effects of the reactor coolant environmental effects on fatigue during the period of extended operation. Therefore, the TLAA associated with environmentally-assisted fatigue has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.2.4 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analyses

The Class 1 boundary encompasses all reactor coolant pressure boundary piping (pipe and fittings) and in-line components subject to ASME Section XI, Subsection IWB, inspection requirements.

FSAR Section 3.9 provides details on the design transients to be considered in the fatigue analyses of reactor coolant pressure boundary (RCPB) components.

The SSES Fatigue Monitoring Program tracks the fatigue usage at the limiting locations throughout the RCPB. The use of FatiguePro and the SSES Fatigue Monitoring Program ensure that the fatigue of RCPB components is maintained below the ASME Code design limits.

All Class 1 valves are required to have a fatigue analysis. A review of a representative sample of Class 1 valve stress reports found the fatigue analyses to be conservatively simplistic, and the predicted fatigue was extremely low (less than 0.1). The simplified analyses for the valves do not provide the detailed information required to track fatigue usage by cycle counting or similar means. As an alternative, since the fatigue usage is typically much higher on the associated piping systems, and fatigue monitoring is performed for the limiting piping locations, the fatigue usage on the Class 1 valves is assumed to be bounded by the Class 1 piping locations. The fatigue on the valves will be managed indirectly by monitoring fatigue on the piping. If a piping system accumulates sufficient fatigue usage to indicate that design values are being approached, the Fatigue Monitoring Program will require a review of the valve fatigue analyses and other fatigue-related TLAA (such as flued head analyses and high energy line break evaluations) to determine if additional actions are required to address any of these additional fatigue-related concerns on the affected piping system.

Metal fatigue for all Class 1 reactor coolant pressure boundary piping and in-line components is managed by the SSES Fatigue Monitoring Program. This program includes requirements for continued monitoring and periodic updates to current and projected CUFs for the limiting piping locations. The program will be enhanced to include an approach to address CUFs that will exceed the allowable before the end of the period of extended operation. The aging management approach will include one or more of the following, which is similar to the approach documented in ASME Code Section III Non-mandatory Appendix L:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with fatigue of the reactor coolant pressure boundary piping and components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.3 Non-Class 1 Component Fatigue Analyses

Calculation of cumulative fatigue usage, i.e., CUFs, is not required for non-Class 1 components designed in compliance with the codes and standards for non-Class 1 components. For non-Class 1 components stresses due to thermal expansion and anchor movement, which are important for fatigue evaluations, are analyzed using stress intensification factors and stress allowables. Allowable stresses are defined for 7000 full temperature cycles with reductions in allowable stresses as cycles increase beyond 7000. In addition, temperature thresholds above which fatigue should be considered for carbon steel and austenitic stainless steel are established.

The fatigue evaluation of non-Class 1 components determined whether the associated operating temperature exceeded threshold values for the affected materials and, if so, evaluated

the number of transient cycles expected. In every case, the number of projected cycles for 60 years was found to be less than 7000 for piping and in-line components whose temperatures exceed threshold values. Therefore, fatigue for non-Class 1 piping and in-line components remains valid for the period of extended operation.

None of the non-Class 1 vessels, heat exchangers, storage tanks, or pumps were designed to ASME Section VIII, Division 2 or ASME Section III, Subsection NC-3200. Therefore, there is no fatigue TLAA for these components.

The TLAA associated with the fatigue of non-Class 1 components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.4 Environmental Qualification of Electric Equipment (NUREG 1931, Section 3.0.3.1.28)

Environmental Qualification analyses for those components with a qualified life of 40 years or greater are identified as TLAA for SSES. NRC regulation 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants” requires licensees to identify electrical equipment covered under this regulation and to maintain a qualification file demonstrating that the equipment is qualified for its application and will perform its safety function up to the end of its qualified life.

10 CFR 50.49 requires EQ components that are not qualified for the current license term to be refurbished, replaced, or have their qualifications extended prior to reaching the aging limits established in the aging evaluation. Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ Program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, corrective actions (if acceptance criteria are not met), and the time remaining to the end of qualified life.

The SSES EQ Program is an existing program that implements the requirements of 10 CFR 50.49 and will be used to manage the effects of aging on the intended function(s) of the components associated with EQ TLAA for the period of extended operation.

3.14.3.5 Fatigue of Primary Containment, Attached Piping, and Components

3.14.3.5.1 ASME Class MC Components

FSAR Section 3.8.2.3.2.4 states the design thermal cycles for containment ASME Class MC stainless steel components, which includes the containment penetrations, hatches, and drywell head, to be 500 cycles for plant startup and shutdown and one cycle for a design basis accident. The reactor pressure vessel assembly and internal components are designed for 117 startups and 111 shutdowns for a combined total of 228 events. The maximum projected cycles for extended life to 60 years includes 148 startups and 148 shutdowns for a total of 296 events. Therefore, the Class MC component design value of 500 cycles for startups and shutdowns remains well above the projected value. Also, the one cycle allowed for a design basis accident is a value assumed in the design for a faulted condition for the life of the plant, whether that is 40 years or 60 years. Hence, the performance of these components will not be impacted by extending the life of the plant to 60 years.

The TLAA associated with thermal cycles on the ASME Class MC components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.5.2 Downcomer Vents and Safety Relief Valve Discharge Piping

Downcomer vents and safety relief valve (SRV) discharge piping penetrate the drywell / suppression pool diaphragm slab with the purpose of transporting steam and non-condensable gases to the suppression pool from the reactor and from the drywell during SRV lifts and under accident conditions. To ensure the integrity of the downcomers and SRV discharge piping for the original 40-year life of the plant, extensive analyses were performed. These analyses satisfy the definition for TLAA.

The significant area analyzed for the downcomers in the suppression pool air space was the downcomer penetration through the diaphragm slab. Structural analyses of all the SRV discharge lines from the diaphragm slab penetration to the quencher were performed, including flued head connections, elbows, and three-way restraint attachments.

The design rules, as set forth in the ASME Section III, Subsection NB were used for the fatigue assessment. The downcomers and SRV discharge lines were analyzed for the appropriate load combinations and their associated number of cycles. The combined stresses and corresponding equivalent stress cycles were computed to obtain the fatigue usage factors in accordance with the equations of Subsection NB-3600 of the ASME Code. The maximum cumulative usage factors for the downcomers and SRV discharge lines for the 40-year plant lifetime were determined from these analyses.

The minimum number of SRV actuations assumed in any of the fatigue analyses was 1100. The projected number of events for 60 years is less than the number assumed in the design basis (40 year) analysis. Therefore, the design basis analysis remains valid for the period of extended operation

The TLAA associated with stress cycles on the downcomer vents and safety relief valve discharge piping have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.5.3 Safety Relief Valve Quenchers

Quenchers provide proper dispersion of reactor steam into the suppression pool upon lifts of SRVs and discharge of the steam through the SRV discharge piping.

Analyses for fatigue of the quenchers satisfy 10 CFR 54.3 criteria as TLAA. Fatigue evaluations for the original 40-year life of the plant list 7000 cycles as the expected number of cycles for each quencher component analyzed. The evaluations calculate the number of allowable cycles for the components and give the expected CUF for each analysis.

Since a quencher can experience up to seven cycles each time its associated SRV actuates (lifts), the worst case number of cycles is seven times the number of actuations projected for 40 years and for 60 years. These projected cycles were compared with analysis data results.

The design cycles exceed the number of cycles projected to 60 years for all components which were analyzed for the quencher. Therefore, the CUFs calculated in the fatigue evaluation remain valid for the period of extended operation.

The TLAA associated with stress cycles on the safety relief valve quenchers have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.6 Other Plant-Specific Time-Limited Aging Analyses

3.14.3.6.1 Main Steam Flow Restrictor Erosion Analyses

A flow restrictor is incorporated in each main steam line to limit flow to 200 percent of rated flow in the event that a main steam line ruptures outside containment. Erosion of a flow restrictor is a safety concern since it could impair the ability of the flow restrictor to limit vessel blowdown following a main steam line break.

FSAR Section 5.4.4.4 discusses an evaluation of the effect of potential erosion of main steam line flow restrictors on radiological dose resulting from a main steam line break accident.

Operating for another 20 years will allow further erosion and, therefore, increased opening in the throat of the flow restrictors. The erosion can be linearly extrapolated from 40 to 60 years. This is conservative since the rate of erosion would be expected to decrease as the throat area of the restrictor increased due to erosion. (It has been determined that operation at Extended Power Uprate conditions will not significantly affect the erosion rate.) Therefore, it can be concluded that erosion for the 20 years of extended operation will be no more than half the erosion for the first 40 years, and the corresponding increase in steam flow will be no more than half of the increase in steam flow due to erosion at the end of 40 years (namely, 5 percent). This means that by the end of 60 years, the increase in flow compared to flow at the beginning of life will be no more than 7.8 percent. Therefore, the released dose for the accident case at 60 years would be no more than a 7.8 percent increase. Such an increase in dose over the analyzed case remains within regulatory limits, as indicated in FSAR Section 15.6.4.5.3.

Hence, the performance of the main steam line flow restrictors is not significantly impacted by the additional erosion during the period of extended operation.

3.14.3.6.2 High Energy Line Break Cumulative Fatigue Usage Factors

High energy line breaks have been postulated and analyzed for potential effects on surrounding equipment and systems. FSAR Section 3.6 provides criteria for determining break locations and types of breaks that could occur, descriptions of analysis methodologies, and results for significant attached piping showing where breaks could develop and where restraints were to be installed. Cumulative fatigue usage factors (CUFs) for the high energy lines are included in the criteria to determine postulated breaks. The CUFs, as calculated in the design fatigue analyses, account for the design transients assumed for the original 40-year life of the plant.

The postulated breaks are in piping for systems important to safety and integrity of the reactor coolant pressure boundary. The restraints designed for these potential breaks are significant for protection of systems and equipment important to plant safety. Therefore, the CUF calculations used in the selection of postulated high energy line break locations are TLAA.

Since these breaks are postulated to occur only once in the lifetime of the plants and restraints were installed appropriately to mitigate these potential breaks, the results of analyses for the potential breaks and the restraints installed in the plants remain unchanged for the extended life

of 60 years. However, it is possible that other locations that had 40-year CUFs below the criteria for postulated breaks, could exceed that CUF criteria in 60 years. The possibility of these additional postulated breaks will need to be managed based on the actual fatigue accumulation encountered as the plant ages.

Presently, SSES utilizes the EPRI FatiguePro software to monitor fatigue at the critical bounding locations of piping systems in the plant. The SSES Fatigue Monitoring Program will identify when piping systems are approaching their original 40-year design CUFs. Prior to any piping system exceeding its' original maximum design CUF, the pertinent design calculations for the affected system will be reviewed to determine if any additional locations should be designated as postulated high energy line breaks, under the original criteria of FSAR Section 3.6. If other locations are determined to require consideration as postulated break locations, appropriate actions will be taken to address the new break locations.

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with high energy line break cumulative fatigue have been dispositioned in accordance with10CFR54.21(c)(1)(iii).

3.14.3.6.3 <u>Core Plate Rim Hold-Down Bolts</u>

The NRC safety evaluation report that references BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," for license renewal identifies loss of preload on the core plate rim hold-down bolts as one of the TLAA that must be addressed by applicants seeking license renewal.

PPL will address the loss of preload on the core plate rim hold-down bolts by one of the following two actions:

a) PPL will perform a SSES plant-specific evaluation consistent with BWRVIP-25 to demonstrate that the core plate rim hold-down bolts will be capable of preventing lateral displacement of the core plate for the period of extended operation. The evaluation will determine the maximum expected reduction in the bolt preload at the end of the period of extended operation, considering all applicable parameters (i.e., operating temperature, operating loads, and irradiation effects) and demonstrate the acceptability of the final preload at the end of the period of extended operation. Using the methodology of BWRVIP-25 Appendix A, the evaluation will also determine the primary membrane and bending stresses for the limiting bolt(s) to demonstrate that ASME Code allowables are not exceeded as a result of the reduction in bolt preload at the end of the period of extended operation. The evaluation will also provide either a) justification for not inspecting the core plate hold-down bolts, or b) an inspection strategy to ensure an adequate number of bolts are intact to prevent lateral displacement of the core plate. The evaluation will be submitted to the NRC for review no less than two years prior to the period of extended operation.

b) PPL will install core plate wedges to structurally replace the lateral load resistance provided by the hold-down bolts. With wedges installed, any loss of preload on the core plate rim hold-down bolts during the period of extended operation will have no effect on the lateral stability of the core plate. The wedges will be installed prior to entering the period of extended operation.

If the evaluation described as Action 1 above is unable to demonstrate acceptable bolt preload or bolt stress values at the end of the period of extended operation, appropriate corrective action will be taken prior to entering the period of extended operation. The installation of core plate wedges described as Action 2 above is considered an appropriate and acceptable corrective action.

3.14.3.6.4 Irradiation Assisted Stress Corrosion Cracking (IASCC)

Austenitic stainless steel reactor internal components exposed to a neutron fluence of greater than 5 x 10^{20} n/cm^2 (E > 1 MeV) are susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. Analyses were performed to determine neutron fluence for extended power uprate (EPU) conditions and for extended operation out to 54 effective full power years (EFPY). The projected fluence values are used to identify the components that exceed the threshold fluence for IASCC.

The following reactor internal components have been identified as being susceptible to IASCC for the period of extended operation for SSES Units 1 and 2:

a) Top Guide

b) Core Shroud

c) In-Core Flux Monitoring Dry Tubes

d) Core Plate

The components identified as being susceptible to IASCC require aging management to identify and address potential degradation (crack initiation and growth) prior to any loss of intended function.

All identified components have been evaluated for IASCC by the BWRVIP, as described in the inspection and evaluation guideline reports for each component: BWRVIP-26-A for the Top Guide; BWRVIP-76 for the Core Shroud; BWRVIP-47-A for the In-Core Flux Monitoring Dry Tubes; and BWRVIP-25 for the Core Plate. The inspection and evaluation guidelines of the identified BWRVIP reports are implemented by the BWR Vessel Internals Program for SSES.

3.14.4 LICENSE RENEWAL COMMITMENT LIST

The listing of commitments identified for SSES license renewal is provided in Table 3.14-1. These commitments are tracked within PPL's regulatory commitment management program.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
1. Inservice Inspection (ISI) Program	Existing program is credited.	3.14.2.23	Ongoing
2. BWR Water Chemistry Program	Existing program is credited.	3.14.2.11	Ongoing
3. Reactor Head Closure Studs Program	Existing program is credited.	3.14.2.40	Ongoing
4. BWR Vessel ID Attachment Welds Program	Existing program is credited.	3.14.2.9	Ongoing
5. BWR Feedwater Nozzle Program	Existing program is credited.	3.14.2.6	Ongoing
6. BWR CRD Return Line Nozzle Program	Existing program is credited. • PPL will implement weld overlay repairs in accordance with ASME Section XI and NRC-approved Code Cases. If no NRC-approved Code Case exists for the weld overlay, PPL will obtain NRC approval prior to implementing the repair in accordance with 10 CFR 50.55a.	3.14.2.5	Ongoing
7. BWR Stress Corrosion Cracking (SCC) Program	Existing program is credited.	3.14.2.8	Ongoing

Table 3.14-1
SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
8. BWR Penetrations Program	Existing program is credited.	3.14.2.7	Ongoing
9. BWR Vessel Internals Program	Existing program is credited. • PPL will continue to perform inspections on at least 10% of the top guide grid beam cells containing control rod drives/blades every twelve years during the period of extended operation. Inspections on at least 5% of the top guide locations will be performed within the first six years of each twelve year interval. The top guide locations to be inspected are those subject to neutron fluence levels that exceed the IASCC threshold of 5.0E+20 n/cm^2. The inspections will be performed using the enhanced visual inspection technique, EVT-1.	3.14.2.10	Ongoing
10. Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	Program is new. The new program for SSES will be consistent with the program described in NUREG-1801 Section XI.M13, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program. The SSES program will identify susceptible components, evaluate those components to determine their susceptibility to loss of fracture toughness, and examine those components that are evaluated to be susceptible.	3.14.2.48	Prior to the period of extended operation.
11. Flow-Accelerated Corrosion (FAC) Program	Existing program is credited.	3.14.2.20	Ongoing
12. Bolting Integrity Program	Existing program is credited with the following enhancement: • Include specific precautions against the use of sulfur (sulfide) containing compounds as a lubricant for bolted connections.	3.14.2.2	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
13. Piping Corrosion Program	Existing program is credited with the following enhancements: • Include the Standby Gas Treatment System loop seals within the scope of the program. • Incorporate performance, documentation and trending of opportunistic visual inspections (during normal maintenance/repair activities) in addition to existing Piping Corrosion Program inspections.	3.14.2.38	Prior to the period of extended operation.
14. Crane Inspection Program	Existing program is credited.	3.14.2.17	Ongoing
15. Fire Protection Program	Existing program is credited.	3.14.2.18	Ongoing
16. Buried Piping Surveillance Program	Program is new. The scope of the Buried Piping Surveillance Program includes only the portions of the buried piping in the Residual Heat Removal Service Water (RHRSW) and Emergency Service Water (ESW) common return header known to have damaged coatings. The program is credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel piping components with damaged coatings.	3.14.2.4	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
17. Condensate and Refueling Water Storage Tanks Inspection	Program is a new one-time inspection. The scope of the Condensate and Refueling Water Storage Tanks Inspection includes the base (bottom surface and foundation pad interface) of the Condensate Storage Tanks (CSTs) and Refueling Water Storage Tank (RWST) that are in the scope of license renewal and included in the Condensate Storage and Transfer and the Refueling Water Storage and Transfer systems. An appropriate combination of volumetric (including thickness measurement) and visual examinations will be conducted, for a unit's CST (or RWST), to detect evidence of a loss of material due to crevice, general, or pitting corrosion or to confirm a lack thereof. Results will be applied to the other unit's tank(s) based on engineering evaluation.	3.14.2.14	Within the 10-year period prior to the period of extended operation.
18. Reactor Vessel Surveillance Program	Existing program is credited with the following enhancement: • Address the additional requirements specified in the NRC safety evaluation dated March 1, 2006, for BWRVIP-116. The program will include a requirement that, if a standby capsule is removed from either of the SSES Unit 1 or Unit 2 reactor vessels without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation if necessary.	3.14.2.41	Prior to the period of extended operation.

Table 3.14-1
SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
19. Chemistry Program Effectiveness Inspection	Program is a new one-time inspection. The Chemistry Program Effectiveness Inspection includes the internal surfaces of aluminum, copper and copper alloy, carbon and low alloy steel, cast iron, stainless steel, and nickel alloy components in systems that contain treated water or fuel oil. A representative sample of components in low flow and stagnant areas (i.e., locations that are isolated from the flow stream and possibly prone to gradual accumulation/concentration of contaminants) will be examined for evidence of a loss of material (due to crevice, galvanic, general, or pitting corrosion or to erosion, and to MIC in fuel oil), or to confirm a lack thereof, and the results applied to the rest of the system(s) based on engineering evaluation.	3.14.2.12	Within the 10-year period prior to the period of extended operation.
20. Cooling Units Inspection	Program is a new one-time inspection. The Cooling Units Inspection activities focus on a representative sample population of subject components at susceptible locations, to be defined in the implementing documents. These inspection activities provide symptomatic evidence of cracking, loss of material, or reduction in heat transfer at all other susceptible locations due to the similarities in materials and environmental conditions.	3.14.2.16	Within the 10-year period prior to the period of extended operation.
21. Heat Exchanger Inspection	Program is a new one-time inspection. The Heat Exchanger Inspection detects and characterizes conditions to determine whether, and to what extent a loss of heat transfer due to fouling is occurring (or likely to occur) for heat exchangers within the scope of license renewal. The Heat Exchanger Inspection is also credited for managing cracking due to stress corrosion cracking / inter-granular attack in the treated water (internal) environment of the admiralty brass tubes.	3.14.2.22	Within the 10-year period prior to the period of extended operation.
22. Not Used		3.14.2.30	

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
23. Monitoring and Collection System Inspection	Program is a new one-time inspection. The scope of the Monitoring and Collection System Inspection includes the internal surfaces of subject carbon steel (and low alloy steel) and cast iron piping and valve bodies that are exposed to potentially radioactive drainage water (untreated water) and potentially other contaminants/fluids during normal plant operations. A representative sample of components in the system, to be defined in the implementing documents, and to include containment isolation piping and/or valve bodies, will be examined for evidence of a loss of material (due to crevice, general, or pitting corrosion or to MIC), or to confirm a lack thereof, and the results applied to the rest of the system based on engineering evaluation.	3.14.2.33	Within the 10-year period prior to the period of extended operation.
24. Supplemental Piping/Tank Inspection	Program is a new one-time inspection. The Supplemental Piping/Tank Inspection is credited for managing loss of material due to crevice and pitting corrosion on carbon steel surfaces at air-water interfaces. The inspection is also credited for managing loss of material due to microbiologically influenced corrosion (MIC) at the air-water interface with the mist eliminator loop seal, which is filled with raw water from the Service Water System, and galvanic corrosion at points of contact between the mist eliminator housing and the SGTS filter enclosure, where condensation and water pooling may occur. Additionally, the Supplemental Piping/Tank Inspection detects and characterizes whether, and to what extent, a loss of material due to crevice and pitting corrosion is occurring (or is likely to occur) for stainless steel surfaces at air-water interfaces. The Supplemental Piping/Tank Inspection also detects and characterizes loss of material due to crevice, galvanic, general, and pitting corrosion on internal carbon steel surfaces within the scram discharge volume (piping and valve bodies) of the Control Rod Drive Hydraulic System, within the air space of the condensate storage tanks and within the Diesel Generator	3.14.2.46	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	starting air receiver tanks and E diesel compressor skid air receiver tanks to determine whether, and to what extent, degradation is occurring (or is likely to occur). In addition, the Supplemental Piping/Tank Inspection is credited to detect and characterize loss of material due to general, crevice, and pitting corrosion on the internal surfaces of carbon steel and cast iron diesel exhaust piping, piping components, and turbocharger casings. The inspection is also credited to detect and characterize cracking and loss of material due to crevice and pitting corrosion on the internal surfaces of stainless steel diesel exhaust piping components.		
25. Selective Leaching Inspection	Program is a new one-time inspection. The Selective Leaching Inspection detects and characterizes conditions to determine whether, and to what extent a loss of material due to selective leaching is occurring (or likely to occur) for susceptible components including piping and tubing, valve bodies, pump and turbocharger casings, heat exchanger, cooler, and chiller components, hydrants, sprinkler heads, strainers, level gauges, orifices, and heater sheaths. The components within the scope of the program are formed of cast iron, brass, bronze, and copper alloy materials. The components are subject to raw water, treated water, groundwater (buried), indoor air with condensation, outdoor air, and fuel oil environments. The components within the scope of this program are located in twenty-six different plant systems.	3.14.2.43	Within the 10-year period prior to the period of extended operation.

Table 3.14-1
SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
26. Buried Piping and Tanks Inspection Program	Program is new. The scope of the Buried Piping and Tanks Inspection Program includes buried components that are within the scope of license renewal for SSES. The program is credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel piping components. In addition, the program is credited with managing loss of material for buried stainless steel piping components. The Buried Piping and Tanks Inspection Program is also credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel tanks in the Diesel Fuel Oil System.	3.14.2.3	Prior to the period of extended operation.
27. Small-Bore Class 1 Piping Inspection	Program is a new one-time inspection. The SSES program will include measures to verify that cracking is not occurring in Class 1 small-bore piping, thereby validating the effectiveness of the Chemistry Program to mitigate cracking and confirming that no additional aging management programs are needed for the period of extended operation.	3.14.2.44	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
28. System Walkdown Program	Existing program is credited with the following enhancements: • The governing procedure for the System Walkdown Program must be revised to add the listing of systems crediting the program for license renewal and to explicitly include inspection of other metals, copper alloy and stainless steel. o It may be determined by engineering evaluation that these components do not require monitoring every two weeks, and the basis for a different walkdown frequency must be documented on the appropriate procedure form. • The governing procedure for the System Walkdown Program must be enhanced to address the license renewal requirement for opportunistic inspections of normally inaccessible components (e.g., those that are insulated), and those that are accessible only during refueling outages. For underground vaults/pits/manholes, an initial sample of at least one vault/pit/manhole from each grouping of components with identical material and environment combinations will be inspected prior to entering the period of extended operation. A representative sample of the entire population will be inspected within the first 6 years of the period of extended operation. Results of the inspection activities that require further engineering evaluation/resolution (e.g., sample expansion and inspection frequency changes if degradation is detected), if any, will be evaluated using the SSES corrective action process. • The governing procedure for the System Walkdown Program will be enhanced to include a visual and ultrasonic inspection of the external surfaces of piping passing into structures through penetrations (underground piping) for those penetrations with a history of leakage. These inspections will be focused on penetrations that are leaking at that time and will include a representative population of each material, environment	3.14.2.47	Prior to the period of extended operation.

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	combination from those piping systems within the scope of license renewal (which includes those for the RHRSW, ESW, and Fire Protection systems) that enter structures below grade. • A routine activity to supplement the existing plant program will be generated, and based at least in part on EPRI 1007933, "Aging Assessment Field Guide," to inspect elastomers and polymers for cracking and/or change in material properties. ○ Evidence of surface degradation, such as cracking or discoloration, as well as physical manipulation and/or prodding, will be used as a measure of the material condition. ○ A representative sample will be determined by engineering evaluation with a focus on components considered to be most susceptible to aging, such as due to their time in service, the severity of conditions during normal plant operations, and any pertinent design margins.		
29. Inservice Inspection (ISI) Program – IWE	Existing program is credited.	3.14.2.24	Ongoing
30. Inservice Inspection (ISI) Program – IWF	Existing program is credited.	3.14.2.25	Ongoing
31. Inservice Inspection (ISI) Program - IWL	Existing program is credited.	3.14.2.26	Ongoing

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32. Containment Leakage Rate Test Program	Existing program is credited.	3.14.2.15	Ongoing
33. Masonry Wall Program	Existing program is credited with the following enhancement: • Specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking and steel degradation is sufficient to invalidate the evaluation basis.	3.14.2.31	Prior to the period of extended operation.
34. Structures Monitoring Program	Existing program is credited with the following enhancements: • Include additional structures requiring aging management for license renewal to the scope of the inspections. • Specify that if a below grade structural wall or structural component becomes accessible through excavation; a follow-up action is initiated for the responsible engineer to inspect the exposed surfaces for age-related degradation. • Clarify “structural component” for inspection includes each of the component types identified as requiring aging management. • Include degradation mechanisms for elastomer and earthen embankment inspection. • Include RG 1.127 inspection elements for water-control structure. • Specify that the responsible engineer shall review site groundwater and raw water pH, chlorides, and sulfates results prior to inspection to validate that the below-grade or raw water	3.14.2.45	Prior to the period of extended operation.

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	environment remains non-aggressive during the period of extended operation. • Specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking and steel degradation is sufficient to invalidate the evaluation basis. • Include additional direction for quantifying, monitoring and trending of inspection results; Include additional guidance for inspection reporting, data collection and documentation; Specify acceptance criteria and critical parameters to monitor degradation and to trigger level of inspection and initiation of corrective action; and provide better alignment with referenced Industry codes, standards and guidelines. • Include specific qualification requirements for the inspector.		

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
35. RG 1.127 Water-Control Structures Inspection	Existing program is credited with the following enhancements: • Add the Spray Pond (including concrete liner, emergency spillway, riser encasements and earthen embankments) to its scope for inspection. • Include RG 1.127 Revision 1 Section C.2 inspection elements and degradation mechanisms for water-control structure inspection. • Include acceptance criteria as delineated in NUREG-1801 Section XI.S7 for water-control structures. Evaluation criteria provided in Chapter 5 of ACI 349.3R-96 provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects and specifies criteria for further evaluation.	3.14.2.42	Prior to the period of extended operation.
36. Non-EQ Electrical Cables and Connections Visual Inspection Program	Program is new. The Non-EQ Electrical Cables and Connections Visual Inspection Program is credited with detecting aging effects from adverse localized environments in non-EQ cables and connections at SSES. The program is applicable to non-EQ cables and connections found in the Reactor Buildings, Circulating Water Pumphouse and Water Treatment Building, Control Structure, Diesel Generator Buildings, Turbine Building, Engineered Safeguards Service Water Pumphouse, and various yard structures (manholes, duct banks, valve vaults, instrument pits, etc.). This program is also applicable to the cables and connections within the scope of license renewal located in the yard areas and control cubicles of the T10 230 kV Switchyard, the 500 kV Switchyard, and the 230 kV Switchyard.	3.14.2.35	Prior to the period of extended operation.

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
37. Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program	Program is new. The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program is credited with identifying aging effects for sensitive, high-voltage, low-current signal applications that are in-scope for license renewal at SSES. These sensitive circuits are potentially subject to reduction in insulation resistance (IR) when found in adverse localized environments.	3.14.2.34	Prior to the period of extended operation.
38. Non-EQ Inaccessible Medium-Voltage Cables Program	Program is new. The Non-EQ Inaccessible Medium-Voltage Cables Program involves two parts: first, the actions to inspect the applicable plant manholes (and to drain them, if necessary) on a periodic basis; and second, the development of a testing program to confirm that the conductor insulation on the applicable cables is not degrading. This program applies to six cables associated with the offsite power supply for SSES. These are the only inaccessible medium-voltage cables at SSES that are within the scope of license renewal and are exposed to significant moisture simultaneously with significant voltage.	3.14.2.36	Prior to the period of extended operation.
39. Metal-Enclosed Bus Inspection Program	Program is new. The Metal-Enclosed Bus Inspection Program is credited with detecting aging effects for in-scope metal-enclosed bus at SSES. The applicable components for the metal-enclosed bus will be listed in the program implementing document(s), with their locations specified, as appropriate. The in-scope bus is limited to non-segregated metal-enclosed bus in the 13.8 kV and 4 kV electrical systems associated with the off-site power supply at SSES.	3.14.2.32	Prior to the period of extended operation.

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
40. Area-Based NSAS Inspection	Program is a new one-time inspection. The Area-Based NSAS Inspection includes confirming the environmental and/or internal surfaces conditions of subject nonsafety-related carbon steel (includes low alloy steel), cast iron, copper alloy and stainless steel components in systems that (frequently or continuously during normal plant operations) contain raw water, potable water, non-radioactive equipment/area drainage water, or in some select cases, treated water. The program is plant-specific.	3.14.2.1	Within the 10-year period prior to the period of extended operation.
41. Leak Chase Channel Monitoring Activities	The existing program is credited. The program is plant-specific.	3.14.2.27	Ongoing
42. Preventive Maintenance Activities – RCIC/HPCI Turbine Casings	Existing program is credited with the following enhancements: • Include a specific step to perform a visual inspection of the RCIC turbine casing. • Add requirements to have inspections performed by qualified personnel using VT-3 or equivalent inspection methods, and to document and trend inspection results. • Establish specific acceptance criteria for inspection results. The program is plant-specific.	3.14.2.39	Prior to the period of extended operation.
43. Fatigue Monitoring Program	Existing program is credited with the following enhancements: • Provisions will be made in the Fatigue Monitoring Program to validate that components which have satisfied ASME Section III, Paragraph N-415.1 requirements (i.e., RPV nozzles N6A, N6B, and N7) continue to satisfy these requirements prior to and during the period of extended operation, thereby allowing fatigue to be	3.14.2.49	Prior to the period of extended operation.

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	continued to be addressed under N-415.1. • The Fatigue Monitoring Program will be enhanced to ensure that the fatigue usage at all monitored locations, including those locations that account for the effect of the reactor water environment, is managed such that an adequate margin against fatigue cracking is maintained. PPL will implement one or more of the following actions, if fatigue usage at a monitored location, including any location that accounts for the effect of the reactor water environment, is projected to reach the design basis limit prior to the end of the period of extended operation: 1. Further refinement of the fatigue analyses to lower the CUFs to less than the allowable; 2. Repair of the affected components; 3. Replacement of the affected components; 4. Management by an inspection program that has been reviewed and approved by the NRC. • The Fatigue Monitoring Program will be enhanced to include the review of Class 1 valve fatigue analyses and other fatigue-related TLAA, such as flued head analyses and high energy line break evaluations, when sufficient fatigue accumulation has occurred, to determine if additional actions are required to address fatigue-related concerns. • The Fatigue Monitoring Program will be enhanced to include fatigue monitoring of the additional locations required to bound the limiting locations applicable to SSES, as identified in NUREG/CR-6260. • The Fatigue Monitoring Program will be enhanced to establish		

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	monitoring criteria to ensure that the fatigue usage at all monitored locations, including those locations that account for the effect of the reactor water environment, is managed such that design basis limits are not exceeded during the period of extended operation. The Fatigue Monitoring Program will define specific fatigue usage values for all monitored locations that, if reached, will require further action. These fatigue usage values shall be conservatively set to values that will allow for not less than 4 years of additional plant operation before the actual fatigue usage at any location would reach the design basis limit. Upon reaching the defined usage at a location, the Fatigue Monitoring Program will require an action request to be generated. The action request will require further engineering evaluation to resolve the issue.		
44. Environmental Qualification (EQ) Program	Existing program is credited. For those EQ components that do not show a minimum 60-year life, the EQ Program will ensure qualified life is not exceeded by directing refurbishment, replacement, or reanalysis to extend the qualification.	3.14.3.4	Ongoing
45. Closed Cooling Water Chemistry Program	Existing program is credited.	3.14.2.13	Ongoing.

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
46. Fire Water System Program	Existing program is credited with the following enhancements: • The Fire Water System Program will be revised to incorporate sprinkler head sampling/replacements, in accordance with NFPA 25. • The Fire Water System Program will be revised to incorporate ultrasonic testing of representative above ground portions of water suppression piping that are exposed to water but which do not normally experience flow, are associated with a dry-pipe sprinkler system and may contain stagnant water, or is pre-action or deluge piping that is normally dry but may have been wetted and not completely dried. • Perform at least one visual inspection (opportunistic or focused) of the internal surface of buried fire water piping, within the 10 year period prior to the period of extended operation. • Perform at least one inspection per year of 'wet' fire protection piping for wall thickness and pipe blockage, if no opportunistic inspection has been completed.	3.14.2.19	Prior to the period of extended operation.
47. Fuel Oil Chemistry Program	Existing program is credited.	3.14.2.21	Ongoing

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
48. Lubricating Oil Analysis Program	Existing program is credited with the following enhancements: • The Lubricating Oil Analysis Program will be enhanced to include sampling of the lubricating oil from the Control Structure Chiller and Diesel Engine Driven Fire Pump when the oil is changed. The oil will be tested for water and for particle count. • The Lubricating Oil analysis Program will be revised to include sampling of the lubricating oil from the Reactor Building Chiller when the oil is changed. The oil will be tested for water and particle count.	3.14.2.28	Prior to the period of extended operation.
49. Lubricating Oil Inspection	Program is a new one-time inspection. The Lubricating Oil Inspection detects and characterizes the condition of materials in systems and components for which the Lubricating Oil Analysis Program is credited with aging management. The inspection provides direct evidence as to whether, and to what extent, a loss of material or a reduction in heat transfer due to fouling has occurred.	3.14.2.29	Within the 10-year period prior to the period of extended operation.
50. Non-EQ Electrical Cable Connections Program	Program is new. The Non-EQ Electrical Cable Connections Program manages the aging for the metallic parts of non-EQ electrical cable connections within the scope of license renewal. The program addresses cable connections that are used to connect cable conductors to other cables or electrical devices. Aging management for the metallic parts of the non-EQ electrical cable connections that are subject to aging stressors will be provided by testing.	3.14.2.37	Prior to the period of extended operation.
51. New P-T Curves	Revised Pressure-Temperature (P-T) limits will be submitted to the NRC for approval when necessary to comply with 10 CFR 50 Appendix G.	3.14.3.1.4	Ongoing
52. OE Review at	Perform an Operating Experience (OE) review for the period of operation at EPU conditions and its impact on aging management	---------	Prior to the period of extended

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
EPU Conditions	programs for systems, structures and components (SSCs).		operation.
53. Incorporate FSAR Supplement	Incorporate FSAR Supplement into the SSES FSAR as required by 10 CFR 54.21(d).	3.14.1	Following issuance of the renewed operating licenses.
54. Re-apply for relief request	PPL will process a relief request for circumferential vessel shell weld volumetric examinations for the period of extended operation.	3.14.3.1.5	Prior to the period of extended operation.
55. Core Plate Hold Down Bolts	PPL will either: (1) obtain NRC approval of a SSES plant-specific evaluation consistent with BWRVIP-25 to demonstrate that the core plate rim hold-down bolts will be capable of preventing lateral displacement of the core plate for the period of extended operation (the plant-specific evaluation will be submitted for NRC review no less than 2 years prior to the period of extended operation and will address the inspection strategy for the hold-down bolts); or (2) install core plate wedges to structurally replace lateral load resistance provided by the bolts.	3.14.3.6.3	Prior to the period of extended operation.
56. BWRVIP-76	PPL will address any future conditions, requirements, or limitations imposed by the NRC's safety evaluation for license renewal for BWRVIP-76.	3.14.3.6.4	Prior to the period of extended operation.
57. Preventative Maintenance Activities-Main Turbine Casing	Existing program is credited with the following enhancement: • Specify that the inspection of the high pressure turbine shell will consist of a visual inspection (VT-3 or equivalent) of accessible surfaces and an ultrasonic examination of selected locations for wall thickness. The program is plant specific.	3.14.2.50	Prior to the period of extended operation.

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Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
58. Activities in Response to NRC Generic Letter 88-14	Activities credited in the SSES response to NRC Generic Letter 88-14 will be continued throughout the period of extended operation.	---------	Ongoing
59. Fuse Holders Program	Program is new. The Fuse Holders Program is credited with identifying increased connection resistance between the fuse holder metallic clamp and fuse due to fatigue of the metallic clamp. The program provides for periodic inspection of fuse holder clamps within the scope of license renewal that are not in enclosures containing active components and whose fuses are scheduled for removal once every 12 months, or more frequently.	3.14.2.51	Prior to the period of extended operation.
60. Activities in Response to NRC Concerns Regarding Fatigue Analyses	PPL will either (1) implement fatigue monitoring software that satisfactorily addresses all issues raised in Regulatory Information Summary (RIS) 2008-30, "Fatigue Analysis of Nuclear Power Plant Components", or (2) perform a confirmatory ASME Code, Section III fatigue evaluation for the SBF-monitored locations to justify the existing FatiguePro methodology used at SSES Units 1 and 2.	-----------	Prior to the period of extended operation.
61. Boral Coupon Testing	Spent fuel pool Boral coupon testing will be continued in the period of extended operation with one set of coupons being tested during the tenth or eleventh year after Unit 1 enters the period of extended operation. SSES FSAR section 9.1.2.3.3 Inservice Inspection will be revised to identify the coupon testing schedule during the period of extended operation. SSES FSAR section 9.1.2.3.3.2 Test Coupon Inspection will be revised to require neutron attenuation testing as part of the inspection of test coupons removed from the spent fuel pool.	-----------	Revise the FSAR prior to the period of extended operation, with coupon testing ongoing as indicated.