APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

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TABLE OF CONTENTS

APPENDIX 3A

TABLE OF CONTENTS (Cont'd.)

APPENDIX 3A

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

APPENDIX 3A

TABLE OF CONTENTS (Cont'd.)

APPENDIX 3A

LIST OF TABLES

APPENDIX 3A

LIST OF TABLES (Cont'd.)

APPENDIX 3A

LIST OF FIGURES

LIST OF FIGURES (Cont'd.)

APPENDIX 3A

LIST OF FIGURES (Cont'd.)

APPENDIX 3A

COMPUTER PROGRAMS FOR DYNAMIC AND STATIC ANALYSIS OF CATEGORY I STRUCTURES, EQUIPMENT, AND COMPONENTS

INTRODUCTION

This introduction details the principal use(s) of the most common computer programs for dynamic and static analysis of Category I structures, equipment and components within the balance of plant (BOP) scope of supply (originally supplied by Stone & Webster Engineering Corporation [SWEC]). See Section 3.9B for a listing and a description of the corresponding nuclear steam supply system (NSSS) computer programs (originally supplied by General Electric Company [GE]).

The following computer programs are used for the stress analysis of the containment system:

1. SHELL 1
2. ASAAS
3. TAC2D
4. CWL

The following computer programs are used in dynamic and static analysis of Category I structures:

1. MAT 6
2. STRUDL
3. TIME HISTORY (TIMHIS6) PROGRAM
4. SHELL 1
5. GHOSH-WILSON
6. TRANFUN AND INVATRAN

The following computer programs are used in the analysis of Category I equipment and components:

1. ME 121
2. DINASAW
3. LIMITA II
4. LIMITA III
5. STARDYNE
6. MISSILE
7. BIJLAARD
8. SLOSH
9. ANSYS
10. IMAGES
11. STAAD-III
12. APLAN
13. DYNARACK

The following computer programs are used for the analysis of Category I piping systems, including pipe supports:

1. NUPIPE
2. HTLOAD
3. WATAIR
4. STEHAM
5. WATHAM
6. PITRUST
7. PILUG
8. ANSYS*
9. LUGSTR
10. SNUFFE
11. PITRIFE
12. SUPERPIPE
13. PIPSYS

The following computer programs are used to calculate submerged structure loads due to various disturbances in the suppression pool:

1. SSLAM
2. SSLOAD

The following computer program is used for the analysis of the primary containment airlock:

1. NASTRAN

For each computer program, there is a brief description of the program's theoretical basis, the assumptions and references used in the program, and the extent of its application.

3A.1 SHELL 1

3A.1.1 General Description

SHELL 1 is a finite difference stress analysis computer code. It can be used to determine the forces, moments, shears, displacements, rotations, and stresses in a thin shell of revolution subject to arbitrary loads expanded in Fourier series of up to 150 terms. Single-layer shells with up to 30 simply connected branches may be analyzed. Poisson's ratio may change at discontinuity points, and Young's modulus and the thermal coefficient of expansion may be different at each point. The allowed types of loading include elastic restraints, pressures in three orthogonal directions, temperature changes that may have a gradient through the shell thickness, and simplified input for weight of the shell or earthquake forces.

3A.1.2 Program Verification

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*ANSYS is also used in the bearing pad and liner analyses for racks described in Chapter 9.

The equilibrium equations for a thin shell are based on the linear theory of Sanders[1]. Sanders' equations are expanded and modified slightly to handle a broader range of problems. All pertinent load, stress, and deformation variables are expanded into Fourier series. The individual Fourier components of stress and deflection are found separately by solution of the finite difference forms of the appropriate differential equations. The algorithm used to solve these equations is a minor modification of the Gaussian elimination method.

Sample Problem: Thin Wall Cylinder

A long thin-walled circular cylinder is subjected to a constant internal pressure distribution. A solution of this problem may be obtained from Reference 2.

The pertinent parameters of the cylinder are:

Dimension and Properties	Loading and Boundary Conditions
$R = 25$ in	$F_R\big\|_{z=0} = M\big\|_{z=0} = \delta z\big\|_{z=0} = 0$
$\ell = 20$ in $t = 0.5$ in	
$E = 28 \times 10^6$ psi $\nu = 0.3$	$F_R\big\|_{z=1} = M\big\|_{z=1} = F_z\big\|_{z=1} = 0$
	$P_i = 75$ psi

The following solution can be verified by consulting Reference 2:

$$\delta_R = \frac{pR^2}{Et} \qquad (3A.1\text{-}1)$$

$$\sigma_\theta = \frac{pR}{t} \qquad (3A.1\text{-}2)$$

The cylinder is idealized by 10 elements (Figure 3A.1-1).

In Table 3A.1-1, the computer results are compared with the results obtained from Equations 3A.1-1 and 3A.1-2. The results compare very favorably. Therefore, this problem serves to demonstrate the accuracy of SHELL 1.

3A.1.3 References

1. Sanders, Jr., J. L. An Improved First Approximation Theory for Thin Shells. NASA Technical Report R-24.

2. Roark, R. J. Formulas for Stress and Strain. McGraw-Hill Book Company, Fourth Edition, New York, NY, 1965.

TABLE 3A.1-1
(Sheet 1 of 1)
EXACT AND COMPUTER STRESSES FOR THIN-WALL CYLINDER
SHELL 1 COMPUTER PROGRAM

Variable	Exact	SHELL 1
δR	3.348×10^{-3} in	3.342×10^{-3} in
$\sigma\theta$	3,750 psi	3,750 psi

3A.2 ASAAS

3A.2.1 General Description

ASAAS is a finite element computer code that can be used to determine stresses and displacements in arbitrary axisymmetric solids, including problems involving asymmetric mechanical and thermal loads and asymmetric temperature-dependent mechanical properties. All dependent variables, including the mechanical properties, are input by Fourier series expansions of the circumferential coordinate. The mechanical loads can be surface pressures, surface shears, and nodal point forces.

3A.2.2 Program Verification

The explicit stiffness relations for the axisymmetric solid ring elements of triangular cross section are based on the classical theorem of potential energy and the assumption that within any element the displacement variation in the R-Z plane is linear. All dependent variables, including the material properties, are expanded into Fourier series. The harmonics are coupled and all the equilibrium equations are solved simultaneously. The algorithm used to solve the equations is a block-modified, square root, Cholesky method with iterative refinement. ASAAS is a recognized program in the public domain[(1)].

<u>Sample Problem: Harmonic Axisymmetric Plane Strain</u>

An infinitely long, solid, circular cylinder is subjected to cos θ and cos θ pressure distributions. A closed-form solution of this problem may be obtained by the use of Reference 2.

The pertinent parameters of the cylinder are:

<u>Dimension and Properties</u>	Loading and <u>Boundary Conditions</u>
$r_o = a$	$\sigma_r = P_o (\cos\theta + \cos 2\theta)$
$\bar{\ell} = a$	$\sigma_{r\theta} = P_o \sin\theta$
$E = 10\text{x}10^6$ psi	$u_z = 0$
$\dot{\nu} = 0.25$	$u_r\|_{r=0} = 0$
$a = 1$ in	$P_o = 10{,}000$ psi

The following solution can be verified by consulting Reference 2.

$$\sigma_r = p_o \left(\frac{\gamma}{a} \cos\theta + \cos 2\theta\right) \qquad (3A.2\text{-}1)$$

$$\sigma_\theta = p_o \left[3\frac{\gamma}{a}\cos\theta + \frac{2r^2 - a^2}{a^2}\cos 2\theta \right]$$

(3A.2-2)

$$\sigma_{r\theta} = p_o \left[\frac{\gamma}{a}\sin\theta + \frac{r^2 - a^2}{a^2}\sin 2\theta \right]$$

(3A.2-3)

$$u_r = p_o \left[\frac{(1-4\upsilon)\ (1+\upsilon) r^2}{2Ea}\cos\theta + \frac{1+\upsilon}{E}\left(r - \frac{2\upsilon r^3}{3a^2} \right)\cos 2\theta \right]$$

(3A.2-4)

$$u_\theta = p_o \left\{ \frac{(5-4\upsilon)\ (1+\upsilon) r^2}{2Ea}\sin\theta + \frac{1+\upsilon}{E}\left[\left(1 - \frac{2\upsilon}{3} \right)\frac{r^3}{a^2} - r \right]\sin 2\theta \right\}$$

(3A.2-5)

The cylinder is idealized by 16 elements (Figure 3A.2-1). Computer results are depicted on Figure 3A.2-2, along with the exact results obtained from Equations 3A.2-4 and 3A.2-5. The computer results are very close to the exact results. Therefore, this problem serves to verify the accuracy of ASAAS for mechanical loading problems where material properties are not variable.

3A.2.3 References

1. Crose, J. G. ASAAS Asymmetric Stress Analysis of Axisymmetric Solids with Orthotropic Temperature-Dependent Material Properties That Can Vary Circumferentially. Air Force Report No. SAMSO-TR-71-297, Aerospace Report No. TR-0172 (S2816-15)-1, December 29, 1971.

2. Love, A. E. H. A Treatise on the Mathematical Theory of Elasticity. Dover Publications, New York, NY, 1944.

3A.3 TAC2D

3A.3.1 General Description

TAC2D is a general purpose, two-dimensional, finite difference heat transfer computer code that is used to determine steady-state and transient temperatures in two-dimensional problems. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular (polar) coordinate system by orthogonal lines of constant coordinate called grid lines. These grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in terms of its capacitance, heat generation, and heat flow paths to neighboring nodal points. The equations for all the nodal points are assembled and solved using an implicit alternating gradient algorithm. TAC2D is a recognized program in the public domain(1).

3A.3.2 Program Verification

<u>Sample Problem</u>

A sample problem is presented to compare the results from TAC2D with an analytical solution. The objective is to show that the TAC2D program yields the correct solution.

<u>Problem Description</u>

The problem is to determine the transient temperature distribution in a right circular cylinder that is initially at temperature T_1. At time t = 0, the temperature at the surface is instantaneously changed to T_2 and maintained at that value. Mathematically, the problem is defined by the following equations:

$$\frac{1}{r}\frac{\delta}{\delta r}\, r\, \frac{\delta T}{\delta t} + \frac{\delta^2 T}{\delta_z^{\ 2}} = \frac{1}{\kappa}\frac{\delta T}{\delta t}; 0 \le r \le R \qquad (3A.3\text{-}1)$$

$$T\left(r, z, 0\right) = T_1 \qquad (3A.3\text{-}2)$$

$$T\left(R, z, t\right) = T_2 \qquad (3A.3\text{-}3)$$

$$T\left(r, \pm\frac{L}{2}, t\right) = T_2 \qquad (3A.3\text{-}4)$$

Where:

t = Time

r = Radius

z = Axial coordinate

R = Outside radius of cylinder

L = Length of cylinder

K = Diffusivity

Further,

$$\kappa = \frac{k}{\rho c} \tag{3A.3-5}$$

Where:

k = Thermal conductivity

ρ = Density

c = Specific heat capacity

For the specific problem analyzed, the following numerical values are used:

R = 12.0 in

L = 48.0 in

k = 20.0 Btu/hr-ft-°F

ρc = 40.0 Btu/cu ft-°F

T_1 = 0.0 °F

T_2 = 1,000.0 °F

<u>Analytical Solution</u>

It may be shown[2] that the solution is:

$$\frac{T - T_1}{T_2 - T_1} = 1 - f(z,t)g(r,t) \tag{3A.3-6}$$

$$f(z,t) = \frac{4}{\pi} \sum_{n=1}^{\infty} \frac{(-1)^{n-1}}{(2n-1)} \cos\left[(2n-1)\frac{\pi z}{L}\right] e^{-\kappa\left(\frac{(2n-1)\pi}{L}\right)^2 t} \tag{3A.3-7}$$

$$g\left(r,t\right)=2\sum_{n=1}^{\infty}\frac{J_0\left(\frac{r}{R}\right)\gamma_m}{\gamma_m J_1\left(\gamma_m\right)}e^{-\kappa y\frac{2}{m}t} \qquad \text{(3A.3-8)}$$

Where:

γ_m = Roots of J_0 (γ_m) = 0 (3A.3-9)

The roots γ_m of Equation 3A.3-9 and the functions J_1 are tabulated in Reference 3 and need not be computed.

From the definition of the problem, there is symmetry about the geometric center of the cylinder and the origin of the coordinate system taken at that point, as is reflected in the boundary conditions, Equations 3A.3-3 and 3A.3-4.

Numerical Solution with TAC2D

A cross section of the problem model for TAC2D is shown on Figure 3A.3-1. The model extends only to the axial midplane of the cylinder, where an adiabatic boundary may be specified by virtue of the symmetry condition described previously. The solid material is represented by one material block. The boundary conditions on the four external boundaries are described by Coolants 1 through 4 (specifically, Coolant Blocks 1 through 4). The material and coolant thermal parameters, as specified by the input functions, are given in Table 3A.3-1. All coolants have the standard specific heat of 1.0 Btu/lb-°F. Coolants 1 and 2, which represent the adiabatic external boundaries, have the standard heat transfer coefficient of 10^{-6} Btu/hr-sq ft-°F and the standard flow rate of 10^{6} lb/hr.

Comparison of the TAC2D Solution with the Analytical Solution

A comparison of the output from the code with the series solution is shown on Figure 3A.3-2. The temperature versus time function is plotted at three representative points within the cylinder. It can be seen that the results from TAC2D are almost identical to the series solution results. The maximum difference between the two sets of results is about 2°F out of a mean magnitude of 100°F.

3A.3.3 References

1. Peterson, J. F. TAC2D-A General Purpose Two-Dimensional Heat Transfer Computer Code. AEC Research and Development Report, Gulf General Atomic Inc., GA-9262, September 6, 1969.

2. Carlslaw, H. S. and Jaeger, J. C. Conduction of Heat in Solids. Oxford at the Clarendon Press, 1959, p 227.

3. Jahnke, E. and Ende, F. Tables of Functions. Dover Publications, Fourth Edition, 1945.

TABLE 3A.3-1
(Sheet 1 of 1)
INPUT THERMAL PARAMETER FUNCTIONS FOR
TAC2D SAMPLE PROBLEM

```
C   MATERIAL THERMAL PARAMETERS

    SPEC1(X)  =  40.0
    RCON1(X)  =  20.0
    ACON1(X)  =  20.0


C   COOLANT THERMAL PARAMETERS

    H3A(X)    =  1.0E+08
    FLO3A(X)  =  1.0E+08
    TIN3A(X)  =  1460.0
    H4A(X)    =  1.0E+08
    FLO4A(X)  =  1.0E+08
    TIN4A(X)  =  1460.0
```

3A.4 MAT 6

3A.4.1 General Description

MAT 6 analyzes a symmetrically loaded circular plate on an elastic foundation and maintains compatibility between: 1) the plate (foundation mat) and the subgrade, and 2) the plate and the circular walls supported thereon. The program computes the discontinuity effects at the interface of the mat and circular walls and includes those effects in the analysis.

The general method is described by Zhemochkin[1]. The displacements and stresses of the subgrade are derived from a Boussinesq solution for a circular plate supported on a semi-infinite, elastic half space[2]. The elastic behavior of the mat and circular wall (or walls) is basically described by Timoshenko and Woinowsky-Krieger[3].

3A.4.2 Program Verification

MAT 6 is used to analyze the containment foundation mat and provide contact pressure and discontinuity forces at the junction of the mat and superstructure (i.e., the containment wall, crane wall, and reactor support wall).

Included are plots (Figures 3A.4-1 through 3A.4-4) of radial and tangential bending moments and the radial shear in the mat for a MAT 6 solution versus a SHELL 1 solution (Section 3A.1). Also shown are the discontinuity forces at the interface of the mat and the containment wall.

3A.4.3 References

1. Zhemochkin, B. N. Practical Methods for Analysis of Beams and Plates on Elastic Foundations. 2nd Edition, Moscow, 1962.

2. Timoshenko, S. P. and Goodier, J. N. Theory of Elasticity. 3rd Edition, McGraw-Hill Book Company, Inc., New York, NY, 1970.

3. Timoshenko, S. and Woinowsky-Krieger, S. Theory of Plates and Shells. 2nd Edition, McGraw-Hill Book Company, Inc., New York, NY, 1959.

3A.5 STRUDL II, PD-STRUDL and STRUDL-SW

3A.5.1 General Description

STRUDL II (Structural Design Language) has been designed as a modified subsystem of the Integrated Civil Engineering System (ICES) which was designed and formulated by the Department of Civil Engineering at Massachusetts Institute of Technology[1].

3A.5.2 Program Verification

The finite element method provides for the solution of a wide range of solid mechanics problems[2]. Its implementation within the context of the STRUDL II analysis facilities expands these for the treatment of plane stress, plane strain, plate bending, shallow shell, and three-dimensional stress analysis problems.

STRUDL II also provides a dynamic analysis capability for linear elastic structures undergoing small displacements. Either free or forced vibrational response may be obtained, and in the latter case, the forcing functions may be in the form of time-histories or response spectra.

The three-dimensional finite element capability of STRUDL II is used to analyze the drywell at the region of the equipment hatch and personnel door assembly and other regions of interest.

Category I structures are analyzed for seismic effect using the dynamic analysis capability of STRUDL II. The analysis yields frequencies of vibration, mode shapes, displacements, velocities, accelerations, and forces. STRUDL is a recognized program in the public domain. The software system is IBM-MVS, Release 3.8. The hardware configuration is IBM-3033.

STRUDL-SW, a static analysis subset of STRUDL II, is primarily used in pipe support design. It can design or check structural members of a structure, according to various codes such as 1974 ASME Section III Subsection NF, and the 1970 AISC manual. Input consists of structural geometry and topology, material constants, member properties, joint and member loads, and design constraints and parameters. STRUDL-SW output includes: joint displacements, loads, and reaction; member distortion, forces, and stresses; printer plot of structure; and design data.

PD-STRUDL, a static and dynamic analysis subset of STRUDL II, is primarily used in structural design. It can design or check structural members of a structure, according to various codes such as 1974 ASME Section III Subsection NF, and the 1970 AISC Manual. Inputs consist of structural geometry and topology, material constants, member properties, joint and member loads, and design constraints and parameters. PD-STRUDL output includes: joint displacements, loads, and reactions; member distortion, forces, and stresses; printer plot of structures; and design data.

PD-STRUDL is a proprietary version of STRUDL which is maintained and enhanced by P-DELTA, Inc. PD-STRUDL is based on the material placed in the public domain by Massachusetts Institute of Technology. PD-STRUDL has been verified via the X/PD STRUDL verification manual, November 1990 edition, which was produced by P-DELTA, Inc., 42 St. Ann's Road, Quincy, Massachusetts 02170.

3A.5.3 References

1. ICES STRUDL - II Structural Design Language Engineering Users' Manual, Vol. I Frame Analysis, November 1968, Vol. II Addition Design and Analysis Facilities (Chapters III and IV), June 1971.

2. Zienkiewicz, O. C. and Cheung, Y. K. The Finite Element Method. McGraw-Hill Book Company, Inc., New York, NY, 1967.

3A.6 TIME-HISTORY (TIMHIS6) PROGRAM

3A.6.1 General Description

The TIMHIS6 program computes time-history response and amplified response spectra (ARS) at any mass point location of a lumped-mass, spring-connected system due to a synthetic earthquake, time-motion record input. The program calculates the time-history response at the selected mass locations by standard modal superposition. The responses are computed by integration of the modal equations of the system by the exact method[1]. The analytical procedure is described in Section 3.7A.2.1. The program's main application is the generation of ARS used for design of Category I equipment and piping.

3A.6.2 Program Verification

TIMHIS6's solution to a test problem is substantially identical to the solution obtained using STRUDL II. The test problem uses an actual containment structure subjected to an earthquake, time-motion record input of Helena East-West normalized to 0.06 g. The time-history response of the structure is computed at the operating floor level by TIMHIS6 and STRUDL II. The results of these two analyses (Figures 3A.6-1 and 3A.6-2, respectively) agree extremely well with each other.

3A.6.3 Reference

1. Wigan, N. C. and Jennings, P. C. Digital Calculation of Response-Spectrum from Strong-Motion Earthquake Records. National Science Foundation, June 1968.

3A.7 HAS BEEN DELETED

3A.8 DINASAW

3A.8.1 General Description

DINASAW (Dynamic Inelastic Nonlinear Analysis by Stone & Webster) is a modification and extension of a lumped mass elastic-plastic dynamic analysis code used to predict the large-deflection behavior of beams and rings[1]. DINASAW extends this analysis to cover pipes (tubular cross sections) that may impact walls or restraints.

The analysis, as derived, employs the spatial finite-element method in which the tangential and normal displacement fields are represented by cubic interpolations[1,2]. By applying the principle of virtual work in conjunction with D'Alembert's principle, the equations of motion may be derived in the following form:

$$[M]\ [\ddot{q}] = (F)-(P)-(H)\ [q] \qquad (3A.8\text{-}1)$$

Where:

$[q]$ and $[\ddot{q}]$	=	Generalized displacements and generalized accelerations, respectively, for the complete assembled discretized structure, defined with respect to a global coordinate system
[M]	=	Lumped mass matrix for the complete assembled discretized structure
(F)	=	Assembled vector of externally applied generalized loading
(P)	=	Assembled internal generalized force matrix (replaces conventional stiffness matrix)
(H) [q]	=	Generalized loads arising from both large deflection and plastic behavior

3A.8.2 Program Verification

Two examples are discussed here. The first involves a ring subjected to a radial blast wave over a portion of its circumference[1]. The resulting deformation severely distorts the ring, flattening it considerably. The computer code results closely follow both the displacement field and the strain time history.

The second case involves the impact of a rotor segment onto a ring or shroud[2]. Again the program, in conjunction with the collision-imparted velocity method (CIVM), follows experimental results very closely.

3A.8.3 References

1. Wu, R. and Witmer, E. Finite-Element Analysis of Large Transient Elastic-Plastic Deformations of Simple Structures, with Application to the Engine Rotor Fragment Containment-Deflection Problem. Aeroelastic and Structures Research Laboratory, Department of Aeronautics and Astronautics, Massachusetts Institute of Technology, January 1972.

2. Collins, T. and Witmer, E. Application of the Collision Imparted Velocity Method for Analyzing the Responses of Containment and Reflector Structures to Engine Rotor Fragment Impact. Aeroelastic and Structures Research Laboratory, Department of Aeronautics and Astronautics, Massachusetts Institute of Technology, August 1973.

3A.9 LIMITA2

3A.9.1 General Description

LIMITA2 (ST-223) is a two-dimensional, nonlinear, transient dynamic computer code developed and fully documented by Stone & Webster Engineering Corporation (SWEC) for in-house use. A plane frame is simulated as a lumped parameter system, represented mathematically by an assembly of discrete lumped masses connected by beam members. Under any loading, the equilibrium at mass point r is ensured by the equation of motion:

$$m_r \ddot{q}_r + \sum^{i} K_{ri} q_i = f_{\bar{r}} \tag{3A.9-1}$$

Where:

Σ = A series with one term for each of the i displacements

K_{ri} = The member stiffness, which is defined as the force necessary to hold the structural member from moving in degree of freedom r when degree of freedom i is given a unit displacement when all other degrees of freedom are restrained from moving[1,2]

$f_{\bar{r}}$ = External load factor

m_r = Discrete mass point r of the structure

q_i, $\ddot{q}_r$ = The generalized displacement and accelerations, respectively, for the complete assembled discretized structure defined with respect to a global system

To take account of nonlinear effects, such as plasticity and large deflections, Equation 3A.9-1 is solved by an incremental method[3]. At any particular time, t, the displacement increment is obtained from:

$$m_r \overset{t}{\ddot{q}_r} + \sum^{i} \overset{t}{K_{ri}} \Delta \overset{t}{q_i} = \overset{t}{f_{ri}} - \sum_{s=o}^{t-\Delta t} \left(\sum^{i} \overset{s}{K_{ri}} \Delta \overset{s}{q_i} \right) \tag{3A.9-2}$$

Where:

K^t_{ri} = The member stiffness

f^t_{ri} = The forcing function

which are calculated based on the current deformed structure[4] and assumed constant through the time step, Δt. The displacement and member forces are thus given by:

$$\overset{t}{q_r} = \sum_{s=o}^{t} \Delta \overset{s}{q_r}$$

$$Q_r^t = \sum_{s=o}^{t} \left(\overset{i}{\Sigma} \overset{s}{K_i} \Delta \overset{s}{q_i} \right)$$

(3A.9-3)

Where:

q_r = Member r displacement vector

Q_r = Member r force vector

The second order differential system equations (Equation 3A.9-2) are solved by a linear acceleration method[5].

Since no external loading is applied to a member between nodes, the maximum value of the internal force acting on a member occurs at its end sections. The transition from the elastic to the fully plastic state is disregarded, and the end sections are assumed to remain linearly elastic up to the full plastic yield surface.

The yield surface is defined by a scalar function of the internal member forces, Q, of the form:[6-8]

$$\Phi(Q^t) = 1 \qquad (3A.9\text{-}4)$$

Here the function, Φ, is obtained by integrating the stress across the section with the stress fully developed over the section and satisfying the von Mises (or Tresca) yield criterion:

$$\sigma^2 + \gamma^2 \tau^2 = \sigma_y^2$$

(3A.9-5)

Where:

σ = Normal stress

τ = Shear stress

σ_y = Yield stress in simple tension

γ^2 = 3 (von Mises) or 4 (Tresca)

Thus, the function, ϕ, depends on the shape of the cross section and the force components being considered.

For a frame structure, the yielding normally occurs due to either a predominant bending moment or to a predominant tension or compression. Thus, two plastic models are provided:

1. Bending Predominant Members

   Since a section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic,

   b. End A is yielding and B is elastic,

   c. End A is elastic and B is yielding, or

   d. Both ends A and B are yielding.

   A plastic hinge is introduced at any end section that is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear strain-hardening curve[9,10]. In situations where force unloads, the elastic stiffness of the hinged member is restored (isotropic strain-hardening model).

2. Tension or Compression Predominant Members

   There are only two possible states:

   a. Entire member is elastic, or

   b. Entire member is plastic.

   When the member yields, Young's Modulus is replaced by a plastic tangent modulus and the force-displacement curve follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.9.2 Program Verification

SWEC sponsored an experimental investigation performed by the Massachusetts Institute of Technology (MIT)[11]. The problem consisted of the cantilevered pipe (Figure 3A.9-1) subjected to an impulsive load at its free end. The impulse is imparted by the detonation of a sheet of high explosive, separated from the pipe by a buffer material. A nearly uniform initial velocity is

produced in the loaded region and is determined by high-speed photography. This problem was analyzed by LIMITA2. The results were compared with experimental data and output from another computer program, DINASAW (Section 3A.8).

The stress-strain curves used in the LIMITA2 and DINASAW calculations are shown on Figure 3A.9-2 with the experimentally derived curve. Figure 3A.9-3 shows the lumped-mass models used for both computer solutions. The impulsive load, idealized as initial nodal velocities, is also shown on Figure 3A.9-3. Time-history plots of the x and y displacements of the free end of the pipe for the LIMITA2 and DINASAW runs are shown on Figures 3A.9-4 and 3A.9-5, respectively. The moment reaction at the clamped end of the pipe is shown on Figure 3A.9-6. A comparison of the permanent pipe deformations predicted by the experiment, DINASAW, and LIMITA2 is illustrated on Figure 3A.9-7. Agreement is good in all cases, as seen on Figures 3A.9-4 through 3A.9-7. Additional problems were also evaluated to ensure that all program options were exercised, and thus demonstrate the function and adequacy of this program.

3A.9.3 References

1. Martin, H. C. Introduction to Matrix Methods of Structural Analysis. McGraw-Hill Book Company, Inc., New York, NY, 1966.

2. Przemieniecki, J. S. Theory of Matrix Structural Analysis. McGraw-Hill Book Company, Inc., New York, NY, 1968.

3. Clough, R. W. and Wilson, E. L. Dynamic Response by Step-by-Step Matrix Analysis. Symposium on Use of Computers in Civil Engineering, Lisbon, Portugal, 1962, p 45.1-45.14.

4. Martin, H. C. On the Derivation of Stiff Matrices for the Analysis of Large Deflection and Stability Problems. Proc. Conf. Matrix Methods Structure Mech., Wright-Patterson Air Force Base, Ohio, October 26-28, 1965, AFFBL TR 66-80, 1966.

5. Hildebrand, F. B. Introduction to Numerical Analysis. McGraw-Hill Book Company, Inc., New York, NY, 1956.

6. Hodge, P. G. Plastic Analysis of Structures. McGraw-Hill Book Company, Inc., New York, NY, 1959.

7. Neal, B. G. The Effect of Shear and Normal Forces on the Fully Plastic Moment of a Beam of Rectangular Cross Section. Journal of Applied Mechanics, June 1961, p 269-274.

8. Stokey, W. F.; Peterson, D. B.; and Wruder, R. A. Limit for Tubes Under Internal Pressure, Bending Moment, Axial Force and Torsion. Nuclear Engineering and Design, 4, North-Holland Publishing Company, Amsterdam, 1966, p 193-261.

9. Clough, R. W.; Benuska, K. L.; and Wilson, E. L. Inelastic Earthquake Response of Tall Buildings. Proceedings of the Third World Conference on Earthquake Engineering, Vol. II, Auckland and Wellington, New Zealand, January 1965, p 68-69.

10. Giberson, M. F. The Response of Non-Linear Multi-Story Structures Subjected to Earthquake Excitation. Earthquake Engineering Research Lab, California Institute of Technology, Pasadena, CA, June 1967.

11. Pirotin, S. O. and East, G. H. Large Deflector, Elastic-Plastic Response of Piping: Experiment, Analysis, and Application. Transactions of the Fourth SMIRT Conference, Paper F3/1, San Francisco, CA, August 1977.

3A.10 LIMITA3

3A.10.1 General Description

LIMITA3 (ST-225) is a computer code developed and fully documented by SWEC for in-house use. Its formulation is identical to that of LIMITA2 (Section 3A.9.1), with the exception that the equations are applicable to a general three-dimensional problem. For a space frame, yielding normally occurs due to either a predominant bending moment or a predominant torsion (combined with axial load). Therefore, two plastic models are provided:

1. Bending Yield Model

   Since a beam section is either elastic or fully plastic, there are four possible states:

   a. Both ends A and B are elastic,

   b. End A is plastic, end B is elastic,

   c. End A is elastic, end B is plastic, or

   d. Both ends A and B are plastic.

   A plastic hinge is introduced at any end section that is yielding. The force-displacement relation of the plastic hinge follows an ideal bilinear strain-hardening curve[1,2]. In situations where the force unloads, the elastic stiffness of the hinged member is restored (isotropic strain-hardening model).

2. Torsional-Axial Yield Model

   There are only two possible states:

   a. The entire member is elastic, or

   b. The entire member is plastic.

   When the member yields, the Young's Modulus is replaced by a plastic tangent modulus and the force-displacement relation follows a bilinear curve. If the member unloads, the elastic modulus is restored.

3A.10.2 Program Verification

<u>Elastic Example</u>

Consider the dynamic response of a space frame (Figure 3A.10-1) subjected to a step load of 30 kips at joint 6. This problem was analyzed by LIMITA3 and STRUDL II elastically. The results of displacements and moment z at joint 6 were plotted against each

other on Figures 3A.10-2 and 3A.10-3, respectively. As shown, there is excellent agreement.

Plastic Example

This example is provided to illustrate the ability of the program to determine the inelastic transient response of a three-dimensional structure. The structure considered consists of cantilevered steel tubes (Figure 3A.10-4) subjected to force transients, causing bending and torsion in the structure. The results obtained from an analysis using the LIMITA3 code are compared with data obtained experimentally[3].

The experiment was a drop test in which the cantilevered tubes were loaded by weights at each tube end. The results tabulated were the peak deflections and their corresponding times and the permanent deflections. These results are compared to those obtained using the LIMITA3 code in Table 3A.10-1.

Additional problems, elastic and inelastic, were analyzed to ensure that all program options were exercised, and thus demonstrate the function and adequacy of this program.

3A.10.3 References

1. Clough, R. W.; Benuska, K. L.; and Wilson, E. L. Inelastic Earthquake Response of Tall Buildings. Proceedings of the Third World Conference on Earthquake Engineering, Vol. II, Auckland and Wellington, New Zealand, January 1965, p 68-69.

2. Giberson, M. F. The Response of Non-Linear, Multi-Story Structures Subjected to Earthquake Excitation. Earthquake Engineering Research Lab, California Institute of Technology, Pasadena, CA, June 1967.

3. Larson, L. D. Inelastic Response of Pressurized Tubes Under Dynamic Bending and Torsional Loads. Ph.D. Thesis, Mechanical Engineering Department, Carnegie-Mellon University, University Microfilm Order No. 73-22872, 1973.

TABLE 3A.10-1
(Sheet 1 of 1)
COMPARISON OF EXPERIMENTAL DATA AND
ANALYTICAL DATA USING LIMITA3

	Experimental Values	LIMITA3 Computer Results	Difference (%)
Peak deflection-in node 6	0.297	0.310	4.2
Peak deflection-in node 9	(Not determined)	0.870	-
Time at peak deflection-sec node 6	0.004	0.0042	4.8
Permanent deflection-in node 6	0.144	0.140	2.8
Permanent deflection-in node 9	0.302	0.310	2.6

3A.11 STARDYNE

3A.11.1 General Description

The STARDYNE Structural Analysis System, written by Mechanics Research, Inc., of Los Angeles, CA, is a fully warranted and documented computer program available at Control Data Corporation.

The MRI STARDYNE Analysis System consists of a series of compatible, digital computer programs designed to analyze linear and nonlinear elastic structural models. The system encompasses the full range of static and dynamic analyses.

3A.11.2 Program Verification

The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal-applied loads and prescribed displacements.

Utilizing the normal mode technique, linear dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady-state harmonic, random, and shock spectra excitation types. Dynamic response results can be presented as structural deformations and internal member loads.

The nonlinear dynamic analysis program is integrated in the rest of the STARDYNE system. The equations of motion for the linear portion of the structural model are generated and modified to account for the nonlinear springs. The resulting nonlinear equations of motion are directly integrated, using either the Newmark or Wilson implicit integration operators. The user may enter sets of structural loadings, which vary with time, and specify time points at which the program is to output the structural response.

3A.12 SSLAM

3A.12.1 General Description

This is a computer code used to calculate submerged structure loads due to disturbances in the suppression pool. The model assumes the fluid to be compressible and the disturbances to be isolated point sources. The velocity potential of the fluid is governed by the inhomogeneous wave equation, constant potential at the pool surface, and homogeneous normal derivatives at the pool solid boundaries.

The code computes the numerical results of the flow field from the analytical solution. The solution method used is the Green's function solution. It is an extension of the Improved Chugging Methodology used to compute the pressures on the boundary of the Mark II suppression pool[1].

The total load on the structure is then calculated from the flow field by vectorially adding the contribution from standard drag, acceleration drag, and lift[2].

3A.12.2 Program Verification

Verification of the flow field calculation is accomplished by a comparison with a manual calculation of the problem shown on Figure 3A.12-1 and in Table 3A.12-1.

3A.12.3 References

1. Mark II Improved Chugging Methodology, NEDE-24822-P, Class III, May 1980.

2. NUREG-0487, Supplements 1 and 2, USNRC.

TABLE 3A.12-1
(Sheet 1 of 2)

SSLAM PROGRAM OUTPUT DATA VERIFICATION

Node Number	Flow Field at Submerged Structure Node Segment (Time = 0.038 sec)	Computer Value	Hand-Calculated Value
5	Acceleration (ft/s^2)		
	X	-0.010	-0.010
	Y	-0.140	-0.140
	Z	-0.004	-0.004
5	Velocity (ft/s)		
	X	-0.107×10^{-4}	-0.107×10^{-4}
	Y	0.389×10^{-5}	0.389×10^{-5}
	Z	-0.145×10^{-3}	-0.145×10^{-3}

NOTE: The following input data were used for the SSLAM sample problem.

Pool Modes

Radial	1
Angular	3
Axial	1

Pool Dimensions (ft)

Outer radius	45.5
Inner radius	14.1667
Depth	24.0

Source Information

Time step	0.002 sec
Pool and wall damping	0.0744
Fluid sound speed	500 m/s
Vent damping	0.065
Source frequency	3.0 Hz
Source amplitude	0.8 m^3/s^2

TABLE 3A.12-1
(Sheet 2 of 2)

SSLAM PROGRAM OUTPUT DATA VERIFICATION

Source Locations

Source Number	Ring Radius (ft)	$\bar{\theta}$ (deg)	Y (ft)
1	19.5	4.5	14.0
2	19.5	49.5	14.0
3	19.5	94.5	14.0
4	19.5	139.5	14.0
5	19.5	184.5	14.0
6	19.5	229.5	14.0
7	19.5	274.5	14.0
8	19.5	319.5	14.0
Target location	20.0	0.0	19.2

3A.13 SSLOAD

3A.13.1 General Description

SSLOAD is used to determine loads on structures submerged in the suppression pool due to loss-of-coolant accident (LOCA) events of chugging, condensation oscillations (CO), and air bubble. The program calculates flow fields based on potential flow with point sources and determines acceleration, velocity drag, and lift forces normal to structures of cylindrical cross section[1-4].

The pool boundaries are either rectangular in shape or are approximated by rectangular surfaces to enable solution by the method of images (MOI) technique. The program also calculates interference effects on acceleration drag based on two-dimensional potential flow[7]. It modifies the inertial coefficient and combines convective and standard drag coefficients by the square root of the sum of the squares (SRSS).

SSLOAD input requires information on pool geometry, sources, structure, and load coefficients. If interference effects are to be calculated, the location and size of the interfering structures are required. The output provides magnitude and direction of the flow fields at the center of each structural segment and the acceleration, velocity drag, and lift loads. Information on source and load coefficient adjustments is also output in addition to locations of sources and structure segments in the approximated pool.

3A.13.2 Program Verification

Verification of the flow fields, loads, and source adjustments is accomplished by a comparison with a manual calculation of the problem given in Figures 3A.13-1 and 3A.13-2. Verification of the interference factors for acceleration drag is accomplished by comparison with known numerical values[5,6]. These comparisons are presented in Tables 3A.13-1 and 3A.13-2.

3A.13.3 References

1. Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety Relief Valve Ramshead Air Discharges. NEDO-21471, September 1977.

2. Mark II Pressure Suppression Containment Systems - Loads on Submerged Structures; An Application Memorandum. NEDE-21730, December 1977.

3. Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria. NUREG-0487, October 1979 and Supplement, September 1980.

4. Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety Relief Valve

Ramshead Air Discharge. Appendix A, NEDE-21471P Class III, September 1977.

5. Reformulation of Equation A-64, in NEDE-21471 Attachment to GE Letter, Mark II-583-E, March 7, 1978.

6. Mark II Containment Dynamic Forcing Functions Information Report. NEDO-21061, Revision 3, June 1978, p 3-34.

7. Dalton, C. and Helfinstein, R. A. Potential Flow Past a Group of Circular Cylinders, Journal of Basic Eng., ASME, p 636-642, December 1971.

8. Yamamoto, T. Hydrodynamic Forces on Multiple Circular Cylinders. Journal of the Hydraulics Division, ASCE, p 1193-1210, September 1976.

TABLE 3A.13-1
(Sheet 1 of 1)

SSLOAD PROGRAM OUTPUT DATA VERIFICATION - COMPUTER VALUES

Segment Number	Force on Submerged Structure Segments	Computer Value (lbf)
1	Acceleration drag	
	X	24.52
	Y	0.0
	Z	0.0
1	Velocity drag	
	X	0.96
	Y	0.0
	Z	0.0
1	Lift	
	X	0.0
	Y	0.48
	Z	0.0
Acceleration drag coefficient (C_M)		2.0948*
Convective drag coefficient (C_V)		0.2593*

* See Figure 3A.13-2.

TABLE 3A.13-2
(Sheet 1 of 1)

SSLOAD PROGRAM OUTPUT DATA VERIFICATION -
HAND-CALCULATED VALUES

Segment Number	Force on Submerged Structure Segments	Hand-Calculated Value (lbf)
1	Acceleration drag	
	X	24.52
	Y	0.0
	Z	0.0
1	Velocity drag	
	X	0.96
	Y	0.0
	Z	0.0
1	Lift	
	X	0.0
	Y	0.48
	Z	0.0
Acceleration drag coefficient (C_M)		2.1*
Convective drag coefficient (C_V)		0.25*

* See Figure 3A.13-2.

3A.14 MISSILE

3A.14.1 General Description

MISSILE calculates the impact probability (P_2) of postulated turbine missiles on specified targets. The solid angle method is used to calculate P_2:

$$P_2 = \frac{1}{\Omega_m} \int d\Omega \qquad (3A.14\text{-}1)$$

Where:

Ω = Solid angle subtended by the target

Ω_m = Total solid angle subtended by all possible missile trajectories

The integral is evaluated by numerical integration, with consideration of the missile ejection velocity and the relative positions of the turbine and target (Figure 3A.14-1).

3A.14.2 High Trajectory Verification

Westinghouse has derived a formula to predict the probability of impact for high-trajectory missiles[1]. Some adjustments to the formula are necessary to enable direct comparison with the program results. The formula has been derived on the basis that the initial velocity is random and uniformly distributed between V_1 and V_2. The program uses a deterministic initial velocity. The formula may be specialized to this condition by setting V_1 equal to V_2 after applying L'Hopital's Rule. Also, the formula has been derived assuming that a missile fragment occurs in the quadrant of the target; whereas, the program assumes that a missile fragment can occur in any of the four quadrants. These differing assumptions can be reconciled by using four fragments for program input.

After making the above adjustments, the high-trajectory formula becomes:

$$P = G^2 / \left(2\pi\Delta V^4\right) \qquad (3A.14\text{-}2)$$

Where:

P = Impact probability/sq ft of target

G = Acceleration of gravity, ft/sec^2

Δ = Deflection angle range, radians

V = Initial velocity, fps

Comparison of the probability calculated by the formula and the results of the computer program are given in Table 3A.14-1.

3A.14.3 Low-Trajectory Verification

The probability of impact for a low-trajectory missile (LTM) calculated by this program was verified by comparison with Bush[2]. The LTM was identified as four fragments of the outer disk resulting from a turbine failure. The two different initial ejection velocities were 300 and 600 fps. The geometry is shown on Figure 3A.14-2.

Since only half of Bush's 4,800-sq ft target lies in the reported interval ($0 \leq \delta < 25°$), only a 2,400-sq ft portion was modeled in MISSILE.

A comparison of the probability listed in Bush and the results of the computer program is provided in Table 3A.14-1.

3A.14.4 References

1. Westinghouse Electric Corporation. Analysis of the Probability of the Generation and Strike of Missiles from a Nuclear Turbine. Steam Turbine Division, March 1974, p 48.

2. Bush, S. H. Probability of Damage to Nuclear Components Due to Turbine Failure. Nuclear Safety, Vol. 14, No. 3, May-June 1973, p 197.

TABLE 3A.14-1
(Sheet 1 of 1)

MISSILE PROGRAM VERIFICATION

Comparison of High-Trajectory Probabilities

Velocity (fps)	Deflection Angle (deg)	Formula[1]	Program
300	5	0.254E-3	0.259E-3
300	25	0.508E-4	0.534E-4
600	5	0.162E-4	0.160E-4
600	25	0.318E-5	0.330E-5

Comparison of Low-Trajectory Probabilities

Velocity (fps)	Deflection Angle (deg)	Bush[2]	Program
300	$0=\delta<25$	0.113	0.111
600	$0=\delta<25$	0.11	0.112

Sources:

[1] Reference 1.
[2] Reference 2.

3A.15 NUPIPE

3A.15.1 General Description

NUPIPE was developed by the Nuclear Services Corporation and is fully documented. The SWEC version of NUPIPE[3] differs slightly from the public domain program NUPIPE in the postprocessing of the analytical results.

NUPIPE performs a linear elastic analysis of three-dimensional piping systems subjected to thermal, static, and dynamic loads. It utilizes the finite element method of analysis with special features incorporated to accommodate specific requirements in piping analysis. In addition, it checks analytical conformance to ASME Section III and ANSI B31.1. This program accepts the complete geometric and physical description of the piping system, provides a complete error and coordinate check for the inputs, and computes internal forces and moments, support and equipment reactions, and displacements and stress values for a variety of loading cases.

NUPIPE-SWPC is a personal computer (PC) version of NUPIPE. This PC version of NUPIPE includes the same features as the mainframe version referenced above, with some enhancements.

3A.15.2 Program Verification

NUPIPE has been verified with ADLPIPE[1] for thermal, weight, and response spectrum seismic analyses. The results from both programs are presented in Tables 3A.15-1 through 3A.15-7. The model used for this comparison is presented on Figure 3A.15-1.

The comparison is also made with ASME Benchmark solution for force time history dynamic response[2]. The model used for this comparison is shown on Figure 3A.15-2. The results for comparisons are presented in the form of plots on Figure 3A.15-2. The natural frequencies are given in Table 3A.15-8.

The Safety Class 1 piping stress computation conforms with the hand calculations. The model used is shown on Figure 3A.15-3. The results are tabulated in Tables 3A.15-9 and 3A.15-10.

NUPIPE-SWPC[4] is a proprietary PC version of NUPIPE and has been verified by demonstrating that similar results are obtained for several identical problems benchmarked on the mainframe version. These results are documented in the verification manual[5].

In addition, whenever a new version of NUPIPE (mainframe or PC) is installed or the hardware configuration is changed, appropriate verification problem(s) are rerun. The old and new results are compared to ensure that program integrity has been maintained.

3A.15.3 References

1. Arthur D. Little, Inc. ADLPIPE: Static, Dynamic, Thermal Pipe Stress Analysis.

2. American Society of Mechanical Engineers. Pressure Vessel and Piping; 1972 Computer Programs Verification. Problem No. 5.

3. NUPIPE-SW Program User's Manual, ME-110 Versions 03, 04, 05 and 06, Stone & Webster Engineering Corporation.

4. NUPIPE-SWPC Program User's Manual, ME-110.01U, Version 01, Stone & Webster Engineering Corporation.

5. NUPIPE-SWPC Program Verification Manual, ME-110.01, Version 01, Stone & Webster Engineering Corporation.

TABLE 3A.15-1
(Sheet 1 of 1)

COMPARISON OF SUPPORT REACTION DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	Program	Forces (lb)			Moments (in/lb)		
		FX	FY	FZ	MX	MY	MZ
170	NUPIPE ADLPIPE	-9,154 -9,178	7,541 7,540	4,492 4,492	-5,952 -5,529	-823,420 -823,420	1,241,512 1,241,512
218	NUPIPE ADLPIPE			16,650 16,622			
330	NUPIPE ADLPIPE	34,532 34,511	-33,620 -33,608	-31,750 -31,736	-486,338 -486,386	-1,516,811 -1,519,359	573,673 573,438
390	NUPIPE ADLPIPE		8,631 8,678				
430	NUPIPE ADLPIPE	1,702 1,746	798 768	12,553 12,541	-28,147 -26,917	164,346 166,180	248,852 250,956

TABLE 3A.15-2
(Sheet 1 of 1)

COMPARISON OF DEFLECTIONS AND ROTATIONS DUE TO THERMAL, ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

Node	Program	Deflection (in)			Rotation (rad)		
		DX	DY	DZ	RX	RY	RZ
197	NUPIPE	0.348	-0.141	0.230	-0.0026	0.0025	-0.0084
	ADLPIPE	0.348	-0.141	0.229	-0.0026	0.0025	-0.0084
212	NUPIPE	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
	ADLPIPE	1.120	0.052	-0.023	-0.0092	-0.0051	-0.0115
230	NUPIPE	1.276	-0.028	-0.548	-0.0066	-0.0044	0.0024
	ADLPIPE	1.276	-0.027	-0.548	-0.0066	-0.0044	0.0024
260	NUPIPE	0.512	-0.001	-0.520	-0.0034	-0.0005	0.0035
	ADLPIPE	0.512	-0.000	-0.520	-0.0035	-0.0005	0.0035
390	NUPIPE	0.066	-0.000	0.249	-0.0010	0.0026	-0.0020
	ADLPIPE	0.067	-0.000	0.248	-0.0010	0.0026	-0.0020
420	NUPIPE	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007
	ADLPIPE	-0.029	-0.079	0.011	-0.0002	-0.0002	-0.0007

TABLE 3A.15-3
(Sheet 1 of 1)

COMPARISON OF STRESS DUE TO THERMAL,
ANCHOR MOVEMENT, AND EXTERNAL FORCE LOADING

	Stress (psi)	
Node	NUPIPE	ADLPIPE
180	18,989	19,013
199	17,703	17,731
214	23,958	23,955
236	14,427	14,416
265	6,254	6,251
305	12,539	12,532
344	11,845	11,838
370	6,295	6,296
395	3,476	3,473
430	3,282	3,308

TABLE 3A.15-4
(Sheet 1 of 1)

COMPARISON OF INTERNAL FORCES DUE TO DEADWEIGHT ANALYSIS

Node	Program	Forces (lb)			Moments (in/lb)		
		FX	FY	FZ	MX	MY	MZ
197	NUPIPE	295	2,337	14	-35,864	5,218	51,979
	ADLPIPE	290	2,341	15	-35,108	5,231	52,081
212	NUPIPE	295	3,306	14	59,390	5,394	14,010
	ADLPIPE	299	3,310	15	59,735	-5,500	14,542
360	NUPIPE	330	2,781	-29	30,930	-22,748	-84,971
	ADLPIPE	326	2,783	-32	31,920	-23,105	-82,784
390	NUPIPE	330	4,933	-29	-255,351	701	126,476
	ADLPIPE	336	4,707	-32	-256,444	916	126,716
420	NUPIPE	330	-492	-29	-8,972	27,075	82,202
	ADLPIPE	336	-497	-32	-9,181	27,724	80,676

TABLE 3A.15-5
(Sheet 1 of 1)

COMPARISON OF DEFLECTIONS AND ROTATION DUE TO
DEADWEIGHT ANALYSIS

Node	Program	Deflections (in)			Rotations (rad)		
		DX	DY	DZ	RX	RY	RZ
197	NUPIPE	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
	ADLPIPE	0.007	-0.014	-0.004	0.0001	0.0001	0.0002
212	NUPIPE	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
	ADLPIPE	-0.005	-0.013	0.013	0.0006	0.0001	0.0004
360	NUPIPE	-0.008	-0.068	0.024	0.0004	-0.0000	-0.0004
	ADLPIPE	-0.009	-0.069	0.024	0.0004	0.0000	-0.0004
390	NUPIPE	-0.014	-0.000	-0.003	0.0002	-0.0003	-0.0005
	ADLPIPE	-0.015	-0.000	-0.003	0.0002	-0.0002	-0.0005
420	NUPIPE	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002
	ADLPIPE	-0.001	0.002	-0.001	-0.0000	-0.0001	-0.0002

TABLE 3A.15-6
(Sheet 1 of 1)

COMPARISON OF STRESSES DUE TO DEADWEIGHT ANALYSIS

Node	NUPIPE (psi)	ADLPIPE (psi)
180	685	694
199	448	458
214	667	679
236	2,472	2,449
265	530	524
305	515	522
344	635	631
370	679	677
395	575	580
430	1,101	1,091

TABLE 3A.15-7
(Sheet 1 of 1)

COMPARISON OF NATURAL FREQUENCIES
(Hz)

	Mode				
	1st	2nd	3rd	4th	5th
NUPIPE	7.109	9.328	12.297	14.681	18.043
ADLPIPE	7.118	9.329	12.492	14.427	17.714

TABLE 3A.15-8
(Sheet 1 of 1)

COMPARISON OF NATURAL FREQUENCIES
(Hz)

	Mode	
	1st	2nd
NUPIPE	2.407	13.537
Benchmark program	2.3288	13.0808

TABLE 3A.15-9
(Sheet 1 of 1)

NUPIPE VERSUS HAND CALCULATION

Point No. 20	Hand Calculation	NUPIPE
Minimum wall thickness	0.032 in	0.032 in
Primary stress (Eq. 9)	3,713 psi	3,712 psi
Primary and secondary stress (Eq. 10)	16,041 psi	16,038 psi
Alternating stress (Eq. 11 and 14)	13,468 psi	13,465 psi
Usage factor	0.0654	0.0631
Point No. 30		
Minimum wall thickness	0.047 in	0.047 in
Primary stress (Eq. 9)	8,748 psi	8,741 psi
Primary and secondary stress (Eq. 10)	117,655 psi	117,546 psi
Expansion stress (Eq. 12)	99,884 psi	99,781 psi
(Eq. 13)	18,252 psi	18,246 psi
Alternate stress (Eq. 14)	218,258 psi	217,811 psi
Usage factor	Out of Range	

NOTE: Equation numbers refer to Subsubarticle NB-3650 of ASME Section III.

TABLE 3A.15-10
(Sheet 1 of 1)

INDIVIDUAL PAIR USAGE FACTOR FOR POINT NO. 30

Pair	Hand Calculation	NUPIPE
1,5	0.183	0.1803
1,8	1.660	1.7361
1,9	0.0001	0.0001
1,10	Not in Range	
5,8	Not in Range	
5,9	0.221	0.2646
5,10	0.747	0.8051
8,9	0.857	0.8832
8,10	5.5518	5.8608
9,10	0.0001	0.0001

3A.15A SUPERPIPE

3A.15A.1 General Description

SUPERPIPE was developed for and is maintained by Impell Corporation. The program is fully documented.

SUPERPIPE is a comprehensive computer program for the structural analysis and design checking of piping systems, with particular emphasis on nuclear power piping. SUPERPIPE performs a linear elastic analysis of three-dimensional piping systems subjected to thermal, static, and dynamic loads, utilizing the finite element method of analysis. Design checking options include checks for analytical conformance to ASME Section III and ANSI B31.1, among others.

The software license for SUPERPIPE applies to Version 22C, and upwardly compatible versions, for use on VAX hardware running the VMS operating system. The applicable User's Manual is the SUPERPIPE User's Manual 22C/Revision 6.

3A.15A.2 Program Verification

SUPERPIPE has been verified in accordance with the Impell Corporation Quality Assurance (QA) Program. SUPERPIPE has been verified for a comprehensive set of sample problems, including extensive comparison with other public domain programs and ASME benchmark problems. SUPERPIPE theory, modeling, and analysis options are fully described in Reference 1.

SUPERPIPE has also been verified by comparing SUPERPIPE generated solutions to known solutions of NUREG benchmark problems[2]. SUPERPIPE generated solutions are contained in References 3 and 4.

In addition, whenever a new version of SUPERPIPE is installed at Unit 2, or whenever the hardware configuration is changed, a set of verification problems prepared by the vendor are re-run. The old and new results are compared to ensure that program integrity is maintained in transition stages.

3A.15A.3 References

1. SUPERPIPE User's Manual 22C/Revision 6. Impell Corporation, Walnut Creek, CA

2. Piping Benchmark Problems. Dynamic Analysis Uniform Support Motion Response Method, Volume 1. Brookhaven National Laboratory, Upton, NY. NUREG/CR-1677-V-1, August 1980.

3. SUPERPIPE Verification to Benchmark Problems Contained in NUREG/CR-1677, Report No. 01-0160-1187, Revision 0. EDS Nuclear, Inc., San Francisco, CA, September 1981.

4. SUPERPIPE Verification to Benchmark Problems for Multiple Response Spectra Method, Report No. 01-0310-1413, Revision 0. Impell Corporation, Walnut Creek, CA, July 1985.

3A.16 HAS BEEN DELETED

3A.17 HTLOAD

3A.17.1 General Description

HTLOAD performs a finite difference method analysis of piping system response to thermal transients of its contained fluid. The output gives overall thermal growth, linear and nonlinear temperature distribution through the pipe wall, gross discontinuity information (T_a-T_b), and Equations 10 and 11 results of Subarticle NB-3600 of ASME Section III.

HTLOAD can analyze piping with or without thermal sleeve that is subject to changes in fluid temperature, velocity, and/or state. The properties of subcooled or saturated water and superheated or saturated steam are taken from the ASME steam tables(1). The pressure range is from 0.45 to 6,210 psia.

HTLOAD also performs thermal analysis for pipes with different insulating conditions ranging from noninsulated to perfectly insulated. It has stored properties for insulation such as unibestos, asbestos, reflective aluminum, reflective stainless, and calcium silicate. Provision is further made for hand input properties of other insulation types.

Also stored in the program are the piping material properties of carbon steel, austenitic stainless steel, low chrome steel, high chrome steel, and nickel-chrome iron for the temperature range of 32°F to 1,600°F.

Program input includes piping material insulation information, time lapse for initial to final fluid temperature, calculation time limit, fluid velocities, initial and final temperature and pressure, and pipe and thermal sleeve dimensions.

HTLOAD requires that each thermal transient be input as a step change, a ramp change, or a 50-point arbitrary function.

Output results are used in the calculation of piping stress in accordance with Article NB-3600 of ASME Section III. HTLOAD also performs the primary plus secondary stress intensity range check (Equation 10) and the peak stress intensity range calculation (Equation 11) from Article NB-3600.

3A.17.2 Program Verification

The sample problem selected for solution by HTLOAD consists of a 2-in Schedule 160 stainless steel pipe, with one end connected to a 1/2-in thick, socket-welded fitting. Saturated water flowing within the piping system changes temperature from 400°F to 500°F in a period of 10 sec. Velocity of fluid is 7,560 ft/hr. Input properties are listed in Tables 3A.17-1 and 3A.17-2.

Reynolds number and heat transfer coefficients are compared with hand calculations(2) and are given in Table 3A.17-3.

Comparison between HTLOAD and Brock and McNeill's charts[3] for ΔT_1 and ΔT_2 are given in Table 3A.17-4. Table 3A.17-5 represents the comparison between TRHEAT[4] and HTLOAD for ΔT_1, ΔT_2, and T_a-T_b.

3A.17.3 References

1. Meyer, McClintock, et al. 1967 ASME Steam Tables.

2. Kreith, F. Principles of Heat Transfer. International Textbook Company, 1964.

3. McNeill, D. R. and Brock, J. E. Charts for Transient Temperature in Pipes, Heating/Piping/Air Conditioning, November 1971.

4. TRHEAT: Computer Code for Transient Heat Analysis of Nuclear Piping, Nuclear Services Corporation, 1972.

TABLE 3A.17-1
(Sheet 1 of 1)

PIPE MATERIAL PROPERTIES

Property	Temperature (°F)	Value
Thermal conductivity	450	10.01 Btu/°F-hr-ft
Thermal diffusivity	450	0.164 ft^2/hr
Young's modulus	70	$28.3x10^6$ psi
Coefficient of thermal expansion	70	$9.11x10^6$ in/in/°F

TABLE 3A.17-2
(Sheet 1 of 1)

FLUID MATERIAL/THERMAL PROPERTIES

Property	Value at 450°F
Density	51.300 lb/ft^3
Viscosity	0.2920 lb/hr/ft
Specific heat	1.135 Btu/lb-°F
Conductivity	0.3650 Btu/°F-hr-ft
Volume expansion coefficient	0.0009/°F

TABLE 3A.17-3
(Sheet 1 of 1)

COMPARISON OF HTLOAD WITH HAND CALCULATION

	HTLOAD	Hand Calculation
Reynolds number	186,700	186,700
Heat transfer coefficient	946.8 Btu/ °F-hr-ft^2	946.8 Btu/ °F-hr-ft^2

TABLE 3A.17-4
(Sheet 1 of 1)

COMPARISON OF HTLOAD WITH CHARTS OF BROCK AND MCNEILL

Parameter	Charts	HTLOAD
Maximum ΔT_1 (°F)	43.31	45.14
Maximum ΔT_2 (°F)	8.50	8.36

TABLE 3A.17-5
(Sheet 1 of 1)

COMPARISON OF HTLOAD WITH TRHEAT

Parameter	TRHEAT	HTLOAD
Maximum ΔT_1(°F)	44.70	45.14
Maximum ΔT_2(°F)	8.69	8.36
Maximum T_a - T_b(°F)	19.03	19.08

3A.18 GHOSH-WILSON

3A.18.1 General Description

Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loadings, known as the GHOSH-WILSON computer code, is a finite element-based computer program developed by S. Ghosh and E. Wilson[1] and modified by SWEC as Code ST-200.

The GHOSH-WILSON program is capable of performing static and dynamic analysis of complex axisymmetric structures subjected to any arbitrary static (mechanical and temperature) and dynamic loading.

The method used to represent the three-dimensional continuum is either as an axisymmetric thin shell, a solid of revolution, or a combination of both. The arbitrary loading in the circumferential direction is represented by a Fourier series, and the analysis is carried out for each term and summed up for the total response.

3A.18.2 Program Verification

Hamilton's variational principle is used to derive the equation of motion. This leads to a diagonal mass matrix and a stiffness matrix and load vector that is consistent with the assumed displacement field. The equations of motion are solved numerically in the time-domain by direct integration using the Wilson method[2].

The input required by the GHOSH-WILSON program is a description of geometry, materials, and boundary conditions. Loadings, damping factors, and time intervals for integration should be provided for each Fourier term. Additional inertias can be added at joints during a dynamic analysis.

The GHOSH-WILSON program provides time-history responses of the resultant forces, moments, shears, displacements, rotations, accelerations, and stresses at each node for the dynamic analysis. Maximum responses can also be obtained for each Fourier term.

3A.18.2.1 Sample Problem No. 1: <u>Cylinder Under Internal Pressure</u>

A cylinder is subjected to a constant internal pressure. The cylinder is modeled using the shell element, rectangular element, and the triangular element. The solution to this problem is found in References 3 and 4.

<u>Pertinent parameters</u>

1. Dimensions and properties

Pressure	P = 1 ksf
Mean radius	R = 40 ft
Height	l = 20 ft
Thickness	t = 2 ft
Young's Modulus	E = 3 x 10^6 psi
Poisson's ratio	ν = 0.15

2. Loading and boundary conditions

   l = 0, M = δz = 0
   l = 20, M = F = 0

From Reference 3 using thin shell solution.

$$\sigma_\theta = \frac{PR}{t}$$

$$\sigma_R = \frac{PR^2}{Et}$$

From reference 4 based on the theory of elasticity:

$$\sigma_\theta = \frac{a^2 p}{b^2 - a^2}\left(1 + \frac{b^2}{r^2}\right)$$

$$\sigma_r = \frac{a^2 p}{b^2 - a^2}\left(1 - \frac{b^2}{r^2}\right)$$

$$\Delta R = \frac{a^2 p}{E}\ \frac{1}{b^2 - a^2}\left[b^2\frac{(1+\nu)}{r} + (1-\nu)r\right]$$

Where:

a = inside radius

b = outside radius

r = radius where results are computed

p = internal pressure

Table 3A.18-1 shows the results of the GHOSH-WILSON solution compared to the theoretical solution. The results compare favorably.

3A.18.2.2 Sample Problem No. 2: Cylinder Subjected to Suddenly Applied Load

A cylinder simply supported at both ends is subjected to a suddenly applied load at midspan. The solution of the equations of motion is obtained by the direct integration method. The dimensions of the cylinder and the loading time-history are shown on Figure 3A.18-1. The cylinder is modeled using rectangular elements. The GHOSH-WILSON solution (displacement under the applied load) is compared to the solution using the ANSYS computer code. The results are shown on Figure 3A.18-2 and they compare favorably.

3A.18.3 References

1. Ghosh, S. and Wilson, E. Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading. Report EERC-69-10, University of California at Berkeley, September 1969. Modified as Stone & Webster Engineering Corporation Computer Code ST-200, September 1973.

2. Bathe, K. J. and Wilson, E. Numerical Methods in Finite Element Analysis. Prentice-Hall, Inc., Englewood Cliffs, NJ, 1976.

3. Roark, R. J. Formulas for Stress and Strain, Fourth Ed., McGraw-Hill Book Company, Fourth Ed., NY, 1965.

4. Timashenko, S. and Goodier, J. N. Theory of Elasticity, McGraw-Hill Book Company, NY, pp 58-60, 1951.

TABLE 3A.18-1
(Sheet 1 of 1)

COMPARISON OF GHOSH-WILSON RESULTS VERSUS THEORETICAL SOLUTIONS FOR A CYLINDER UNDER STATIC INTERNAL PRESSURE

	Radius (R)-ft	Theoretical Solution	Thin-Shell Theory Shell Element	GHOSH-WILSON Results Rectangular Element	GHOSH-WILSON Results Triangular Element
	R=39.5	19.748	-	19.74	19.755
σ_θ	R=40	20.00	20.27	-	-
Ksf	R=40.5	19.249	-	19.24	19.255
σ_γ	R=39.5	-0.7357	-	-0.7365	-0.683
Ksf	R=40.5	-0.236	-	-0.2369	-0.188
	R=39	1.82×10^{-3}	-	1.8197×10^{-3}	1.8192×10^{-3}
ΔR	R=40	1.812×10^{-3}	1.87×10^{-3}	1.8111×10^{-3}	1.8110×10^{-3}
ft	R=41	1.804×10^{-3}	-	1.8039×10^{-3}	1.8039×10^{-3}

σ_θ = hoop stress

σ_γ = radial stress

ΔR = displacement in radial direction

3A.19 HAS BEEN DELETED

3A.20 STEHAM

3A.20.1 General Description

STEHAM is used to determine the steamhammer transients of piping systems. This program uses the method of characteristics with finite difference approximations in both space and time[1-3]. It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as the stop valve and the safety/relief valve (SRV). Flow characteristics of piping components are mathematically formulated as boundary conditions in the program. These components include the flow control valve (FCV), the stop valve, the SRV, the steam manifold, and the steam reservoir. Frictional effects are taken into consideration.

This program accepts the following as input: 1) the flow network representation of the piping system, 2) initial flow conditions along the piping system, and 3) time-dependent flow characteristics of piping components. The output consists of time-histories of flow pressures, flow densities, flow velocities, inertia forces, and momentum functions.

3A.20.2 Program Verification

STEHAM is verified by comparing its solutions of a test problem (Figures 3A.20-1 and 3A.20-2) to the results of the same problem obtained by an independent analytical approach, as well as an experimental measurement[4,5]. A comparison of results for time-history pressure responses is plotted on Figures 3A.20-3 through 3A.20-5. The forcing functions developed for nodal points of the piping system calculated from the relation $F = (p+\rho V^2/g)A-p_aA$ has also been checked by hand calculations as tabulated in Table 3A.20-1.

3A.20.3 References

1. Jonsson, V. K.; Matthews, L.; and Spalding, D. B. Numerical Solution Procedure for Calculating the Unsteady One-Dimensional Flow of Compressible Fluid. ASME Paper No. 73-FE-30.

2. Luk, C. H. Effects of the Steam Chest on Steamhammer Analysis for Nuclear Piping Systems. ASME Paper No. 75-PVP-61.

3. Moody, F. J. Time-Dependent Pipe Forces Caused by Blowdown and Flow Stoppage. ASME Paper No. 73-FE-23.

4. Progelhof, R. C. and Owczarek, J. A. The Rapid Discharge of a Gas from a Cylindrical Vessel Through a Nozzle. AIAA Journal, Vol. 1, No. 9, September 1963, p 2182-2184.

5. Progelhof, R. C. and Owczarek, J. A. The Rapid Discharge of a Gas from a Cylindrical Vessel Through an Orifice. ASME Paper No. 63-WA-10.

TABLE 3A.20-1
(Sheet 1 of 1)

NODAL FORCE COMPARISON

Diameter D = 0.25 ft

Area $A = \pi D^2/4 = 0.0490874\ ft^2$

Nodal Force = $(p + \rho V^2/g)\ A - p_a\ A$

p = Pressure lb/ft^2

ρ = Density lb/ft^3

V = Velocity ft/sec

g = Gravitational constant 32.2 ft/sec^2

p_a = Ambient pressure (14.7x144 lb/ft^2)

at time t = 0.00650 sec

Node No.	Pressure (psia)	Velocity (fps)	Density (lb/ft^3)	Force STEHAM (lb)	Force Hand Calculation (lb)
1	42.523	0.0	0.23954	186.57	196.67
5	42.785	5.7843	0.24076	198.43	198.53
10	44.231	31.219	0.24647	209.00	209.11
15	47.003	78.172	0.25737	230.62	230.73
20	50.214	129.89	0.26979	257.84	257.97
25	52.095	159.43	0.27697	274.93	275.06
30	52.209	161.97	0.27742	276.09	276.23
35	52.168	162.21	0.27731	275.83	275.97

3A.21 WATHAM

3A.21.1 General Description

WATHAM is used to determine the flow-induced forcing functions acting on piping systems due to waterhammer. These forcing functions are then used as input to a structural dynamic analysis such as a NUPIPE program run.

WATHAM is applicable to a waterhammer problem or, more generally, any unsteady, incompressible fluid flow. These events may be caused by normal or abnormal operational changes of piping components such as the startup and trip of pumps or the rapid opening and closing of valves.

The analysis is based upon the method of characteristics with finite difference approximations both in time and space for the solution of one-dimensional liquid flows. Influences of piping components, including flow valves, pipe connections, reservoirs, and pumps have been considered in the analysis.

WATHAM input requires the geometry of the piping system, pipe properties, water properties, operational characteristics of pump and valve, flow frictional coefficients, and the initial water flow conditions. The output provides the time-history functions of piezometric heads, velocities, and nodal forces for all nodes and the inertial unbalanced force for each segment. It also gives the maximum value of all the preceding functions and their occurring time in the process of flow transient.

3A.21.2 Program Verification

Figure 3A.21-1 depicts a flow network with nine pipes, its geometrical properties, and steady-state flow conditions. The flow transient mode analyzed is the sudden closure of a valve at the downstream end. Figure 3A.21-2 shows the hydraulic network for WATHAM. Table 3A.21-1 illustrates the input data needed for WATHAM run. Figures 3A.21-3 and 3A.21-4 show a comparison of head-time curves[1,2] with WATHAM. Table 3A.21-2 presents the comparison of nodal forces between hand calculation and WATHAM computation.

In general, WATHAM results are in agreement with Streeter's results[1]. The small discrepancy is attributed to the modeling of the reservoir boundary condition. In WATHAM, the energy equation between the reservoir is utilized, rather than assuming the head of pipe entrance to be the same as that of the reservoir.

3A.21.3 References

1. Streeter, V. L. and Wylie, E. G. Hydraulic Transients, McGraw-Hill Book Company, New York, NY, 1967.

2. Fabic, S. Computer Program WHAM for Calculation of Pressure, Velocity, and Force Transients in Liquid Filled Piping Networks. Report No. 67-49-R, Kaiser Engineers, November 1967.

TABLE 3A.21-1
(Sheet 1 of 1)

INPUT DATA FOR WATHAM

Pipe No.	Total Length (ft)	Inside Diameter (ft)	Friction Factor	No. Nodes	Nodal Span (ft)	Thickness (in)	Velocity (fps)
1	2,000	3.0	0.03	7	333.33	0.30824	4.24413
2	3,000	2.5	0.028	9	375	0.44	2.92132
3	2,000	2.0	0.024	6	400	0.50026	4.98473
4	1,800	1.5	0.02	7	300	0.11108	3.59336
5	1,500	1.5	0.022*	5	375	0.264	4.52142
6	1,600	1.5	0.025	6	320	0.13796	2.29183
7	2,200	2.5	0.04	8	314.29	0.21534	3.65878
8	1,500	2.0	0.03	6	300	0.14811	3.83245
9	2,000	3.0	0.024	7	333.33	0.30824	4.24413

* Friction factor in Pipe 5, 0.022, differs slightly from that of hand calculation, 0.020.

NOTE: The initial heads of all nodes are calculated by using the Darcy-Weisbach equation.

TABLE 3A.21-2
(Sheet 1 of 1)

COMPARISON OF NODAL FORCE CALCULATION AT TIME = 2.34 SEC

Pipe No.	Node No.	Force (kip) WATHAM	Force (kip) Hand Calculation
1	1	276.34	276.48
1	2	300.46	300.62
1	3	317.78	317.94
1	4	329.59	329.76
1	5	341.39	341.56
1	6	355.31	355.49
1	7	369.52	369.71

NOTE: Nodal force calculation is based on the following equation:

$$F = A\left(\rho H + \frac{\rho}{g}\ V^2\right)$$

Where:

F = Nodal force, lb

ρ = Density, lb/ft^3

H = Nodal head, ft

g = 32.2 ft/sec^2

V = Nodal velocity, fps

A = Pipe area, ft^2

3A.22 PITRUST

3A.22.1 General Description

PITRUST calculates local stresses in the pipe caused by cylindrical welded attachments under external loadings. This program uses the Bijlaard method to calculate local stresses in the pipe wall caused by cylindrical welded attachments under external loadings, including pressure, dead load, thermal load, and combinations of maximum dynamic loads[1].

3A.22.2 Program Verification

PITRUST has been verified by comparing its solution of a test problem to the solution of the same problem by an independently written piping local stress program, CYLNOZ, in the public domain. The CYLNOZ piping local stress program was written by Franklin Institute (Philadelphia, PA) and is presently used by engineering companies. The test problem is a 72.375-in outside diameter by 0.375-in thick run pipe, reacting under an external loading condition of 1,000 lb force (normal and shear) and 1,000 in-lb bending and torsional moments transmitted by a 16-in outside diameter nozzle. A comparison of results is tabulated in Table 3A.22-1. The forces and moments are defined in Table 3A.22-2. PITRUST has also been verified by comparing its solution of the test problem to the experimental results obtained in Reference 2. A comparison of these results is tabulated in Table 3A.22-2.

3A.22.3 References

1. Local Stress in Spherical and Cylindrical Shells due to External Loading. Welding Research Council Bulletin, WRC-107, 1965.

2. Corum, J. M. and Greenstreet, W. L. Experimental Elastic Stress Analysis of Cylinder to Cylinder Shell Models and Comparison with Theoretical Predictions. First International Conference on Structural Mechanics in Reactor Technology (Berlin, Preprints Vol. 3, Part G, 1971).

TABLE 3A.22-1
(Sheet 1 of 1)

COMPARISON OF PITRUST WITH FRANKLIN INSTITUTE PROGRAM, CYLNOZ, AND HAND CALCULATION

Source of Stress	Stress (psi) Franklin Institute Corrected Values	Output from PITRUST	Hand Calculation
Circumferential			
P (normal)	395	399	399.99
P (bending)	1,875	1,833	1,877.3
M_c (normal)	35.85	35.57	36.06
M_c (bending)	364.7	366.6	354.3
M_L (normal)	79.05	79.66	79.54
M_L (bending)	90.52	80.57	79.42
Axial			
P (normal)	813	812	814.8
P (bending)	812.3	827	810.6
M_c (normal)	91.79	105	95.45
M_c (bending)	158.8	160	158.8
M_L (normal)	37.06	37	37.12
M_L (bending)	117.9	105	103.85
Shear stress by M_T	6.63	6.63	6.63
Shear stress by V_c	106.1	106.1	106.1
Shear stress by V_L	106.1	106.1	106.1

NOTE: For illustration of forces and moments see figure on Table 3A.22-2.

TABLE 3A.22-2
(Sheet 1 of 1)

COMPARISON OF PITRUST WITH REFERENCE 2 RESULTS

Location and Cause	PITRUST Results (psi)	Experimental Results[2] (psi)
Element A		
Longitudinal moment, M_L		
Circumferential stress	20,438.9	20,000
Axial stress	26,292.6	25,000
Element B		
Circumferential moment, M_c		
Circumferential stress	22,016.2	24,000
Axial stress	13,105.8	13,000

3A.23 PILUG

3A.23.1 General Description

PILUG calculates local stresses in the pipe wall caused by rectangular welded attachments under external loadings. This program uses the Bijlaard method to calculate local stresses in pipe wall caused by rectangular welded attachments under external loadings, including pressure, dead load, thermal load, and combinations of maximum dynamic loads[1].

3A.23.2 Program Verification

PILUG has been verified by comparing its solution of a test problem to results obtained by hand calculations using the formulations of Reference 1. A comparison of results is tabulated in Table 3A.23-1.

3A.23.3 Reference

1. Local Stress in Spherical and Cylindrical Shells due to External Loading. Welding Research Council Bulletin, WRC-107, 1965.

TABLE 3A.23-1
(Sheet 1 of 2)

COMPARISON OF PILUG OUTPUT WITH HAND CALCULATIONS

Test Problem: Run pipe outside diameter = 17 in
Run pipe thickness = 0.812 in
Axial length of LUG = 12 in
Width of LUG along circumference = 3 in
Loads: P = 3,300 lb; V_c = -1,788 lb;
V_L = 2,478 lb; M_c = 81,834 in-lb;
M_L = 103,320 in-lb; M_T = 76,284 in-lb

Stress in Circumferential Direction (psi):

Figure*	β	Stress from Hand Calculation	Stress from Computer Output	Remarks
3C	0.5485	387	330	Membrane stress due to P
1C	0.326	2,165	2,160	Bending stress due to P
3A	0.294	671	629	Membrane stress due to M_c
1A	0.388	18,976	19,904	Bending stress due to M_c
3B	0.467	3,014	2,961	Membrane stress due to M_L
1B	0.416	6,143	5,969	Bending stress due to M_L

Stress in Axial Direction (psi):

Figure*	β	Stress from Hand Calculation	Stress from Computer Output	Remarks
4C	0.4447	683	690	Membrane stress due to P
2C	0.4632	773	792	Bending stress due to P
4A	0.294	1,897	1,864	Membrane stress due to M_c
2A	0.550	6,357	5,942	Bending stress due to M_c
4B	0.467	2,365	2,328	Membrane stress due to M_L
2B	0.582	4,989.7	4,842	Bending stress due to M_L

TABLE 3A.23-1
(Sheet 2 of 2)

COMPARISON OF PILUG OUTPUT WITH HAND CALCULATIONS

Figure*	β	Stress from Hand Calculation	Stress from Computer Output	Remarks
Shear Stress (psi):				
--	--	1,304.8	1,304.8	Shear stress due to M_T
--	--	-366.99	-366.99	Shear stress due to V_L
--	--	127.15	127.16	Shear stress due to V_c

* All the terms used in the test problem are defined in Reference 1.

3A.24 WATAIR

3A.24.1 General Description

WATAIR is used to determine the waterhammer load on piping systems with trapped air. It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as pump startup and valve opening.

The analysis is based on a one-dimensional separated two-phase flow model with ideal gas trapped between two incompressible liquids. Numerical integration is used to obtain the solution of the governing equations.

WATAIR input requires the geometry of the piping system, flow frictional coefficients, operational characteristics of pump and valve, and the initial flow conditions. The output provides the time-history functions of the flow velocities, the pressure head of the air pocket, the pump discharge head, and the inertial unbalanced force for each segment. It also lists the maximum value, and the time of its occurrence for each of the above parameters.

3A.24.2 Program Verification

WATAIR is verified by comparing its solution of a test problem (Figure 3A.24-1) to the results of the same problem obtained by an independent and verified computer program WATHAM[1]. Figure 3A.24-2 gives the plot of the forcing function produced from WATAIR, while Figure 3A.24-3 is from WATHAM. Table 3A.24-1 lists the input data of the sample problem. Table 3A.24-2 compares the peak values of the unbalanced force and their time of occurrence. The WATAIR results are in good agreement with those from WATHAM both in shape and in values. Minor differences are due to the modeling differences. WATAIR uses incompressible flow solution for water; therefore, the effects of the acoustic waves are lost.

3A.24.3 Reference

1. WATHAM, Stone & Webster Engineering Corporation Computer Program ME 168, Version 02.

TABLE 3A.24-1
(Sheet 1 of 1)

INPUT DATA FOR WATAIR AND WATHAM

Pump	Suction head	42.9868 ft
	Rated head	2,980 ft
	Rated discharge velocity	7.5824 ft/sec
	Rated speed	4,550 rpm
	Accelerating time	5 sec
Pipe	Inside diameter	0.4801 ft
	Total length	287.3396 ft

TABLE 3A.24-2
(Sheet 1 of 1)

COMPARISON OF THE FIVE LARGEST WATER HAMMER LOADS ON PIPING SEGMENTS

Segment No.	From WATAIR (lb_f)	Time (sec)	From WATHAM (lb_f)	Time (sec)
4	114.33	4.465	110.0	4.476
5	78.296	4.465	75.04	4.476
11	100.549	4.479	99.25	4.479
14	582.652	4.479	590.1	4.449
19	154.176	4.479	166.9	4.431

3A.25 ANSYS

3A.25.1 General Description

The ANSYS engineering analysis system, developed by the Swanson Analysis System, Inc., is a fully warranted and documented computer program.

The ANSYS computer program, which has been used for production analysis since early 1970, is a large-scale, general purpose computer program for the solution of several classes of engineering analysis problems. Analysis capabilities include: static and dynamic; plastic, creep, and swelling; small and large deflections; steady-state and transient heat transfer; and steady-state fluid flow.

ANSYS is used in various analyses including those performed for the spent fuel racks (spent fuel pool structure gamma heating thermal evaluation), the drywell head assembly, the primary containment access hatches, and other complex structures.

3A.25.2 Program Verification

The ANSYS engineering analysis system is a nationally recognized computer program available in the public domain through Swanson Analysis Systems, Inc.

The matrix displacement method of analysis, based upon finite element idealization, is employed throughout the program. The library of finite elements available contains more than 30 elements for static and dynamic analyses and more than 10 for heat transfer and fluid flow analyses. This variety of elements gives the ANSYS program the capability of analyzing frame structures (two-dimensional frames, grids, and three-dimensional frames), piping systems, two-dimensional plane and axisymmetric solids, flat plates, three-dimensional solids, axisymmetric and three-dimensional shells and nonlinear problems, including interfaces and cables.

Loading on the structure may be forces, displacements, pressures, temperatures, or response spectra. Loadings may be arbitrary time functions for linear and nonlinear dynamic analyses. Loadings for heat transfer analyses include: internal heat generation, convection and radiation boundaries, and specified temperatures or heat flows.

3A.26 CONTAINMENT WALL LOADING (CWL)

3A.26.1 General Description

CWL is a SWEC program documented for in-house use. It is designed to compute pressure time-history on containment walls of cylindrical or annulus suppression pools due to point disturbances in the pool. It determines the chugging load during a postulated LOCA in a boiling water reactor (BWR) containment. During an air-poor blowdown of steam into a suppression pool, unsteady condensation (chugging) occurs. The steam bubbles collapse completely in a chugging event thus creating point disturbances in the pool. These events are treated as a mathematical model of a boundary value problem of the inhomogeneous wave equation. The solution technique used is the Inhomogeneous Wave Equation Green's Function Solution (IWEGS).

3A.26.2 Program Verification

CWL is verified by test data comparison. Figure 3A.26-1 shows 4TCO test data for chug 30 of Reference 1. Figure 3A.26-2 shows the CWL output for the chug 30 source definition given in Reference 1 and reproduced in Table 3A.26-1.

3A.26.3 Reference

1. Mark II Containment Program, Generic Chugging Load Definition Report, NEDE-24302-P Class III, April 1981, Nuclear Power System Division, General Electric Company, San Jose, CA.

TABLE 3A.26-1
(Sheet 1 of 1)

INPUT DATA USED IN CONTAINMENT WALL LOADINGS

Pool	Geometry	Cylindrical
	Radius	3.5 ft
	Depth	21 ft
	Source location	Pool center - 12 ft above basement
Water property	Density	1,000 kg/m^3
	Sound speed	634 m/sec
Tank property	Pool and wall damping	0.045
	Vent damping	0.05
South strength	Impulse amplitude	2.8 m^3/sec^2
	Impulse duration	0.018 sec
	Vent harmonics	-0.85 m^3/sec^2 @ 5.0 Hz -0.65 m^3/sec^2 @ 12.6 Hz -0.35 m^3/sec^2 @ 20.9 Hz -0.65 m^3/sec^2 @ 28.7 Hz -1.0 m^3/sec^2 @ 39.1 Hz

3A.27 LUGSTR

3A.27.1 General Description

LUGSTR performs stress analysis and fatigue evaluation at lugs attached to ASME Section III, Safety Class 1 piping. It is based on ASME Code Case N-122(1745)[1]. LUGSTR determines lug size and computes stress intensities due to lug loads for inclusion in Equations 9 through 14 of ASME Section III, Subsubarticle NB-3650.

Program input includes pipe and lug geometry, lug loads for each load case, thermal and material properties of pipe/lug, fluid film coefficient, combined coefficient of convection and radiation for noninsulated pipe, thermal transient data, and Safety Class 1 stress information from NUPIPE ASME Section III Subsubarticle NB-3650 analysis.

By adding local stress intensities at lugs to Safety Class 1 stress intensities computed by NUPIPE, LUGSTR computes resultant intensities at lugs and performs stress analysis in accordance with ASME Section III Subsubarticle NB-3650.

3A.27.2 Program Verification

Hand and LUGSTR calculations were performed on a sample problem that covered all features of the LUGSTR program. Hand calculation was based on ASME Section III Code Case 1745[1] and Welding Research Council Bulletin No. 198[2]. Comparison between hand calculation and LUGSTR is presented in Table 3A.27-1.

3A.27.3 References

1. Code Case No. 1745, Stress Indices for Integral Structural Attachments, Class 1, Section III, Division 1, ASME Boiler and Pressure Vessel Code, Section III, 1976.

2. Welding Research Council Bulletin No. 198, Stress Indices at Lug Supports on Piping Systems, American Welding Society, September 1974.

TABLE 3A.27-1
(Sheet 1 of 1)

HAND CALCULATION VERSUS LUGSTR

Design No.	(PL+PB) Lug Stress* (psi) Computer	(PL+PB) Lug Stress* (psi) Hand Calc.	Equation 9 Stress (psi) Computer	Equation 9 Stress (psi) Hand Calc.	Lug Dimensions (in) D Circu	Lug Dimensions (in) D Axial
1	21045	21071	18243	18243	2.000	11.39
2	19360	-	18243	18243	2.228	11.39
3	17959	17977	18243	18243	2.456	11.39

Sne and Stress Indices

	Sne (psi)	Mc Stress	Q1 Stress	Ct	Cl	Cc
Computer value	29070	13707	15363	11.829	3.518	2.276
Hand calculation	29910	13749	15360	11.84	3.522	2.284

Equation (10) and (11) Stresses

	Load Case 23 SNL	Load Case 23 SPL	Load Case 25 SNL	Load Case 25 SPL
Computer value	12096	31757	2562	24066
Hand calculation	12112	31766	2562	24075

Fatigue Evaluation for Lug Loads Only

Load Range	Allowable Fatigue Cycle LUGSTR	Allowable Fatigue Cycle Hand Calculation	Usage Factor LUGSTR	Usage Factor Hand Calculation
6-7	367236	367230	0.0003	0.00027
6-8	366137	366107	0.0001	0.00013

* This and following terms are defined in ASME Section III Code Case 1745.

3A.28 COMPUTER PROGRAMS FOR STRUCTURAL RESPONSES TO HYDRODYNAMIC LOADING (TRANFUN AND INVTRAN)

3A.28.1 Introduction

The design ARS of the nuclear power plant containment structures due to hydrodynamic loads are obtained from response time-histories. The response time-histories due to hydrodynamic loads can be efficiently obtained by combining a single time domain analysis due to impulse loading time-history and frequency domain analyses (using Fourier transform of the multiple input forcing functions due to hydrodynamic loads). Detailed steps required to obtain pertinent ARS are:

1. The GHOSH-WILSON[1] program is used to obtain response time-histories at various locations due to impulse loading time-history (3A.18).

2. Response time-histories and the input impulse time-history loadings are used to generate transfer functions using TRANFUN[2] (3A.28.2).

3. Transfer functions are used to determine response time-histories for the actual forcing function time-histories using INVTRAN[3] (3A.28.3). These time-histories are then used to generate the ARS at various locations on the structure.

3A.28.2 TRANFUN

TRANFUN computes the transfer function (TF) as a function of frequency of any nodal response value using the output of dynamic finite element program GHOSH-WILSON (3A.18).

Transfer function at a given node is defined as the ratio of Fourier transform[4] of the specified nodal response time-history and the Fourier transform of the input forcing function, and is a function of the frequency parameter. To enable a rapid evaluation of TF, a technique known as Fast Fourier Transform[5] (FFT) is employed in this program.

The output of TRANFUN becomes the input of the INVTRAN program described in 3A.28.3.

3A.28.3 INVTRAN

The program computes nodal response time-histories for a given input forcing function using nodal TF generated in the TRANFUN program.

This is achieved in three steps. First, the program determines the Fourier transform of the input forcing function. Second, the nodal response is obtained as a function of frequency by multiplying the nodal TF and the Fourier transform of the input

forcing function. Finally, an inverse Fourier transform of the product is taken to get the nodal response time-histories.

The output of INVTRAN is utilized to generate ARS at various locations on the structures.

3A.28.4 References

1. Ghosh, S. and Wilson, E. L. Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading, University of California, Berkeley, Report No. EERC69-1, EERC69-10, September 1969, Program No. ST-200, Stone and Webster Engineering Corporation, Cherry Hill, NJ, 1978.

2. TRANFUN - A Program to Compute Transfer Function at a Given Node and Angle, Program No. ST-340, Stone and Webster Engineering Corporation, Cherry Hill, NJ, 1979.

3. INVTRAN - A Program to Compute Response Time History of Forcing Function, Program No. ST-341, Stone and Webster Engineering Corporation, Cherry Hill, NJ, 1979.

4. Craig, R. R., Jr. Structural Dynamics An Introduction to Computer Methods, John Wiley & Sons, New York, NY, 1981, p 175-179.

5. Cooley, J. W.; Lewis, P. A. W.; and Welch, P. D. The Fast Fourier Transform and Its Applications, IEEE Transactions on Education, Vol. KE-12, No. 1, March 1960, p 27-34.

3A.29 NASTRAN

3A.29.1 General Description

NASTRAN is a general-purpose, large digital computer program for static and dynamic structural analyses using the finite element approach. The program has been specifically designed to analyze large problems with many degrees of freedom.

NASTRAN, developed by McDonnell Douglas Automation Company, is used for the structural analysis of the primary containment personnel airlock employing a finite-element method. The program embodies a lumped-element approach, wherein the distributed physical properties of a structure are represented by a model consisting of a finite number of elements that are interconnected at a finite number of grid points, to which the loads are applied.

3A.29.2 Program Verification

NASTRAN is a nationally recognized program available in the public domain through McDonnell Douglas Automation Company. The program is capable of performing different types of static and dynamic analyses, buckling analysis, transient analysis, and linear analysis.

3A.30 BIJLAARD: VESSEL PENETRATION ANALYSIS

3A.30.1 General Description

This computer code performs various analyses on tanks and pressure vessels. All of the analyses are concerned with local stresses at penetrations. Typical problems which can be handled include the following:

1. Stresses at vessel-nozzle junction for:

    a. Rigid attachment to cylinder

    b. Rigid attachment to sphere

2. Pressure discontinuity analyses for thin shell interaction

3. Allowable nozzle load functions for each case

Local stresses due to nozzle loads are found by the method prescribed by P. P. Bijlaard[1]. The method prescribed by Johns and Orange is used for pressure discontinuity stresses[2].

3A.30.2 Program Verification

BIJLAARD has been qualified in accordance with the SWEC Engineering Assurance Program, with complete documentation and verification.

The computer program produces accurate results consistent with the mathematical model used during its development and with the user's manual. All program options used are qualified. A sufficient number and variety of test problems are used to establish that the program will provide accurate results when used to solve problems which are within the limits specified in the user's manual. The qualification demonstrates that the computer code performs as designed and that the program adequately solves the intended problem.

3A.30.3 References

1. Wichman, K. R.; Hopper, A. G.; and Mershon, J. L. Local Stresses in Spherical and Cylindrical Shells Due to External Loading. Welding Research Council Bulletin, WRC-107, 1965.

2. Johns, R. H. and Orange, T. W. Theoretical Elastic Stress Distribution Arising from Discontinuities and Edge Loads in Several Shell Type Structures. NASA Technical Report R-103, 1961.

3A.31 SLOSH: SIMPLIFIED TANK SLOSHING ANALYSIS

3A.31.1 General Description

SLOSH is a computer program used to compute the seismically induced liquid pressures and the maximum vertical displacement of the liquid surface in a container under horizontal acceleration. The program is designed to handle input for both cylindrical and rectangular tanks which can be either ground supported or tower supported.

The mathematical procedures and formulas used in developing the program were taken from TID-7024[1], Nuclear Reactors and Earthquakes (Reactor Technology, TID-4500, 22nd Ed.), issuance date: August 1963, United States Atomic Energy Commission, Division of Reactor Development, Washington, DC, Chapter 6, Dynamic Pressure on Fluid Containers. The analysis uses data for intensity of ground motion taken from the average-acceleration-spectrum curves rather than average-velocity-spectrum curves.

3A.31.2 Program Verification

SLOSH has been qualified in accordance with the SWEC Engineering Assurance Program, with complete documentation and verification. The computer program produces accurate results consistent with the mathematical model used during its development and with the user's manual. All program options used are qualified. A sufficient number and variety of test problems are used to establish that the program will provide accurate results when used to solve problems which are within the limits specified in the user's manual. The qualification demonstrates that the computer code performs as designed and that the program adequately solves the intended problem.

3A.31.3 Reference

1. TID-7024, Nuclear Reactors and Earthquakes (Reactor Technology, TID-4500, 22nd Edition), issuance date: August 1963, United States Atomic Energy Commission, Division of Reactor Development, Washington, DC.

3A.32 SNUFFE: SUPPLEMENT TO NUPIPE FOR FATIGUE EVALUATION

3A.32.1 General Description

SNUFFE is a computer program which performs the stress evaluation of selected piping components in accordance with ASME III Code Subarticle NB-3650 for Code Class 1 piping. It is intended to be a supplement to NUPIPE program, in particular, while fatigue evaluation of a few nodes is of interest.

The program performs the stress evaluation of Class 1 piping components based on the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Subarticle NB-3650.

The output begins with an echo print of the input data. The minimum pipe wall thickness and sectional properties are printed for design check. Moment ranges for all load permutations are printed for Equations 10, 12, and 13 requirements. The program prints the stress contributing terms for Equations 10, 11, and 13 if the option is used. Stress intensities for all load permutations are also printed for Equations 10 through 14. The Class 1 fatigue usage factors for the significant load pairs and cumulative usage factor (CUF) are printed.

3A.32.2 Program Verification

SNUFFE has been qualified in accordance with SWEC Engineering Assurance Program, with complete documentation and verification. The computer program produces accurate results consistent with the mathematical model used during its development and with the user's manual. All program options used are qualified. A sufficient number and variety of test problems are used to establish that the program will provide accurate results when used to solve problems which are within the limits specified in the user's manual. The qualification demonstrates that the computer code performs as designed and that the program adequately solves the intended problem.

3A.33 PITRIFE

3A.33.1 General Description

PITRIFE (Pipe Trunnion Interpolated Stresses by SWEC) is a program for calculating the local discontinuity stresses in a pipe at the intersection with a circular trunnion due to loads applied to the trunnion. It is a post processor program that uses the results of a finite element model of two intersecting cylinders. Based upon the stresses calculated with the finite element model, nondimensional stress coefficients were computed for a size-on-size pipe-trunnion configuration for three different values of average pipe radius to wall thickness (R/t = 5, 10, 20). Additionally, nondimensional stress coefficients were computed for a trunnion radius equal to 0.707 times the pipe radius (0.707 size-on-size) for the three values of R/t. To facilitate the determination of nondimensional stress coefficients for other values of R/t, a rotated parabola curve that fits the three R/t data points was generated for both the size-on-size and the 0.707 size-on-size data. The PITRIFE program reconstructs these curves and uses them to interpolate and extrapolate for stress coefficients for different values of R/t. The finite element models were analyzed using the ICES STRUDL II Program.[1]

3A.33.2 Program Verification

The PITRIFE program has been verified by demonstrating that the maximum stress intensities as given by PITRIFE equal the values given by the finite element analysis for specific size-on-size and 0.707 size-on-size models. A comparison of these results is tabulated in Table 3A.33-1. The program was verified for other ratios of trunnion-to-pipe radius by demonstrating that the stress coefficients and maximum stress intensities derived by hand calculation equal the coefficients used in the program to calculate maximum stress intensity. A comparison of these results is given in Table 3A.33-2.

3A.33.3 References

1. STRUDL II - Structural Design Language.

TABLE 3A.33-1
(Sheet 1 of 1)

COMPARISON OF PITRIFE RESULTS WITH STRUDL II RESULTS

	Test Problem	Size-on-Size	0.707 Size-on-Size
(Graphic Here)	Ave pipe radius (in)	3.00	3.00
	Ave trunnion radius (in)	3.00	2.12
	Pipe wall thickness (in)	0.30	0.30
	Trunnion wall thickness (in)	0.30	0.21

Size-on-Size Maximum Stress Intensity-psi ($\alpha = 30°$)

Load	PITRIFE Output	STRUDL II Output
Fx = 10,000 (lb)	5,763	5,768
Fy = 10,000 (lb)	7,844	7,864
Fz = 10,000 (lb)	6,507	6,506
Mx = 10,000 (in-lb)	1,329	1,329
My = 10,000 (in-lb)	1,688	1,687
Mz = 10,000 (in-lb)	4,066	4,068

0.707 Size-on-Size Maximum Stress Intensity-psi ($\alpha = 30°$)

Load	PITRIFE Output	STRUDL II Output
Fx = 10,000 (lb)	13,471	13,458
Fy = 10,000 (lb)	9,616	9,611
Fz = 10,000 (lb)	20,105	20,030
Mx = 10,000 (in-lb)	4,371	4,368
My = 10,000 (in-lb)	2,467	2,467
Mz = 10,000 (in-lb)	6,178	6,176

TABLE 3A.33-2
(Sheet 1 of 1)

COMPARISON OF PITRIFE RESULTS WITH HAND CALCULATIONS

Test Problem:

Ave pipe radius	=	1.5 in
Ave trunnion radius	=	1.35 in
Pipe wall thickness	=	0.30 in
Trunnion wall thickness	=	0.27 in

Loads for Each Load Type Combined (DL, OBEI, THER, OCCU, etc):

Fx = Fy = Fz = 10,000 lb
Mx = My = Mz = 10,000 in-lb
MNS Stress = 200 psi
Internal Pressure = 100 psi

Stress Coefficients - 0.9 Size-on-Size Fx Loading, $\alpha = 30°$

Stress Type	Coefficient by Hand Calculation	Coefficient from PITRIFE
Longitudinal - Inside Fiber	-1.2652	-1.2652
Circumferential - Inside Fiber	-0.2764	-0.2764
Shear - Inside Fiber	0.2041	0.2041
Longitudinal - Outside Fiber	0.7454	0.7454
Circumferential - Outside Fiber	1.3509	1.3509
Shear - Outside Fiber	0.2041	0.2041

Maximum Stress Intensity - 0.9 Size-on-Size, $\alpha = 30°$

Load Condition	Maximum Stress Intensity - psi Hand Calculation	PITRIFE
P + DL + MNS_1	28,181	28,182
P + DL + SRSS (OBEI, OCCU) + MNS_2	73,220	73,220
P + DL + OBEA + THER + MNS_3	88,216	88,216
P + DL + OCCE + MNS_4	59,853	59,853
P + DL + SRSS (SSEI, OCCF) + MNS_5	73,220	73,220

3A.34 IMAGES

3A.34.1 General Description

IMAGES is a three-dimensional general purpose PC-based finite element analysis program. It is composed of four different computer modules: the static, modal, and dynamic modules (IMAGES-3D)[1,2], and the thermal module (IMAGES-THERMAL)[3]. These modules work either independently or in various combinations.

The static module of IMAGES-3D has the capability to handle separate load cases with various loading types which include concentrated loads, prescribed nodal displacements, inertial loads, distributed loads, pressure loads and thermal loads. Output can be requested in terms of displacements, forces, reactions, stresses and plots.

The modal module can be used to generate structural weights and lumped masses. This module can calculate frequencies and mode shapes, then reports the modal weight and participation factor for each mode.

The dynamic module has the capacity to calculate nodal displacements, elements stresses and reactions due to time-dependent loads or ground excitations. In addition, it accepts input of response spectra for seismic analysis and provides a choice of combination methods for multiple directing spectra: ABS (absolute summation), SRSS (square root of the sum of the squares), ten percent, double sum and CQC (complete quadratic combination) modal combination methods.

The thermal module can provide steady-state solutions for heat transfer problems and the resulting temperatures can be used by other modules to calculate member thermal stresses.

The software license for IMAGES applies to Version 2.0 and upwardly compatible and qualified versions for use on the PC.

3A.34.2 Program Verification

IMAGES is a computer program produced by Celestial Software Inc. (CSI). This program has been verified in accordance with the CSI quality assurance (QA) program, which has been audited and accepted by Unit 2 QA. CSI issues notifications of software errors on a monthly basis.

IMAGES-THERMAL has been verified via the Heat Transfer Finite Elements Analysis Verification Manual[4], and IMAGES-3D has been verified by the static and dynamic Finite Elements Analysis Verification Manual[5]. These verification manuals were generated by CSI to ensure high reliability of the program. The method used in both of these verification manuals is comparison of the IMAGES program solutions to those solutions published in the literature in accordance with Standard Review Plan (SRP) Section

3.9.1.II.2.c.(2). Additional verification was also conducted by Unit 2 Engineering. The results of this supplemental verification closely match results previously obtained using STRUDL II.

In addition, prior to using a new version of IMAGES or whenever the hardware configuration is changed, the program will be subjected to the same QA requirements listed above.

3A.34.3 References

1. IMAGES-3D, Version 2.0, User's Manual, July 1990.

2. IMAGES-3D, Technical Reference Manual, Version 2.0, 1990.

3. IMAGES-THERMAL, Version 2.0, User's Manual, July 1990.

4. IMAGES-THERMAL, Version 2.0, Heat Transfer Finite Element Analysis Verification Manual, July 1990.

5. IMAGES-3D, Version 2.0, Static Dynamic Finite Element Analysis Verification Manual, July 1990.

3A.35 PIPSYS

3A.35.1 General Description

PIPSYS was developed by and is maintained by Sargent & Lundy[LLC] (S&L). The program is fully documented.

PIPSYS combines a linear, three-dimensional finite element analysis program with menu-driven interactive modules to pre- and post-process data for the analysis and stress evaluation of piping systems. The program consists of interactive modules to generate, review and modify piping data; modules to perform static and dynamic analyses; and modules to evaluate piping stress and determine support loads and equipment reactions. Examples of static analyses include distributed and concentrated weight loadings, thermal expansion and displacement analyses. Dynamic analyses include seismic response spectrum and time-history analyses. Design checking options include checks for analytical conformance to ASME Section III and ANSI B31.1.

3A.35.2 Program Verification

PIPSYS has been verified in accordance with the S&L Quality Assurance Program. PIPSYS has been verified for a comprehensive set of sample problems, including comparison with other public domain programs, such as DYNAL[1] and NASTRAN[2,3], hand calculations, and previously verified S&L programs. PIPSYS theory, modeling, and analysis options are fully described in Reference 4.

Whenever modifications to the program are made, the same procedure is used to verify their validity. To demonstrate that previous capabilities are not adversely affected by the modifications, test problems from previous verifications are rerun and the results verified. All verification calculations are documented and filed in S&L's Computer Services Library.

3A.35.3 References

1. ICES DYNAL User's Manual, McDonnell Douglas Automation Co., September 1971.

2. NASTRAN Theoretical Manual, NASA SP-221, September 1970.

3. NASTRAN User's Manual, NASA SP-222, September 1970.

4. PIPSYS User's Manual, S&L Program No. 03.7.026, Version 2.2, Sargent & Lundy[LLC].

3A.36 STAADIII

3A.36.1 General Description

STAAD-III was developed and is maintained by Research Engineers Inc. of Yorba Linda, California. The program verification is maintained by Sargent & Lundy LLC (S&L). The program is fully documented.

STAAD-III performs structural analysis and design of 2D and 3D structures and models with plate elements. The program performs stiffness, P-Delta and dynamic analysis. Extensive design capabilities are available for steel and concrete design[2].

The software license for STAAD-III applies to Version 22.3 and upwardly compatible and qualified versions for use on the PC, subject to the limitations noted below.

3A.36.2 Program Verification

STAAD-III has been verified in accordance with S&L's Quality Assurance Program requirements. Only selected portions of the program are validated for use at S&L. These include: 1) frame and plate element static analysis, 2) steel design (AISC/ASD), 3) P-Delta analysis, 4) response spectra analysis with beam-type elements, and 5) frequency analysis with plate elements. The program was verified by comparing sample problems with the output from another validated program, SAP90[1].

Whenever modifications to the program are made, or the scope of the validation is expanded, the same procedure is used to verify their validity. All verification calculations are documented and filed in S&L's Computer Services Library.

3A.36.3 References

1. SAP90 User's Manual, S&L Program No. 03.7.224, Version 5.4, Sargent & Lundy LLC.

2. STAAD-III Reference Manual, Version 22.3, Research Engineers Inc. (S&L Program No. 03.7.065, Version 2.23).

3A.37 APLAN

3A.37.1 General Description

APLAN was designed to analyze rectangular attachment plates mounted on concrete or masonry by means of expansion anchors, head welding studs, or wire embedments. The finite element method is used to perform decoupled bending and plane stress analysis of the attachment plate and interfacing surface. Plane stress analysis is performed to determine the shear reactions on the anchors. Results from both analyses are combined to determine the maximum stress in the plate[(1)]. The program is fully documented.

The software license for APLAN applies to Version 1.1o and upwardly compatible and qualified versions for use on the PC.

3A.37.2 Program Verification

APLAN has been verified in accordance with S&L's Quality Assurance Program requirements. The program was verified by comparing a comprehensive set of sample problems with the output from another validated program, ADINA[(2)].

Whenever modifications to the program are made, or the scope of the validation is expanded, the same procedure is used to verify their validity. All verification calculations are documented and filed in S&L's Computer Services Library.

3A.37.3 References

1. APLAN User's Manual, S&L Program No. 03.7.282, Version 1.1o, Sargent & Lundy LLC.

2. ADINA User's Manual, S&L Program No. 09.7.199, Version 2.0, Sargent & Lundy LLC.

3A.38 DYNARACK

3A.38.1 General Description

DYNARACK is a proprietary Holtec International computer program with capability to perform dynamic simulation on a wide variety of linear and nonlinear systems and structures. Here it is used to simulate the spent fuel pool rack structure response to seismic excitation. DYNARACK utilizes the classical Component Element method[1] for simulation of dynamic response. The Component Element method emphasizes the overall dynamic behavior of systems while de-emphasizing the detailed stress field in the structure in contrast to the Finite Element method[2] that focuses on the stress distribution. As a result, the Component Element method has been a powerful analysis tool in the study of dynamic problems involving friction, damping, impact, and other nonlinear phenomena. The method has been shown to be particularly suited for treating a diverse array of problems, such as those of earthquake response in nonlinear systems, vehicle collision dynamics, aircraft landing simulation, and railroad car motion[1,3-9]. In essence, the Component Element method entails modeling the structure as an assemblage of lumped masses, springs, gap elements, friction elements, and dampers.

3A.38.2 Program Verification

All computer programs utilized by Holtec International to perform analyses for rack installation are benchmarked and verified in accordance with Holtec International's Quality Procedure HQP 11.0. The validation of DYNARACK is documented in Holtec Report HI-92844[10].

3A.38.3 References

1. Levy, S. and Wilkinson, J. The Component Element Method in Dynamics, McGraw Hill, 1976.

2. Zienkiewiczs, O. C. The Finite Element Method in Engineering Science, McGraw Hill Book Company (UK) Ltd., London, 1971.

3. Moreadith, F. L., Patterson, G. J., Angstadt, C. R., and Glova, J. F. Structural Design of Nuclear Plant Facilities, Vol. II, pp 201-342 (particularly pp 282-283), American Society of Civil Engineers, New York, 1974.

4. Patterson, G. J. Lumped-Parameter Model of Nonlinear Dynamic Analysis of Pipe Whip Restraints, Proc. 2nd ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, Vol. 1-B, pp 1444-1484, American Society of Civil Engineers, New York, 1975.

5. Hurty, W. C. Dynamic Analysis of Structural Systems Using Component Modes, AIAA J., 3, 678-685, 1965.

6. Benfield, W. A. and Hruda, R. F. Vibration Analysis of Structures by Component Mode Substitution, AIAA J., 9, 1255-1261, 1971.

7. Potts, G. R. and Walker, H. S. Nonlinear Truck Ride Analysis, J. Eng. for Industry, Trans. ASME, 96, Ser. B, 597-602, 1974.

8. Wickens, A. H. General Aspects of the Lateral Dynamics of Railway Vehicles, J. Eng. for Industry, Trans. ASME, 91, Ser. B, 869-878, 1969.

9. Meachan, H. C. and Ahlbeck, D. R. A Computer Study of Dynamic Loads Caused by Vehicle-Track Interaction, J. Eng. for Industry, Trans. ASME, 91, Ser. B, 808-815, 1969.

10. Holtec International Report HI-92844, QA Documentation Computer Program MR216.

3A.39 AutoPIPE

3A.39.1 General Description

AutoPIPE was developed and is maintained by Bentley Corporation (formerly Rebis Industrial Workgroup Software), Exton, Pennsylvania. The program verification is also maintained by the same company.

Bentley AutoPIPE is a native Windows program for calculating piping code stresses, loads and deflections under static and dynamic loading conditions. AutoPIPE analyzes systems of any complexity, with special features for buried pipeline analysis, wave loading, fluid transients, FRP/GRP pipe, and pipe/structure interaction.

AutoPIPE combines object-oriented graphics technology with advanced analytical capabilities for piping analysis and design. The software license for AutoPIPE applies to Version 09.00.00.08 for use on the personal computer.

3A.39.2 Program Verification

AutoPIPE has been verified in accordance with Bentley Corporation's quality assurance program requirements. Bentley's Document AutoPIPE Acceptance Test Set (D/N #QA25API-9.0.0) documents the validity of this AutoPIPE computer program. The accuracy of this program is verified by Bentley during execution of alpha tests which are performed prior to any commercial or quality assured release.

APPENDIX 3B

PRESSURE ANALYSIS FOR SUBCOMPARTMENTS OUTSIDE CONTAINMENT

APPENDIX 3B

TABLE OF CONTENTS

APPENDIX 3B

LIST OF TABLES

APPENDIX 3B

LIST OF TABLES (Cont'd.)

APPENDIX 3C

FAILURE MODE ANALYSIS
FOR PIPE BREAKS AND CRACKS

TABLE OF CONTENTS

APPENDIX 3B

LIST OF FIGURES (Cont'd.)

APPENDIX 3B

LIST OF FIGURES (Cont'd.)

APPENDIX 3B

PRESSURE ANALYSIS FOR SUBCOMPARTMENTS OUTSIDE CONTAINMENT

3B.1 DESIGN BASES

Pressure response analyses were performed for the structural design basis of the main steam tunnel and other subcompartments in the reactor building for postulated ruptures of high-energy piping. The definition for high energy and criteria for protection against dynamic effects associated with postulated rupture of piping are given in Section 3.6A. The analyses were performed using the Stone & Webster Engineering Corporation (SWEC) computer code THREED (Appendix 6B). The impact of extended power uprate (EPU) on this analysis is discussed in Section 3B.4.

The reactor building was divided into 45 separate subcompartments for this analysis. A dummy node was added to represent the atmosphere. A reactor water cleanup (RWCU) filter demineralizer room (Volume SC328190) was added to the reactor building model only for the postulated breaks in that room and in Volume 328-02. The main steam tunnel was divided into seven separate subcompartments for its design evaluation. An eighth node was used to represent the turbine building and the outside atmosphere. Reactor building and main steam tunnel subcompartment boundaries were chosen to represent physical restrictions to flow and to reflect additional detail in the vicinity of high-energy lines.

Breaks were postulated in each reactor building volume containing a high-energy line. Breaks were postulated in main steam tunnel nodes where the main steam piping bends occurred. All breaks, except for a postulated main steam line (MSL) crack in the main steam tunnel, were considered to be instantaneous circumferential double-ended ruptures (DER), i.e., the break area was equal to twice the effective cross-sectional flow area of the pipe.

Reactor building high-energy lines were identified in the RWCU system, the reactor core isolation cooling (RCIC) system, and the residual heat removal (RHR) system. A total of 35 breaks were postulated and analyzed. Peak calculated pressure differentials for the 46 subcompartments were generated by 18 of the 35 postulated breaks. Table 3B-1 lists all of the postulated breaks and identifies the breaks that determined the design differential pressures.

During RCIC isolation valve closure, the flow area used for mass and energy release calculations was assumed to be constant until the valve area equaled the flow limiting area. Subsequently the limiting flow area was linearly reduced to zero.

Some RWCU isolation valve closures did not assume the limiting flow area to always reduce linearly to zero. In some cases, the flow area remained constant and zero flow was assumed to occur instantaneously at the isolation valve closure time. The instantaneous cases were done to prevent numerical instabilities from appearing in the analyses.

The main steam tunnel analysis considered feedwater and MSL breaks. MSL break analyses were performed assuming an all-steam blowdown and a two-phase blowdown. Six combinations of break location and blowdown conditions were analyzed. Peak differential pressure values were generated by the two-phase blowdown breaks. Table 3B-2 lists all of the postulated breaks and identifies the two breaks that determined the design differential pressures for the steam tunnel.

A DER of the 8-in RWCU line in the main steam tunnel was not postulated. A 28-in MSL DER in the main steam tunnel would envelop the 8-in RWCU line's short-term peak pressure results.

Figures 3B-1 and 3B-2 show the nodalization schemes used in the reactor building, and identify the volume numbers referred to in the remainder of this section. Figure 3B-49 shows the nodalization scheme for the main steam tunnel.

Table 3B-3 provides the nodal descriptions and gives the peak calculated and design differential pressures within the reactor building. Table 3B-4 shows the subcompartment nodal descriptions for the main steam tunnel and identifies the calculated and design peak differential pressures.

Tables 3B-5 and 3B-6 give the vent flow path data for the reactor building based on the nodalization schemes described in Figures 3B-1 and 3B-2. Table 3B-7 presents the vent path description shown on Figure 3B-49 for the main steam tunnel.

Tables 3B-8 through 3B-16 provide the mass and energy release data for the 18 breaks that determine the reactor building design differential pressures. Except for the breaks in Volumes 328-02 and SC328190 and all RCIC breaks, frictionless saturated Moody flow[1] or subcooled Henry-Fauske flow[2] was assumed at the limiting downstream and upstream flow areas. Mass and energy release data were calculated using the methodology of NEDO-24548[3]. Figures 3B-3 through 3B-48 provide the absolute pressure transient plots for the 46 reactor building subcompartments.

Breaks in Volumes 328-02 and SC328190 and all RCIC breaks considered the effects of pipe friction on the blowdown results. Moody flow with friction was used to determine blowdown mass flux values.

Fluids with a temperature less than 250°F were not considered in these RWCU blowdowns considering this fluid would have lowered the average energy of the blowdowns. This "cold" fluid is discharged during the first 13.40 sec after the postulated DERs occur.

The mass and energy release data used for the postulated main steam tunnel pipe breaks are presented in Tables 3B-17 and 3B-18. These blowdowns were based entirely on frictionless Moody flow with a constant reservoir pressure. The blowdown was assumed to be all steam for the first second after the accident. After 1 sec, the reactor pressure vessel (RPV) was assumed to depressurize, and the froth level rising in the vessel was discharged through the MSLs.

The quality of this part of the blowdown was assumed to be 15 percent.

The mass and energy release data for the main steam tunnel were calculated using the methodology of ANS-58.2, Appendix E[4]. Figures 3B-50 through 3B-56 provide the absolute pressure transient plots for the seven main steam tunnel subcompartments.

Pressure differentials across reactor building subcompartment walls were calculated by subtracting the reactor building's initial pressure from each calculated nodal absolute pressure. This method simplified the number of differential pressures to be calculated. It is also conservative because any counterbalancing pressurization on the other side of the wall(s) was not included. Peak pressure differential values for the main steam tunnel subcompartments were calculated by subtracting 14.7 psia from the peak pressure values. No heat sink credit was taken for the reactor building and the main steam tunnel analysis. The initial environmental conditions within the reactor building and the main steam tunnel were normally assumed to be minimum pressure, maximum temperature, and zero percent relative humidity. Twenty percent relative humidity was used in some of the RWCU analyses to prevent numerical instabilities in the analyses. Twenty percent relative humidity is the minimum design relative humidity in the plant.

3B.2 DESIGN FEATURES

Except for the RWCU breaks in Volumes 328-2 and SC328190, area temperature elements are used to signal RCIC and RWCU isolation valve closures during a high-energy line rupture. Valve closure begins within 2.85 sec after the DER occurs. The valve closure signal is initiated by a local high temperature signal.

RWCU breaks in Volumes 328-2 and SC328190 are isolated by a high differential flow signal between flow elements FE-115 and FE-119. Valve closure begins 45 sec after the differential flow signal is

generated. A 2-sec instrumentation delay is assumed between the valve closure signal and the actual start of valve closure.

Reactor building high-energy lines are routed through pipe chases. Pressure buildup from postulated high-energy pipe breaks would be restricted primarily to these pipe chases and to the subcompartments containing the system pumps and heat exchangers. The RCIC pump room (Volume 175-1) and the pipe chase above it (Volume 175-2) have blowout panels to relieve the pressure buildup from a high-energy line rupture. The RCIC pump room blowout panel is located on the roof (Volume 175-1 to Volume 175-9). The pipe chase above Volume 175-1 has two blowout panels that connect to Volume 175-3 (Volume 175-2 to Volume 175-3). The single-failure criterion is applied to only two of the three blowout panels which are assumed to be available for venting. Unless otherwise noted, one of the blowout panels between Volume 175-2 to Volume 175-3 is assumed not to open. Each blowout panel has an opening area of 10 sq ft. They open up 0.3 sec after a 0.5 psid force is applied across them. They open only in the directions described above.

No credit is taken for any potential reactor building metal siding buckling under the pressure generated by high-energy line ruptures. No vent paths to the atmosphere are postulated through the siding joints. Any potential leakage through these joints would relieve the reactor building pressure buildup but would not result in offsite doses in excess of the 10CFR100 limit.

Vent curtains are used at certain openings to prevent air flow into specific areas during normal operation. These curtains are affixed to wire-mesh frames and are positioned to allow air flow during a high-energy line rupture but not during normal operation. They are modeled in the analyses as flaps at a 30-deg angle with respect to the horizontal. A friction loss of 3.2 is taken in the direction of allowed air flow. No flow is permitted in the opposite direction.

Some horizontal vent openings have wire-mesh door frames across them. These wire-mesh doors are assumed to obstruct one-third of the total opening area. A friction loss of 0.224 is taken across them.

Some vertical vent openings have grating across them. The grating is assumed to obstruct 19.7 percent of the total opening area. A friction loss of 0.200 is used for 1 1/4-in grating. Two-in grating has a friction loss of 0.188.

For the 6-in RWCU DER in a RWCU filter/demineralizer cubicle (Volume SC328190), the concrete plug on top of the cubicle is assumed to rise under sufficient pressure to permit venting to the refueling floor (Volume 353-1). This movement begins 0.31 sec after "hot" fluid (fluid above 250°F) enters the cubicle. At this moment, the pressure force in Volume SC328190 being applied on the plug equals the weight of the plug. It then takes an

additional 0.60 sec for the plug to rise through the cubicle's 4-ft ceiling and fully clear the opening to the refueling floor.

It is assumed that the concrete plug rises without any rotation. The initial pressure rise rate in Volume SC328190 during the plug's movement is assumed to be linear. No plug movement from any jet impingement loading is considered and the effects of friction on the plug's rise were not modeled.

The main steam tunnel also has blowout panels to relieve the pressure buildup during high-energy line ruptures. Eight panels are located on the roof of the main steam tunnel and vent directly to the outside atmosphere. Seven of the eight blowout panels are assumed to be available for venting. The main steam tunnel also vents directly to the turbine building.

Each main steam tunnel blowout panel has an opening area of 21.3 sq ft and opens up within 0.3 sec after a 0.9 psid force is applied across them. 150.0 of the total 170.6 sq ft of vent area are modeled in the analysis.

A high main steam flow signal is used to trigger main steam isolation valve (MSIV) closure. MSIV closure takes 5.5 sec.

RPV flow from a feedwater DER in the main steam tunnel would be stopped by check valves within the lines. Flow from the feedwater pump side and the RWCU line continue until the end of the analysis.

3B.3 DESIGN EVALUATION

3B.3.1 Elevation 175 ft 0 in to Elevation 215 ft 0 in

Four cases were modeled on this elevation. A 4-in RCIC steam line DERs were postulated in the RCIC pump and turbine room (Volume 175-1) and in the pipe chase directly above this room (Volume 175-2). Each DER was modeled with two different nodalization models. One model assumed the blowout panel from Volume 175-1 to Volume 175-9 and a blowout panel from Volume 175-2 to Volume 175-3 to be available. The other model assumed both blowout panels connecting Volume 175-2 to Volume 175-3 to be available.

RCIC valve isolation occurs at 32.85 sec based on a 2.85-sec delay between the time of the break and valve actuation and a 30-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed.

The peak calculated absolute pressures with their respective design values and margins are listed in Table 3B-3. Graphs showing the calculated pressures in Volumes 175-1 through 175-11 with respect to time are given on Figures 3B-3 through 3B-13.

The design pressure of Volume 175-1 is based on a 4-in RCIC DER in that node with the two blowout panels between Volumes 175-2 and 175-3 being available. Volume 175-2's design pressure is based on a 4-in RCIC DER in Volume 175-2. The blowdown listings for these two RCIC breaks are given in Tables 3B-8 and 3B-9. Both nodalization models produced the same peak pressure in Volume 175-2.

The peak pressure values for Volumes 175-3 through 175-10 are determined by a 10-in RCIC DER in Volume 215-1. Volume 175-11's peak pressure is based on an 8-in RWCU DER in Volume 215-5. The 8-in DER occurs on the suction side of the RWCU recirculation pumps. These breaks are discussed in the section that analyzes breaks between el 215 ft 0 in and el 240 ft 0 in.

3B.3.2 Elevation 215 ft 0 in to Elevation 240 ft 0 in

Thirteen cases were modeled on this elevation. A 10-in RCIC steam line DER was postulated in the spent fuel pool cooling and cleanup (SFC) heat exchanger room B (Volume 215-1). An 8-in RHR steam line DER was postulated in the horizontal pipe tunnels at Azimuth 135° (Volume 215-4) and Azimuth 164° (Volume 215-5). The 8-in RWCU DERs on the suction and discharge sides of the RWCU recirculation pumps were analyzed in the SFC heat exchanger room B (Volume 215-1) and the horizontal pipe tunnel at Azimuth 164° (Volume 215-5). The 4-in RWCU DERs on the suction and discharge sides of the RWCU recirculation pumps were examined in RWCU pump rooms A (Volume 215-9) and B (Volume 215-10).

RCIC valve isolation occurs at 32.85 sec based on a 2.85-sec delay between the time of the break and valve actuation and a 30-sec valve closure time. RWCU valve isolation occurs at 16.5 sec based on a 2.5-sec delay between the time of the break and valve actuation and a 14-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed.

Two nodalization models were used to analyze the 10-in RCIC DER and 8-in RWCU DER in Volume 215-1. One model assumed a fire damper between Volume 215-1 and the pipe chase below it (Volume 175-2) remained open. This was done to see how the blowdown from these two cases could affect Volume 175-1.

Vent curtains are across certain junctions to prevent air flow from entering volumes during normal operation. Vent curtains exist at the opening between the horizontal pipe tunnel at Azimuth 230° (Volume 215-6) and the general volume (Volume 215-8), and between the RWCU pump rooms and the general volume (Volume 215-9 to Volume 215-8 and Volume 215-10 to Volume 215-8). All 13 cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open.

The peak calculated absolute pressures with their respective design values and margins are listed in Table 3B-3. Graphs

showing the calculated pressures in Volumes 215-1 through 215-10 with respect to time are given on Figures 3B-14 through 3B-23.

The peak pressures of Volumes 215-1 and 215-6 are based on the 8-in RWCU DER in Volume 215-1 on the suction side of the RWCU recirculation pumps. This blowdown is listed in Table 3B-11. Volume 215-2's peak pressure is based on the 10-in RWCU DER in the vertical pipe chase at Azimuth 29° (Volume 261-5). This break is described in Section 3B.3.4.

The peak pressures of Volumes 215-3 and 215-8 are based on the 10-in RCIC DER in Volume 215-1. Volumes 215-4, 215-5, and 215-7's peak pressures are calculated from an 8-in RWCU DER in Volume 215-5 on the suction side of the RWCU recirculation pumps. The peak pressures of Volume 215-9 and Volume 215-10 come from 4-in RWCU suction side DERs within the respective volumes. The blowdown listings for these breaks are given in Tables 3B-10 and 3B-12.

3B.3.3 Elevation 240 ft 0 in to Elevation 261 ft 0 in

Four cases were modeled on this elevation. The 8-in RWCU line DERs on the suction and discharge sides of the RWCU recirculation pumps were analyzed in the vertical pipe chase at Azimuth 180° (Volume 240-1). A 10-in RCIC steam line DER and an 8-in RWCU DER were postulated in the cubicle containing penetrations Z-11 and Z-13 (Volume 240-2).

RCIC valve isolation occurs at 32.85 sec based on a 2.85-sec delay between the time of the break and valve actuation and a 30-sec valve closure time. RWCU valve isolation occurs at 16.5 sec based on a 2.5-sec delay between the time of the break and valve actuation and a 14-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed.

Vent curtains are attached to the wire-mesh door that leads from the TIP cubicle at Azimuth 45° (Volume 240-3) to the general volume (Volume 240-5). All four cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open.

The peak calculated absolute pressures with their respective design values and margins are listed in Table 3B-3. Graphs showing the calculated pressures in Volumes 240-1 through 240-8 with respect to time are given on Figures 3B-24 through 3B-31.

The peak pressures of Volume 240-1 and Volume 240-2 are based on an 8-in RWCU DER in Volume 240-1 on the suction side of the RWCU recirculation pumps. The blowdown listing for this break is shown in Table 3B-13.

The peak pressures of Volume 240-3 and 240-4 are determined from a 10-in RWCU DER in the vertical pipe chase at Azimuth 29°

(Volume 261-5). This break is reviewed in the section that studies breaks between el 261 ft 0 in and el 289 ft 0 in.

The peak pressures of Volume 240-5, Volume 240-7, and Volume 240-8 come from a 6-in RWCU DER in a RWCU filter/demineralizer cubicle (Volume SC328190). The peak pressure of Volume 240-6 is determined by the 10-in RWCU DER in the horizontal pipe tunnel at Azimuth 45° (Volume 306-4). These breaks are respectively reviewed in the section that studies breaks above el 328 ft 10 in and the section that studies breaks between el 306 ft 6 in and el 328 ft 10 in.

3B.3.4 Elevation 261 ft 0 in to Elevation 289 ft 0 in

Four cases were modeled on this elevation. A 10-in RCIC steam line DER and an 8-in RWCU line DER were postulated in the vertical pipe chase at Azimuth 180° (Volume 261-1). An 8-in RWCU line DER is also postulated in the vertical pipe chase directly above Volume 261-1 (Volume 261-2). A 10-in RWCU line DER is analyzed in the vertical pipe chase at Azimuth 29° (Volume 261-5).

RCIC valve isolation occurs at 32.85 sec based on a 2.85-sec delay between the time of the break and valve actuation and a 30-sec valve closure time. RWCU valve isolation occurs at 16.5 sec based on a 2.5-sec delay between the time of the break and valve actuation and a 14-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed.

Vent curtains are across the door opening that connects Volume 261-1 to the general volume (Volume 261-4). All four cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open.

The peak calculated absolute pressures with their respective design value and margins are listed in Table 3B-3. Graphs showing the calculated pressures in Volumes 261-1 through 261-5 with respect to time are given on Figures 3B-32 through 3B-36.

The peak pressure of Volume 261-1 is based on an 8-in RWCU DER in Volume 261-1. Volume 261-2's peak pressure is determined by the 8-in RWCU DER in Volume 261-2. Volume 261-3's and Volume 261-5's peak pressures are determined from a 10-in RWCU DER in the vertical pipe chase at Azimuth 29° (Volume 261-5). The blowdown listing for these breaks is given in Tables 3B-14 and 3B-15.

Volume 261-4's peak pressure is calculated from a 6-in RWCU steam line DER in a filter/demineralizer cubicle (Volume SC328190). This break is reviewed in the section that studies breaks above el 328 ft 10 in.

3B.3.5 Elevation 289 ft 0 in to Elevation 306 ft 6 in

Two cases were modeled on this elevation. An 8-in RWCU line DER was postulated in the vertical pipe chase at Azimuth 180° (Volume 289-1). A 10-in RWCU line DER is also postulated in the vertical pipe chase at Azimuth 29° (Volume 289-2).

RWCU valve isolation occurs at 16.5 sec based on a 2.5-sec delay between the time of the break and valve actuation and a 14-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed. Both cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open.

The peak calculated absolute pressures with their respective design values and margins are listed in Table 3B-3. Graphs showing the calculated pressures in Volumes 289-1 through 289-3 with respect to time are given on Figures 3B-37 through 3B-39.

The peak pressure of Volume 289-1 is based on an 8-in RWCU line DER in Volume 289-1. Volume 289-2's peak pressure is determined by the 10-in RWCU DER in Volume 289-2. The blowdown listings for these breaks are given in Tables 3B-14 and 3B-15.

Volume 289-3's peak pressure is calculated from a 6-in RWCU DER in a RWCU filter/demineralizer cubicle (Volume SC328190). This break is reviewed in the section that studies breaks above el 328 ft 10 in.

3B.3.6 Elevation 306 ft 6 in to Elevation 328 ft 10 in

Six cases were modeled on this elevation. An 8-in RWCU line DER was postulated in the horizontal pipe tunnel at Azimuth 180° (Volume 306-1). An 8-in and 10-in RWCU line DERs were modeled in the RWCU heat exchanger room (Volume 306-2). Two 10-in RWCU line DERs were analyzed in the horizontal pipe tunnels at Azimuth 315° (Volume 306-3) and Azimuth 45° (Volume 306-4).

The sixth case models the DER in Volume 306-2 with lower initial temperatures. This case was submitted to ensure that the lower temperatures in the volumes would not affect the temperature elements' ability to detect the break within 2.5 sec. The initial temperatures used here are 40° lower than the maximum normal design temperatures in the volumes.

RWCU valve isolation occurs at 16.5 sec based on a 2.5-sec delay between the time of the break and valve actuation and a 14-sec valve closure time. Flow was maximized by assuming a constant reactor pressure until the valve was fully closed. All six cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open.

The peak calculated absolute pressures with their respective design values and margins are listed in Table 3B-3. Graphs

showing the calculated pressures in Volumes 306-1 through 306-5 with respect to time are given on Figures 3B-40 through 3B-44.

The peak pressures of Volume 306-1, Volume 306-2, Volume 306-3, and Volume 306-4 are based on the 10-in RWCU DER case in Volume 306-2 with the lower initial temperatures. The blowdown listing for this break is given in Table 3B-15.

Volume 306-5's peak pressure is determined by the 6-in RWCU DER in a RWCU filter/demineralizer cubicle (Volume 328-2). This break is reviewed in the section that discusses breaks above el 328 ft 10 in.

3B.3.7 Elevation 328 ft 10 in and Above

The two cases modeled were 6-in RWCU line DERs in the RWCU pump holdup room (Volume 328-2) and in a RWCU filter/demineralizer cubicle (Volume SC328190). The nodalization diagram and subcompartment vent path descriptions for these cases are given on Figure 3B-2 and in Table 3B-6.

Both cases assumed that one of the blowout panels between Volume 175-2 and Volume 175-3 fails to open. For the break in Volume SC328190, the concrete plug on top of the volume was calculated to move and create a vent opening 0.91 sec after "hot" fluid emerges from the DER and enters the volume. This action is described in Section 3B.2.

RWCU valve isolation occurs at 61 sec. This time is derived from the 45 sec it takes for a high differential flow signal to be generated, the 2 sec assumed for instrumentation delay, and a 14 sec valve closure time.

Upstream flow was maximized by assuming a constant reactor pressure until the RWCU isolation valves fully close. Downstream flow assumed the driving pressure to be reducing linearly with respect to time. Other assumptions made about the breaks in these two volumes are stated in Section 3B.1.

The peak calculated absolute pressures with their respective design values and margins are listed in Table 3B-3. Graphs showing the calculated pressures in Volume 328-1, Volume 328-2, Volume SC328190, and Volume 353-1 with respect to time are given on Figures 3B-45 through 3B-48.

The peak pressures of Volume 328-1, Volume SC328190, and Volume 353-1 are based on a 6-in RWCU DER in the RWCU filter/demineralizer cubicle (Volume SC328190). Volume 328-2's peak pressure is determined by the 6-in RWCU DER in Volume 328-2. The blowdown listing for these breaks is given in Table 3B-16.

3B.4 IMPACT OF EXTENDED POWER UPRATE (EPU)

The impact of EPU on the mass and energy release rates used to determine sub-compartment pressurization was evaluated. The results of the EPU evaluation demonstrate that the controlling mass and energy release rates, for power levels up to 102 percent of EPU thermal power(for both normal and reduced feedwater temperature cases), are bounded by the calculation of record mass and energy release rates.

For a feedwater line break in the main steam tunnel, EPU implementation results in approximately a 4 percent increase in the flashing energy release flux above the flashing energy flux associated with the analysis of record mass and energy release rates. However, the limiting MSLB in the steam tunnel remains bounding for steam tunnel pressurization.

The evaluations of a RCIC line break and a MSLB in the steam tunnel are unchanged, based on the NEDC-33004P-A generic disposition for high-energy lines containing steam.

For RWCU, a comparison of 102 percent stretch power uprate (SPU) and 102 percent EPU reactor heat balances determined that there is a negligible change (0.2 Btu/lbm) in RWCU fluid conditions. Additionally, there is no change in reactor pressure. Thus, there is a negligible change in RWCU mass and energy releases from SPU to EPU conditions.

3B.5 REFERENCES

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2. Henry, R. E. and Fauske, H. K. The Two-Phase Critical Flow of One Component Mixtures in Nozzles, Orifices, and Short Tubes, Journal of Heat Transfer, Trans. ASME, 93, May 1971, pp 179-187.

3. NEDO-24548, Technical Description Annulus Pressurization Load Adequacy Evaluation, Section 2, Short Term Mass Energy Release, D. F. Sharma, January 1979.

4. ANSI/ANS-58.2-1980, American National Standard Design Bases for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture, Appendix E, Acceptable Simplified Methods for Calculating Mass and Energy Release Rates for Compartment Pressurization Results, December 31, 1980.

5. NEDC-33004P-4, Licensing Topical Report Constant Pressure Power Uprate, July 2003.

TABLE 3B-1
(Sheet 1 of 3)

HIGH-ENERGY LINE BREAKS ANALYZED IN
THE REACTOR BUILDING

Break Number	Line	Break in Volume Number	Design Break for Volume Number(s)	Comments
1	4" RCIC	175-1	175-1	A blowout panel between nodes 175-2 and 175-3 is assumed not to open.
2	4" RCIC	175-1	N/A	The blowout panel between nodes 175-1 and 175-9 is assumed not to open.
3	4" RCIC	175-2	175-2	A blowout panel between nodes 175-2 and 175-3 is assumed not to open.
4	4" RCIC	175-2	175-2	The blowout panel between nodes 175-1 and 175-9 is assumed not to open.
5	10" RCIC	215-1	175-3, 175-4, 175-5, 175-6, 175-7, 175-8, 175-9, 175-10, 215-3, 215-8	
6	10" RCIC	215-1	N/A	The fire damper junction between nodes 175-2 and 215-1 is included in the nodalization.
7	8" RWCU	215-1	215-1, 215-6	The break is modeled on the suction side of the RWCU recirculation pump.
8	8" RWCU	215-1	N/A	The break is modeled on the discharge side of the RWCU recirculation pump.
9	8" RWCU	215-1	N/A	The fire damper junction between nodes 175-2 and 215-1 is included in the nodalization.
10	8" RHR	215-4	N/A	This line is connected to a high-energy RCIC line.
11	8" RHR	215-5	N/A	This line is connected to a high-energy RCIC line.
12	8" RWCU	215-5	175-11, 215-4, 215-5, 215-7	The break is modeled on the suction side of the RWCU recirculation pump.
13	8" RWCU	215-5	N/A	The break is modeled on the discharge side of the RWCU recirculation pump.

TABLE 3B-1
(Sheet 2 of 3)

HIGH-ENERGY LINE BREAKS ANALYZED IN
THE REACTOR BUILDING

Break Number	Line	Break in Volume Number	Design Break for Volume Number(s)	Comments
14	4" RWCU	215-9	215-9	The break is modeled on the suction side of the RWCU recirculation pump.
15	4" RWCU	215-9	N/A	The break is modeled on the discharge side of the RWCU recirculation pump.
16	4" RWCU	215-10	215-10	The break is modeled on the suction side of the RWCU recirculation pump.
17	4" RWCU	215-10	N/A	The break is modeled on the discharge side of the RWCU recirculation pump.
18	8" RWCU	240-1	240-1, 240-2	The break is modeled on the suction side of the RWCU recirculation pump.
19	8" RWCU	240-1	N/A	The break is modeled on the discharge side of the RWCU recirculation pump.
20	10" RCIC	240-2	N/A	
21	8" RWCU	240-2	N/A	
22	10" RCIC	261-1	N/A	
23	8" RWCU	261-1	261-1	
24	8" RWCU	261-2	261-2	
25	10" RWCU	261-5	215-2, 240-3, 240-4, 261-3, 261-5	
26	8" RWCU	289-1	289-1	
27	10" RWCU	289-2	289-2	
28	8" RWCU	306-1	N/A	
29	8" RWCU	306-2	N/A	
30	10" RWCU	306-2	N/A	
31	10" RWCU	306-2	306-1, 306-2, 306-3, 306-4	Lower initial nodal temperatures are used here.

TABLE 3B-1
(Sheet 3 of 3)

HIGH-ENERGY LINE BREAKS ANALYZED IN
THE REACTOR BUILDING

Break Number	Line	Break in Volume Number	Design Break for Volume Number(s)	Comments
32	10" RWCU	306-3	N/A	
33	10" RWCU	306-4	240-6	
34	6" RWCU	328-2	328-2	
35	6" RWCU	SC328190	240-5, 240-7, 240-8, 261-4, 289-3, 306-5, 328-1, SC328190, 353-1	The concrete plug on top of the filter/demineralizer cubicle moves and creates a vent opening.

TABLE 3B-2
(Sheet 1 of 1)

HIGH-ENERGY LINE BREAKS ANALYZED IN THE MAIN STEAM TUNNEL

Break Number	Line	Break in Volume Number	Design Break for Volume Number(s)	Comments
1	28" MSS	MST-5	N/A	All-steam blowdown. Double-ended rupture.
2	28" MSS	MST-5	MST-5, MST-6, MST-8	Two-phase blowdown (15% steam). Double-ended rupture.
3	28" MSS	MST-3	N/A	All steam blowdown. Double-ended rupture.
4	28" MSS	MST-3	MST-1, MST-2, MST-3, MST-4	Two-phase blowdown (15% steam). Double-ended rupture.
5	28" MSS	MST-2	N/A	Longitudinal break - all-steam blowdown.
6	24" FWS	MST-3	N/A	Subcooled liquid blowdown. Double-ended rupture.

NOTE: Subcompartment volumes are defined in Table 3B-4 and Figure 3B-49.
MSS: Main steam system
FWS: Feedwater system

TABLE 3B-3
(Sheet 1 of 4)

SUBCOMPARTMENT NODAL DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 AND 46 NODE MODELS)

Volume Number	Net Volume (ft^3)	Description of Volume	Break Location	Break Line[(2)] (All Double-Ended Ruptures)	Break Number(s)	Absolute Peak Pressure	Calculated Peak Differential Pressure[(1)] (psid)	Design Peak Differential Pressure[(1)] (psid)	Initial Air Temperature (deg F)
175-1	12,800	RCIC pump and turbine room (el 175'-0" to 197'-0")	175-1	4" RCIC	1	17.53	2.84	3.80	120
175-2	504	Vertical pipe chase at AZ 183 (el 194'-9" to 215'-0")	175-2	4" RCIC	3,4	17.78	3.09	3.80	120
175-3	1,530	Penetration Z-19 cubicle at AZ 171 (el 197'-0" to 215'-0")	215-1	10" RCIC	5	17.14	2.45	3.80	120
175-4	371,900	General volume of el 175'-0" to 215'-0"	215-1	10" RCIC	5	17.14	2.45	3.00	104
175-5	5,160	Flood trough west of AZ 90 (el 188'-6" to 201'-0")	215-1	10" RCIC	5	17.14	0.00[(a)]	1.00	104
175-6	5,490	Flood trough east of AZ 90 (el 188'-6" to 201'-0")	215-1	10" RCIC	5	17.14	0.00[(a)]	1.00	104
175-7	5,100	Flood trough west of AZ 270 (el 188'-6" to 201'-0")	215-1	10" RCIC	5	17.14	0.00[(a)]	1.00	104
175-8	4,970	Flood trough east of AZ 270 (el 188'-6" to 201'-0")	215-1	10" RCIC	5	17.14	0.00[(a)]	1.00	104
175-9	1,200	Penetration area at AZ 155 (el 197'-0" to 204'-0")	215-1	10" RCIC	5	17.14	0.23[(a)]	2.00	104
175-10	3,320	Vertical pipe chase at AZ 105 (el 188'-6" to 215'-0")	215-1	10" RCIC	5	17.23	2.54	3.50	120
175-11	3,120	Vertical pipe chase at AZ 245 (el 188'-6" to 212'-9")	215-5	8" RWCU suction	12	18.59	3.90	4.50	120
215-1	8,740	Spent fuel chamber (SFC) heat exchanger room B at AZ 200 (el 215'-0" to 240'-0")	215-1	8" RWCU suction	7	17.97	3.28	4.00	120
215-2	3,310	Horizontal pipe tunnel at AZ 45 (el 227'-0" to 240'-0")	261-5	10" RWCU	25	17.40	2.71	3.50	120
215-3	11,300	SFC heat exchanger room A at AZ 90 (el 215'-0" to 240'-0")	215-1	10" RCIC	5	17.22	2.53	3.50	120

TABLE 3B-3
(Sheet 2 of 4)

SUBCOMPARTMENT NODAL DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 AND 46 NODE MODELS)

Volume Number	Net Volume (ft^3)	Description of Volume	Break Location	Break Line[2] (All Double-Ended Ruptures)	Break Number(s)	Absolute Peak Pressure	Calculated Peak Differential Pressure[1] (psid)	Design Peak Differential Pressure[1] (psid)	Initial Air Temperature (deg F)
215-4	2,710	Horizontal pipe tunnel at AZ 135 (el 227'-0" to 240'-0")	215-5	8" RWCU suction	12	19.01	4.32	5.00	120
215-5	4,480	Horizontal pipe tunnel at AZ 164 above the RWCU pump rooms (el 227'-0" to 240'-0")	215-5	8" RWCU suction	12	19.05	4.36	5.00	120
215-6	5,400	Horizontal pipe tunnel at AZ 230 (el 227'-0" to 240'-0")	215-1	8" RWCU suction	7	18.12	3.43	3.50	120
215-7	1,490	Vertical pipe chase at AZ 245 (el 212'-9" to 227'-0")	215-5	8" RWCU suction	12	18.08	3.39	3.50	120
215-8	213,900	General volume of el 215'-0" to 240'-0"	215-1	10" RCIC	5	17.13	2.44	3.00	104
215-9	1,540	RWCU pump room A at AZ 152 (el 215'-0" to 227'-0")	215-9	4" RWCU suction	14	22.79	8.10	9.50	120
215-10	1,750	RWCU pump room B at AZ 172 (el 215'-0" to 227'-0")	215-10	4" RWCU suction	16	22.81	8.12	9.50	120
240-1	783	Vertical pipe chase at AZ 180 (el 240'-0" to 261'-0")	240-1	8" RWCU suction	18	28.98	14.29	16.50	120
240-2	4,280	Penetrations Z-11 and Z-13 cubicle at AZ 193 (el 240'-0" to 261'-0")	240-1	8" RWCU suction	18	29.02	14.33	16.50	120
240-3	4,570	TIP cubicle at AZ 45 (el 250'-0" to 261'-0")	261-5	10" RWCU	25	19.12	4.43	5.00	120
240-4	575	Vertical pipe chase at AZ 29 (el 240'-0" to 261'-0")	261-5	10" RWCU	25	18.64	3.95	4.50	120
240-5	181,000	General volume of el 240'-0" to 261'-0"	SC328190	6" RWCU	35	17.01	2.32	3.00	104
240-6	1,809	Radiation pipe chase at AZ 52 (el 240'-0" to 315'-10")	306-4	10" RWCU	33	20.21	5.52	10.20	120
240-7	3,100	Penetration Z-10A cubicle at AZ 100 (el 240'-0" to 261'-0")	SC328190	6" RWCU	35	17.01	2.32	3.00	134.5
240-8	2,890	Penetration Z-10B cubicle at AZ 260 (el 240'-0" to 261'-0")	SC328190	6" RWCU	35	17.01	2.32	3.00	120

TABLE 3B-3
(Sheet 3 of 4)

SUBCOMPARTMENT NODAL DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 AND 46 NODE MODELS)

Volume Number	Net Volume (ft^3)	Description of Volume	Break Location	Break Line(2) (All Double-Ended Ruptures)	Break Number(s)	Absolute Peak Pressure	Calculated Peak Differential Pressure(1) (psid)	Design Peak Differential Pressure(1) (psid)	Initial Air Temperature (deg F)
261-1	1,650	Vertical pipe chase at AZ 180 (el 261'-0" to 272'-0")	261-1	8" RWCU	23	25.26	10.57	13.80	120
261-2	1,080	Vertical pipe chase at AZ 180 (el 272'-0" to 289'-0")	261-2	8" RWCU	24	25.32	10.63	13.80	120
261-3	1,150	Vertical pipe chase at AZ 29 (el 261'-0" to 279'-0")	261-5	10" RWCU	25	21.32	6.63	11.70	120
261-4	303,800	General volume of el 261'-0" to 289'-0"	SC328190	6" RWCU	35	17.00	2.31	3.00	104
261-5	1,340	Vertical pipe chase at AZ 29 (el 279'-0" to 295'-0")	261-5	10" RWCU	25	21.58	6.89	11.70	120
289-1	872	Vertical pipe chase at AZ 180 (el 289'-0" to 306'-6")	289-1	8" RWCU	26	23.36	8.67	13.80	120
289-2	2,530	Vertical pipe chase at AZ 29 (el 295'-0" to 315'-10")	289-2	10" RWCU	27	21.48	6.79	11.70	120
289-3	181,600	General volume of el 289'-0" to 306'-6"	SC328190	6" RWCU	35	16.99	2.30	3.00	104
306-1	29,700	Horizontal pipe tunnel at AZ 180 (el 306'-6" to 328'-10")	306-2 (Lower initial temperatures)	10" RWCU	31	19.53	4.84	10.20	120
306-2	22,400	RWCU heat exchanger room at AZ 270 (el 306'-6" to 328'-10")	306-2 (Lower initial temperatures)	10" RWCU	31	19.56	4.87	10.20	120
306-3	4,810	Horizontal pipe tunnel at AZ 315 (el 312'-5 1/2" to 328'-10")	306-2 (Lower initial temperatures)	10" RWCU	31	19.47	4.78	10.20	120
306-4	7,360	Horizontal pipe tunnel at AZ 45 (el 315'-10" to 328'-10")	306-2 (Lower initial temperatures)	10" RWCU	31	19.48	4.79	10.20	120
306-5	187,300	General volume of el 306'-6" to 328'-10"	SC328190	6" RWCU	35	16.97	2.28	3.00	104
328-1	68,700	General volume of eastern portion of el 328'-10" to 353'-10"	SC328190	6" RWCU	35	16.96	2.27	3.00	104
328-2	14,300	RWCU holdup pump room (el 328'-10" to 353'-10")	328-2	6" RWCU	34	21.07	6.38	8.20	120

TABLE 3B-3
(Sheet 4 of 4)

SUBCOMPARTMENT NODAL DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 AND 46 NODE MODELS)

Volume Number	Net Volume (ft^3)	Description of Volume	Break Location	Break Line(2) (All Double-Ended Ruptures)	Break Number(s)	Absolute Peak Pressure	Calculated Peak Differential Pressure(1) (psid)	Design Peak Differential Pressure(1) (psid)	Initial Air Temperature (deg F)
SC328190(b)	1,610	RWCU filter/demineralizer cubicle (el 328'-10" to 353'-10")	SC328190	6" RWCU	35	27.15	12.46	15.00	120
353-1	1,590,000	General volume of el 353'-10" to 426'-3"	SC328190	6" RWCU	35	16.93	2.24	3.00	104
---	1.00E+30	Outside atmosphere (dummy node)	N/A	N/A	N/A	N/A	N/A	N/A	104

NOTES: The initial air pressure for all the subcompartment volumes is 14.69 psia.
The initial relative humidity for the RCIC breaks and the 6" RWCU breaks in Volume Numbers 328-2 and SC328190 is zero percent.
The other RWCU breaks use an initial relative humidity of 20% to prevent numerical instabilities in the analyses.
Additional information about the initial environmental conditions can be found in Section 3B.1: Design Bases.

(1) This is calculated by subtracting 14.69 psia (-0.25 in W.G.) from the absolute pressure of the volume.
(2) The break line information comes from Table 3B-1.
(a) This volume is openly exposed to Volume Number 175-4. Calculated differential pressures across these walls are taken with respect to Volume Number 175-4 to get the correct results. The value listed is the calculated peak differential pressure for the entire transient.
(b) Volume Number SC328190 is the added volume in the 46 node model.

TABLE 3B-4
(Sheet 1 of 2)

SUBCOMPARTMENT NODAL DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE MAIN STEAM TUNNEL

Volume Number	Net Volume (ft^3)	Description of Volume	Break Location	Break Line[2] (All Double-Ended Ruptures)	Break Number(s)	Absolute(*) Peak Pressure	Calculated Peak(*) Differential Pressure[1] (psid)	Design Peak Differential Pressure[1] (psid)	Initial air temperature (deg F)
MST-1	32,300	Volume between el 261'-0" and el 289'-0" against the primary containment	MST-3	28" MSS - two phase blowdown	4	39.55	24.85	27.35	130
MST-2	25,000	Volume between el 240'-0" and el 261'-0" against the primary containment	MST-3	28" MSS - two phase blowdown	4	39.55	24.85	27.35	130
MST-3	12,000	Volume between el 238'-6" and el 258'-9" against the jet impingement wall	MST-3	28" MSS - two phase blowdown	4	39.55	24.85	27.35	130
MST-4	12,000	Vertical pipe chase between el 258'-9" and el 292'-0"	MST-3	28" MSS - two phase blowdown	4	28.20	13.50	27.35	130
MST-5	7,340	Volume below blowout panels subcompartment (el 292'-0" to el 304'-8.5")	MST-5	28" MSS - two phase blowdown	2	26.71	12.01	27.35	130
MST-6	7,450	Volume leading to the turbine building (el 292'-0" to el 301'-7.5")	MST-5	28" MSS - two phase blowdown	2	20.85	6.15	27.35	130
MST-7	1.00E+ 20	Turbine building and outside atmosphere (dummy node)	N/A	N/A	N/A	N/A	N/A	N/A	130
MST-8	7,470	T-section volume containing blowout panels (el 304'-8.5" to el 318'-9")	MST-5	28" MSS - two phase blowdown	5	23.90	9.20	23.00	130

TABLE 3B-4
(Sheet 2 of 2)

SUBCOMPARTMENT NODAL DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE MAIN STEAM TUNNEL

(*) The maximum initial normal main steam tunnel (MST) air temperature has been increased from 120°F to 130°F. The absolute peak pressures and peak calculated differential pressures are based on an initial MST air temperature of 120°F. A sensitivity study was performed which showed insignificant changes. The above pressures are still valid. USAR Figures 3B-50 through 3B-56, the main steam tunnel absolute pressure curves, are unchanged.

(1) This is calculated by subtracting 14.70 psia from the absolute pressure of the volume.

(2) The break line information comes from Table 3B-2.

NOTES: The initial air pressure for all subcompartment volumes is 14.70 psia.
The initial relative humidity for all breaks is zero percent.
Additional information about the initial environmental conditions can be found in Section 3B.1, Design Bases.

TABLE 3B-5
(Sheet 1 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction[1]	Turning Loss	Friction	Sharp Edge Loss	Total
1	175-1 175-2	175-2 175-1	25.88 25.88	0.327 0.327	-- --	-- --	-- --	-- --	-- --	0.631 1.411	0.631 1.411
2(a)	175-1 175-9	175-9 175-1	10.00 10.00	0.258 0.258	0.4932 0.4715	0.8891 0.9729	-- --	0.3368 0.3445	0.0260 0.0260	-- --	1.745 1.815
3(a)	175-2 175-3	175-3 175-2	10.00 10.00	0.366 0.366	0.4517 0.4646	0.8635 0.8161	-- --	0.1568 0.1546	0.0347 0.0347	-- --	1.507 1.470
4(a)	175-2 175-3	175-3 175-2	10.00 10.00	0.366 0.366	0.4517 0.4646	0.8635 0.8161	-- --	0.1568 0.1546	0.0347 0.0347	-- --	1.507 1.470
5(b)	175-2 215-1	215-1 175-2	1.047 1.047	2.648 2.648	0.4773 0.4982	0.9930 0.9113	0.200 0.200	-- --	0.0920 0.0920	-- --	1.762 1.702
6	175-3 175-4	175-4 175-3	21.00 21.00	0.215 0.215	-- --	-- --	0.224 0.224	-- --	-- --	0.898 1.584	1.122 1.808
7	175-4 175-5	175-5 175-4	242.86 242.86	0.015 0.015	-- --	-- --	0.200 0.200	-- --	-- --	0.994 1.847	1.194 2.047
8	175-4 175-6	175-6 175-4	265.09 265.09	0.014 0.014	-- --	-- --	0.200 0.200	-- --	-- --	0.992 1.840	1.192 2.040
9	175-4 175-7	175-7 175-4	241.26 241.26	0.014 0.014	-- --	-- --	0.200 0.200	-- --	-- --	0.994 1.848	1.194 2.048
10	175-4 175-8	175-8 175-4	240.31 240.31	0.015 0.015	-- --	-- --	0.200 0.200	-- --	-- --	0.994 1.848	1.194 2.048
11	175-4 175-9	175-9 175-4	140.18 140.18	0.020 0.020	-- --	-- --	0.200 0.200	-- --	-- --	0.814 1.698	1.014 1.898
12(d)	175-4 215-8	215-8 175-4	62.42 62.42	0.039 0.039	0.4964 0.4962	0.9849 0.9856	0.200 0.200	1.1548 1.1549	0.0066 0.0066	-- --	2.843 2.843
13	175-10 215-3	215-3 175-10	81.30 81.30	0.121 0.121	-- --	-- --	0.200 0.200	-- --	-- --	1.337 0.763	1.537 0.963

TABLE 3B-5
(Sheet 2 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction[1]	Turning Loss	Friction	Sharp Edge Loss	Total
14	175-11 215-7	215-7 175-11	68.66 68.66	0.153 0.153	-- --	-- --	0.200 0.200	-- --	-- --	0.438 0.506	0.638 0.706
15	215-1 215-5	215-5 215-1	113.40 113.40	0.213 0.213	0.2542 0.1484	0.0881 0.2585	-- --	0.2331 0.3051	0.0091 0.0091	-- --	0.585 0.721
16	215-1 215-6	215-6 215-1	103.20 103.20	0.228 0.228	-- --	-- --	-- --	-- --	-- --	0.599 0.951	0.599 0.951
17	215-1 215-8	215-8 215-1	23.25 23.25	0.209 0.209	-- --	-- --	0.224 0.224	-- --	-- --	1.949 2.344	2.173 2.568
18	215-1 240-1	240-1 215-1	15.46 15.46	0.637 0.637	0.4832 0.1500	0.0900 0.9339	0.200 0.200	0.0919 0.1649	0.0212 0.0212	-- --	0.886 1.470
19	215-2 215-3	215-3 215-2	29.03 29.03	0.357 0.357	0.3500 0.4601	0.8469 0.4899	-- --	0.3243 0.2829	0.0184 0.0184	-- --	1.540 1.251
20	215-2 240-4	240-4 215-2	32.52 32.52	0.220 0.220	0.4358 0.0500	0.0100 0.7597	0.200 0.200	0.1033 0.3050	0.0089 0.0089	-- --	0.758 1.324
21	215-3 215-4	215-4 215-3	43.00 43.00	0.171 0.171	-- --	-- --	-- --	-- --	-- --	1.731 2.016	1.731 2.016
22	215-3 215-8	215-8 215-3	21.00 21.00	0.176 0.176	-- --	-- --	0.224 0.224	-- --	-- --	0.815 1.682	1.039 1.906
23	215-4 215-5	215-5 215-4	90.71 90.71	0.245 0.245	-- --	-- --	-- --	-- --	-- --	0.606 0.505	0.606 0.505
24	215-6 215-7	215-7 215-6	68.66 68.66	0.074 0.074	-- --	-- --	0.200 0.200	-- --	-- --	0.776 1.401	0.976 1.601
25	215-6 215-8	215-8 215-6	21.00 N/A	0.321 N/A	0.3698 --	0.9655 --	3.424 (c)	0.9010 --	0.0098 --	-- --	5.670 (c)
26	215-8 215-9	215-9 215-8	N/A 14.33	N/A 0.313	-- --	-- --	(c) 3.424	-- --	-- --	-- 2.087	(c) 5.511
27	215-8 215-10	215-10 215-8	N/A 14.33	N/A 0.313	-- --	-- --	(c) 3.424	-- --	-- --	-- 2.185	(c) 5.609

TABLE 3B-5
(Sheet 3 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction[(1)]	Turning Loss	Friction	Sharp Edge Loss	Total
28(d)	215-8 240-5	240-5 215-8	52.03 52.03	0.045 0.045	0.4969 0.4970	0.9878 0.9876	0.200 0.200	1.1419 1.1418	0.0072 0.0072	-- --	2.834 2.834
29	215-9 215-10	215-10 215-9	15.75 15.75	0.414 0.414	0.4379 0.4379	0.7669 0.7669	-- --	0.0587 0.0587	0.0366 0.0366	-- --	1.300 1.300
30	240-1 240-2	240-2 240-1	101.05 101.05	0.064 0.064	-- --	-- --	-- --	-- --	-- --	0.835 0.557	0.835 0.557
31	240-1 261-1	261-1 240-1	28.38 28.38	0.336 0.336	-- --	-- --	0.200 0.200	-- --	-- --	0.652 0.379	0.852 0.579
32	240-3 240-4	240-4 240-3	37.83 37.83	0.110 0.110	-- --	-- --	0.200 0.200	-- --	-- --	0.504 1.093	0.704 1.293
33	240-3 240-5	240-5 240-3	16.38 N/A	0.234 N/A	-- --	-- --	3.424 (c)	-- --	-- --	1.688 --	5.112 (c)
34	240-3 240-3	261-3 261-3	39.68 39.68	0.143 0.143	-- --	-- --	0.200 0.200	-- --	-- --	0.867 1.617	1.067 1.817
35	240-5 240-7	240-7 240-5	2.299 2.299	0.091 0.091	0.4993 0.4962	0.9847 0.9971	-- --	1.2672 1.2711	0.0033 0.0033	-- --	2.755 2.768
36	240-5 240-8	240-8 240-5	2.299 2.299	0.093 0.093	0.4993 0.4958	0.9834 0.9971	-- --	1.2663 1.2707	0.0033 0.0033	-- --	2.752 2.767
37(d)	240-5 261-4	261-4 240-5	67.45 67.45	0.029 0.029	0.4960 0.4967	0.9868 0.9840	0.188 0.188	1.1615 1.1606	0.0064 0.0064	-- --	2.839 2.836
38	240-6 306-4	306-4 240-6	21.15 21.15	1.674 1.674	-- --	-- --	-- --	0.768 0.181	-- --	1.146 0.663	1.914 0.844
39	261-1 261-2	261-2 261-1	33.26 33.26	0.169 0.169	-- --	-- --	0.200 0.200	-- --	-- --	0.938 1.246	1.138 1.446
40	261-1 261-4	261-4 261-1	15.93 N/A	0.213 N/A	-- --	-- --	3.424 (c)	-- --	-- --	1.765 --	5.189 (c)
41	261-2 289-1	289-1 261-2	26.30 26.30	0.293 0.293	-- --	-- --	0.200 0.200	-- --	-- --	0.775 0.897	0.975 1.097

TABLE 3B-5
(Sheet 4 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction[1]	Turning Loss	Friction	Sharp Edge Loss	Total
42	261-3 261-5	261-5 261-3	49.95 49.95	0.227 0.227	-- --	-- --	0.200 0.200	-- --	-- --	0.220 0.198	0.420 0.398
43(d)	261-4 289-3	289-3 261-4	503.31 503.31	0.010 0.010	0.4802 0.4796	0.9199 0.9224	-- --	1.0607 1.0614	0.0023 0.0023	-- --	2.463 2.466
44	261-5 289-2	289-2 261-5	65.75 65.75	0.174 0.174	-- --	-- --	0.200 0.200	-- --	-- --	0.282 0.235	0.482 0.435
45	289-1 306-1	306-1 289-1	40.82 40.82	0.174 0.174	-- --	-- --	0.200 0.200	-- --	-- --	1.168 0.520	1.368 0.720
46	289-2 306-4	306-4 289-2	36.86 36.86	0.169 0.169	0.3166 0.4676	0.8746 0.4010	0.400 0.400	0.7511 0.6182	0.0108 0.0108	-- --	2.353 1.898
47(d)	289-3 306-5	306-5 289-3	2079.0 2079.0	0.010 0.010	0.4156 0.3910	0.6116 0.6907	-- --	0.9317 0.9605	0.0015 0.0015	-- --	1.960 2.044
48(d)	289-3 306-5	306-5 289-3	2590.0 2590.0	0.010 0.010	0.3948 0.3643	0.5307 0.6235	-- --	0.8459 0.8858	0.0013 0.0013	-- --	1.773 1.875
49	306-1 306-2	306-2 306-1	245.99 245.99	0.014 0.014	-- --	-- --	-- --	-- --	-- --	2.311 2.390	2.311 2.390
50(e)	306-1 328-2	328-2 306-1	8.956 8.956	0.401 0.401	0.4969 0.4932	0.9731 0.9875	-- --	0.1645 0.1651	0.0272 0.0272	-- --	1.662 1.673
51	306-2 306-3	306-3 306-2	136.80 136.80	0.035 0.035	-- --	-- --	-- --	-- --	-- --	1.645 1.959	1.645 1.959
52	306-3 306-4	306-4 306-3	92.48 92.48	0.445 0.445	-- --	-- --	-- --	-- --	-- --	0.324 0.390	0.324 0.390
53	306-5 328-1	328-1 306-5	508.58 508.58	0.010 0.010	0.4733 0.4200	0.7057 0.8962	-- --	1.0504 1.1151	0.0023 0.0023	-- --	2.232 2.434
54	328-1 353-1	353-1 328-1	502.88 502.88	0.010 0.010	0.4209 0.4887	0.9554 0.7087	-- --	1.1522 1.0693	0.0023 0.0023	-- --	2.531 2.269

TABLE 3B-5
(Sheet 5 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(45 NODE MODEL)

(1) The obstruction loss includes losses from a grill, grating, wire mesh door, and/or vent curtains. These losses are described in Section 3B.2, Design Features.

(a) This vent path is a blowout panel. The values in this table are for after the panel has fully opened (if it is not assumed to fail closed under the single-failure criteria). The blowout panel is fully open 0.3 sec after a 0.5 psid force is applied across it. The panel opens in the direction shown in the top listing for the vent path.

(b) This junction is used only for Break Numbers 6 and 9. This is explained in Table 3B-1, High-Energy Line Breaks Analyzed in the Reactor Building, and Section 3B.3.2, Elevation 215'-0" to Elevation 240'-0".

(c) This junction has vent curtains. The curtain allows flow in only one direction. Vent curtains are discussed in Section 3B.2, Design Features, Section 3B.3.2, Elevation 215'-0" to Elevation 240'-0", Section 3B.3.3, Elevation 240'-0" to Elevation 261'-0", and Section 3B.3.4, Elevation 261'-0" to Elevation 289'-0".

(d) Additional parallel flow paths could be created (approximately 100 to 200 sq ft opening) if the hatch covers at azimuth 318° are removed. This modification does not significantly change the differential pressure values.

(e) The junction area is reduced by approximately 20%, along with changes to inertia and resistance coefficients, as a result of the grating modification. This change does not significantly alter the differential pressure values.

NOTE: The homogeneous equilibrium model (HEM) for choked flow is used for all breaks and nodalizations. Appendix 6B, THREED Subcompartment Analytical Model, gives a description of the HEM used in the SWEC computer code THREED.

TABLE 3B-6
(Sheet 1 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(46 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction(1)	Turning Loss	Friction	Sharp Edge Loss	Total
1	175-1 175-2	175-2 175-1	25.88 25.88	0.327 0.327	-- --	-- --	-- --	-- --	-- --	0.631 1.411	0.631 1.411
2(a)	175-1 175-9	175-9 175-1	10.00 10.00	0.258 0.258	0.4932 0.4715	0.8891 0.9729	-- --	0.3368 0.3445	0.0260 0.0260	-- --	1.745 1.815
3(a)	175-2 175-3	175-3 175-2	10.00 10.00	0.366 0.366	0.4517 0.4646	0.8635 0.8161	-- --	0.1568 0.1546	0.0347 0.0347	-- --	1.507 1.470
4	175-2 215-1	215-1 175-2	1.047 1.047	2.648 2.648	0.4773 0.4982	0.9930 0.9113	0.200 0.200	-- --	0.0920 0.0920	-- --	1.762 1.702
5	175-3 175-4	175-4 175-3	21.00 21.00	0.215 0.215	-- --	-- --	0.224 0.224	-- --	-- --	0.898 1.584	1.122 1.808
6	175-4 175-5	175-5 175-4	242.86 242.86	0.015 0.015	-- --	-- --	0.200 0.200	-- --	-- --	0.994 1.847	1.194 2.047
7	175-4 175-6	175-6 175-4	265.09 265.09	0.014 0.014	-- --	-- --	0.200 0.200	-- --	-- --	0.992 1.840	1.192 2.040
8	175-4 175-7	175-7 175-4	241.26 241.26	0.014 0.014	-- --	-- --	0.200 0.200	-- --	-- --	0.994 1.848	1.194 2.048
9	175-4 175-8	175-8 175-4	240.31 240.31	0.015 0.015	-- --	-- --	0.200 0.200	-- --	-- --	0.994 1.848	1.194 2.048
10	175-4 175-9	175-9 175-4	140.18 140.18	0.020 0.020	-- --	-- --	0.200 0.200	-- --	-- --	0.814 1.698	1.014 1.898
11(c)	175-4 215-8	215-8 175-4	62.42 62.42	0.039 0.039	0.4964 0.4962	0.9849 0.9856	0.200 0.200	1.1548 1.1549	0.0066 0.0066	-- --	2.843 2.843
12	175-10 215-3	215-3 175-10	81.30 81.30	0.121 0.121	-- --	-- --	0.200 0.200	-- --	-- --	1.337 0.763	1.537 0.963
13	175-11 215-7	215-7 175-11	68.66 68.66	0.153 0.153	-- --	-- --	0.200 0.200	-- --	-- --	0.438 0.506	0.638 0.706

TABLE 3B-6
(Sheet 2 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(46 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction(1)	Turning Loss	Friction	Sharp Edge Loss	Total
14	215-1 215-5	215-5 215-1	113.40 113.40	0.213 0.213	0.2542 0.1484	0.0881 0.2585	-- --	0.2331 0.3051	0.0091 0.0091	-- --	0.585 0.721
15	215-1 215-6	215-6 215-1	103.20 103.20	0.228 0.228	-- --	-- --	-- --	-- --	-- --	0.599 0.951	0.599 0.951
16	215-1 215-8	215-8 215-1	23.25 23.25	0.209 0.209	-- --	-- --	0.224 0.224	-- --	-- --	1.949 2.344	2.173 2.568
17	215-1 240-1	240-1 215-1	15.46 15.46	0.637 0.637	0.4832 0.1500	0.0900 0.9339	0.200 0.200	0.0919 0.1649	0.0212 0.0212	-- --	0.886 1.470
18	215-2 215-3	215-3 215-2	29.03 29.03	0.357 0.357	0.3500 0.4601	0.8469 0.4899	-- --	0.3243 0.2829	0.0184 0.0184	-- --	1.540 1.251
19	215-2 240-4	240-4 215-2	32.52 32.52	0.220 0.220	0.4358 0.0500	0.0100 0.7597	0.200 0.200	0.1033 0.3050	0.0089 0.0089	-- --	0.758 1.324
20	215-3 215-4	215-4 215-3	43.00 43.00	0.171 0.171	-- --	-- --	-- --	-- --	-- --	1.731 2.016	1.731 2.016
21	215-3 215-8	215-8 215-3	21.00 21.00	0.176 0.176	-- --	-- --	0.224 0.224	-- --	-- --	0.815 1.682	1.039 1.906
22	215-4 215-5	215-5 215-4	90.71 90.71	0.245 0.245	-- --	-- --	-- --	-- --	-- --	0.606 0.505	0.606 0.505
23	215-6 215-7	215-7 215-6	68.66 68.66	0.074 0.074	-- --	-- --	0.200 0.200	-- --	-- --	0.776 1.401	0.976 1.601
24	215-6 215-8	215-8 215-6	21.00 N/A	0.321 N/A	0.3698 --	0.9655 --	3.424 (b)	0.9010 --	0.0098 --	-- --	5.670 (b)
25	215-8 215-9	215-9 215-8	N/A 14.33	N/A 0.313	-- --	-- --	(b) 3.424	-- --	-- --	-- 2.087	(b) 5.511
26	215-8 215-10	215-10 215-8	N/A 14.33	N/A 0.313	-- --	-- --	(b) 3.424	-- --	-- --	-- 2.185	(b) 5.609
27(c)	215-8 240-5	240-5 215-8	52.03 52.03	0.045 0.045	0.4969 0.4970	0.9878 0.9876	0.200 0.200	1.1419 1.1418	0.0072 0.0072	-- --	2.834 2.834
28	215-9 215-10	215-10 215-9	15.75 15.75	0.414 0.414	0.4379 0.4379	0.7669 0.7669	-- --	0.0587 0.0587	0.0366 0.0366	-- --	1.300 1.300

TABLE 3B-6
(Sheet 3 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(46 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction[(1)]	Turning Loss	Friction	Sharp Edge Loss	Total
29	240-1 240-2	240-2 240-1	101.05 101.05	0.064 0.064	-- --	-- --	-- --	-- --	-- --	0.835 0.557	0.835 0.557
30	240-1 261-1	261-1 240-1	28.38 28.38	0.336 0.336	-- --	-- --	0.200 0.200	-- --	-- --	0.652 0.379	0.852 0.579
31	240-3 240-4	240-4 240-3	37.83 37.83	0.110 0.110	-- --	-- --	0.200 0.200	-- --	-- --	0.504 1.093	0.704 1.293
32	240-3 240-5	240-5 240-3	16.38 N/A	0.234 N/A	-- --	-- --	3.424 (b)	-- --	-- --	1.688 --	5.112 (b)
33	240-3 240-3	261-3 261-3	39.68 39.68	0.143 0.143	-- --	-- --	0.200 0.200	-- --	-- --	0.867 1.617	1.067 1.817
34	240-5 240-7	240-7 240-5	2.299 2.299	0.091 0.091	0.4993 0.4962	0.9847 0.9971	-- --	1.2672 1.2711	0.0033 0.0033	-- --	2.755 2.768
35	240-5 240-8	240-8 240-5	2.299 2.299	0.093 0.093	0.4993 0.4958	0.9834 0.9971	-- --	1.2663 1.2707	0.0033 0.0033	-- --	2.752 2.767
36(c)	240-5 261-4	261-4 240-5	67.45 67.45	0.029 0.029	0.4960 0.4967	0.9868 0.9840	0.188 0.188	1.1615 1.1606	0.0064 0.0064	-- --	2.839 2.836
37	240-6 306-4	306-4 240-6	21.15 21.15	1.674 1.674	-- --	-- --	-- --	0.768 0.181	-- --	1.146 0.663	1.914 0.844
38	261-1 261-2	261-2 261-1	33.26 33.26	0.169 0.169	-- --	-- --	0.200 0.200	-- --	-- --	0.938 1.246	1.138 1.446
39	261-1 261-4	261-4 261-1	15.93 N/A	0.213 N/A	-- --	-- --	3.424 (b)	-- --	-- --	1.765 --	5.189 (b)
40	261-2 289-1	289-1 261-2	26.30 26.30	0.293 0.293	-- --	-- --	0.200 0.200	-- --	-- --	0.775 0.897	0.975 1.097
41	261-3 261-5	261-5 261-3	49.95 49.95	0.227 0.227	-- --	-- --	0.200 0.200	-- --	-- --	0.220 0.198	0.420 0.398
42	261-4 289-3	289-3 261-4	503.31 503.31	0.010 0.010	0.4802 0.4796	0.9199 0.9224	-- --	1.0607 1.0614	0.0023 0.0023	-- --	2.463 2.466
43(d)	261-4 289-3	289-3 261-4	197.42 197.42	0.010 0.010	0.4922 0.4920	0.9682 0.9692	-- --	1.2078 1.2080	0.0040 0.0040	-- --	2.672 2.673

TABLE 3B-6
(Sheet 4 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(46 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction(1)	Turning Loss	Friction	Sharp Edge Loss	Total
44	261-5 289-2	289-2 261-5	65.75 65.75	0.174 0.174	-- --	-- --	0.200 0.200	-- --	-- --	0.282 0.235	0.482 0.435
45	289-1 306-1	306-1 289-1	40.82 40.82	0.174 0.174	-- --	-- --	0.200 0.200	-- --	-- --	1.168 0.520	1.368 0.720
46	289-2 306-4	306-4 289-2	36.86 36.86	0.169 0.169	0.3166 0.4676	0.8746 0.4010	0.400 0.400	0.7511 0.6182	0.0108 0.0108	-- --	2.353 1.898
47	289-3 306-5	306-5 289-3	2079.0 2079.0	0.010 0.010	0.4156 0.3910	0.6116 0.6907	-- --	0.9317 0.9605	0.0015 0.0015	-- --	1.960 2.044
48	289-3 306-5	306-5 289-3	2590.0 2590.0	0.010 0.010	0.3948 0.3643	0.5307 0.6235	-- --	0.8459 0.8858	0.0013 0.0013	-- --	1.773 1.875
49(d)	289-3 306-5	306-5 289-3	202.16 202.16	0.010 0.010	0.4918 0.4894	0.9581 0.9674	-- --	1.2022 1.2050	0.0039 0.0039	-- --	2.656 2.666
50	306-1 306-2	306-2 306-1	245.99 245.99	0.014 0.014	-- --	-- --	-- --	-- --	-- --	2.311 2.390	2.311 2.390
51	306-1 328-2	328-2 306-1	7.192 7.192	0.493 0.493	0.4969 0.4932	0.9731 0.9875	0.200 0.200	0.1645 0.1651	0.0272 0.0272	-- --	1.862 1.873
52	306-2 306-3	306-3 306-2	136.80 136.80	0.035 0.035	-- --	-- --	-- --	-- --	-- --	1.645 1.959	1.645 1.959
53	306-3 306-4	306-4 306-3	92.48 92.48	0.445 0.445	-- --	-- --	-- --	-- --	-- --	0.324 0.390	0.324 0.390
54	306-5 328-1	328-1 306-5	508.58 508.58	0.010 0.010	0.4733 0.4200	0.7057 0.8962	-- --	1.0504 1.1151	0.0023 0.0023	-- --	2.232 2.434
55	328-1 353-1	353-1 328-1	502.88 502.88	0.010 0.010	0.4209 0.4887	0.9554 0.7087	-- --	1.1522 1.0693	0.0023 0.0023	-- --	2.531 2.269
56	SC328190 306-1	306-1 SC328190	0.170 0.170	20.65 20.65	0.5000 0.5000	1.0000 1.0000	-- --	-- --	0.2618 0.2618	-- --	1.762 1.762
57	SC328190 306-1	306-1 SC328190	0.049 0.049	71.51 71.51	0.5000 0.5000	1.0000 1.0000	-- --	-- --	0.5862 0.5862	-- --	2.086 2.086
58	SC328190 306-1	306-1 SC328190	0.863 0.863	4.098 4.098	0.5000 0.5000	1.0000 1.0000	-- --	-- --	0.1458 0.1458	-- --	1.646 1.646

TABLE 3B-6
(Sheet 5 of 5)

SUBCOMPARTMENT VENT PATH DESCRIPTION HIGH-ENERGY LINE BREAK ANALYSIS IN THE REACTOR BUILDING
(46 NODE MODEL)

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction(1)	Turning Loss	Friction	Sharp Edge Loss	Total
59	SC328190	328-2	0.718	4.915	0.5000	1.0000	--	--	0.1458	--	1.646
	328-2	SC328190	0.718	4.915	0.5000	1.0000	--	--	0.1458	--	1.646
60	SC328190	328-2	0.718	4.915	0.5000	1.0000	--	--	0.1458	--	1.646
	328-2	SC328190	0.718	4.915	0.5000	1.0000	--	--	0.1458	--	1.646
61	SC328190	328-2	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854
	328-2	SC328190	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854
62	SC328190	328-2	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854
	328-2	SC328190	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854
63	SC328190	328-2	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854
	328-2	SC328190	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854
64	SC328190	328-2	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854
	328-2	SC328190	0.181	19.390	0.5000	1.0000	--	--	0.3537	--	1.854

This nodalization model is used for the 6" RWCU breaks in Volumes 328-2 and SC328190. The fire damper junction between Volumes 175-2 and 215-1 is assumed to be open for the entire transient.

(1) The obstruction loss includes losses from a grill, grating, wire mesh door, and/or vent curtains. These losses are described in Section 3B.2, Design Features.

(a) This vent path is a blowout panel. The values in this table are for after the panel has fully opened. The blowout panel is fully open 0.3 sec after a 0.5 psid force is applied across it. The panel opens in the direction shown in the top listing for the vent path.

(b) This junction has vent curtains. The curtain allows flow in only one direction. Vent curtains are discussed in Section 3B.2, Design Features, Section 3B.3.2, Elevation 215'-0" to Elevation 240'-0", Section 3B.3.3, Elevation 240'-0" to Elevation 261'-0", and Section 3B.3.4, Elevation 261'-0" to Elevation 289'-0".

(c) Additional parallel flow paths could be created (approximately 100 sq ft opening) if the hatch covers at azimuth 318° are removed. This modification does not significantly change the differential pressure values.

(d) The parallel flow path could be unavailable if the hatches at azimuth 318° are covered. This change does not significantly affect the differential pressure values.

NOTES: The homogeneous equilibrium model (HEM) for choked flow is used for all breaks and nodalizations. Appendix 6B, THREED Subcompartment Analytical Model, gives a description of the HEM used in the SWEC computer code THREED.

TABLE 3B-7
(Sheet 1 of 1)

SUBCOMPARTMENT VENT PATH DESCRIPTION: HIGH-ENERGY LINE BREAK ANALYSIS IN THE MAIN STEAM TUNNEL

Vent Path Number	From Volume Number	To Volume Number	Vent Area (ft^2)	Inertia Factor L/A (1/ft)	Head Loss Coefficients						
					Contraction	Expansion	Obstruction[1]	Turning Loss	Friction	Sharp Edge Loss	Total
1	MST-1	MST-2	1054.4	0.017	0.3819	0.2510	--	--	0.0102	--	0.643
	MST-2	MST-1	1054.4	0.017	0.3931	0.2415	--	--	0.0102	--	0.645
2	MST-2	MST-3	157.2	0.013	0.4117	0.6149	--	0.0790	0.0019	--	1.108
	MST-3	MST-2	157.2	0.013	0.4117	0.6149	--	0.1155	0.0019	--	1.144
3	MST-3	MST-4	181.8	0.060	0.3695	0.2711	--	--	0.0091	--	0.650
	MST-4	MST-3	181.8	0.060	0.2715	0.5128	--	--	0.0091	--	0.793
4	MST-4	MST-5	255.5	0.054	0.2495	0.3737	--	--	0.0176	--	0.641
	MST-5	MST-4	255.5	0.054	0.3787	0.1395	--	--	0.0176	--	0.536
5	MST-5	MST-6	175.6	0.051	0.5580	0.2418	--	0.2018	0.0068	--	1.008
	MST-6	MST-5	175.6	0.051	0.5580	0.2418	--	0.2018	0.0068	--	1.008
6	MST-6	MST-7	264.8	0.071	0.5500	1.6847	--	--	0.0548	--	2.290
	MST-7	MST-6	264.8	0.071	0.5500	1.6847	--	--	0.0548	--	2.290
7	MST-5	MST-8	342.0	0.051	0.2771	0.0015	--	2.6872	--	--	2.966
	MST-8	MST-5	342.0	0.051	0.2771	0.0015	--	2.6872	--	--	2.966
8[a]	MST-8	MST-7	150.0	0.014	--	1.0000	--	0.5169	--	--	1.517
	MST-7	MST-8	150.0	0.014	--	1.0000	--	0.5169	--	--	1.517

(1) Obstruction losses in the main steam tunnel from grating, ductwork, and piping were modeled as sudden contractions and expansions. These losses are included in the Contraction and Expansion columns.

(a) This vent path consists of eight blowout panels. One of the eight blowout panels is assumed to fail closed under the single-failure criteria. The above values apply after the seven panels are fully open. The blowout panels are fully open 0.3 sec after a 0.9 psid force is applied across it. The panel opens in the direction shown in the top listing of the vent path.

NOTE: The homogeneous equilibrium model (HEM) for choked flow is used for all breaks and nodalizations. Appendix 6B: THREED Subcompartment Analytical Model gives a description of the HEM used in the SWEC computer code THREED.

TABLE 3B-8
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 4-IN RCIC DER IN VOLUME 175-1 BREAK NUMBERS 1 AND 2

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
0.00000	236.2	2.813E+05
0.05320	236.2	2.813E+05
0.05321	275.6	3.282E+05
0.05360	275.6	3.282E+05
0.05361	118.1	1.406E+05
0.06850	118.1	1.406E+05
0.06851	157.5	1.876E+05
15.23000	157.5	1.876E+05
17.50000	0	0
100.00000	0	0

TABLE 3B-9
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 4-IN RCIC DER IN VOLUME 175-2 BREAK NUMBERS 3 AND 4

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
0.00000	236.2	2.813E+05
0.04280	236.2	2.813E+05
0.04281	275.6	3.282E+05
0.07890	275.6	3.282E+05
0.07891	315.0	3.751E+05
0.07950	315.0	3.751E+05
0.07951	157.5	1.876E+05
15.23000	157.5	1.876E+05
17.50000	0.0	0
100.00000	0.0	0

TABLE 3B-10
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 10-IN RCIC DER IN VOLUME 215-1 BREAK NUMBERS 5 AND 6

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
0.00000	1562	1.860E+06
0.00830	1562	1.860E+06
0.00831	938.3	1.117E+06
0.22400	938.3	1.117E+06
0.22401	1199	1.427E+06
0.42150	1199	1.427E+06
0.42151	1041	1.240E+06
2.50000	1041	1.240E+06
17.50000	0	0
100.00000	0	0

TABLE 3B-11
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR AN 8-IN RWCU DER UPSTREAM OF THE RWCU RECIRCULATION PUMPS IN VOLUMES 215-1 AND 215-5 BREAK NUMBERS 7 AND 12

UPSTREAM FLOW			DOWNSTREAM FLOW		
Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)	Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.000	2426	1.293E+06	0.000	1996	1.064E+06
0.453	2426	1.293E+06	0.251	1996	1.064E+06
0.454	1154	6.150E+05	0.252	351	1.870E+05
11.467	1154	6.150E+05	11.932	351	1.870E+05
16.500	0	0	11.933	351	1.293E+05
100.000	0	0	17.630	351	1.293E+05
			17.631	351	7.424E+04
			25.322	351	7.424E+04
			25.333	0	0
			100.000	0	0

TABLE 3B-12
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 4-IN RWCU DER UPSTREAM OF THE RWCU RECIRCULATION PUMPS IN VOLUMES 215-9 AND 215-10 BREAK NUMBERS 14 AND 16

UPSTREAM FLOW			DOWNSTREAM FLOW		
Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)	Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.000	611	3.256E+05	0.000	611	3.256E+05
0.115	611	3.256E+05	0.115	611	3.256E+05
0.116	733	3.906E+05	0.116	421	2.244E+05
16.500	733	3.906E+05	16.500	421	2.244E+05
16.501	611	3.256E+05	16.501	351	1.870E+05
18.955	611	3.256E+05	28.181	351	1.870E+05
18.956	0	0	28.182	351	1.294E+05
100.000	0	0	33.879	351	1.294E+05
			33.880	351	7.424E+04
			41.571	351	7.424E+04
			41.572	0	0
			100.000	0	0

TABLE 3B-13
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR AN 8-IN RWCU DER UPSTREAM OF THE RWCU RECIRCULATION PUMPS IN VOLUMES 240-1 AND 240-2 BREAK NUMBERS 18 AND 21

UPSTREAM FLOW			DOWNSTREAM FLOW		
Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)	Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.000	2426	1.293E+06	0.000	2426	1.293E+06
0.330	2426	1.293E+06	0.330	2426	1.293E+06
0.331	1154	6.150E+05	0.331	351	1.870E+05
11.467	1154	6.150E+05	12.011	351	1.870E+05
16.500	0	0	12.012	351	1.294E+05
100.000	0	0	17.709	351	1.294E+05
			17.710	351	7.424E+04
			25.401	351	7.424E+04
			25.402	0	0
			100.000	0	0

TABLE 3B-14
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR AN 8-IN RWCU DER IN
VOLUMES 261-1, 261-2, 289-1, 306-1, AND 306-2
BREAK NUMBERS 23, 24, 26, 28, AND 29

UPSTREAM FLOW			DOWNSTREAM FLOW		
Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)	Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.000	2157	1.149E+06	0.000	2157	1.149E+06
0.719	2157	1.149E+06	0.719	2157	1.149E+06
0.720	1262	6.725E+05	0.720	1449	6.184E+05
1.511	1262	6.725E+05	3.963	1449	6.184E+05
1.512	928	4.945E+05	3.964	1449	5.341E+05
16.500	928	4.945E+05	5.343	1449	5.341E+05
16.511	0	0	5.344	1449	4.104E+05
100.000	0	0	6.723	1449	4.104E+05
			6.724	1449	3.065E+05
			8.586	1449	3.065E+05
			8.587	0	0
			100.000	0	0

TABLE 3B-15
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 10-IN RWCU DER IN
VOLUMES 261-5, 289-2, 306-2, 306-3, AND 306-4
BREAK NUMBERS 25, 27, 30, 31, 32, AND 33

UPSTREAM FLOW			DOWNSTREAM FLOW		
Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)	Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.000	2303	9.829E+05	0.000	2303	9.829E+05
0.673	2303	9.829E+05	0.673	2303	9.829E+05
0.674	2157	1.149E+06	0.674	1449	6.184E+05
2.110	2157	1.149E+06	1.156	1449	6.184E+05
2.111	1262	6.725E+05	1.157	0	0
2.902	1262	6.725E+05	100.000	0	0
2.903	928	4.945E+05			
16.500	928	4.945E+05			
16.501	1449	6.184E+05			
17.121	1449	6.184E+05			
17.122	1449	5.341E+05			
18.501	1449	5.341E+05			
18.502	1449	4.104E+05			
19.881	1449	4.104E+05			
19.882	1449	3.065E+05			
21.744	1449	3.065E+05			
21.745	0	0			
100.000	0	0			

TABLE 3B-16
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 6-IN RWCU DER IN
VOLUMES 328-2 AND SC328190
BREAK NUMBERS 34 AND 35

UPSTREAM FLOW			DOWNSTREAM FLOW		
Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)	Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.000	0	0	0.000	0	0
13.400	0	0	81.600	0	0
13.401	400	2.132E+05	81.601	179.2	7.650E+04
61.000	400	2.132E+05	144.900	0	0
115.700	0	0	200.000	0	0
200.000	0	0			

TABLE 3B-17
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 28-IN MAIN STEAM LINE DER WITH A TWO-PHASE BLOWDOWN IN VOLUME MST-5 BREAK NUMBER 2

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (BTU/sec)
0.0000	11458	1.3635E+07
0.0200	11458	1.3635E+07
0.0201	11753	1.3985E+07
0.1910	11753	1.3985E+07
0.1911	8032	9.557E+06
0.9999	8032	9.557E+06
1.0000	22205	1.4350E+07
4.9692	22205	1.4350E+07
5.5000	0	0
15.0000	0	0

TABLE 3B-18
(Sheet 1 of 1)

MASS AND ENERGY RELEASE RATES FOR A 28-IN MAIN STEAM LINE DER WITH A TWO-PHASE BLOWDOWN IN VOLUME MST-3 BREAK NUMBER 4

Time (sec)	Total Mass Flow Rate (lbm/sec)	Total Enthalpy Flow Rate (Btu/sec)
0.000	11458	1.3635E+07
0.0799	11458	1.3635E+07
0.0800	11753	1.3985E+07
0.1270	11753	1.3985E+07
0.1271	8032	9.557E+06
0.9999	8032	9.557E+06
1.0000	22205	1.4350E+07
4.9692	22205	1.4350E+07
5.5000	0	0
15.0000	0	0

APPENDIX 3C

FAILURE MODE ANALYSIS FOR PIPE BREAKS AND CRACKS

APPENDIX 3C

TABLE OF CONTENTS

APPENDIX 3C

TABLE OF CONTENTS (Cont’d)

APPENDIX 3C

FAILURE MODE ANALYSIS FOR PIPE BREAKS AND CRACKS

3C.1 INTRODUCTION

This appendix describes the specific pipe failure protection provided to satisfy the requirements of Section 3.6A.1 and demonstrates that the essential systems, components, and equipment are not adversely affected by pipe breaks or cracks.

The information provided by this appendix is in four sections: 3C.2, a discussion of high-energy pipe breaks and the effects of pipe whip; 3C.3, a discussion of the effects of jet impingement; 3C.4, a discussion of moderate-energy pipe cracks and the effects of spraying; and 3C.5, a discussion of flooding as a result of breaks or cracks.

Subcompartment pressurization is discussed in detail in Appendix 3B.

This appendix does not address the specific protection of field-routed essential instrument tubing or electrical conduit. However, these items are protected in accordance with the requirements of Section 3.6A.1.

For a detailed discussion of break/crack locations and types, break exclusion areas, guard pipes, and whip restraints that are frequently mentioned in this appendix, refer to Sections 3.6A.1 and 3.6A.2.

3C.2 HIGH-ENERGY PIPE BREAKS AND EFFECTS OF PIPE WHIP

The following systems are described in the noted sections:

Main steam piping system	3C.2.1
Feedwater piping system	3C.2.2
Reactor recirculation	3C.2.3
RCIC system	3C.2.4
LPCS/HPCS system	3C.2.5
RHR system (LPCI mode)	3C.2.6
RHR system (shutdown mode)	3C.2.7
RWCU system	3C.2.8
RPV vent line/RCIC head spray	3C.2.9

Each section references appropriate isometric drawings with break location and restraints. In addition, composite drawings showing pipe/equipment/room configurations are provided in Figures 3.6A-52 through 3.6A-60 but are not specifically referenced.

The only pipe breaks of concern in the non-Category I turbine building were those having a potential to impact safety-related equipment in adjacent buildings. Although the reactor protection sensors for turbine stop and control valve scram initiation are located in the turbine building, they are not considered essential in the evaluation of low probability events such as pipe breaks. Other scram signals, not related to line breaks in the turbine building, are available as backup.

3C.2.1 Main Steam Piping System (MSS)

Each of the four main steam lines (MSLs) is welded to the appropriate reactor nozzle (el 321 ft 11 7/8 in) above the top of the shield wall. After the first elbow, each line runs downward to approximate el 293 ft 9 in, and then horizontally to the third elbow where it runs downward to approximate el 251 ft 6 in. It then passes through the containment penetrations, the reactor building steam tunnel, and into the turbine building.

The break exclusion zone, as described in Section 3.6A.2.1.5, High-Energy Fluid Systems, Item 2, starts inboard of the zero-gap restraint adjacent to the inboard isolation valve and extends just beyond the jet impingement wall of the reactor building steam tunnel which contains the outboard zero-gap restraint.

A total of 18 safety/relief valves (SRVs) are mounted on the horizontal runs between the reactor and the first isolation valves inside the containment. The discharge piping from these valves is normally unpressurized; therefore, there is no potential for pipe whip or similar hazards. (Pipe whip prevention of the relief valve discharge lines is discussed in Section 3C.2.10.)

In addition, a 10-in line branching from MSL 2MSS-026-44-1 supplies steam to the reactor core isolation cooling (RCIC) turbine and to the residual heat removal (RHR) heat exchanger. This line passes through the RCIC pipe chase and is discussed in the analysis of the RCIC system.

<u>Inside Containment</u>

The main steam piping, if allowed to whip, can impact targets such as the 12-in LPCI piping, feedwater piping, structural steel at various elevations, the containment liner, and the biological shield wall (BSW).

To preclude any likelihood of loss of an essential system, or structural integrity of the containment or Category I walls, a total of 31 restraints have been installed inside the containment for the MSS system. An additional 5 restraints have been installed on the main steam drain piping as discussed in Section 3C.2.11.3.

For example, low-pressure coolant injection (LPCI) line 2RHS-012-8-1 is protected by restraints 2MSS*PRR021A, 2MSS*PRR024, and 2MSS*PRR031A. LPCI line 2RHS-012-125-1 is protected by restraints 2MSS*PRR004 and 2MSS*PRR016.

Restraints 2MSS*PRR005, 2MSS*PRR006, 2MSS*PRR017, 2MSS*PRR018, 2MSS*PRR025, 2MSS*PRR026, 2MSS*PRR036, and 2MSS*PRR037 protect the MSIV for breaks inside containment and ensure that stress allowables are within the limits defined for the break exclusion region.

Inside Steam Tunnels

The main steam piping, from the zero-gap restraints inboard of the first isolation valve (inside the containment) to and including zero-gap restraints 2MSS*PRR102, 2MSS*PRR112, 2MSS*PRR122 and 2MSS*PRR132 at the jet impingement wall, meets the stress criteria for no postulated breaks, as discussed in Section 3.6A, and, therefore, is defined as a break exclusion zone.

From the jet wall, the four 28-in MSLs (north inner and outer loops, south inner and outer loops) run straight for approximately 8 ft and then run vertically upward to el 295 ft 1 in before making another turn above the 12-line wall, where they connect to the main steam headers.

Pipe whip of the MSLs in the steam tunnel area has been precluded by the placement of 20 restraints.

Turbine Building

There are no breaks postulated in the turbine building because there are no essential targets in the turbine building.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the MSLs can cause additional damage that could impair the ability to safely shut down the reactor, or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.2 Feedwater Piping System (FWS)

The two feedwater loops each consist of three 12-in discharge lines that connect to the RPV at el 309 ft 1 in. The lines pass through the BSW openings and drop down to the 24-in/18-in/12-in header line at el 292 ft 8 in. The header lines then drop down to el 257 ft 0 in and penetrate the primary containment into the main steam tunnel. The lines run straight through the tunnel, then turn up and loop to the turbine building.

The feedwater piping, from the zero-gap restraints inboard of the first isolation valve (inside containment) to and including zero-gap restraints at the jet impingement wall, meets the criteria for no postulated breaks, as discussed in Section 3.6A, and, therefore, is defined as a break exclusion zone.

Inside Containment

The feedwater piping, if allowed to whip, can impact targets such as the 12-in LPCI piping, high-pressure core spray (HPCS) piping, structural steel at various elevations, the containment liner, and the BSW.

To preclude any likelihood of loss of an essential system or structural integrity of the containment, restraints have been installed inside the containment for the FWS system.

For example, LPCI line 2RHS-012-125-1 is protected by restraints 2FWS*PRR010, 2FWS*PRR010A, 2FWS*PRR015, 2FWS*PRR016, and 2FWS*PRR014. The HPCS piping is protected by restraint 2FWS*PRR001.

Restraints 2FWS*PRR017, 018, 035, and 036 protect the FWS check valves (2FWS*V12A and V12B) for breaks inside the containment and ensure that the stresses in the break exclusion region are within allowable limits.

Main Steam Tunnel

The break exclusion zone starts inboard of the zero-gap restraint adjacent to the inboard check valve and extends just beyond the jet impingement wall of the reactor building steam tunnel which contains the outboard zero-gap restraint. These zero-gap restraints also are used to prevent overstressing the penetration or disabling the outboard isolation valves (2FWS*V23A and B) following postulated breaks.

From the jet wall, the two 24-in feedwater lines run straight until passing to the west side of 12-line wall. Pipe whip of the feedwater lines between 11-line and 12-line walls has been precluded by the placement of restraints 2FWS*PRR112, 2FWS*PRR113, 2FWS*PRR114, and 2FWS*PRR117.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the feedwater lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.3 Reactor Recirculation System (RCS)

The force results, pipe whip restraints, break locations, and types are the responsibility of General Electric Company (GE) and are described in Section 3.6B.

The two recirculation loops consist of a 24-in suction line that exits the reactor pressure vessel (RPV) at el 282 ft 4 3/8 in. The line penetrates the BSW opening (which contains a flow diverter to minimize annulus pressurization [AP]) and drops to el 249 ft 5 3/4 in. The lines then run horizontally to the RCS pump. On the discharge side of the pump the lines run horizontally at el 253 ft 4 3/4 in and turn up at 90 deg and 270 deg. At el 275 ft 0 7/8 in, 16-in headers loop around the RPV, and 12-in risers branch off the headers at 30-deg increments and discharge into the RPV.

The recirculation piping, if allowed to whip, can impact targets such as structural steel platforms, SRV lines, and the RCS pump supports.

To preclude any likelihood of loss of an essential system or structural integrity of the containment, restraints have been installed for the recirculation system. Additionally, the BSW doors have been designed for pipe rupture loads.

For example, pipe whip due to nozzle breaks on the 12-in discharge lines will be prevented by the GE strap-type restraints just after the first elbow. The BSW doors also will be impacted as the pipe whips away from the RPV and will provide additional control.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the RCS lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.4 Reactor Core Isolation Cooling System (RCIC)

The RCIC 10-in line branches off the MSL (2MSS-026-44-1) at el 302 ft 2 in. The line runs through the primary containment and drops to el 263 ft 7 1/2 in, passing through its penetration at Azimuth 185°. Once outside the primary containment, the line

drops into the RCIC pipe chase. At el 234 ft there is an interconnection between the RCIC and RHR systems, where the RCIC line reduces in size from 10 in to 4 in and leads into the RCIC turbine room at el 175 ft. The high-energy portion of the line stops at normally closed valve 2ICS*MOV120. The RCIC head spray 6-in line attaching to the RPV head is high energy up to normally closed valve 2ICS*V157 (Section 3C.2.9).

Inside Containment

The RCIC piping, if allowed to whip, can impact targets such as the containment liner, LPCS system, and structural steel.

To preclude any likelihood of loss of an essential system or structural integrity of the containment, restraints have been installed inside the containment for the RCIC system.

For example, pipe whip due to a break at the connection to the MSL will be prevented by 2ICS*PRR001.

Outside Containment

To preclude target impacts and pipe whip of the RCIC piping outside containment, restraints 2ICS*PRR007, 2ICS*PRR008, 2ICS*PRR009, 2ICS*PRR010, and 2ICS*PRR012 have been installed.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the RCIC lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.5 LPCS/HPCS System

3C.2.5.1 Low-Pressure Core Spray (LPCS)

The LPCS system description is detailed in Section 6.3.2.2.3.

The LPCS 10-in line is welded to the reactor nozzle (el 308) and passes through the BSW. In the primary containment the line expands to 12 in and runs to 2CSL*V101, which is the end of the high-energy portion of the line. After the valve, the line drops to el 295 ft where it passes through its primary containment penetration.

The LPCS piping, if allowed to whip, can impact targets such as the containment liner and BSW. To preclude any likelihood of loss of structural integrity of the containment, a total of three restraints are installed for the LPCS system.

3C.2.5.2 High-Pressure Core Spray (HPCS)

The HPCS system description is detailed in Section 6.3.2.2.1.

The HPCS 10-in line is welded to the reactor nozzle (el 307 ft 11 3/16 in) and passes through the BSW. In the primary containment the line expands to 12 in and runs to 2CSH*V108, which is the end of the high-energy portion of the line. After the valve, the line drops to el 291 ft 11 11/16 in where it passes through its primary containment penetration.

The HPCS piping, if allowed to whip, can impact targets such as the containment liner and BSW. To preclude any likelihood of loss of structural integrity of the containment, a total of five restraints are installed inside containment for HPCS.

For example, pipe whip due to a break at 2CSH*V108 will be prevented by restraint 2CSH*PRR001 and 2CSH*PRR002; 2CSH*PRR003A at the BSW will provide additional control.

<u>Conclusions</u>

Using very conservative assumptions and criteria, no postulated failure of the LPCS/HPCS lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.6 Residual Heat Removal System (RHR-LPCI Mode)

The LPCI subsystem is an operating mode of the RHR system. The system description is detailed in Section 6.3.2.2.4.

The three high-energy lines for the LPCI mode are 12-in lines welded to the reactor nozzles at el 299 ft 0 3/8 in. The lines pass through the BSW openings and loop to their isolation valves, which end the high-energy portion of this system. After the valves, the lines drop to their primary containment penetrations.

The RHR, if allowed to whip, can impact targets such as the containment liner, BSW, and star truss. To preclude any likelihood of loss of structural integrity of the containment, restraints are installed for the LPCI mode of the RHR system.

3C.2.7 Residual Heat Removal (Shutdown Mode)

The three high-energy portions of this system branch off the reactor recirculation piping at el 266 ft 10 7/8 in and el 271 ft 1 7/8 in and run to their isolation valve inside the primary containment. The suction line is 20 in, and the two discharge lines are 12 in. After isolation valves 2RHS*V39A, 2RHS*V39B, and 2RHS*MOV112, these lines drop to el 249 ft 6 in and el 247 ft where they penetrate the primary containment wall.

To preclude target impacts and pipe whip of the RHR piping, six restraints, two per line, have been installed. The two discharge RHR lines have jet impingement source shields around the first elbow after they branch off the RCS pipes. These source shields protect the control rod drive (CRD) from jet impingement loads (Section 3C.3).

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the RHR lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.8 Reactor Water Cleanup System (RWCU)

This system has three suction lines: two 4-in lines branching from the recirculation piping and a 2-in line from the RPV bottom head. The 2-in line expands to 2.5 in and passes through the reactor pedestal at el 263 ft; this line expands again to 4 in inside the primary containment. The three 4-in suction lines lead into an 8-in header line which passes through the primary containment wall at Azimuth 185°. Outside containment the RWCU lines branch into various process equipment and discharge into the feedwater piping.

Inside Containment

The RWCU piping, if allowed to whip, can impact targets such as reactor pedestal, CRD, and main steam SRV piping.

To preclude any likelihood of loss of an essential system or structural integrity of containment, restraints have been installed inside the containment for the RWCU system.

Outside Containment

Most breaks in the RWCU outside containment are isolated with nonsafety-related targets such as walls, ceilings, floors, and the failed piping system itself. Restraints are provided to establish a break exclusion zone and protect the floor at el 240 ft. A study was done to determine the pipe whip and jet impingement consequences for a hypothetical break at the terminal end of the RWCU line connection to the feedwater thermal tee. The conclusions of the study are discussed below:

1. The RWCU pipe would impact a maintenance platform at el 251 ft. This is acceptable, since the platform's failure causes no secondary damage to other safety-related equipment and does not impair plant shutdown.

2. The main steam penetration Z1A and the MSL (spool piece 2MSS-026-154-1) would be struck by the whipping RWCU pipe. In accordance with Section 3.6A.1.1., this will not result in the rupture of the MSL or penetration, since the whipping pipe is smaller in diameter and thickness than the pipes it strikes. Additionally, the function of the outside main steam isolation valve (MSIV) will not be impaired.

3. The feedwater line is the only jet impingement target affected. Stresses resulting in the feedwater line are acceptable and within allowable limits. In addition, the feedwater isolation valves, penetration, and jet impingement wall will not be overloaded due to the effects of jet impingement. Since the stresses induced by the jet are less than the design allowable, this condition is acceptable.

In summary, plant safety would not be jeopardized for the hypothetical RWCU break. No additional pipe whip restraints are required in this area.

<u>Conclusions</u>

Using very conservative assumptions and criteria, no postulated failure of the RWCU lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.9 RPV Vent Line/RCIC Head Spray

3C.2.9.1 RCIC Head Spray

The RCIC head spray (2ICS-006-67-1) is a 6-in line which originates in the secondary containment from the reactor core isolation cooling system (ICS) pumps. The pipe runs through the primary containment at el 292 ft 0 in through penetration Z-22. Once inside the primary containment it rises to el 330 ft 7/8 in after penetrating through the refueling bulkhead. From that elevation the pipe continues as 2ICS-006-33-1. This pipe continues to rise and has valve 2ICS*V157 mounted on it at el 338 ft 9 1/16 in and finally leads into the RPV head. The section of the pipe attaching the RPV head to normally closed valve 2ICS*V157 is considered a high-energy line.

The ICS piping, if allowed to whip, can impact targets such as the drywell head and the insulation support frame.

To preclude any likelihood of loss of an essential system or structural integrity of the drywell head, structural steel of the insulation support framing is erected and functions as a barrier between the ruptured pipe and the drywell head.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the ICS line can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation beyond the limits of 10CFR100.

3C.2.9.2 RPV Steam Vent Line

The main steam 2-in vent line branches off the MSL (2MSS-026-43-1) at el 314 ft 10 in. It then loops its way up to el 342 ft 8 11/16 in and drops vertically into the RPV head.

The MSS piping, if allowed to whip, could impact targets such as drywell head, vessel dome, insulation support structure, MSL 2MSS-026-43-1, and rupture restraint 2RHS*PRR014. Since the piping is a 2-in diameter line, the impacting forces of the piping system are relatively small and will not cause detrimental results to the targets mentioned above. The pipe whip due to a break in this line will not cause any loss of structural integrity to the targets mentioned above.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the MSS vent line can cause damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.10 Main Steam Safety Relief Valve Piping (SVV)

The SVV piping for each of the 18 SRVs consists of an 8-in connection off the MSL to the SRV and a 10-in discharge line from the SRV to the suppression pool. The high-energy portion of the system is the 1 ft length for the connection from main steam pipe to the normally closed SRV. Two breaks are postulated for each SRV; they are terminal point, circumferential breaks for the 8-in connection.

Targets from pipe whip are the 12-in residual heat removal system (RHS) lines. Impact analysis shows failure of SVV piping cannot cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation beyond the limits of 10CFR100.

3C.2.11 3-Inch and Smaller High-Energy Piping

Included in 3-in and smaller high-energy piping are the main steam drains, standby liquid control (SLC) system, and the CRD.

3C.2.11.1 Control Rod Drive System (CRD)

Inside the primary containment this system has no postulated breaks from the penetration to the reactor interface, since all the lines are less than or equal to 1-in diameter pipe size.

3C.2.11.2 Standby Liquid Control System (SLC)

The SLC 2-in line branches off the HPCS line in the primary containment at el 307 ft 11 3/16 in.

The high-energy portion of this system runs straight to normally closed check valve 2SLS*V10.

There is no pipe whip associated with the circumferential breaks in this straight section of 2-in piping; therefore, there are no targets or protection requirements.

3C.2.11.3 Main Steam Drain Line

The main steam drain piping inside containment tap off the four 26-in MSLs. The 10-in ICS line is normally pressurized and ASME Code Class 1.

The four 2-in drain lines branching out from the MSLs connect into the 6-in horizontal line at el 250 ft. This 6-in line (2MSS-006-150-1) runs through the primary containment at el 246 ft 3 in passing through its penetration at Z-2.

The main steam drain piping, if allowed to whip, can impact targets such as the platforms at el 247 ft 6 in, and el 261 ft 0 in, and various structural steel.

To preclude any likelihood of loss of an essential system or structural integrity, restraints 2MSS*PRR201, PRR 202, PRR204A, PRR204B, and PRR203 are installed for the system.

<u>Conclusions</u>

Using very conservative assumptions and criteria, no postulated failure of the main steam drain lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.3 EFFECTS OF JET IMPINGEMENT

3C.3.1 Jet Impingement Analysis

3C.3.1.1 Introduction

To ensure the integrity of structures, systems, and components whose failures could impact safety and which are potential targets of jet impingement, it is necessary to calculate the jet intensity and the impingement loadings on targets. The analytical tools required for making these estimates are provided

in this section. Jet impingement loadings are determined as follows:

1. The jet force at the exit plane of a pipe break is calculated as discussed in Section 3.6A.2.2.2. This jet force is dependent on the fluid condition in the system, which varies with time. For jet impingement analysis, only the peak force is used unless a complete jet time-history is required to reduce conservatism. A rise time of 1 msec is used.

2. The jet expands as it travels along its path. The jet shape is assumed to be conical at a 10-deg half-angle expansion for subcooled water. Moody's asymptotic expansion model[1] is adopted for saturated water and saturated steam (Figure 3C.3-1).

3. The impinging jet proceeds along a straight path that is normal to the break area.

    a. Circumferential breaks result in pipe severance, the break area is circular in shape, and the centerline of the jet is coincident with the pipe centerline (Figure 3C.3-1).

    b. Longitudinal breaks result in axial split without pipe severance. The break area is circular in shape and equal to the effective cross-sectional flow area of the pipe at the break location. The jet centerline is normal to the opening and the pipe centerline (Figure 3C.3-1).

4. If the ruptured pipe is not restrained, the path of jet impingement is defined by the trajectory of the whipping pipe.

5. The total jet force on any cross section is assumed distance-invariant, with a total magnitude equivalent to the jet force. The jet pressure is uniformly distributed across the cross section of the jet.

3C.3.1.2 Jet Force

The maximum value of the jet pressure from a pipe break can be expressed as:

$$P_J = C_J P_O$$

Where:

P_O = Fluid pressure inside pipe

C_J = Jet coefficient = 1.26 for steam and saturated water[1]

= 2.0 for subcooled, nonflashing water

(See Section 3.6A.2.2.3 for jet pressure discussion).

Total jet force is:

$$F_J = P_J A$$

Where:

A = Pipe break area = $\frac{\pi}{4} Di^2$

D_i = Inside diameter of pipe

It is assumed that the total jet force remains constant throughout its traveling distance and the jet force is uniformly distributed across the cross-sectional area of the jet stream. Thus, the jet intensity at a distance L from the pipe break location is:

$$P_L = F_J/A_L$$

Where:

A_L is a cross-sectional area of the jet stream at L.

As the jet propagates away from the pipe break area, it expands due to the expansion of the fluid from a higher exit pressure to ambient pressure. Therefore, the jet intensity decreases with distance from the break location, whereas the total jet force is assumed to remain constant throughout its traveling distance.

3C.3.1.3 Target Impingement Load

The impingement load on a target is defined as the normal force exerted on a target surface due to a jet hitting this surface and being turned or diverted to a different direction. This load can be calculated based on the average jet intensity established in Section 3C.3.1.2 by the equation:

$$F_i = C_s \; P_1 \; A_t$$

Where:

C_s = Shape factor

A_t = Projected impingement area of the target

Values of C_s and A_t depend on the target geometry and orientation with respect to the jet stream.

For a target with a flat surface area normal to the center of the jet stream, the impingement load is the product of the pressure and the intercepted jet area. For cases where the target area is such that the intercepted jet stream is deflected rather than totally stopped, a shape factor which is a function of the target geometry is used to calculate the total jet impingement load. Shape factors are calculated according to the method described in ANSI/ANS-58.2-1980[2].

3C.3.1.4 Target Evaluation

All potential targets in jet paths are identified and examined for jet impingement effects. Targets are classified in five categories:

1. Structural targets.

2. Fluid piping targets.

3. Control system and instrumentation targets.

4. Electrical system targets.

5. Equipment targets.

All essential targets are identified for further evaluation. Nonessential targets in the jet paths are excluded from further consideration unless jet damage to them could initiate or escalate failure of an essential target.

Jet impingement loadings on affected targets are evaluated to determine whether:

1. Structural integrity or operability, if required, can be demonstrated.

2. Loss of function is acceptable, considering all target damages due to each jet in conjunction with the loss of offsite power (LOOP) and the postulated worst single active failure.

3C.3.1.5 Target Protection

If required, jet impingement shields may consist of two types:

1. Target Shield

   The target shield is a flat plate at an angle to the jet or a pair of plates assembled in a wedge shape (Figure 3C.3-2). This shape provides a shape factor which reduces the effective jet intensity on the shield. The shield geometry, size, and location are such that the target does not directly intercept any of the postulated jets.

2. Source Shield

The source shield is a removable casing which forms an annulus about the process pipe in the region of a postulated longitudinal pipe break location (Figure 3C.3-3). When the pipe ruptures, the source shield redirects the escaping fluid jet parallel to the piping axis.

3C.3.1.6 Nomenclature

A = Pipe break area.

A_L = Jet cross section area at a distance L from the pipe break location.

A_t = Projected impingement area on target.

C_j = Jet coefficient.

C_s = Shape factor.

D_i = Inside diameter of pipe.

D_∞ = Diameter of asymptotic jet cross section.

F_i = Jet impingement load on target.

F_j = Jet force.

L = Distance, length.

L_2 = Total length of Regions 1 and 2 of asymptotic jet.

P_J = Maximum value of initial jet pressure.

P_L = Jet intensity at a distance L from pipe break location.

P_o = System pressure prior to pipe break.

3C.3.1.7 References

1. Moody, F. J. Prediction of Blowdown Thrust and Jet Forces. ASME Paper No. 69-HT-31, August 6, 1969.

2. ANSI/ANS-58.2-1980, Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Ruptures.

3C.3.2 High-Energy Pipe Breaks and the Effects of Jet Impingement

The following systems are evaluated in the noted sections:

Main Steam Piping System	3C.2.1
Feedwater Piping System	3C.2.2
Reactor Recirculation System	3C.2.3
RCIC System	3C.2.4
HPCS/LPCS System	3C.2.5
RHR System - (LPCI Mode)	3C.2.6
RHR System (Shutdown Mode)	3C.2.7
RWCU System	3C.2.8
RCIC Head Spray - Main Steam Safety Relief Valve Piping (Between Main Steam Line and First Relief Valve)	3C.2.10
3-In and Smaller High-Energy Piping	3C.2.11

The above systems have been described and evaluated for the effects of pipe breaks in the corresponding 3C.2 sections. In many cases, pipe whip restraints have been designed to minimize the effects of pipe whip and resulting jet impingement.

This section evaluates the effects of jet impingement considering single active failure, LOOP (if loss of normal onsite power is a direct consequence of the event), and other appropriate assumptions, criteria, etc., as explained in Section 3.6A.1.1.

For each target, a section is referenced to explain the protective measure. In cases where more than one target can fail, the protective measure reference will address the complete event.

In certain cases, protective measures can be grouped to apply to all events. The following deal with protective measures common throughout this section.

3C.3.2.1 Jet Impingement Targets and Protection Measures

<u>Targets Designed to Maintain Containment Integrity</u>

The primary containment, drywell floor, and drywell head have been designed for applicable jet loads, including the effects of temperature. These structures are required to maintain containment integrity following a LOCA inside containment.

Targets Designed to Prevent Generation of Missiles

Items located in the primary containment, although not directly essential for safe shutdown and containment integrity, have been designed for jet loads since their failure could lead to unacceptable consequences. These targets are as follows:

Structural Steel - All structural steel identified as targets has been designed for jet loads since it supports piping and equipment essential for safe shutdown.

Pipe Rupture Restraints - Pipe rupture restraints have been designed for jet loads primarily because of their size and weight and proximity to other high-energy piping systems. In cases where the jet load acts on the restraint that is preventing pipe motion due to the same break, the two loads are considered concurrently.

Biological Shield Wall and Pedestal - The BSW and pedestal have been designed for jet loads to ensure overall integrity of containment items that are being supported. Of particular significance are the RPV nozzle breaks, where AP, jet impingement, and pipe whip are considered concurrently on the BSW doors, BSW, and RPV. For these events, the BSW and BSW doors are designed for the load. The star truss has also been considered.

Other items such as the primary containment instrument air tanks and unit coolers have been designed such that they will not become missiles. They are of sufficient mass that they could adversely affect essential equipment if they were to fly free.

Reactor Pressure Vessel

The RPV has been evaluated for jet impingement loads primarily due to the nozzle breaks. Similar to the event where AP and pipe whip are considered concurrently (Section 3C.3.b), GE has shown that the RPV will withstand the loads. Additional details are in Section 3.9B.

Nonessential Piping

Piping such as closed loop cooling, service air, breathing air, containment purge, instrument air, fire protection, reactor building drains, equipment drains, and containment leakage and atmosphere monitoring systems are not required for safe shutdown. Consideration is given to maintaining containment integrity to ensure that offsite dose limits are not exceeded.

Primary Containment Ventilation

The entire primary containment ventilation system is contained inside the containment. Therefore, containment integrity is not a concern. The ductwork is composed of light sheet metal that is

not of sufficient mass to be considered a viable missile which could damage an essential target.

3C.4 HIGH- AND MODERATE-ENERGY PIPE CRACKS AND EFFECTS OF SPRAYING

3C.4.1 Discussion

The components and/or equipment required for safe shutdown of the reactor were evaluated for the effects of spraying from through-wall leakage cracks in high- and moderate-energy systems. The evaluation demonstrates that the plant can be safely shut down, assuming a concurrent single active failure in systems necessary to mitigate the consequences of the postulated piping failure. Where necessary, measures have been provided to protect and ensure component operability. Flooding effects from cracks in high- and moderate-energy systems are discussed in Section 3C.5.

Moderate-energy piping, as defined in Section 3.6A.2.1.2, includes piping systems where the maximum operating temperature is 200°F or less and maximum operating pressure is 275 psig or less. It also includes some systems that qualify as high-energy systems for short operational periods and moderate-energy for major operational periods.

Only high-energy piping is capable of producing breaks (Section 3.6A.2.1.1). Moderate-energy piping produces only through-wall leakage cracks.

The effects of high-energy cracks are enveloped by the moderate-energy pipe failures. The most limiting moderate-energy piping crack, i.e., RHR system, produces environmental conditions as severe as high-energy cracks.

The criteria used to define the location of cracks in moderate-energy systems outside containment are defined in Section 3.6A.2.1.5, and the criteria for calculating crack flow rates are given in Section 3.6A.2.1.6.

3C.4.2 Evaluation Procedure - Spraying Failure Modes and Effects Analysis (FMEA)

The evaluation was conducted in accordance with Nuclear Regulatory Commission (NRC) Branch Technical Position (BTP) ASB 3-1, which states that a leakage crack in moderate-energy piping is considered separately as a single, postulated initial event occurring during normal plant conditions. The essential equipment that must operate under these conditions is that required to bring the plant to a safe shutdown condition and maintain long-term cooling. In this case, the essential components, which have been located by fire zones, are the same as those identified in the Appendix R safe shutdown analysis (SSA) (Appendix 9B and Figures 9B.4-1 and 9B.4-2). The

evaluation of effects of spraying from moderate-energy cracks proceeded in all fire zones containing targets. Included in the evaluation were the reactor building, reactor building auxiliary bays, control and screenwell buildings, and piping and electrical tunnels. Excluded were the turbine, radwaste, normal switchgear, and auxiliary boiler buildings.

The following summary outlines the procedure used to evaluate spraying effects from moderate-energy cracks.

1. List by fire zone all components and/or equipment (targets) required for safe shutdown in all buildings.

2. Identify water sources in fire zones that contain potential targets (cracks are not postulated for spray evaluation in zones without targets).

3. If there is a water source in the zone, assume that all potential targets are sprayed. Evaluate all components and/or equipment to determine if they are waterproof (not susceptible to failure from spraying) and can withstand the effects of water temperature. Table 3C.4-1 shows the maximum spray temperatures in each building.

   If there is no water source in the zone, evaluate the susceptibility of the equipment to failure as the result of dripping water from other zones.

4. Assume the failure of all targets in the zone that are not waterproof and identify available paths for safe shutdown and maintenance of long-term cooling. Figures 9B.4-1 and 9B.4-2 depict the safe shutdown trains.

   If it is concluded through this evaluation that the plant could not be shut down safely, a more detailed approach is taken to determine if components are actually sprayed and rendered inoperable. Using this basis, a reexamination of paths for safe shutdown is then conducted.

5. The spraying evaluation is conducted in conjunction with a flooding evaluation (Section 3C.5). If a spray source in a given zone is large enough to cause potential flooding problems in the given zone (or other zones), failures from flooding are combined with failures from spraying to evaluate available safe shutdown equipment.

6. In addition to the direct consequences of pipe crack, a single active failure is assumed in those systems required to mitigate the consequences of the piping failure.

3C.4.3 Evaluation Guidelines - Spraying FMEA

The basic guidelines used to evaluate the effects of spraying were:

1. If a water pipe is within a fire zone, all targets within that zone are assumed to be sprayed. If this assumption yields unacceptable results, a more detailed review of spraying and component shielding is conducted.

2. All Class 1E valve motor operators and solenoid valves have NEMA 4 (or equivalent) enclosures and are not assumed to fail as the result of water spray.

3. All motors other than valve operators are evaluated on an individual basis. If the motors are in open dripproof enclosures, it is assumed that they will fail when exposed to a spray.

4. Unit cooler and fan motors are not assumed to fail since they are enclosed within the unit cooler housing or ductwork, which shields them from direct spraying.

5. Cables are waterproof and unaffected by water spray.

6. If subject to spray, junction and terminal boxes required for safe shutdown equipment have upgraded NEMA 12 (or equivalent) dripproof enclosures and are not assumed to fail as a result of direct water spray.

7. All QA Category I instruments have NEMA 4 (or equivalent) enclosures and are not affected by water spray. Control panels are assumed to fail as the result of water spray.

8. Motor control centers (MCCs), switchgear, and load centers are in dripproof enclosures, but are assumed to fail as a result of water spray.

9. If the postulated piping failure results in a reactor or turbine trip, LOOP is assumed.

10. Guidelines for single-failure evaluation are as follows:

    a. A single active failure is assumed in any one of the safe shutdown systems.

    b. Where the postulated piping failure is assumed to occur in one train of a dual-purpose, moderate-energy, safe shutdown system (e.g., safety-related RHR and service water are subsystems comprising such a safe shutdown

system), a single failure is not postulated in the redundant safety-related train of that system or subsystem.

3C.4.4 Analytical Methods

As described in the spraying evaluation procedure (Section 3C.4.2), all targets in a given zone were assumed to be sprayed by any water sources in the zone. Analytical calculations of spraying distance were not utilized in reevaluating problem areas. In these instances, shielding, moving equipment, and other modifications were considered.

3C.4.5 Results of Evaluation - Spraying FMEA

The following subsections present, building-by-building, the results of the spraying evaluation using the procedures and guidelines discussed in Sections 3C.4.2 and 3C.4.3.

The evaluation verifies that the plant can be safely shut down in the event of pipe cracks in fluid systems. As noted below, protective measures have been implemented to ensure the required system functional capability is maintained. A list of moderate-energy piping systems and system parameters is provided in Tables 3C.4-1 through 3C.4-6 for those buildings housing equipment required for safe shutdown.

3C.4.5.1 Reactor Building (Including Auxiliary Bays)

In the reactor building, spray sources include both safety-related and nonsafety-related systems. Components susceptible to failure from spray are motors and MCCs for RCIC, HPCS, spent fuel pool cooling and cleanup (SFC), RHR, and low-pressure core spray (LPCS) system pumps. A single spray source will not affect more than one of these pump motors. Failure of a RCIC, HPCS, SFC, or LPCS motor is acceptable; sufficient redundancy exists to maintain spent fuel pool cooling and to safely shut down the plant when considering an additional single active failure as described in Section 3C.4.3.10. The RHR pump motors (2RHS*P1A and P1B) are protected from spray as required to ensure safe shutdown of the plant.

MCCs for these pumps are protected from spray.

Junction boxes required to maintain safe shutdown capability are sprayproofed if subject to spraying conditions.

The spray sources which would fail these components will not fail the redundant trains by flooding (Section 3C.5).

3C.4.5.2 Control Building

The spray sources in the control building include chilled, service, domestic, and fire protection water systems (Table

3C.4-2). The spray-susceptible targets are the control panels, switchgear, ventilation systems, and pump motors. The control and relay rooms contain equipment serving all four trains for safe shutdown. These components are shielded from potential spraying, as required, to ensure availability of the system safe shutdown function when considering an additional single active failure as described in Section 3C.4.3.10. Junction boxes required to maintain safe shutdown capability are sprayproofed if subject to spraying conditions.

3C.4.5.3 Diesel Generator Building

The spray sources in the diesel generator building are fire protection, service water, and diesel generator fuel oil systems (Table 3C.4-3).

The diesel fuel pumps and air compressors are spray-susceptible targets. Failure of these motors by spraying is acceptable since this would only result in a loss of one of the two redundant trains. However, failure would not require immediate plant shutdown. Potential flooding from the spray source would not result in loss of the redundant trains of emergency power (Section 3C.5.5.4). Additionally, since the crack would not result in a reactor or turbine trip, offsite power would still be available.

3C.4.5.4 Piping Tunnels

The moderate-energy systems in the piping tunnels are water treatment, service water, component cooling water, floor drains, and turbine building equipment drains. There are no spray-susceptible targets in the pipe tunnel (Table 3C.4-4) that are required for safe shutdown.

3C.4.5.5 Electrical Tunnels

The only water source in the electrical tunnels is fire protection, and the electrical tunnel equipment rooms have fire protection and service water. There are no spray-susceptible targets in any of these areas (Table 3C.4-5).

3C.4.5.6 Screenwell Building

There are ten moderate-energy systems in the screenwell (Table 3C.4-6). The spray-susceptible targets are the service water system MCCs, junction boxes, and pumps located in the service water pump bays. These components are protected as required to ensure availability of the system safe shutdown function when considering an additional single active failure as described in Section 3C.4.3.10. Flooding from the postulated cracks will not affect the redundant trains (Section 3C.5.5.7). Junction boxes required to maintain safe shutdown capability are sprayproofed if subject to spraying conditions.

3C.4.5.7 Standby Gas Treatment Building

The equipment in the standby gas treatment (SGT) building is not needed for mitigation of a moderate-energy line crack or safe shutdown during normal plant operation. Therefore, a review of the equipment in this building is not necessary.

3C.5 COMPARTMENT FLOODING AS A RESULT OF BREAKS OR CRACKS

3C.5.1 Discussion

The components and/or equipment required for safe shutdown of the reactor were evaluated for the effects of flooding from through-wall leakage cracks in moderate-energy systems, breaks and cracks in high-energy lines, and failure of nonseismic tanks and vessels. The evaluation verifies that the plant can be safely shut down, assuming a concurrent single active failure in systems necessary to mitigate the consequences of the postulated component failure. Where necessary, measures have been provided to ensure component operability. Spraying effects from cracks in moderate-energy systems are discussed in Section 3C.4.

A detailed discussion of break/crack locations and types is provided in Sections 3.6A.1 and 3.6A.2.

As discussed in the following sections, flooding effects from high-energy pipe breaks and cracks outside of containment and failure of nonseismic tanks and vessels are enveloped by moderate-energy crack flooding. This is primarily due to rapid detection and isolation of high-energy pipe breaks and cracks based on automatic isolation on area high temperature.

The total mass released by high-energy pipe breaks is shown in Table 3C.5-1, and the capacity of nonseismic tanks and vessels inside buildings containing safe shutdown equipment is shown in Table 3C.5-2. Flooding effects from external water sources are discussed in Section 3.4.

3C.5.2 Evaluation Procedure - Flooding FMEA

The approach for the flooding evaluation was similar to the procedure described in Section 3C.4.2 for the spraying evaluation. The evaluation was conducted utilizing the lists of safe shutdown equipment located by fire zones defined in the Appendix 9B fire hazards analysis (FHA).

The following summary outlines the procedure used to evaluate flooding effects:

1. List by fire zone all components and/or equipment required for safe shutdown in all buildings (see Appendix 9B, Figures 9B.4-1 and 9B.4-2, and Table 9B.8-1).

2. Locate all safe shutdown targets by elevation.

3. Identify the hydraulic boundaries of each area to determine the extent of flooding. These were generally more extensive than the fire zones.

4. Identify flood sources and calculate either maximum mass released or limiting crack flow rate (Section 3C.5.4) from postulated water sources.

5. Determine flood levels within each hydraulic boundary based on either total mass released or balance of flow in/out of the boundary. In this determination no credit is taken initially for the normal plant drainage system.

6. Identify all safe shutdown targets which could possibly be submerged and rendered inoperable. Evaluate all components and/or equipment to determine if they are waterproof (not susceptible to failure from submergence) and can withstand the effects of the water temperature. Tables 3C.4-1 through 3C.4-6 show the maximum spray temperatures in each building.

7. Assume the failure of all targets in the hydraulic boundary that are determined to be below flood level and susceptible to failure. Identify the available paths to safe shutdown and maintenance of long-term cooling.

   If it was concluded through this evaluation that the plant could not be shut down safely, a more detailed evaluation, including consideration of the normal plant drainage systems and possible protective measures, was conducted.

8. In addition to the direct consequences of flooding, a single active failure is assumed in those systems required to mitigate the consequences of the piping failure.

9. Review drainage systems to ensure that leakage from one failed redundant train does not backflow through drains and flood the other train.

3C.5.3 Evaluation Guidelines - Flooding FMEA

The basic guidelines used to evaluate the effects of flooding were:

1. Within a given hydraulic boundary, the largest water source located anywhere in that boundary is used to calculate flood heights for all areas included. In many cases this leads to the largest water source being

used for flood calculations on all floors within a building.

2. Credit is taken for flood protection by doorways and penetrations only if the particular doorway or penetration is specified as watertight.

3. All motors, including valve motor operators and solenoids, are assumed to fail if submerged.

4. All junction and terminal boxes are assumed to fail if submerged.

5. All instruments are assumed to fail if submerged.

6. All cables are nonhydroscopic and are not assumed to fail if submerged. The Unit 2 cable specifications require a 14-day water absorption test (accelerated water absorption) to be performed in accordance with ICEA Standard S-19-81, Section 6.9.

7. MCCs and switchgear are assumed to fail if submerged.

8. Guidelines for single active failure are the same as those assumed for failure due to spraying (Section 3C.4.3).

9. Credit is taken for Operator action to isolate the leak 30 min after detection.

10. Visual leak detection is assumed to occur within 24 hr of leak initiation. This is based on an area walkdown by plant personnel a minimum of once per calendar day, not to exceed a 24-hr time period. In addition to visual detection, the emergency core cooling system (ECCS) pump cubicles each have a safety-related Class 1E cubicle flood level switch with an associated alarm in the main control room. Furthermore, each cubicle has a nonsafety-related level switch with an alarm in the main control room. Other areas that require leak detection have redundant safety-related Class 1E level switches that alarm in the main control room.

3C.5.4 Analytical Methods

For a pipe in any given area, a through-wall leakage crack is assumed to occur at a location that would result in the most severe consequences due to flooding. The flow rate of the fluid is evaluated by assuming that the crack acts as an orifice. The following equation is used:

$$Q = 19.65\, C\, d^2 \sqrt{h_L}$$

Where:

Q = Crack flow (gpm)

C = Orifice coefficient ($C = 0.6$ or higher)

d = Equivalent diameter of crack (in)

h_L = Fluid head (ft)

The diameter of the crack is determined by assuming that the crack area is circular in shape. The area is defined as:

$$A = (D/2)(t/2)$$

Where:

A = Crack area (in^2)

D = Nominal pipe diameter (in)

t = Nominal wall thickness (in)

The equivalent crack diameter is then defined as:

$$d = \left(\frac{4A}{\pi}\right) 1/2$$

In calculating flow over stairways, hatches, and other floor openings or curbs, weir flow is assumed to determine the height of the water above the top of the weir as follows:

$$h_w = \left(\frac{q}{3.33L}\right) 2/3$$

Where:

h_w = Water head above weir (ft)

q = Flow (ft^3/sec)

L = Length of weir (ft)

If there is an intervening door which is not watertight, an additional head loss (modeled as a thick-edged orifice) is assumed for the door.

3C.5.5 Results of Evaluation - Flooding FMEA

The following subsections present, building-by-building, the results of the flooding evaluation using the procedures and guidelines discussed in Sections 3C.5.2 and 3C.5.3. The evaluation verifies that the plant can be safely shut down in the event of pipe cracks in fluid systems. As noted below, protective measures have been implemented to ensure the required system functional capability is maintained.

3C.5.5.1 Reactor Building (Including Auxiliary Bays)

In the reactor building general area, a crack in a moderate-energy line, mass-released from a high-energy pipe break or a nonseismic tank failure, on the various floors above el 175 ft would result in a maximum flood height of approximately 8 in. Buildup above this level is prevented by spillage over the curbs surrounding several pipe chases. None of the flood-susceptible safe shutdown equipment on the upper floor elevations would be submerged under these conditions.

In the reactor building general floor area on el 175 ft, the limiting case for flooding is the moderate-energy line crack (18-in RHS line). The total volume released before isolation envelopes the total mass release of any nonseismic tank in the reactor building (see Table 3C.5-2). Redundant safety-related Class 1E level switches with alarms in the main control room are provided in this area. The maximum flood level in the area after detection and isolation is less than 9 in.

Potential flooding could fail some equipment supporting RCIC or HPCS operation. Failure in these cases is considered acceptable as discussed in the spray analysis of these components in Section 3C.4.5.1.

In the reactor building auxiliary bays, safe shutdown equipment is located above the maximum predicted flood level, except in the ECCS pump rooms. Although these pump rooms are adequately protected against flooding from external sources, flooding from sources inside the rooms could result in water level buildup sufficient to fail the RHS or LPCS pump motors. However, leak detection, visual inspection, and/or watertight separation of redundant safe shutdown trains ensures that the required function of these systems is maintained.

Drainage systems in the auxiliary bays are designed such that leakage in one cubicle will not backflow into another. Each pump room is provided with an independent drainage sump and two level switches which alarm in the control room (one safety-related and one nonsafety). Although sump discharge piping from each cubicle is headered together, backflow is prevented by check valves.

3C.5.5.2 Control Building

There are no high-energy lines or nonseismic tank storage in the control building. The crack flow rate for the relatively small service, domestic, and fire protection lines in the control building is within the design capacity of the normal drain system. Water from the upper floors of the control building would drain to the lowest level of the building which contains the sumps. The only safe shutdown equipment on the lowest level are cables, which are not susceptible to failure from flooding. Safe shutdown equipment which could be affected adversely by water buildup on upper floors is protected as required. Visual detection a minimum of once per calendar day, not to exceed a 36-hr time period, is sufficient to limit water buildup to acceptable levels.

3C.5.5.3 Diesel Generator Building

There are no high-energy lines or nonseismic tanks housed in the diesel generator building. The maximum flood level in any one of the areas containing the diesels is slightly more than 4 in above the floor, which would not be enough to submerge any safe shutdown targets.

3C.5.5.4 Piping Tunnels

There are no high-energy lines or nonseismic tank storage within the piping tunnels. In addition, there are no high-energy lines which are hydraulically connected (e.g., open drains, etc.) to the piping tunnels. Flooding of safety-related service water system equipment in the pipe tunnels will not prevent safe shutdown. Inoperable valves will fail as is to permit the cooling of ECCS systems.

The four areas of the pipe tunnel containing redundant safe shutdown equipment have independent drainage sumps and nonsafety level switches. The sump pump discharges are headered together, but backflow is prevented by check valves and elevation differences in the header and feed lines.

3C.5.5.5 Electrical Tunnels

There are no high-energy lines or nonseismic tank storage in or hydraulically connected to the electrical tunnels or electrical tunnel equipment rooms. Flooding from fire protection or service water lines would result in radiation detectors in one tunnel being submerged. These detectors monitor service water leaving the RHS heat exchangers for radiation from tube leakage. Under the shutdown cooling mode, this function could be served by daily grab samples as allowed by the Technical Specifications, if these monitors were lost.

The electrical tunnels are served by independent drainage sumps. Although sump pump discharge piping is headered together, backflow in the sump pump discharge lines is prevented by check valves and elevation differences in the header and feeder lines.

3C.5.5.6 Screenwell Building

There are no high-energy liquid lines in the screenwell building. The nonseismic tank storage shown in Table 3C.5-2 is located in an area of the screenwell building which is hydraulically isolated from the safe shutdown equipment.

In the screenwell building, the service water pump bays may be flooded by either fire protection or service water. This would result in a loss of one of the redundant trains, but would not require immediate plant shutdown. Flooding of one pump bay will not affect the redundant pump bay. Flooding in either pump bay will be wholly contained within that bay by the physical separation provided. Flood height from service water piping is limited to maximum lake elevation, which is below the level of physical separation. For other sources of flooding, a minimum of once per calendar day, not to exceed a 24-hr time period for visual detection, is sufficient to limit flooding to acceptable levels. Each bay is served by independent floor drain sumps and two level switches which alarm in the control room (one safety grade and one nonsafety). The sump pump discharge piping is headered outside the pump bays. Backflow is prevented by check valves and elevation differences in the header and feeder lines.

3C.5.5.7 Standby Gas Treatment Building

The equipment in the SGT building is not needed for mitigation of a moderate-energy line crack or safe shutdown during normal plant operation. Therefore, a review of the equipment in this building is not necessary.

TABLE 3C.4-1
(Sheet 1 of 1)

LEAKAGE RATES IN REACTOR BUILDING
Moderate-Energy Systems

System	System Design Conditions		Nominal Line Size In/Sch	Line Wall Thickness In	Circular Area (In^2)	Maximum Flooding Leakage Rate (gpm)
	Pressure (psig)	Temperature (°F)				
Component Cooling (CCP)	145	150	16/30	0.375	1.500	570
Condensate (CNS)	240	150	6/40	0.280	0.420	205
High-Pressure Core Spray (CSH)	136*	120	20/20	0.375	1.875	685
Low-Pressure Core Spray (CSL)	103*	170	20/20	0.375	1.875	600
Domestic Water (DWS)	135	200	3/40	0.216	0.162	60
Fire Protection (FPW)	175	120	8/40	0.322	0.644	270
Reactor Core Isolation Cooling (ICS)	150*	104	6/120	0.280	0.420	360
Makeup Water (MWS)	160	104	4/40	0.237	0.237	95
Reactor Rod Drive (RDS)	300	150	4/40	0.237	0.237	130
Residual Heat Removal (RHS)	450**	355	18/SPW	0.500	2.250	1580
Spent Fuel Pool Cleaning and Cleanup (SFC)	300	150	10/40	0.365	0.913	500
Standby Liquid Control (SLS)	160*	104	3/40	0.216	0.162	65
Reactor Plant Sampling (SSR)	160	180	2/40	0.154	0.077	35
Service Water (SWP)	150	134	30/SPW	0.375	2.813	1085
Reactor Water Cleanup (WCS)	150*	150	10/40	0.365	0.9125	355
Alternate Decay Heat Removal (ADH)	200	150	12/STD	0.375	1.125	305

* Standby mode.
** 2 percent of the time this is a high energy system.

TABLE 3C.4-2
(Sheet 1 of 1)

LEAKAGE RATES IN CONTROL BUILDING
Moderate-Energy Systems

System	System Design Conditions		Nominal Line Size In/Sch	Line Wall Thickness In	Circular Area (In^2)	Maximum Flooding Leakage Rate (gpm)
	Pressure (psig)	Temperature (°F)				
Chilled Water (HVK)	150	120	6/40	0.28	0.42	120
Fire Protection (FPW)	175	120	6/40	0.28	0.42	125
Service Water (SWP)	150	130	6/40	0.28	0.42	120
Domestic Water (DWS)	135	200	4/40	0.237	0.237	90

TABLE 3C.4-3
(Sheet 1 of 1)

LEAKAGE RATES IN DIESEL GENERATOR BUILDING
Moderate-Energy Systems

System	System Design Conditions		Nominal Line Size In/Sch	Line Wall Thickness In	Circular Area (In^2)	Maximum Flooding Leakage Rate (gpm)
	Pressure (psig)	Temperature (°F)				
Service Water (SWP)	60*	130	10/40	0.365	0.913	165
Fire Protection (FPW)	175	120	4/40	0.237	0.237	100
Diesel Generator Fuel (EGF)	10	120	4/40	0.237	0.237	30

* Maximum system operating pressure.

TABLE 3C.4-4
(Sheet 1 of 1)

LEAKAGE RATES IN PIPE TUNNELS
Moderate-Energy Systems

System	System Design Conditions		Nominal Line Size In/Sch	Line Wall Thickness In	Circular Area (In^2)	Maximum Flooding Leakage Rate (gpm)
	Pressure (psig)	Temperature (°F)				
Service Water (SWP)	150	130	30/SPW	0.375	2.813	1085
Component Cooling (CCS)	150	150	30/SPW	0.375	2.813	1085
Circulating Water (CWS)	35	134	16/STD	0.375	1.50	280
Condensate Makeup and Drawoff (CNS)	300	150	8/STD	0.322	0.644	355
Offgas (OFG)	50	225	3/STD	0.216	0.162	40

TABLE 3C.4-5
(Sheet 1 of 1)

LEAKAGE RATES IN ELECTRICAL TUNNELS
Moderate-Energy Systems

System	System Design Conditions		Nominal Line Size In/Sch	Line Wall Thickness In	Circular Area (In^2)	Maximum Flooding Leakage Rate (gpm)
	Pressure (psig)	Temperature (°F)				
Fire Protection (FPW)	175	120	6/40	0.280	0.420	175

TABLE 3C.4-6
(Sheet 1 of 1)

LEAKAGE RATES IN SCREENWELL BUILDING
Moderate-Energy Systems

System	System Design Conditions		Nominal Line Size In/Sch	Line Wall Thickness In	Circular Area (In^2)	Maximum Flooding Leakage Rate (gpm)
	Pressure (psig)	Temperature (°F)				
Component Cooling (CCS)	150	150	2/80	0.218	0.109	45
Fire Protection (FPW)	175	120	12/std	0.375	1.125	465
Yard Structures Ventilation (HVY)	50	150	2/80	0.218	0.109	25
Lube Oil (LOS)	70	170	3/40	0.216	0.162	45
Liquid Waste (LWS)	256	150	3/40	0.216	0.162	80
Makeup Water (MWS)	160	104	2/40	0.154	0.077	30
Domestic Water (DWS)	135	104	4/tubing	0.130	0.130	50
Service Water (SWP)	150	130	36/spw	0.500	4.500	1720
Traveling Screens Service Water (SWT)	130	100	8/40	0.322	0.644	230
Water Treating (WTS)	120	104	4/40	0.237	0.237	80

TABLE 3C.5-1
(Sheet 1 of 2)

TOTAL MASS RELEASED BY HIGH-ENERGY
LINE BREAKS (HELBs)

Building	HELB	Total Mass (lb)
Reactor building*	RWCU	15,700
Reactor building auxiliary bays	None	-
Control building	None	-
Diesel generator building	None	-
Piping tunnels	None	-
Electrical tunnels	None	-
Screenwell building	None	-

* Mass released by the high-energy liquid line (RWCU) envelopes the RCIC steam line break releases.

TABLE 3C.5-2
(Sheet 1 of 2)

CAPACITY OF NONSEISMIC TANKS AND VESSELS WITHIN BUILDINGS CONTAINING SAFE SHUTDOWN EQUIPMENT

Building	Mark No.	Capacity (gal) (total)
Reactor building	2DER-TK1	1,000
	2DER-TK2A	1,000
	2DER-TK2B	1,000
	2LWS-TK6A	7,600
	2LWS-TK6B	7,600
	2LWS-TK30	7,600
	2CCP-TK1	3,000
	2CCP-TK2	150
	2DFR-TK1	1,000
	2DFR-TK2A	1,250
	2DFR-TK2B	1,250
	2DFR-TK2C	1,250
	2DFR-TK2D	1,250
	2DFR-TK2E	1,250
	2DFR-TK2F	1,250
	2DFR-TK2G	1,250
	2DFR-TK2H	1,250
Reactor building auxiliary Bays	None	-
Control building	None	-
Diesel generator building	None	-
Piping tunnels	None	-
Electrical tunnels	None	-
Screenwell building (water treatment area)	2FPW-TK2	300
	2MWS-TK1A	30,000
	2MWS-TK1B	30,000
	2FOF-TK1	650(1)
	2WTS-TK1	60,000
	2WTS-TK2	5,000(2)
	2WTS-TK5	30,000
	2WTS-TK6A	30,000
	2WTS-TK6B	30,000
	2WTS-TK7	175(3)

TABLE 3C.5-2
(Sheet 2 of 2)

CAPACITY OF NONSEISMIC TANKS AND VESSELS WITHIN BUILDINGS CONTAINING SAFE SHUTDOWN EQUIPMENT

Building	Mark No.	Capacity (gal) (total)
	2WTS-TK8	125[4]
	2WTS-TK9	1,475[4]

(1) Contains oil.
(2) Contains caustic. Leakage is totally contained.
(3) Contains acid. Leakage is totally contained.
(4) These tanks have been abandoned/retired in place.

APPENDIX 3D

UNIT 2 ASSESSMENT OF GENERAL DESIGN CRITERIA 51 to 10CFR50

APPENDIX 3D

UNIT 2 ASSESSMENT OF GENERAL DESIGN CRITERIA 51 TO 10CFR50

Compliance of the primary containment pressure boundary materials with General Design Criteria (GDC) 51 was evaluated by identifying the materials which were limiting under operation, maintenance, test and postulated accident conditions based on material type, thickness and metallurgical characterization. The fracture toughness of these materials was evaluated on the basis of ASME III NC-2300, 1977 Edition, including the summer 1977 addendum, and NUREG-0577, Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports, October 1979.

Except for the high-pressure core spray (HPCS) system test isolation valve bonnet, the limiting materials were found to have a permissible lowest service metal temperature (PLSMT) at or below the design lowest service metal temperature (LSMT). The PLSMT for the HPCS test isolation valve bonnet was found to be +71°F, whereas the LSMT is +70°F. This LSMT is the lowest possible temperature which could be reached during an extended outage, taking no credit for heat contributions of nearby operating equipment. Since the probability is very remote that the HPCS would be needed before the fluid and environment temperature increased by 1°F, assuming the LSMT had actually been reached, the HPCS test isolation valve bonnet is considered acceptable.

In the case of the feedwater system (FWS) and reactor water cleanup system (WCS), there are situations where the actual metal temperature may be less than +70°F. However, these situations are during startup from the cold shutdown condition, where the loads are very low and the probability of rapidly propagating fracture, as referenced in GDC 51, is minimal. The FWS and WCS are above +70°F for all conditions where the pressure exceeds 660 psi.

In the following discussion, the fracture toughness of each of the specific limiting materials is evaluated in detail. This information also is presented in summary form in Table 1.

1. Equipment Hatch

   SA516 Grade 70 quenched and tempered material with a nominal thickness of 4.875 in was applied for the equipment hatch cover flange. Actual drop weight tests performed on this material indicate a nil ductility transition temperature (NDTT) of -10°F or less. Thus, the PLSMT is +45°F when the rules of ASME III NC-2300, 1977 Edition, including the summer 1977 addendum, are applied.

2. Drywell Head Pins

   SA564 Grade 630, H1075 material with a nominal diameter of 3.25 in was applied for the drywell head pins. The heat of material used had a relatively high nickel content (4.42 percent) and was age hardened at a relatively high temperature (1,075°F minimum). An estimated PLSMT of +70°F for this material is derived from Armco data on H1100 material. Armco report A.I. 71.6-16, Report No. 1, June 11, 1969, shows Charpy transition curves for relatively high nickel heats where the curve midheight temperatures are at or below +5°F. This is consistent with a PLSMT of +70°F.

3. Penetrations

   a. Z-1A sleeve. SA155 Grade CMSH80 applying SA537 Class 2, quenched and tempered and the finished pipe normalized and tempered, with a nominal wall thickness of 1.5 in was applied for the penetration Z-1A sleeve. Due to its similarity to SA516 Grade 65 (SA155 Grade KCF65) with respect to melting practice, chemistry and heat treatment, ASME III NC-2300, 1977 Edition, including the summer 1977 addendum, would assign a TNDT of 0°F and a PLSMT of +30°F.

   b. Z-11 sleeve. SA333 Grade 6 normalized with a nominal wall thickness of 1 in was applied for the penetration Z-11 sleeve. NUREG-0577, in a "worst case" characterization of this material as a "mild" steel, indicates a TNDT at or below the NDT of +40°F. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

   c. Z-1A flued head. SA508 Class 1 quenched and tempered with a nominal web thickness of 6 in was applied for the penetration Z-1A flued head. Actual dropweight tests performed on this material indicate a TNDT of 0°F or lower. Thus, the PLSMT is +62°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

   d. Z-4A flued head. SA508 Class 2 quenched and tempered with a design web thickness of 6.16 in was applied for the penetration Z-4A flued head. Actual dropweight tests performed on this material indicate a TNDT of 0°F or lower. Thus, the PLSMT is +62°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

   e. Z-14 sleeve. SA333 Grade 6 normalized with a nominal wall thickness of 3/4 in was applied for the HPCS injection penetration sleeve. NUREG-0577, in a "worst case" classification of this material as a "mild"

steel, indicates a TNDT at or below the NDT of +40°F. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

f. Z-14 flued head. SA508 Class 1, quenched and tempered with a nominal web thickness of 4 1/4 in was applied for the HPCS injection penetration flued head. Actual dropweight tests performed on this material indicate a TNDT of 0°F or lower. Thus, the PLSMT is +51°F when the rules of ASME III, NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

g. Z-12 and Z-13 penetrations. SA312 type 304 was applied for the HPCS suppression pool suction and return penetrations. This is an austenitic stainless steel which is exempt.

4. Pipe

a. Main steam system (MSS) pipe (mark no. NM-1-85). SA106 Grade C normalized with a manufactured minimum wall thickness of 1.177 in (by Cameron Iron Works) was applied for the main steam piping. NUREG-0577 indicates a TNDT for this material at or below the mean TNDT of +40°F for mild carbon steel. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

b. MSS sockolet (26" x 3/4" x 6000#). SA105 as forged with a design thickness of 0.156 in was applied for the main steam sockolet. Although this material has a design thickness of less than 0.625 in, the philosophy of NC-2300 can still be applied.

NUREG-0577 categorizes this material as an as-hot rolled carbon manganese steel and assigns it a TNDT of +39°F. Thus, the PLSMT is +69°F when the "worst case" rules for 5/8-in thick material of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

c. FWS pipe and WCS pipe (mark nos. NM-47-112, NM-47-113, NM-09-142 and NM-09-144). SA106 Grade B normalized with a nominal wall thickness of 0.906 in and SA106 Grade C normalized with a nominal wall thickness of 2.062 in were applied for the feedwater piping. NUREG-0577 indicates a TNDT for this material at or below the NDT of +40°F for mild carbon steel. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

d. WCS pipe (mark nos. NM-09-98, NM-09-143, NM-09-145 and NM-09-146). SA333 Grade 6 normalized with a nominal wall thickness of 0.906 in was applied for the reactor water cleanup piping. NUREG-0577, in a "worst case" characterization of this material as a "mild" steel, indicates a TNDT at or below the NDT of +40°F. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

e. WCS sockolet (mark no. NM-09-143). SA105 with a design thickness of 0.092 in was applied for the reactor water cleanup sockolet. Although this material has a design thickness of less than 0.625 in, the philosophy of NC-2300 can still be applied. NUREG-0577 categorizes this material as an as-hot rolled carbon-manganese steel and assigns it a TNDT of +39°F. Thus, the PLSMT is +69°F when the "worst case" rules for 5/8-in thick material of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

f. WCS elbows (mark nos. NM-09-98, NM-09-142, NM-09-143, NM-09-144, NM-09-145 and NM-09-146). SA234 Grade WPB fabricated from SA106 and the final fitting normalized, with a nominal thickness of 0.906 in, was applied for the WCS elbows. NUREG-0577 indicates a TNDT for normalized SA106 at or below the NDT of +40°F for mild carbon steel. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

g. High-pressure core spray (CSH) system pipe, ASME III Class 1 (mark no. NM-25-59X). SA106 Grade B manufactured by Phoenix Steel, Phoenixville, PA, with a nominal wall thickness of 0.844 in was applied for the HPCS injection line piping. Information obtained from Phoenix Steel during the preparation of the Salem Nuclear Generating Station Unit No. 2 Safety Evaluation Report indicates that the mill finishes the material at 1600°F. (Reference Salem Unit 2 SER (NUREG-0517) Supplement No. 5 and Public Service Electric & Gas (PSE&G) letter to the Director of Nuclear Reactor Regulation, USNRC, dated December 19, 1980, regarding Containment Boundary Fracture Toughness). Since this finishing temperature is in the range of normalizing temperatures, data for normalized SA106 Grade B can be applied. NUREG-0577 indicates a TNDT for this material at or below the NDT of +40°F for mild carbon steel. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F. In addition, this material was CVN impact tested at +40°F and met the acceptance criteria of ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

h. CSH pipe, ASME III Class 2, except between the HPCS test isolation valve and the HPCS test throttling valve (mark nos. NM-25-25X, NM-25-116X, NM-25-117X, NM-25-118X, NM-25-119X, NM-25-120X, NM-25-121X and NM-25-138X). SA106 Grade B with a nominal wall thickness ranging from 0.237 in to 0.562 in was applied for the HPCS suppression pool suction and the HPCS suppression pool return piping, except for the pipe between the HPCS test isolation and throttling valves. This material is exempt from testing, due to its nominal thickness, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

i. CSH pipe, ASME III Class 2, between the HPCS test isolation valve and the HPCS test throttling valve (mark no. NM-25-115X). SA106 Grade B manufactured by Phoenix Steel, Phoenixville, PA, with a nominal wall thickness of 0.844 in was applied for the pipe between the HPCS test throttling and isolation valves. As in item 4.g above, data for normalized SA106 Grade B can be applied. NUREG-0577 indicates a TNDT for this material at or below the NDT of 40°F for mild carbon steel. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F. In addition, this material was subsequently CVN impact tested at +65°F and met the acceptance criteria of ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

j. CSH elbows (mark nos. NM-25-116X, NM-25-117X, NM-25-118X, NM-25-119X, NM-25-120X and NM-25-121X). SA234 Grade WPB with a nominal wall thickness of 0.280 in to 0.406 in wall applied for the CSH elbows. This material is exempt from testing, due to its nominal thickness, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

k. CSH sockolets/weldolets (mark nos. NM-25-59X, NM-25-120X and NM-25-121X). SA105 with a nominal thickness of 0.237 in and less (applying the ASME III definition) was applied for the CSH sockolets and weldolets. This material is exempt from testing, due to its nominal thickness, in accordance with ASME NC-2300, 1977 edition, including the summer 1977 addendum. However, an analysis of this material was performed as requested by the NRC using the "worst case" thickness of 0.843 in and indicated the following. NUREG-0577 categorizes this material as an as-hot rolled carbon manganese steel and assigns it a TNDT of +39°F. Thus, the PLSMT is +69°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

l. CSH reducer, outlet side of 2CSH*HCV133 (mark no. NM-25-120X). SA234 Grade WPB, fabricated from SA106 and the final fitting normalized, with a nominal wall thickness of 0.375 in was applied for the reducer on the outlet side of the CSH system test throttling valve. This material is exempt from testing due to its nominal thickness, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

m. CSH reducer, inlet side of 2CSH*HCV133 (mark no. NM-25-115X). SA234 Grade WPB, fabricated from SA106 and the final fitting normalized with a nominal wall thickness of 0.844 in was applied for the reducer on the inlet side of the CSH system test throttling valve. NUREG-0577 indicates a TNDT for normalized SA106 at or below the NDT of +40°F for mild carbon steel. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

5. Feedwater Thermal Tees (2FWS*FTG1A)

a. Flued head. SA350 Grade LF2 normalized with a manufactured minimum web thickness of 1.804 in was applied for the feedwater thermal tee flued head. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

b. Extruded outlet fitting. SA420 Grade WPL6, fabricated from SA350 Grade LF2, normalized with a manufactured minimum web thickness of 2.625 in, was applied for the feedwater thermal tee extruded outlet fitting. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +28°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

c. Thermal sleeve. SA350 Grade LF2 normalized with a manufactured minimum wall thickness of 0.793 in was applied for the feedwater thermal tee thermal sleeve. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

d. Reducer. SA350 Grade LF2 normalized with a manufactured minimum wall thickness of 1.804 in was applied for the feedwater thermal tee reducer. NUREG-0577 categorizes this material as a normalized

carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

e. Thermal sleeve run. SA516 Grade 70 normalized, cold formed into pipe and stress relieved, with a nominal thickness of 0.375 in was applied for the feedwater thermal tee thermal sleeve run. Although this material is less than 5/8 in thick, the philosophy of NC-2300 can still be applied. ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a TNDT of 0°F to normalized SA516 Grade 70. According to the discussion on "Effects of Cold Work" in Welding Research Council Bulletin Number 158, January 1971, when this type of material is cold worked 1 percent and then stress relieved, it completely regains its fracture toughness; when it is cold worked 5 percent and then stress relieved, its transition curve midheight temperature increases 20°F. As the material for the thermal sleeve run was strained just under 2 percent, a conservative assumption of a 20° increase in TNDT can be made. Based on a TNDT of 0° + 20°, or 20°F, the PLSMT is +50°F when the "worst case" rules for 5/8-in thick material of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

6. Main Steam Isolation Valve

Fracture toughness requirements for the main steam isolation valves (MSIV) are discussed in Appendix 5A, Section 5A.3.

7. Feedwater Isolation Valve (2FWS*MOV21A)

a. Body. SA105 quenched and tempered with a manufactured minimum wall thickness of 2.28 in was applied for the feedwater isolation valve body. NUREG-0577 indicates a TNDT for quenched and tempered SA105 at or below the NDT of -28°F for normalized carbon manganese steel. Based on a TNDT of -28°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +2°F.

b. Bonnet. SA105 normalized with a manufactured minimum thickness of 2.47 in was applied for the feedwater isolation valve bonnet. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

c. Wedge. SA105 normalized with a design thickness of 1.7128 in was applied for the feedwater isolation valve

wedge. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

d. Bolting. The feedwater isolation valve bolting is not pressure retaining.

e. Thrust ring. SA182 Grade F6 with a design thickness of 1.000 in was applied for the feedwater isolation valve thrust ring. This material was tempered at a relatively high temperature of 1400°F, which serves to enhance its fracture toughness. An estimated PLSMT of +70°F for this material is derived from Republic Steel data, Universal-Cyclops Steel data and other data in the literature that exhibit very good toughness properties after tempering at 1400°F. In addition, the thrust ring is loaded in compression and shear only, which minimizes the possibility of crack propagation.

8. Feedwater Swing Check Valve (2FWS*V23A)

a. Body. SA216 Grade WCB normalized with a manufactured minimum wall thickness of 2.28 in was applied for the feedwater swing check valve body. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

b. Bonnet. SA105 normalized with an actual thickness of 4.498 in was applied for the feedwater swing check valve bonnet. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +50°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

c. Disc. SA105 normalized with a manufactured minimum thickness of 2.28 in was applied for the feedwater swing check valve disc. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

d. Bolting. SA193 Grade B7 and SA194 Grade 2H, both quenched and tempered, with a nominal diameter of 0.625 in, were applied for the feedwater swing check valve bolting. This material is categorized by NUREG-0577 as having least susceptibility to brittle failure.

9. Reactor Water Cleanup Isolation Valve (2WCS*MOV200)

   a. Body. SA105 normalized with a design thickness of 0.880 in was applied for the reactor water cleanup (RWCU) isolation valve body. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

   b. Bonnet. SA105 normalized with a design thickness of 0.875 in was applied for the RWCU isolation valve bonnet. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

   c. Disc. SA105 normalized with a design thickness of 2.25 in was applied for the RWCU isolation valve disc. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

   d. Bolting. The RWCU isolation valve bolting is not pressure retaining.

   e. Thrust ring. SA182 Grade 6 with a design thickness of 1.253 in was applied for the RWCU isolation valve thrust ring. This material was tempered at a relatively high temperature of 1400°F, which serves to enhance its fracture toughness. An estimated PLSMT of +70°F for this material is derived from Republic Steel data, Universal-Cyclops Steel data and other data in the literature that exhibit very good toughness properties after tempering at 1400°F. In addition, the thrust ring is loaded in compression and shear only, which minimizes the possibility of crack propagation.

10. High-Pressure Core Spray Recirculation Isolation Valve (2CSH*MOV105)

    The HPCS recirculation isolation valve is located in and connected to 4-in diameter, 0.337-in nominal wall pipe. Therefore, the materials of this valve are exempt from testing, due to the nominal pipe size and nominal wall thickness of the connecting pipe, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum. However, an analysis of the materials of this valve was performed as requested by the Nuclear Regulatory Commission (NRC) and indicated the following.

a. Body. SA216 Grade WCB normalized with a design thickness of 1 3/4 in was applied for the HPCS recirculation isolation valve body. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

b. Bonnet. SA216 Grade WCB normalized with a design thickness of 1 3/4 in was applied for the HPCS recirculation isolation valve bonnet. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

c. Disc. SA216 Grade WCB normalized with a design thickness of 1/2 in was applied for the HPCS recirculation isolation valve disc. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, using the "worst case" rules for 5/8-in thick material assigns a PLSMT of +65°F.

d. Bolting. SA193 Grade B7 and SA194 Grade 2H, both quenched and tempered, with a nominal diameter of 1 in were applied for the HPCS recirculation isolation valve bolting. These materials are categorized by NUREG-0577 as having least susceptibility to brittle failure.

11. High-Pressure Core Spray Test Isolation Valve (2CSH*MOV111)

a. Body. SA216 Grade WCB normalized and tempered with a design thickness of 2.585 in was applied for the HPCS test isolation valve body. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +67°F. In addition, this material was CVN impact tested at +40°F and met the acceptance criteria of ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

b. Bonnet. SA216 Grade WCB normalized and tempered with a design thickness of 2.836 in was applied for the HPCS test isolation valve bonnet. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +71°F. Also,

this material was CVN impact tested at +40°F and met the acceptance criteria of ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

c. Disc. SA216 Grade WCB normalized and tempered with a design thickness of 1.51 in was applied for the HPCS test isolation valve disc. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F. In addition, this material was CVN impact tested at +40°F and met the acceptance criteria of ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

d. Bolting. SA193 Grade B7 and SA194 Grade 7, both quenched and tempered, with a nominal diameter of 1 3/8 in was applied for the HPCS test isolation valve bolting. This material is categorized by NUREG-0577 as having least susceptibility to brittle failure.

e. Drain connection. SA105 was applied for the 3/4-in nominal pipe size half coupling drain connection on the HPCS test isolation valve. This material is exempt from testing, due to its nominal pipe size and thickness, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

12. High-Pressure Core Spray Injection Valve (2CSH*MOV107)

a. Body. SA216 Grade WCB normalized and tempered with an actual thickness of 2.440 in was applied for the HPCS injection valve body. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

b. Bonnet. SA216 Grade WCB normalized and tempered with an actual thickness of 2.40 in was applied for the HPCS injection valve bonnet. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

c. Disc. SA216 Grade WCB normalized and tempered with a design thickness of 1 3/8 in was applied for the HPCS injection valve disc. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

d. Bolting. SA193 Grade B7 and SA194 Grade 2H, both quenched and tempered, with a nominal diameter of 1 1/4 in were applied for the HPCS injection valve bolting.

   This material is categorized by NUREG-0577 as having least susceptibility to brittle failure.

13. High-Pressure Core Spray Recirculation Check Valve (2CSH*V7)

The HPCS recirculation check valve is located in and connected to 4-in diameter, 0.237-in nominal wall pipe. Therefore, the materials of this valve are exempt from testing, due to the nominal pipe size and nominal wall thickness of the connecting pipe, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum. However, an analysis of the materials of this valve was performed as requested by the NRC and indicated the following.

a. Body. SA105 normalized with a design thickness of 0.250 in was applied for the HPCS recirculation check valve body. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the "worst case" rules for 5/8-in thick material of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied. In addition, this material is exempt due to its thickness.

b. Cover (bonnet). SA105 normalized with a design thickness of 1.038 in was applied for the HPCS recirculation check valve cover. NUREG-0577 categorizes this material as a normalized carbon manganese steel and assigns it a TNDT of -5°F. Thus, the PLSMT is +25°F when the rules of ASME III NC-2300, 1977 edition, including the summer 1977 addendum, are applied.

c. Disc. The HPCS recirculation check valve disc does not perform a primary containment pressure boundary function.

d. Bolting. SA193 Grade B7 and SA194 Grade 2H, both quenched and tempered, with a nominal diameter of 5/8 in were applied for the HPCS recirculation check valve bolting. This material is categorized by NUREG-0577 as having least susceptibility to brittle failure.

14. High-Pressure Core Spray Suppression Pool Suction Isolation Valve (2CSH*MOV118)

The HPCS suppression pool suction isolation valve is located in and connected to 18-in diameter, 0.375-in nominal wall pipe. Therefore, the materials of this valve are exempt

from testing, due to the nominal wall thickness of the connecting pipe, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum. However, an analysis of the materials of this valve was performed as requested by the NRC and indicated the following.

a. Body. SA216 Grade WCB normalized and tempered with an actual thickness of 2 1/2 in was applied for the HPCS suppression pool suction isolation valve body. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

b. Bonnet. SA216 Grade WCB normalized and tempered with a design thickness of 2 1/2 in was applied for the HPCS suppression pool suction isolation valve bonnet. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

c. Disc. SA216 Grade WCB normalized and tempered with an actual thickness of 1 1/2 in was applied for the HPCS suppression pool suction isolation valve disc. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition including the summer 1977 addendum, assigns a PLSMT of +65°F.

d. Bolting. SA193 Grade B7 and SA194 Grade 2H, both quenched and tempered, with a nominal diameter of 1 1/4 in, were applied for the HPCS suppression pool suction isolation valve bolting. This material is categorized by NUREG-0577 as having least susceptibility to brittle failure.

15. High-Pressure Core Spray Recirculation Throttling Valve (2CSH*HCV116)

The HPCS recirculation throttling valve is located in and connected to 4-in diameter 0.337-in nominal wall pipe. Therefore, the materials of this valve are exempt from testing, due to the nominal pipe size and nominal wall thickness of the connecting pipe, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum. However, an analysis of the materials of this valve was performed as requested by the NRC and indicated the following.

a. Body. SA217 Grade C5 normalized and tempered with an actual thickness of 0.500 in was applied for the HPCS recirculation throttling valve body. SA217 Grade C5, because of its alloy content, can be expected to exhibit toughness properties at least as good as a normalized cast carbon steel. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels, and this can also be applied as a "worst case" characterization of normalized and tempered SA217 Grade C5. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, using the "worst case" rules for 5/8-in thick material, assigns a PLSMT of +65°F.

b. Bonnet. SA217 Grade C5 normalized and tempered with an actual thickness of 1.6385 in was applied for the HPCS recirculation throttling valve bonnet. SA217 Grade C5, because of its alloy content, can be expected to exhibit toughness properties at least as good as normalized case carbon steel. NUREG-0577 indicates a TNDT for normalized SA216 Grade WCB at or below the NDT of +35°F for heat-treated cast steels, and this can also be applied as a "worst case" characterization of normalized and tempered SA217 Grade C5. Based on a TNDT of +35°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +65°F.

c. Plug. The HPCS recirculation throttling valve plug does not perform a primary containment pressure boundary function.

d. Nipples. SA106 Grade B with nominal wall thicknesses of 0.337 and 0.237 in was applied for the HPCS recirculation throttling valve inlet and outlet nipples. This material is exempt from testing based on its nominal thickness, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

e. Bolting. SA193 Grade B7 and SA194 Grade 2H, both quenched and tempered, with a nominal diameter of 1 in, were applied for the HPCS recirculation throttling valve bolting. This material is categorized by NUREG-0577 as having least susceptibility to brittle failure.

16. High-Pressure Core Spray Test Throttling Valve (2CSH*HCV133)

a. Body. SA217 Grade WC9 normalized and tempered with a design thickness of 1.07 in was applied for the HPCS test throttling valve body. Metallography was performed on a sample of this material having a more

conservative (greater) thickness and subjected to the same heat treatment as the body. Light microscopy at magnifications up to 1000X showed that the microstructure is tempered bainite/tempered bainite plus ferrite. This is consistent with predictions of continuous cooling diagrams for this cast material. Dropweight TNDT tests performed by Maino, Gomez-Gallardo and Wallace, "Section Size Effects on Toughness of Various Cast Steel," yielded a TNDT of +1°F for the same material, heat treatment, section size and microstructure as the HPCS test throttling valve body. Based on a TNDT of +1°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +31°F.

b. Bonnet. SA217 Grade WC9 normalized and tempered with a design thickness of 3.672 in was applied for the HPCS test throttling valve bonnet. Metallography was performed on a sample of this material having a similar thickness, subjected to the same heat treatment, and cast by the same foundry that produced the bonnet. Light microscopy at magnifications up to 1000X showed that the microstructure is tempered bainite/tempered bainite plus ferrite. This is consistent with predictions of continuous cooling diagrams for this cast material. Dropweight TNDT tests performed by Maino, Gomez-Gallardo and Wallace, "Section Size Effects on Toughness of Various Cast Steel," yielded a TNDT of +1°F for the same material, heat treatment, section size and microstructure as the HPCS test throttling valve body. Based on a TNDT of +1°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +49°F.

c. Inlet nipple. SA106 Grade B manufactured by Phoenix Steel, Phoenixville, PA, with a nominal wall thickness of 0.719 in was applied for the HPCS test throttling valve inlet nipple. As in Item 4.g discussed earlier, data for normalized SA106 Grade B can be applied. NUREG-0577 indicates a TNDT for this material at or below the NDT of +40°F for mild carbon steel. Based on a TNDT of +40°F, ASME III NC-2300, 1977 edition, including the summer 1977 addendum, assigns a PLSMT of +70°F.

d. Outlet nipple. SA106 Grade B with a nominal wall thickness of 0.365 in was applied to the HPCS test throttling valve outlet nipple. The connecting pipe to the outlet side of the valve is 12-in diameter, 0.375-in nominal wall pipe. Therefore, the outlet nipple of this valve is exempt from testing, due to the nominal wall thickness of the connecting pipe, in accordance with ASME III NC-2300, 1977 edition, including the summer 1977 addendum.

e. Plug. SA479 Type 304L was applied for the HPCS test throttling valve plug. This is an austenitic stainless steel which is exempt.

f. Bolting. SA193 Grade B7 and SA194 Grade 2H - both quenched and tempered, with a nominal diameter of 1 3/4 in, were applied for the HPCS test throttling valve bolting. This material is categorized by NUREG-0577 as having least susceptibility to brittle failure.

17. Reactor Building Closed Loop Cooling System Containment Penetrations and Isolation Valves

During spring refueling outages, the CCP supply temperature may be lowered to 55°F. Engineering assessments for the affected components (Containment Penetrations 2CCP*Z33A, 2CCP*Z34A, 2CCP*Z34B, 2CCP*Z46A, and 2CCP*Z47; and valves 2CCP*MOV265, 2CCP*MOV273, 2CCP*MOV122, 2CCP*MOV124, 2CCP*15A/B, 2CCP*MOV16A/B, 2CCP*MOV17A/B, and 2CCP*MOV94A/B) have determined that these CCP components will not be adversely affected by operation of the system with a supply temperature of 55°F during operating Modes 4 and 5, and that GDC 51 requirements continue to be met.

TABLE 3D-1
(Sheet 1 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
Equipment Hatch Cover Flange	SA516-70 Quenched and tempered	4-7/8 in (n)	+45 - Based on dropweight test (DWT) indicating a nil ductility transition temperature (TNDT) of -10.	+70	
Drywell Head Pins	SA564 Grade 630 H1075	3-1/4 in (n)	+70 - Based on chemistry, heat treatment and data from Armco Steel.	+70	Note 3
Penetration Z-1A Sleeve	SA155 CMSH80 Normalized and tempered	1.5 in (n)	+30 - Based on Summer 1977 Class 2 TNDT data for SA516-65.	+70	
Penetration Z-11 Sleeve	SA333 Grade 6 Normalized	1 in (n)	+70 - Based on NUREG-0577 for "worst case" analysis as "mild" steel. Also based on Charpy V-notch tests at -50, which is consistent and adequate for a design lowest service metal temperature of +70.	+70	Note 4
Penetration Z-14 Sleeve	SA333 Grade 6 Normalized	3/4 in (n)	+70 - Based on NUREG-0577 for "worst case" analysis as "mild" steel. Also based on Charpy V-notch tests at -50°F, which is consistent and adequate for a design lowest service metal temperature of +70°F.	+70	Note 4
Penetrations Z-12 Z-13	SA312 Type 304	--	Excluded - Based on austenitic stainless steel.		
Flued Head Penetration Z-1A	SA508 Class 1 Quenched and tempered	6 in (n)	+62 - Based on actual DWT indicating TNDT 0.	+70	
Flued Head Penetration Z-4A	SA508 Class 2 Quenched and tempered	8 in (n), 6.16 in (d)	+62 - Based on actual DWT indicating TNDT 0 and design thickness.	+70	Note 5
Flued Head Penetration Z-14	SA508 Class 1 Quenched and tempered	4-1/4 in (n)	+51 - Based on actual DWT indicating TNDT 0.	+70	

TABLE 3D-1
(Sheet 2 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
MSS Pipe NM-01-85	SA106-C Normalized	1.177 in (m)	+70 - Based on NUREG-0577 for "mild" steel not heat treated (better than).	+70	Note 6
MSS Sockolet	SA105	0.156 in (d)	+69 - Based on NUREG-0577 for C-Mn steel not heat treated.	+70	
FWS Pipe and Reactor Water Cleanup (WCS) Pipe - 24 in and 8 in	SA106B and C Normalized	2.062 in (n) 0.906 in (n)	+70 - Based on NUREG-0577 for "mild" steel not heat treated (better than).	+70	Note 5
WCS Pipe 8 in	SA333 Grade 6 Normalized	0.906 in (n)	+70 - Based on NUREG-0577 for "worst case" analysis as "mild" steel. Also based on Charpy V-notch tests at -50, which is consistent and adequate for a design lowest service metal temperature of +70.	+70	Notes 4,5
WCS Sockolet	SA105	0.092 in (d)	+69 - Based on NUREG-0577 for C-Mn steel not heat treated.	+70	Note 5
WCS Elbows	SA234 WPB (SA106) Normalized	0.906 in (n)	+70 - Based on NUREG-0577 for "mild" steel not heat treated (better than).	+70	Note 5
CSH Pipe ASME III Cl 1	SA106 Grade B	0.844 in (n)	+70 - Based on NUREG-0577 for "mild" steel not heat treated (better than) and Phoenix Steel mill practice. Also, CVNs at +40.	+70	
CSH Pipe ASME III Cl 2	SA106 Grade B	0.237 in (n) to 0.375 in (n)	Exempt based on thickness criteria of Summer 1977 Class 2 and SRP 6.2.7.	--	
CSH Pipe ASME III Cl 2	SA106 Grade B	0.844 in (n)	+70 - Based on NUREG-0577 for "mild" steel not heat treated (better than) and Phoenix Steel mill practice. Also, CVNs at +65.	+70	
CSH Elbows	SA234 Grade WPB	0.280 in (n) to 0.375 in (n)	Exempt - Based on thickness criteria of Summer 1977 Class 2 and SRP 6.2.7.	--	

TABLE 3D-1
(Sheet 3 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
CSH Sockolets/ Weldolet	SA105	0.237 in (n) and less (or 0.843 in "worst case" thickness)	Exempt - Based on thickness criteria of Summer 1977 Class 2 and SRP 6.2.7, or +69 - Based on NUREG 0577 for C-Mn steel not heat treated and worst case thickness.	--	
CSH Reducer, Outlet	SA234-WPB (SA106) Normalized	0.365 in (n)	Exempt - Based on thickness criteria of Summer 1977 Class 2 and SRP 6.2.7.	--	
CSH Reducer, Inlet	SA234-WPB (SA106) Normalized	0.844 in (n)	+70 - Based on NUREG-0577 for "mild" steel not heat treated (better than).	+70	
<u>2FWS*FTG1A</u>					
FWS Thermal Tee Flued Head	SA350 Grade LF2	1.804 in (m)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Thermal Tee Extruded Outlet Fit	SA240 Grade WPL6 (SA350 LF2) Normalized	2.625 in (m)	+28 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Thermal Tee Thermal Sleeve	SA350 Grade LF2 Normalized	0.793 in (m)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Thermal Tee Reducer	SA350 Grade LF2	1.804 in (m)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Thermal Tee Thermal Sleeve Run	SA516 Grade 70 Normalized, cold formed and stress relieved	3/8 in (n)	+50 - Based on Summer 1977 Class 2 TNDT data and NRC Bulletin No. 158.	+70	Notes 5,7

TABLE 3D-1
(Sheet 4 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
<u>2FWS*MOV21A</u>					
FWS Valve Body	SA105 Quenched and tempered	2.28 in (m)	+2 - Based on NUREG-0577 for C-Mn steel normalized (better than).	+70	Note 5
FWS Valve Bonnet	SA105 Normalized	2.47 in (m)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Valve Wedge	SA105 Normalized	1.7128 in (d)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Valve Bolting	--	--	--	--	Note 8
FWS Valve Thrust Ring	SA182 Grade F6 Normalized and tempered	1.253 in (d)	+70 - Based on heat treatment and data from Republic Steel and Universal-Cyclops Steel.	+70	Notes 5,9
<u>2FWS*V23A</u>					
FWS Swing Check Valve Body	SA216 Grade WCB Normalized	2.28 in (m)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	Note 5
FWS Swing Check Valve Bonnet	SA105 Normalized	4.498 in (a)	+50 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Swing Check Valve Disc	SA105 Normalized	2.28 in (m)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
FWS Swing Check Valve Bolting	SA193-B7 SA194-2H Quenched and tempered	5/8 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization as least susceptible to brittle failure.	--	Note 5

TABLE 3D-1
(Sheet 5 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
2WCS*MOV200					
WCS Valve Body	SA105 Normalized	0.880 (d)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
WCS Valve Bonnet	SA105 Normalized	0.875 (d)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
WCS Valve Disc	SA105 Normalized	2.25 (d)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 5
WCS Valve Bolting	--	--	--	--	Note 8
WCS Valve Thrust Ring	SA182 Grade F6 Normalized and tempered	1.0 (m)	+70 - Based on heat treatment and data from Republic Steel and Universal-Cyclops Steel.	+70	Notes 5,9
2CSH*MOV105			Entire valve is exempt based on thickness of connecting pipe.		Note 10
Body	SA216 Grade WCB Normalized	1-3/4 in (d)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	Note 10
Bonnet	SA216 Grade WCB Normalized	1-3/4 in (d)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	Note 10
Disc	SA216 Grade WCB Normalized	1/2 in (d)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	Note 10
Bolting	SA193-B7 SA194-2H Quenched and tempered	1 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization at least susceptible to brittle failure.	+70	Note 10

TABLE 3D-1
(Sheet 6 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
<u>2CSH*MOV111</u>					
Body	SA216-WCB Normalized and tempered	2.585 (d)	+67 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	
Bonnet	SA216-WCB Normalized and tempered	2.836 (d)	+71 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	
Disc	SA216-WCB Normalized and tempered	1.51 (d)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	
Bolting	SA193-B7 SA194-7 Quenched and tempered	1-3/8 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization as least susceptible to brittle failure.	--	
Drain Connection	SA105	3/4" NPS (n)	Exempt - Based on thickness criteria of Summer 1977 Class 2 and SRP 6.2.7.	--	
<u>2CSH*MOV107</u>					
Body	SA216-WCB Normalized and tempered	2.440 in (a)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	
Bonnet	SA216-WCB Normalized and tempered	2.40 in (a)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	
Disc	SA216-WCB Normalized and tempered	1-3/8 in (d)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	
Bolting	SA193-B7 SA194-7 Quenched and tempered	1-1/4 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization as least susceptible to brittle failure.	+70	

TABLE 3D-1
(Sheet 7 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
2CSH*V7			Entire valve is exempt based on connecting pipe nominal pipe size.		Note 11
Body	SA105 Normalized	0.250 in (d)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 11
Cover	SA105 Normalized	1.038 in (d)	+25 - Based on NUREG-0577 for C-Mn steel normalized.	+70	Note 11
Disc	SA216-WCB Normalized and tempered	7/8 in (n)	Exempt - Not a primary containment pressure boundary part.	--	Note 11
Bolting	SA193-B7 SA194-2H Quenched and tempered	5/8 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization as least susceptible to brittle failure.	--	Note 11
2CSH*MOV118			Entire valve is exempt based on thickness of connecting pipe.		Note 10
Body	SA216-WCB Normalized and tempered	2-1/2 in (a)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	Note 10
Bonnet	SA216-WCB Normalized and tempered	2-1/2 in (a)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	Note 10
Disc	SA216-WCB Normalized and tempered	1-1/2 in (a)	+65 - Based on NUREG-0577 for cast steel heat treated (better than).	+70	Note 10
Bolting	SA193-B7 SA194-2H Quenched and tempered	1-1/4 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization as least susceptible to brittle failure.	--	Note 10

TABLE 3D-1
(Sheet 8 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
2CSH*HCV116			Entire valve is exempt based on connecting pipe nominal pipe size.		Note 11
Body	SA217-C5 Normalized and tempered	0.500 in (a)	+65 - Based on NUREG-0577 for heat treated cast steel as worst case.	+70	Note 11
Bonnet	SA217-C5 Normalized and tempered	1.6385 in (a)	+65 - Based on NUREG-0577 for heat treated cast steel as worst case.	+70	Note 11
Plug	SA564 Grade 630 Aged at 1100°F	--	Exempt - Not a primary containment pressure boundary part.	--	Note 11
Nipples	SA106 Grade B	0.337 in (n)	Exempt - Based on thickness criteria of Summer 1977 Class 2 and SRP 6.2.7.	--	Note 11
Bolting	SA193-B7 SA194-2H Quenched and tempered	1 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization as least susceptible to brittle failure.	--	Note 11
2CSH*HCV133					
Body	SA217-WC9 Normalized and tempered	1.07 in (d)	+31 - Based on metallography and data from the literature.	+70	
Bonnet	SA217-WC9 Normalized and tempered	3.672 in (d)	+49 - Based on metallography and data from the literature.	+70	
Nipple (inlet)	SA106 Grade B	0.719 in (n)	+70 - Based on NUREG-0577 for "mild" steel not heat treated (better than) and Phoenix Steel mill practice.	+70	
Nipple (outlet)	SA106 Grade B	0.365 in (n)	Exempt - Based on thickness of connecting pipe criteria of Summer 1977 Class 2 and SRP 6.2.7.	--	Note 10
Plug	SA479 Type 304L	--	Exempt - Based on austenitic stainless steel.	--	

TABLE 3D-1
(Sheet 9 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

Item	Material	Thickness (Note 1)	Permissible Lowest Service Metal Temperature (PLSMT) (°F) and Basis	Lowest Service Metal Temp. (LSMT) (°F) (Note 2)	Remarks
Bolting	SA193-B7 SA194-2H Quenched and tempered	1-3/4 in (n)	Excluded from GDC 51 review based on NUREG-0577 categorization as least susceptible to brittle failure.	--	

TABLE 3D-1
(Sheet 10 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

NOTES

1. The values presented as "thickness" are as noted:

    a. Actual thickness

    d. Minimum design thickness

    (The minimum design thickness for the flued head penetration Z-4A is based on all design loads minus temperature loads. Therefore, this value is conservative since it includes conditions that are not present at the lowest service metal temperature (LSMT), such as safety relief valve discharge loads. Other design thicknesses noted in this table are based on all worst-case design loads including temperature, and are therefore also conservative since they include conditions not present at the LSMT.)

    m. Manufacturer's minimum thickness

    n. Nominal thickness

2. The LSMT is limited either by the minimum local ambient temperature or by the minimum hydrotest temperature. When limited by the local ambient temperature, the LSMT is based on the minimum capacity of HVAC plus heat effects due to plant conditions necessary prior to the time the components are stressed. These heat contributions include, for example, heat from plant lighting and operating mechanical equipment. The following figure serves to clarify the LSMT when the hydrotest is the limiting condition.

2.1 This portion to be hydrotested with reactor pressure vessel 140°F minimum temperature.

2.2 This portion to be hydrotested at 70°F minimum temperature.

2.3 This portion to be hydrotested at 70°F minimum temperature.

3. SA564, Grade 630, is a precipitation hardening steel which cannot be dropweight tested. The deposition of the weld bead, as required by ASTM E208, would alter the material properties and therefore render the test not meaningful.

TABLE 3D-1
(Sheet 11 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

NOTES (continued)

From a metallurgical consideration, the heat used had a relatively high nickel content (4.42 percent) and was age hardened at a relatively high temperature (1,075°F, minimum), both of which serve to enhance the fracture toughness of this material.

4. The PLSMT for penetration Z-11 and for the 8-in feedwater pipe, both fabricated of SA333, Grade 6, were evaluated as follows:

   Generally, SA333 can be expected to perform significantly better than the "mild steel" group of NUREG-0577. SA333 is Specification for Seamless and Welded Steel Pipe for Low Temperature Service. When intended for low temperature service, materials are manufactured with built-in inherent toughness, accomplished mainly by tight controls on cleanliness, chemistry, and heat treatment. This inherent toughness is evidenced by the high CVN absorbed energy values obtained at very low temperatures.

   For the two items in question, CVNs were performed on each heat of material at -50°F in accordance with SA333, Grade 6, and demonstrated absorbed energy values of 60/41 ft/lb and 131/100 ft/lb (average of 3/lowest single value). In accordance with NUREG-0577, paragraph 4.4.1, the temperature at which CVNs demonstrate 20 to 25-ft/lb absorbed energy is considered to approximate the TNDT. Therefore, for the heats above, it can be conservatively assumed that the TNDT is at or below -50°F, and that the SA333, Grade 6, is adequate for these items.

5. There are situations where the FWS and WSS actual metal temperature may be less than +70°F; however, these situations could occur only during startup from the cold shutdown condition where the loads are very low. At very low stresses, the probability of rapidly propagating fracture is also very low. As the pressure increases, the temperature also increases such that the temperature is greater than +70°F for all conditions where the pressure exceeds 600 psi. A pressure of 660 psi correlates to 20

TABLE 3D-1
(Sheet 12 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

NOTES (continued)

percent of the preoperational system hydrostatic test pressure, which is defined by ASME III NC-2300, 1977 Edition, including the Summer 1977 addendum as the reference point to be used in establishing the lowest service temperature.

6. Actual fabrication of main steam piping which serves as primary containment pressure boundary did not include hot bending. The CMTRs included hot bending information as a qualification in case it was elected to hot bend in fabrication. The actual fabrication, however, used miters and the material is therefore in the normalized condition. NUREG-0577, Table 4.4, on the basis of Figure 8 data for normalized SA106, would assign a TNDT at or below +40°F. The ASME III Summer 1977 Addenda, Class 2 rules then would assign a PLSMT of +70°F.

7. The normalized SA516 Grade 70 was cold worked approximately 1.9 percent in forming the thermal sleeve run. Welding Research Council Bulletin No. 158 presents data on the effects of cold work and cold work plus stress relief on the toughness properties of material such as this. The data demonstrate that the transition temperature increases with increasing cold work. After a stress relief of 1150°F, this ranges from an increase of 0°F for 1 percent cold work to an increase of 20°F for 5 percent cold work. Conservatively, a 20°F increase in the TNDT of the thermal sleeve run can be assumed, raising the ASME III Summer 1977, Class 2 TNDT to +20°F and resulting in a PSLMT of +50°F.

8. FWS*MOV21A and WCS*MOV200 bolting are not pressure-retaining parts.

9. SA132, Grade F6, is a martensitic chromium stainless steel which cannot be dropweight tested. The deposition of the weld bead, as required by ASTM E208, would alter the material properties and therefore render the test not meaningful. From a metallurgical consideration, the high tempering temperature of 1400°F produces a material with a significantly high toughness. This is reflected in data from Republic Steel and other sources in the literature.

TABLE 3D-1
(Sheet 13 of 13)

GDC 51 COMPLIANCE REVIEW

Nine Mile Point Nuclear Station - Unit 2

NOTES (continued)

10. ASME III, Summer 1977 Addendum NC-2331, defines the nominal wall thickness for valves as the nominal pipe wall thickness of the connecting pipe. As the nominal pipe wall thickness of the pipe connecting to this valve is 5/8 in or less, no testing is required. Also, NC-2311(a)(5) of the same code exempts testing of material for valves with pipe connections of 5/8-in wall and less. Therefore, this valve meets the Summer 1977 Addendum, Class 2 rules and SRP 6.2.7 acceptance criteria.

11. ASME III, Summer 1977 Addendum NC-2311(a)(4) exempts testing for all thickness of material for valves with a nominal pipe size 6 in and smaller. Therefore, this valve meets the Summer 1977 Addendum, Class 2 rules and SRP 6.2.7 acceptance criteria.

APPENDIX 3E

CLASSIFICATION OF STRESSES IN COMPONENT/PIPING SUPPORTS DUE TO LOADS FROM SEISMIC ANCHOR MOTION AND THERMAL GROWTH OF PIPING

APPENDIX 3E

CLASSIFICATION OF STRESSES IN COMPONENT/PIPING SUPPORTS DUE TO LOADS FROM SEISMIC ANCHOR MOTION AND THERMAL GROWTH OF PIPING

The stresses in pipe supports caused by seismic anchor motion and thermal growth of piping are not required to be considered as primary stress by the plant design basis. However, a discussion of the current plant design basis for supports as it relates to this criteria is provided herein as it was provided in the response to NRC Question F210.57.

QUESTION F210.57 (3.9.3)

It is the staff's position that for the design of component supports, stresses produced by seismic anchor point motion of piping and the thermal expansion of piping should be categorized as primary stresses. Confirm that Nine Mile Point 2 meets this criteria.

RESPONSE

<u>FOR NSSS</u>

For pipe supports, reactions produced by primary and secondary loads are summed and compared to the load rating to ensure that the rating is not exceeded. No distinction is made between primary and secondary loads, and load-rated components are designed to primary limits or qualified by testing. Therefore, the supports meet primary stress criteria for primary and secondary loads combined.

<u>FOR BOP</u>

1. The Unit 2 FSAR states that component supports are in compliance with the 1974 Edition of ASME Section III, including the Summer 1974 Addendum. This edition of the Code classifies stresses caused by restraint of piping thermal expansion and anchor motions as secondary stresses and exempts them from being considered for emergency and faulted conditions. Components supports designed for Unit 2 are in full compliance with the FSAR and the ASME Code. The following is the basis for this position:

    a. Paragraph NF-3213.10 states that "Free End Displacements consist of the relative motions that would occur between an attachment and connected structure or equipment if the two members are separated. Examples of such motions are those that would occur because of relative thermal expansion of piping, equipment and equipment

supports or because of rotations imposed upon the equipment by sources other than the piping."

This classifies thermal expansion (piping) and anchor movements as free-end displacements.

b. Paragraph NF-3213.11 states that "Expansion stresses are those stresses resulting from restraint of free-end displacement of the piping system."

This defines stresses produced by the restraint of thermal expansion and anchor movement loads (from piping) as expansion stresses.

c. Paragraphs NF-3231.1(b) and (c) state explicitly that constrained free-end displacements and differential support motion effects need not be considered for emergency and faulted conditions.

ASME Section III used the principle of elastic shakedown as the basis for classifying thermal expansion and anchor movement stresses as secondary stresses. This principle is well recognized, as exemplified in Regulatory Guide 1.124.

The NRC MEB staff position regarding the classification of stresses would result in higher calculated stresses for most service conditions, and this position is identical to that of the 1983 Code Edition. However, the Code also permits higher allowable stresses than earlier editions.

2. A study was performed to assess the effect of classifying thermal and seismic anchor movement loads as primary loads on existing pipe support designs. The following QA Category I systems were included in the study:

a. Main steam.

b. Residual heat removal.

c. Feedwater.

d. Reactor water cleanup.

e. Service water.

f. Safety valve vents.

g. Low-pressure core spray.

h. Spent fuel cooling.

The pipe sizes vary between 2 in NPS and 26 in NPS. These systems have a total of 2,750 pipe supports, of which 1,490 are anchors, restraints, and struts which can be subjected to thermal expansion and anchor movement loads. Two hundred supports were selected randomly for this study. The sampling size is well above the normal sampling requirements of MIL-STD-105D. These 200 supports were reviewed, and 89 were judged to be potentially sensitive to the increased design loads if the staff's position is implemented. These 89 supports were then evaluated in more detail; finally, 19 worst cases were selected for complete reanalyses. The reanalyses were performed using thermal expansion and seismic anchor movement loads in plant service levels A through D. The results indicated that redefining such loads as primary loads would not require physical modification of the existing designs even if the allowable stresses are maintained at the present levels of the 1974 Code.

3. A review also was performed to assess the effect of classifying thermal and seismic movement loads as primary loads on all other existing component supports such as tank and pump supports.

   The results of this review indicated that redefining such loads as primary loads would not require physical modification of the existing designs.