APPENDIX 3.6A

DISCUSSION OF FINITE DIFFERENCE ANALYSIS FOR ANALYSIS OF PIPE WHIP

E

EFFECTIVE PAGE LISTING

APPENDIX 3.6A

Table of Contents

Page	Amendment
i	E

Text

Page	Amendment
3.6A-1	E
3.6A-2	E
3.6A-3	E

TABLE OF CONTENTS

APPENDIX 3.6A

E

1.0 EQUATIONS OF MOTION

The equations of motion used in the Finite Difference Analysis of Section 3.6.2.2.2.2 are of the form:

$$h\ (p_k - m_k y_k = -M_{k+1} + 2M_k - M_{k-1}) \qquad (3.6A\text{-}1)$$

where:

h = the node spacing

P_k = the externally applied lateral loads at node k

m_k = the lumped mass at node k

y_k = the lateral deflection at node k

M_k = the internal resisting moment in the beam at node k. E

Power law moment-curvature relationship is assumed and the central difference approximation for the curvature,

$$\frac{1}{h^2} = (-Y_{k+1} + 2y_k - Y_{k-1}) \qquad (3.6A\text{-}2)$$

is used.

A timewise central-difference scheme is used to solve the dynamic equations

$$y(t + \Delta t) = \Delta t^2 y(t) + 2y(t) - y\ (t - \Delta t) \qquad (3.6A\text{-}3)$$

and for the first time step

$$y(t) = \Delta t^2 y(0) \qquad (3.6A\text{-}4)$$

A time step equal to 1/10 the shortest period of vibration is used in the integration.

2.0 ELASTIC-PLASTIC MOMENT-CURVATURE LAW

The pipe is assumed to obey an elastic-strain hardening plastic moment-curvature law with isotropic strain hardening. The symbols used are defined as follows:

M = moment

$\bar{M}$ = current yield moment

E = elastic modulus of material at temperature

I = moment of inertia

Z = IE

Φ = curvature

Φ_c = M/Z = elastic curvature

$\Delta\Phi_p$ = increment of plastic curvature

Φ_p = $\epsilon|\Delta\Phi|$ = effective plastic curvature

Φ_o = $\epsilon\Delta\Phi_p$ = permanent set curvature

At the end of each integration step, new values of Φ are calculated at each node.

The known values of Φ_p, Φ_o, and M at the start of the step are used to calculate M, $\bar{M}$ and $\Delta\Phi_p$ by the following procedure:

if $|\Phi - \Phi_o| < \bar{M}/Z$

$$M = Z\ (\Phi - \Phi_o) \qquad (3.6A\text{-}5)$$ E

and

$$\Delta\Phi_p = 0 \qquad (3.6A\text{-}6)$$

if $|\Phi - \Phi_o| > \bar{M}/Z$

$$M = \bar{M} = F(|\Phi - \Phi_o| + \Phi_p)\ \sin\ (\Phi - \Phi_o)$$

and

$$\Delta\Phi_p = \Phi - \Phi_o - M/Z$$

where

$$F\ (\Phi) = K\ (\Phi)n.$$

3.0 <u>POWER LAW MOMENT-CURVATURE RELATIONSHIP</u>

The following stress strain law is assumed in the plastic range:

$$\sigma = K\ (\epsilon)^n \qquad (3.6A\text{-}7)$$

The corresponding moment-curvature law is:

$$M = K\ (\Phi)^n \qquad (3.6A\text{-}8)$$

where:

$$K = \frac{2\Pi}{3+n}\left(R_o^{3+n} - R_i^{3+n}\right)\frac{\Gamma[(1/2)n + 1]}{\Gamma[(1/2)n + 3/2]}\bar{K} \qquad (3.6A\text{-}9)$$

or, to a good approximation,

$$K = \frac{4\bar{K}}{3+n}\left(1 - .291n - .076n^2\right)\left(R_o^{3+n} - R_i^{3+n}\right) \qquad (3.6A\text{-}10)$$

in which:

R_o = pipe outside radius

R_i = pipe inside radius

In the elastic range the moment-curvature law is:

$$M = EI\Phi \qquad (3.6A\text{-}11)$$

The transition from elastic to plastic behavior on initial loading occurs at:

$$\Phi = \frac{(EI)^{\frac{1}{n-1}}}{K} \qquad (3.6A\text{-}12)$$

4.0 STRAIN RATE EFFECTS E

The effect of strain rate in carbon steel is accounted for by using a rate dependent stress strain law of the form:

$$\sigma(\epsilon, \dot{\epsilon}) = \left(1 + \frac{\dot{\epsilon}}{(40.4)}\right)^{1/5} G(\epsilon) \qquad (3.6A\text{-}13)$$

Where $G(\epsilon)$ is the static stress stain relationship. For stainless steel, the effect of strain rate is less pronounced so that a 10% increase in yield and ultimate strength is used.

5.0 RESTRAINT BEHAVIOR

The analysis is capable of handling a bilinear or power law restraint behavior. The behavior of the restraint is unidirectional. The restraint unloads elastically only to zero state, being left with a permanent set, and reloads along a bilinear or power law curve.

3.7 SEISMIC DESIGN

3.7.1 SEISMIC INPUT

3.7.1.1. Design Response Spectra

This section discusses the seismic design of those systems and sub-systems important to safety and classified as Category I in Section 3.2.

The System 80 Standard Design as defined by CESSAR is not based on a specific site, therefore seismic and geologic information cannot be provided. Seismic response spectra which envelope actual design requirements for current System 80 plants are provided in Figures 7.3.1-1 through 3.7.1-4. These spectra reflects responses for several different building types located throughout the continental United States.

The response spectra shown in Figures 3.7.1-1 thru 3.7.1-3 are applicable to the upper most horizontal support on the component named. Figure 3.7.1-4 is applicable to the vertical supports of all of the RCS major components.

The effect of differential seismic displacement on the equipment and supports is included in the site specific analysis of the Reactor Coolant System described in Section 3.7.2.1.

CE provides the following to assist the Applicant in his design of support structures:

a) A simplified mathematical model which accounts for the mass and stiffness properties of the System 80 system, suitable for coupling with the mathematical model of the supporting structures and foundations.

b) A set of design basis seismic loads transmitted from the System 80 systems to the supporting structures at all support locations.

c) The set of design basis floor response spectra at each support location, upon which the design basis loads are based.

The final verification of the design basis seismic loads is performed as described in Section 3.7.2 and 3.7.3, based on the site specific seismic excitations provided by the Applicant.

3.7.1.2 Design Time History

See Applicant's SAR for site specific information.

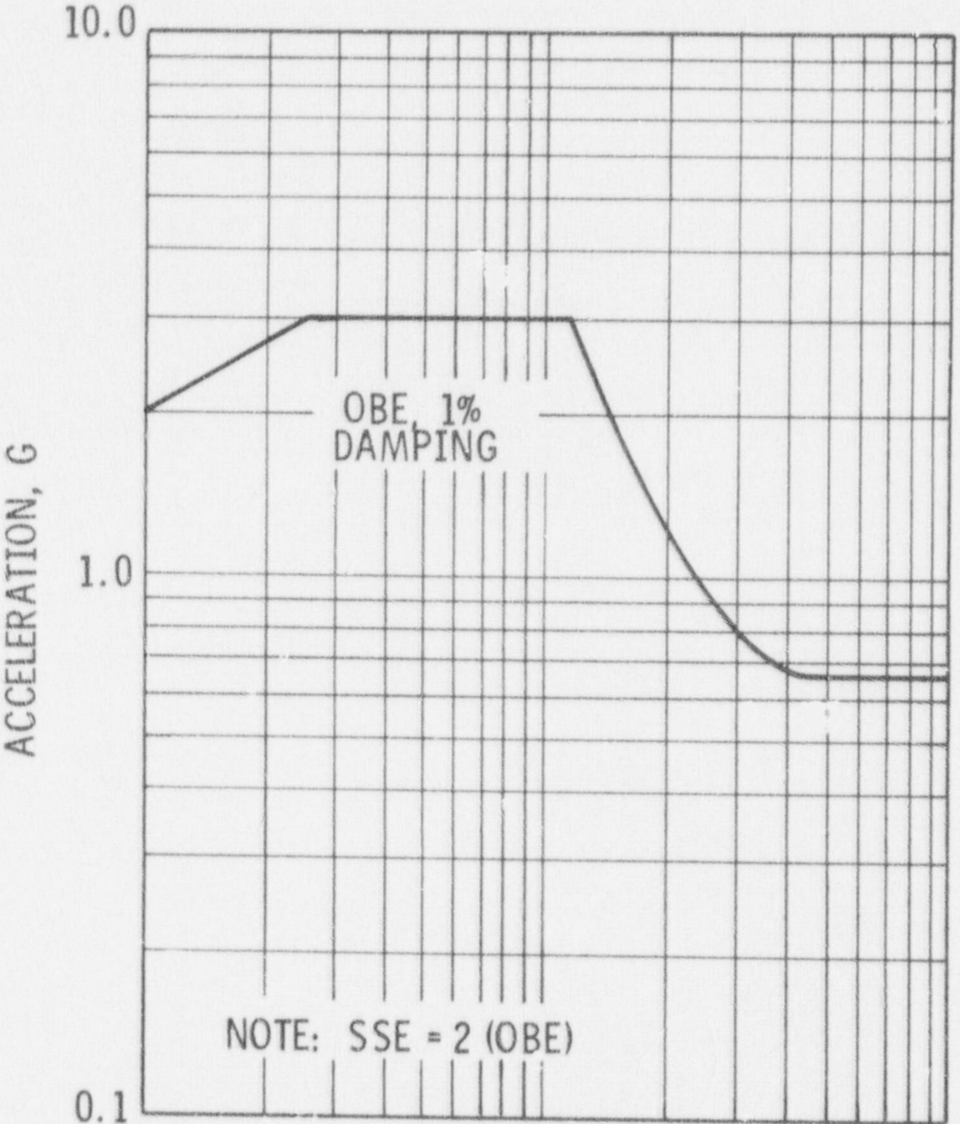
3.7.1.3 Critical Damping Values

Critical damping values used for Category I Systems and Components are given in Table 3.7.2-1.

3.7.1.4 Supporting Media for Category I Structures

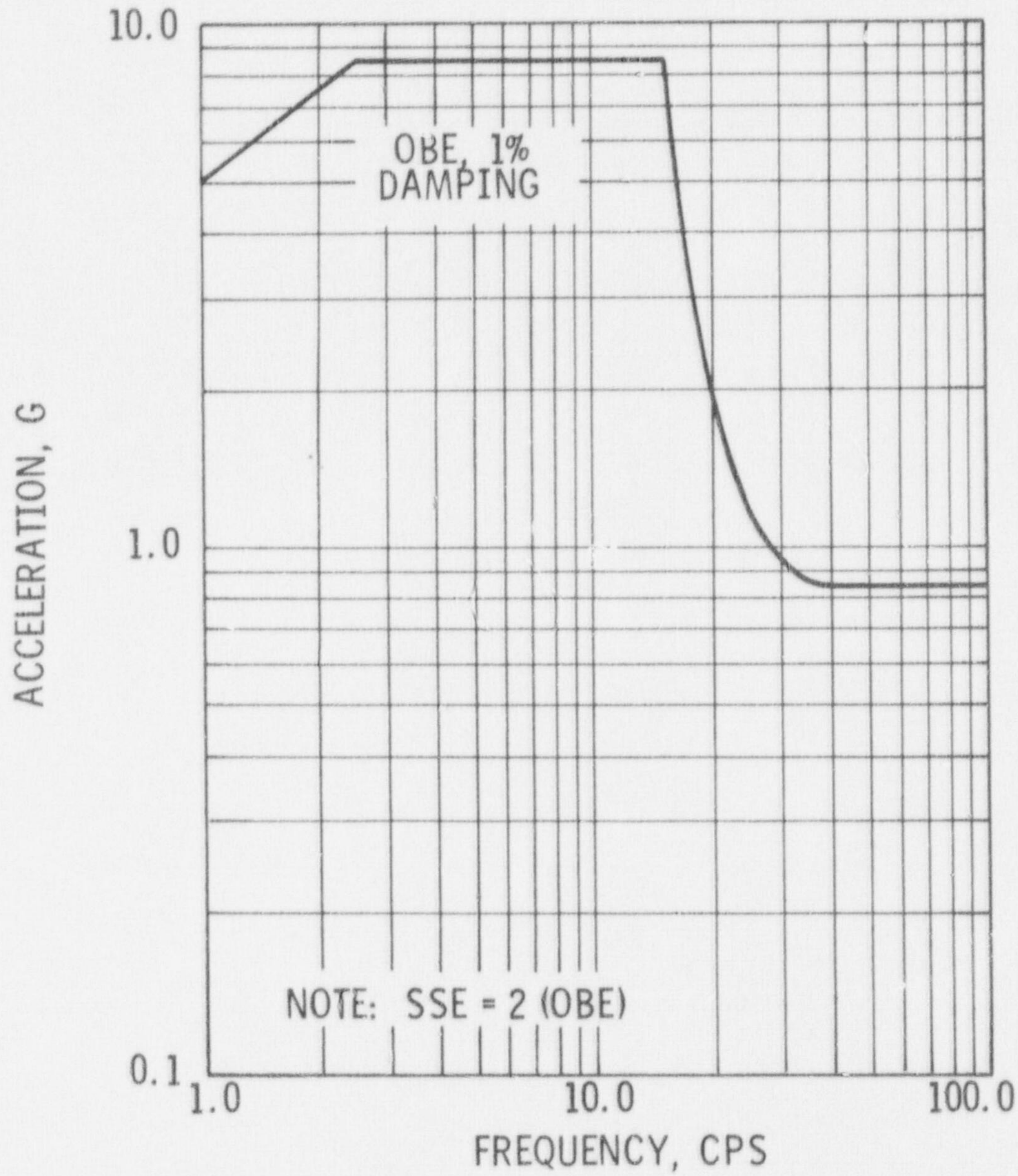
See Applicant's SAR. 1

THIS PAGE INTENTIONALLY BLANK



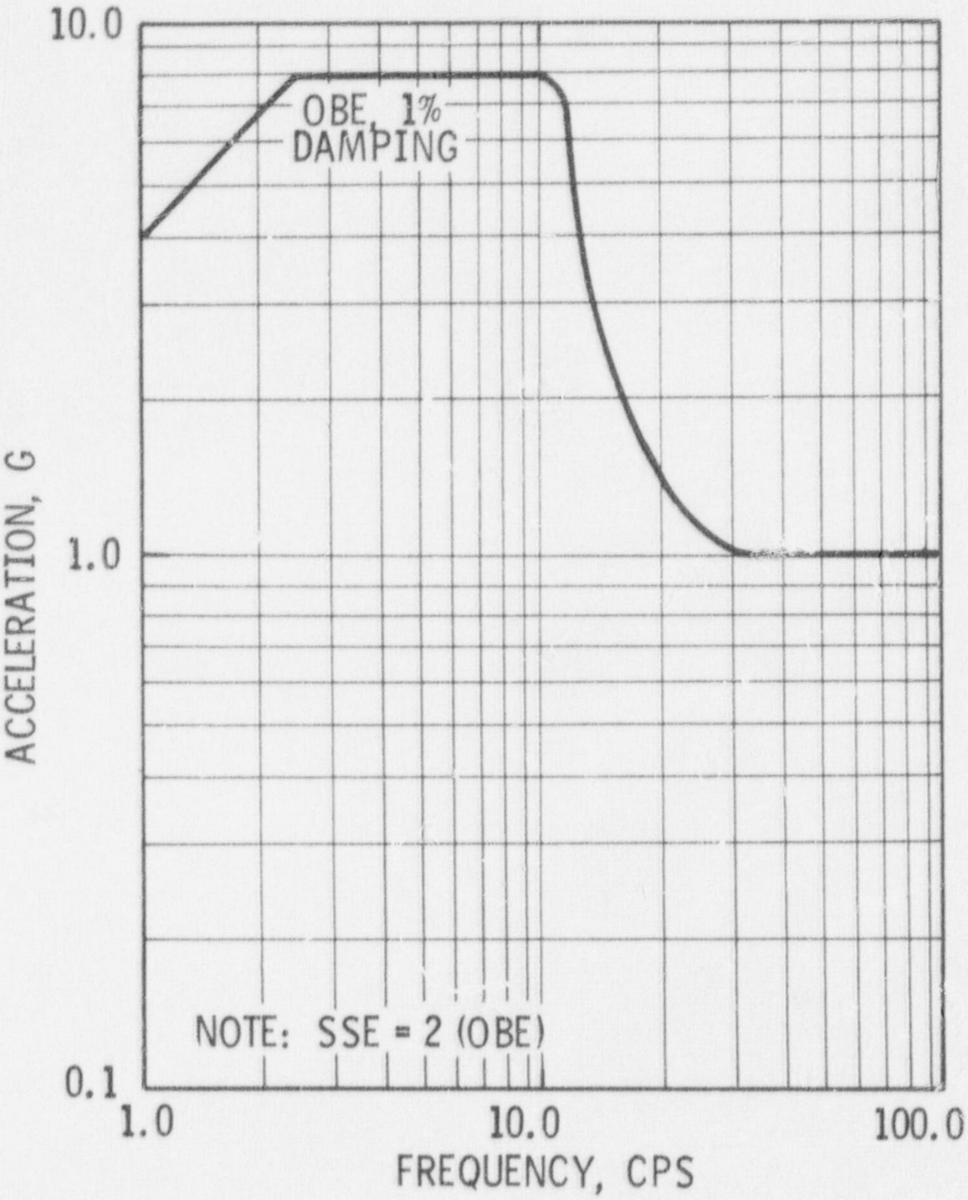
Amendment No. 1
February 20, 1981

C-E SYSTEM 80	RESPONSE SPECTRUM FOR UPPER REACTOR VESSEL SUPPORTS	Figure 3.7.1-1



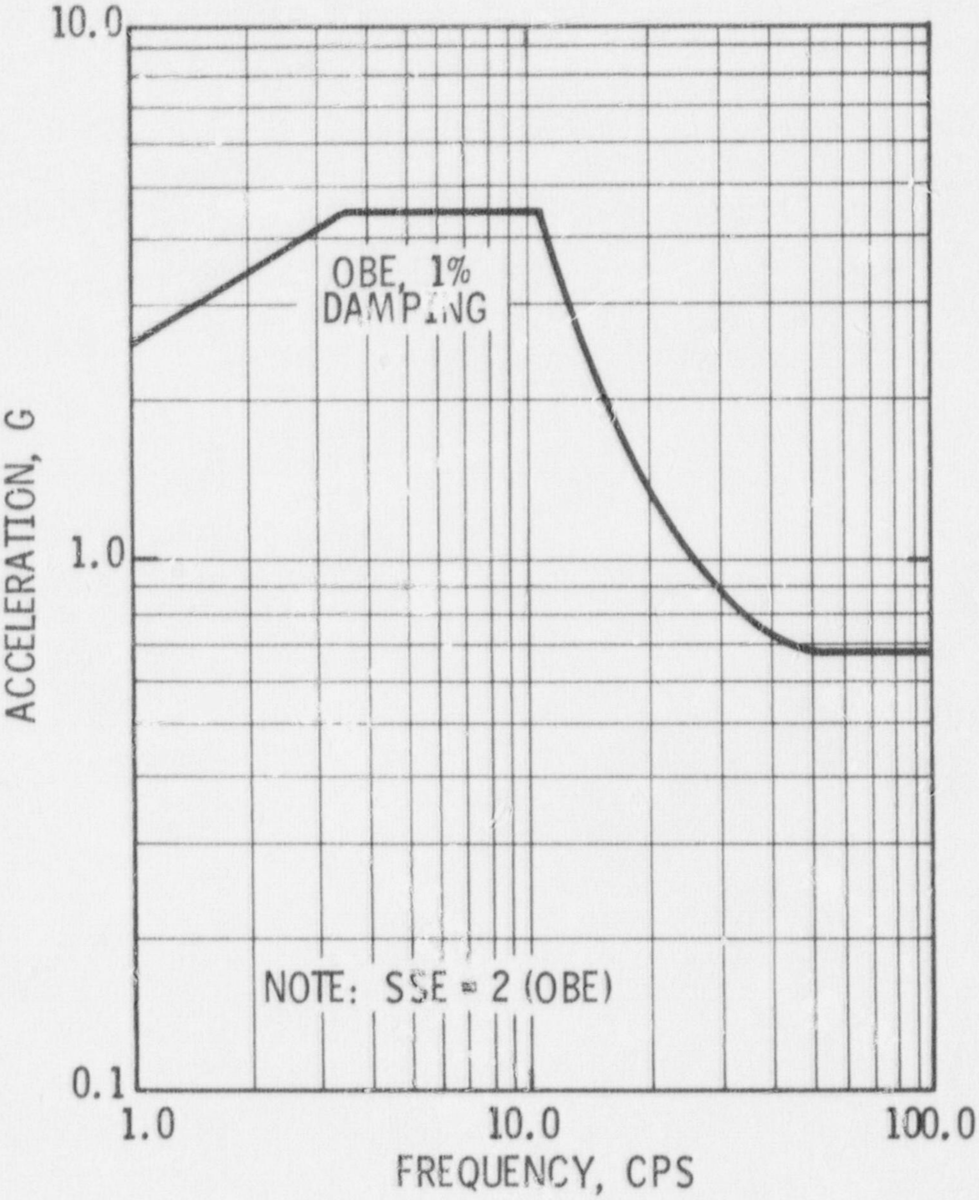
Amendment No. 1
February 20, 1981

C-E SYSTEM 80	RESPONSE SPECTRUM FOR UPPER STEAM GENERATOR SUPPORTS	Figure 3.7.1-2



Amendment No. 1
February 20, 1981

C-E SYSTEM 80	RESPONSE SPECTRUM FOR UPPER REACTOR COOLANT PUMP SUPPORTS	Figure 3.7.1-3



Amendment No. 1
February 20, 1981

C-E SYSTEM 80	RESPONSE SPECTRUM FOR VERTICAL SUPPORTS FOR ALL COMPONENTS	Figure 3.7.1-4

THIS PAGE INTENTIONALLY BLANK.

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2.1 Reactor Coolant System

3.7.2.1.1 Introduction

The major components of the Reactor Coolant System are designed to the appropriate stress and deformation criteria of the ASME Code, Section III for the set of loadings included in the component design specification. The adequacy of seismic loadings used for the design of the major components of the reactor coolnat system are confirmed by the methods of dynamic analysis employing time-history and response spectrum techniques. The major components are the reactor vessel, the steam generators, the reactor coolant pumps, the reactor coolant piping and the pressurizer.

In order to account for possible dynamic interaction effects between the components, a composite coupled model is employed in the dynamic analysis of the reactor vessel, the two steam generators, the four reactor coolant pumps and the interconnecting reactor coolant piping. The analysis of these dynamically coupled multisupported components utilizes different time dependent input excitations applied simultaneously to each support. Thus, the effect of differential seismic displacements on the equipment and supports is accounted for. In addition, the representation of the reactor vessel assembly used in this coupled model includes sufficient detail of the reactor internals to account for possible dynamic coupling from the reactor coolant system supports to the internals. The results of the analysis of the coupled components of the reactor coolant system include an appropriate time history forcing function used in a separate analysis of a more detailed model of the reactor internals.

A representation of the coupled components, of sufficient detail to account for possible dynamic coupling effects from the containment internal support structure to the reactor coolant system components, is supplied to the Applicant for performing the analysis of the containment internal support structure. The results of the analysis of the containment internal support structure include the time history forcing functions for use in the separate analysis of the more detailed model of the coupled components of the reactor coolant system.

The analyses of the pressurizer and the surge line piping employ separate, uncoupled, mathematical models and utilize either response spectrum or time-history techniques.

In the analyses, dynamic responses to vertical seismic excitation are combined with the responses to seismic excitations in two orthogonal horizontal directions. See Section 3.7.2.1.4 for methods of combination of results from excitations in three orthogonal directions.

The square root of the sum of the squares methods will normally be used to combine the modal responses when the response spectrum modal analysis method is employed. In those cases, however, where modal frequencies are closely spaced, the responses of the closely spaced modes will be combined by the sum of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the square root of the sum of the square method.

Contributions from all significant modes of response are retained in the analyses.

The damping factors used in seismic analysis of Category I structures,
systems and equipment are selected from Table 3.7.2-1. Modal damping
factors of 2 to 3 percent of critical and 1 to 2 percent of critical for
the SSE and OBE, respectively, are used in the seismic analysis of the
coupled components of the reactor coolant system. Modal damping factors 1
of 1 to 2 percent of critical and 1/2 to 1 percent of critical for the SSE and
OBE, respectively, are used in separate seismic analyses of branch runs of
piping for which C-E has responsibility for supply, such as the surge line
piping. The damping factors given in Table 3.7.2-1 include and are in
agreement with those recommended in Regulatory Guide 1.61.

The dynamic analyses of the major components of the reactor coolant system to confirm the seismic adequacy of the design are scheduled for completion such that a report of the results will be included in the Applicant's FSAR.

3.7.2.1.2 Mathematical Models

In the descriptions of the typical mathematical models which follow, the spatial orientations are defined by the set of orthogonal axes where Y is in the vertical direction and X and Z are in the horizontal plane, in the directions indicated on the appropriate figure. The mathematical representation of the section properties of the structural elements employs a 12 x 12 stiffness matrix for the three-dimensional space frame models, and employs a 6 x 6 stiffness matrix for the two-dimensional plane frame model. Elbows in piping runs include the in-plane/out-of-plane bending flexibility factors as specified in the ASME Code, Section III.

The methods used to combine the responses due to the different components of Earthquake Motion are described in Section 3.7.2.1.4 for the dynamic seismic system analysis, and in Section 3.7.3.6 for the seismic subsystem analysis.

Reactor Coolant System - Coupled Components

A schematic diagram of the typical composite mathematical model used in the analyses of the dynamically coupled components of the reactor coolant system is presented in Figure 3.7.2-1. This model includes 30 mass points with a total of 83 dynamic degrees of freedom. The mass points and corresponding dynamic degrees of freedom are distributed to provide appropriate representations of the dynamic characteristics of the components, as follows: the reactor vessel, with internals, is represented by 4 mass points with a total of 11 dynamic degrees of freedom; each of the two steam generators is represented by 4 mass points with a total of 10 dynamic degrees of freedom, each of the four reactor coolant pumps is represented by 2 mass points with a total of 6 dynamic degrees of freedom; and each branch of piping is represented by a mass point with 3 dynamic degrees of freedom. The representation of the reactor vessel internals is formulated in conjunction with the analysis of the reactor vessel internals discussed in Section 3.7.3.14, and is designed to simulate the dynamic characteristics of the models used in that analysis.

The mathematical model provides a three-dimensional representation of the dynamic response of the coupled components to seismic excitations in both the horizontal and vertical directions. The mass is distributed at the selected mass points and corresponding translational degrees of freedom are retained to include rotary inertial effects of the components. The toatl mass of the entire coupled system is dynamically active in each of the three coordinate directions.

Pressurizer

The mathematical model employed in the analysis of the pressurizer is shown schematically in Figure 3.7.2-2. This lumped parameter, 3-dimensional model provides a multimass representation of the pressurizer which includes 6 mass points with a total of 13 dynamic degrees of freedom.

Surge Line

The lumped parameter, multimass mathematical model employed in the analysis of a typical surge line is shown schematically in Figure 3.7.2-3. The surge line is modeled as a three dimensional piping run with end points anchored at the attachments to the pressurizer and the reactor vessel outlet piping. In the definition of this particular mathematical model, 10 mass points with a total of 27 dynamic degrees of freedom were selected to provide a three-dimensional representaion of the dynamic response of the surge line. All supports defined for the surge line assembly are included in the mathematical model. The total mass of the surge line is dynamically active in each of the three coordinate directions.

3.7.2.1.3 Calculations

General

The general matrix form of the undamped coupled equations of motion can be written (Reference 8) as follow:

$$M\ddot{X} + KU = F \qquad (1)$$

Where $\ddot{X}$ represents the absolute acceleration of the mass point dynamic degrees of freedom, and U represents the displacements of the mass and support point dynamic degrees of freedom relative to a datum support which is chosen to eliminate free body motion.

Expanding Equation (1) gives:

$$\begin{bmatrix} M_m & 0 \\ 0 & M_s \end{bmatrix} \begin{bmatrix} \ddot{X}_m \\ \ddot{X}_s \end{bmatrix} + \begin{bmatrix} K_{mm} & K_{ms} \\ K_{sm} & K_{ss} \end{bmatrix} \begin{bmatrix} U_m \\ U_s \end{bmatrix} = \begin{bmatrix} 0 \\ F_s \end{bmatrix} \qquad (2)$$

M_m = a diagonal submatrix of the system model lumped masses.

M_a = a submatrix of inertia terms associated with the support joints of the system. For the purposes of this analysis, $M_s = 0$, because there is no mass lumped at support joints.

F_s = the reaction forces at the system support points due to the response of the system to the motion of the supporting structure.

K = the stiffness matrix of the system model condensed in a manner such that only mass point elements (subscript m) and active, non-released, non-datum support elements (subscript s) remain in the matrix. (The method used for this analysis employs the choice of a datum support to eliminate free body motions.)

U_m = displacement of mass point dynamic-degrees-of-freedom relative to the datum support.

U_s = displacement of support points relatvie to the datum support.

$\ddot{X}_m$ = absolute acceleration of the mass point dynamic-degrees-of-freedom of the model.

$\ddot{X}_s$ = absolute acceleration of the system support points.

The first equation of the set of equations (2) yields:

$$M_m \ddot{X}_m + K_{mm} U_m + K_{ms} U_s = 0 \qquad (3)$$

A separation of variables can be achieved by defining the absolute motion of a mass point in terms of motion relative to the datum support, such that:

$$\ddot{U}_m = \ddot{X}_m - \gamma \ddot{X}_{sd} \qquad (4)$$

γ = a vector defining the direction of excitation, such that:

γ_i = 1, if the i^{th} dynamic-degree-of-freedom is in the direction of support motion, or

γ_i = 0, if the i^{th} dynamic-degree-of-freedom is not in the direction of support motion.

$\ddot{X}_{sd}$ = the absolute acceleration of the datum support in a given direction.

Equation (5) then becomes:

$$M_m \ddot{U}_m + K_{mm} U_m = -M_m \gamma \ddot{X}_{sd} - K_{ms} U_s \qquad (5)$$

At this point it is to be noted that the equations of motion are in a form expressing three dimensional response of the system mass points, due to multiple support excitations in a single direction. Methods for determination of the responses due to two or more directions of excitation are discussed later in this presentation.

Introducing the normal mode coordinate transformation:

$$U = \phi q, \tag{6}$$

where: ϕ = the matrix of eigenvectors,

then the equations of motion can be uncoupled and written in the following form:

$$\ddot{q} + 2\xi\omega\dot{q} + \omega^2 q = -(\phi^T M \phi)^{-1} (\phi^T M \gamma \ddot{X}_{sd} + \phi^T K_{ms} U_s), \tag{7}$$

where: ω^2 = diagonal matrix of eigenvalues,
ϕ = matrix of eigenvectors, and
$2\xi\omega$ = diagonal matrix of model damping terms.

Having the stiffness and mass properties of the model, the eigenvalue solution, and the digitized support excitations X_{sd} and U_s, eq (7) can be solved in closed form for the time histories of the mass point responses, U_m and X_m.

Frequency Analysis

An eigenvalue analysis is perfomed utilizing the ICES STRUDL II computer code, Reference 5, to calculate the mode shapes and natural frequencies of the composite mathematical models. Modifications to the standard ICES STRUDL II program have been implemented by Combustion Engineering to include a Jacobi diagonalization procedure in the eigenvalue analysis, and to provide appropriate influence coefficients and stiffness matrices for use in the response and reaction calculations.

The natural frequencies and dominant degrees of freedom calculated for typical systems are shown in Tables 3.7.2-2 and 3.7.2-3 for the Reactor Coolant System, the surge line and the pressurizer.

Mass Point Response Analysis

The time history of mass point responses to seismic excitation are computed using TMCALC, a C-E code. This code performs a closed form integration of the equations of motion for singly or multiply supported dynamic systems utilizing normal mode theory. For the multiply supported systems, the separate time histories of each support are imposed on the system simultaneously. The results are time history responses of the mass points.

The mass point responses resulting from spectrum analysis are found utilizing the ICES STRUDL II computer code. This code performs a normal mode response spectrum analysis resulting in the modal inertial loads at each mass point. The mass point responses of the pressurizer are found using the response spectrum for the pressurizer support. The mass point responses of the surge line are found using an envelope of the spectra for the surge line nozzles on the interconnected major components.

A description of the TMCALC code is given in Appendix 3A.

Seismic Reaction

The dynamically induced loads at all system design points due to the superimposed time history support excitations and mass point responses are calculated utilizing FORCE, a C-E computer code. This code performs a complete loads analysis of the deformed structure at each incremental time step by computing internal and external system reactions (forces and moments) by superposition of the reactions due to the mass point displacements and the non-datum support displacements.

A description of the FORCE code is given in Appendix 3A.

Influence coefficients for each desired reaction are computed for the effect of unit displacements of each mass point and each non-datum support. There is a complete set of mass point and support influence coefficients for each component of force, moment, stress, or displacement to be computed at the locations of interest throughout the system. The given support displacements and computed mass point displacements at each time step are multipled by the set of influence coefficients, to perform a complete reaction analysis of the system at each time step.

The desired components of reaction (force, moment, stress, or deflection) are computed at each time step as follows:

$$R(t) = C_m U_m(t) + C_s U_s(t), \qquad (8)$$

where:

$R(t)$ = a vector of reaction components at time t,

C_m = a matrix of mass point unit displacement influence coefficients (one column per mass point and one row per reaction component),

C_s = a matrix of non-datum support unit displacement influence coefficients (one column per non-datum support and on row per reaction component),

$U_m(t)$ = a vector of mass point relative displacements at time t, and

$U_s(t)$ = a vector of non-datum support relative displacements at time t.

In a similar manner, the absolute acceleration of any point in the system can be computed by multiplying the mass point and support relative accelerations by the influence coefficients for displacement reactions, and adding in the datum support absolute acceleration, as follows:

$$\ddot{R}(t) = \gamma \ddot{X}_{sd}(t) + C_m \ddot{U}_m(t) + C_s \ddot{U}_s(t), \qquad (9)$$

where:

$\ddot{R}(t)$ = a vector of absolute acceleration components at time t,

$\ddot{X}_{sd}(t)$ = the absolute acceleration of the datum support at time t,

C_m = a matrix of mass point unit displacement influence coefficients for components of displacement reactions,

$\ddot{U}_m(t)$ = a vector of mass point accelerations relative to the datum at time t,

C_s = a matrix on non-datum support unit displacement influenc coefficients for components of displacement reactions, and

$\ddot{U}_s(t)$ = a vector of non-datum support relative accelerations at time t.

γ = a vector defining the direction of excitation, such that:

γ_i = 1, if the i^{th} component-of-reaction is in the direction of support motion, or

γ_i = 0, if the i^{th} component-of-reaction is not in the direction of support motion.

This method, therefore permits the calculation of any desired force or moment, or non-mass point motion, on a time-history basis.

Using this linear superposition approach, the simultaneous results from two or more directions of excitation can be combined at each time step. Two or more sets of mass point displacement responses and support point displacement excitations are combined at each time step prior to the influence coefficient multiplications.

The support and mass point displacements due to both horizontal and the vertical seismic excitations are added at each time step. The maximum component of each reaction for the entire time domain, and its associated time of occurrence, are selected.

The maximum reactions for the pressurizer and surge line resulting from the response spectrum analysis are found by applying the modal inertial loads for each mode to the structural model using the STRUDL computer code. The design point reactions due to each modal loading are comboned by STRUDL by summing the absolute values and by Root-Sum-Square of the modal reactions, as appropriate. The surge line analysis includes consideration of the relative end displacements. The reactions found by statically imposing the maximum relative displacemetns of the two ends of the surge line are included with the inertial response from the spectrum analysis.

3.7.2.1.4 Results

The reaction (forces and moments) at all design points in the system, obtained from the dynamic seismic analysis, are compared with seismic loads in each component design specification. The results of this comparison are summarized in tabular form for the points of maximum calculated load.

The maximum response due to each of the three components of the earthquake motion are calculated separately on a time history basis and combined by the SRSS method.

When the three components of earthquake motion are statistically independent, the maximum responses are calculated by a simultaneous application of motion resulting from all three components of earthquake. In either case the maximum seismic loads calculated by the time history techniques are the result of a search and comparison over the entire time domain of each individual component of load. The maximum calculated components of load for each design location do not in general occur at the same time and therefore use results in a conservative worst case.

The maximum seismic loads calculated by the response spectrum techniques are the result of combining the modal reactions due to both horizonatal and vertical excitations. The method of modal combination is discussed in Section 3.7.2.5. The maximum responses due to each of the three earthquake components are then combined by the SRSS method.

3.7.2.1.5 Conclusion

It is concluded that the seismic loadings specified for the design of the reactor coolant system components and supports are adequate, when all seismic loads calculated by the dynamic seismic analysis are less than the corresponding loads in the component design specification.

3.7.2.2 Natural Frequencies and Response Loads

This section is provided in the Applicant's SAR.

3.7.2.3 Procedure Used for Modeling

This procedure used for modeling NSSS components and interconnecting piping are described in Section 3.7.2.1.2.

3.7.2.4 Soil/Structure Interaction

See Applicant's SAR.

3.7.2.5 Development of Floor Response Spectra 1

See Applicant's SAR.

3.7.2.6 Three Components of Earthquake Motion | 1

The procedures for considering the effects of three components of earthquake motion in determining the seismic response of NSSS systems, components and supports are discussed in Section 3.7.2.1.4.

3.7.2.7 Procedure for Combining Modal Responses | 1

The square root of the sum of the squares method is the procedure normally used to combine the modal responses when the modal analysis response spectum method of analysis is employed. The procedure is modified only in two cases:

THIS PAGE INTENTIONALLY BLANK

a. In the analysis of simple system where three or less dynamic degrees-of-freedom are involved, the modal responses are combined by the summation of the absolute values method;

b. In the analysis of complex systems where closely spaced modal frequencies are encountered, the responses of the closely spaced modes are combined by the summation of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the square root of the sum of the squares method. Modal frequencies are considered closely spaced when their difference is less than ± 10 percent of the lower frequency.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structure

See Applicant's SAR.

3.7.2.9 Effects of Parameter Variations in Floor Response Spectra 1

See Applicant's SAR.

3.7.2.10 Use of Constant Vertical Static Factors

A constant seismic vertical load factor is not used for the seismic design of Seismic Category I structures, components and equipment.

3.7.2.11 Torsional Effects of Eccentric Masses 1

The mathematical models used in seismic analysis of Category I systems, components, and piping systems include sufficient mass points and corresponding dynamic degrees-of-freedom to provide a three-dimensional representation of the dynamic characteristics of the system. The distribution of mass and the selected location of mass points account for torsional effects of valves and other eccentric masses.

3.7.2.12 Comparison of Responses 1

See Applicant's SAR.

3.7.2.13 Methods for Seismic Analysis of Dams 1

See Applicant's SAR.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

See Applicant's SAR.

3.7.2.15 Analysis Procedure for Damping

Uniform modal damping factors given in Section 3.7.2.1.1 are used in the analysis of the coupled components of the reactor coolant system. 1

Procedures for accounting for system damping are included in Section 3.7.2.1.3.

TABLE 3.7.2-1

DAMPING VALUES USED IN ANALYSIS OF CATEGORY 1

STRUCTURES, SYSTEMS AND COMPONENTS

Item	Maximum Allowable Damping percent of critical viscous damping: Operational Basis Earthquake	Maximum Allowable Damping percent of critical viscous damping: Safe Shutdown Earthquake
Equipment and large diameter piping systems, pipe diameter greater than 12 inches	2	3
Small diameter piping systems, diameter less than or equal to 12 inches	1	2
Welded steel structures	2	4
Bolted steel structures	4	7

1

TABLE 3.7.2-2

(Sheet 1 of 2)

NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM

REACTOR COOLANT SYSTEM

Mode Number	Frequency (CPS)	Dominant Degrees of Freedom: Joint Number	Direction	Locations
1	1.75	9911	Z	Reactor Internals
2	1.74	9911	X	Reactor Internals
3	12.18	9916	Z	R.V. Top Mass
4	13.06	9916	X	R.V. Top Mass
5	17.43	404, 3404	X	S.G Top Masses
6	17.54	4103, 2103, etc.	X	Top Masses of all RCP
7	17.63	2103, 4103	X	Top Masses of RCP 1B & 2B
8	17.67	1103, 5103	X	Top Masses of RCP 1A & 2A
9	17.83	3408, 408	Z	S.G. Internals
10	17.83	408, 3408	Z	S.G. Internals
11	17.94	4103, 2103	X	Top Masses of RCP 1B & 2B
12	18.00	5103, 1103	X	Top Masses of RCP 1A & 2A
13	19.76	9995	Z	R.V. Lower Mass
14	20.23	9911	Y	Reactor Internals
15	21.02	2103, 4103, etc.	Z	Top Masses for all RCP
16	21.02	5103, 4103, etc.	Z	Top Masses for all RCP
17	21.02	1103, 5103	Z	Top Masses of RCP 1A & 2A
18	21.02	2103, 4103	Z	Top Masses for RCP 1B & 2B
19	22.34	5103, 1103, etc.	Y	All RCP
20	22.36	1103, 5103, etc.	Y	All RCP
21	22.36	4103, 2103, etc.	Y	All RCP
22	22.36	2103, 4103, etc.	Y	All RCP
23	23.11	9905, 9995	X	RV Internals & Externals
24	24.16	404, 3404	Y	S.G. Externals
25	25.23	3404, 404	Y	S.G. Externals
26	24.55	408, 3408	X	S.G. Internals
27	25.89	9905	X	R.V. Internals
28	29.37	404, 3404	Z	S.G. Top Masses
29	29.37	3404, 404	Z	S.G. Top Masses
30	30.05	2580, 4580	Z, Y	C.L. Piping
31	32.12	4580, 2580	Z, X	C.L. Piping
32	32.40	9911	Y	Reactor Internals
33	32.46	1580, 5580	Z, X	C.L. Piping
34	32.53	5580, 1580	Z, X	C.L. Piping
35	36.14	5580, 1580	X	C.L. Piping
36	36.40	4580, 2580	X	C.L. Piping
37	36.44	5580, 1580	X	C.L. Piping
38	36.51	2580, 4580	X	C.L. Piping
39	39.41	2580, 4580	X	C.L. Piping
40	39.79	1580, 5580	X	C.L. Piping

TABLE 3.7.2-2 (Cont'd.) (Sheet 2 of 2)

NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM

REACTOR COOLANT SYSTEM

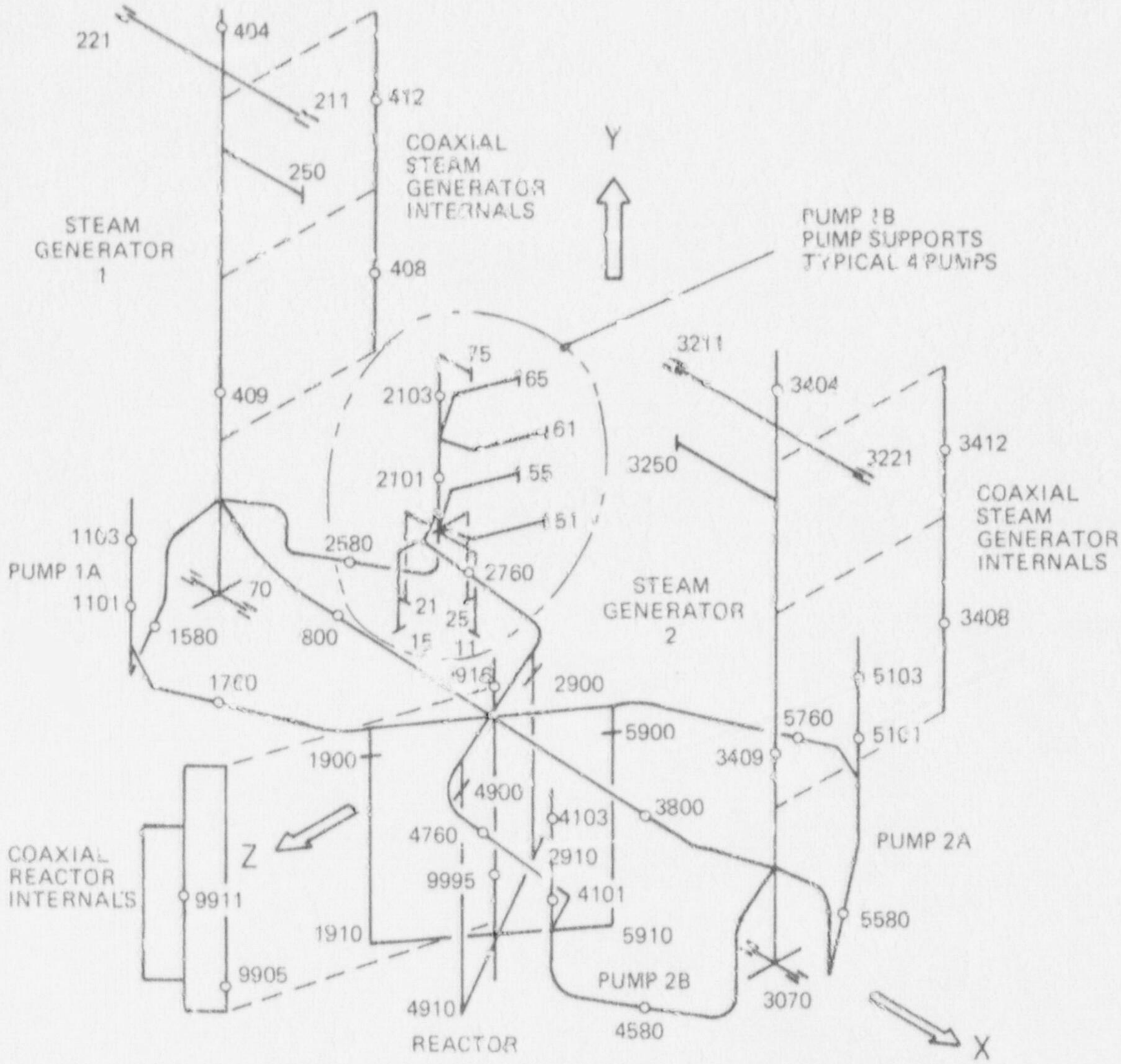
Mode Number	Frequency (CPS)	Dominant Degrees of Freedom: Joint Number	Direction	Locations
41	39.81	4580, 2580	X, Z	C.L. Piping
42	39.91	1580, 5580	X, Z	C.L. Piping
43	41.78	5580, 1580	X, Z	C.L. Piping
44	44.99	9995	X	R.V. Lower Mass
45	47.85	412, 3412, etc.	X	S.G. Internals
46	48.04	3412, 412, etc.	X	S.G. Internals
47	48.39	412, 3412, etc.	Z	S.G. Internals
48	48.39	3412, 412, etc.	Z	S.G. Internals
49	51.05	412, 3412	X	S.G. Internals
50	51.11	3412, 412	X	S.G. Internals

TABLE 3.7.2-3

NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM

PRESSURIZER AND SURGE LINE

Mode Number	Frequency (CPS)	Dominant Degrees of Freedom: Joint Number	Direction	Locations
1	29.17	135	X	Pressurizer
2	29.19	135	Z	Pressurizer
3	40.05	135	Y	Pressurizer
4	54.68	135	X	Pressurizer
5	54.71	135	Z	Pressurizer
6	116.31	110	X	Pressurizer
7	116.34	110	Z	Pressurizer
8	185.72	135	X	Pressurizer
9	185.81	135	Z	Pressurizer
10	221.80	110	X	Pressurizer
11	221.84	110	Z	Pressurizer
12	307.48	125	X	Pressurizer
13	307.48	125	Z	Pressurizer
1	5.75	7, 8, H3	Y	Surge Line
2	11.29	5, 7, H1	X	Surge Line
3	1[illegible].98	5, H1	Y	Surge Line
4	21.79	8, 9, H3	Z	Surge Line
5	25.81	9, H3	Y	Surge Line
6	32.12	8, 9	X	Surge Line
7	40.35	10	X	Surge Line
8	73.64	5	Z	Surge Line
9	108.52	3	X	Surge Line
10	129.12	10	Y	Surge Line
11	151.51	7, H3	X	Surge Line
12	155.03	7	Y	Surge Line
13	174.76	7, H1	Y	Surge Line
14	186.39	H1	X	Surge Line
15	229.30	10, 11	Z	Surge Line
16	260.38	11	X	Surge Line
17	304.86	3	Y	Surge Line
18	320.52	7	X	Surge Line
19	503.60	H1	X	Surge Line
20	525.21	8	Y	Surge Line
21	535.87	8	X	Surge Line
22	542.64	11	Z	Surge Line
23	668.06	4	X	Surge Line
24	752.11	4	Z	Surge Line
25	938.66	11	Y	Surge Line
26	1327.41	8	Z	Surge Line
27	1807.56	4	Y	Surge Line

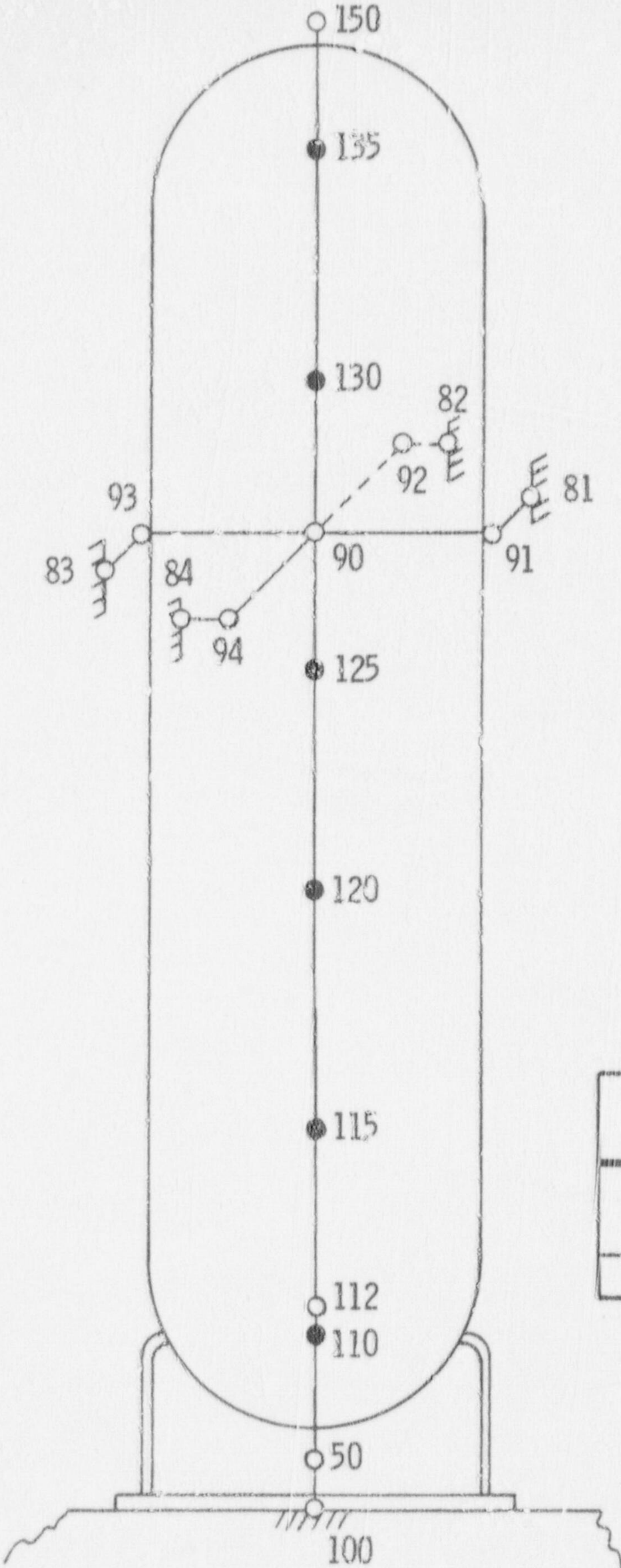


COMPONENT NAME	MASS POINT NUMBER	DEGREES OF FREEDOM
REACTOR	9916, 9911, 9905 9995	X, Y, Z X, Z
STEAM GENERATORS	404, 3404, 412, 3412 409, 3409 408, 3408	X, Y, Z X, Z X, Z
PUMPS	1103, 2103, 4103, 5103 1101, 2101, 4101, 5101	X, Y, Z X, Y, Z
REACTOR COOLANT PIPING	800, 3800 1760, 2760, 4760, 5760 1580, 2580, 4580, 5580	Y, Z X, Y, Z X, Y, Z

COMPONENT NAME	SUPPORT POINT NUMBER	RESTRAINT
REACTOR	1900, 2900, 4900, 5900 1910, 2910, 4910, 5910	FX, FZ FIXED
STEAM GENERATORS	70, 3070 250, 3250 211, 221 3211, 3221	FY, FZ FX FZ FZ
REACTOR COOLANT PUMPS (TYPICAL)	11, 15, 21, 25 51, 55, 61, 65 75	FY FX, FZ FX, FZ

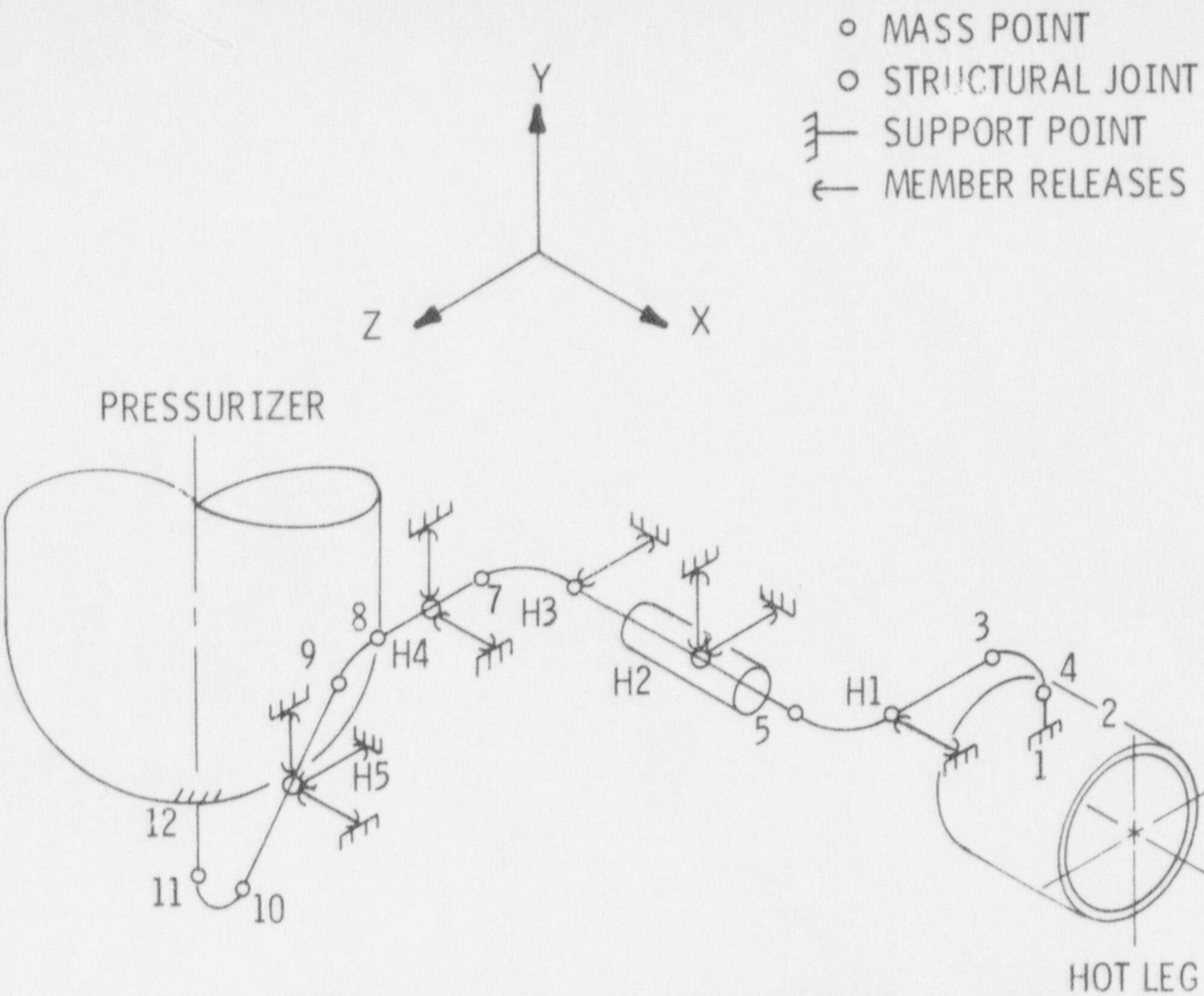
Amendment No. 21
March 3, 1975

C-E SYSTEM 80	TYPICAL REACTOR COOLANT SYSTEM SEISMIC ANALYSIS MODEL	Figure 3.7.2-1



MASS POINT	DEGREE OF FREEDOM
110, 115, 120 125, 130	X, Z
135	X, Y, Z

C - E SYSTEM 80	TYPICAL PRESSURIZER SEISMIC ANALYSIS MODEL	Figure 3.7.2-21



MASS POINT	DEGREE OF FREEDOM
3, 7	X, Y
H3	Z, Y
4, H1, 5, 8, 9, 10, 11	X, Y, Z

SUPPORT NAME	SUPPORT RESTRAINT DIRECTIONS
H1	Fx
H2	Fy, Fz
H3	Fz
H4	Fx, Fy
H5	Fx, Fy, Fz

C-E SYSTEM 80

TYPICAL SURGE LINE SEISMIC ANALYSIS MODEL

Figure 3.7.2-3

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

The discussion presented in this section describes procedures employed in seismic subsystem analysis.

3.7.3.1 Seismic Analysis Methods

Analysis methods used for the major components of the reactor coolant system, defined and analyzed as a "seismic system", are described in Section 3.7.2.

Seismic analysis methods used for the reactor internals, fuel assemblies, and control element drive mechanism are given in Section 3.7.3.14. 6

Mechanical systems and components and Category I instrumentation and electrical equipment supplied by the NSSS vendor are analyzed in accordance with the methods described in Sections 3.9 and 3.10, respectively.

3.7.3.2 Determination of Number of Earthquake Cycles

The procedure used to account for the fatigue effect of cyclic motion associated with the OBE recognizes that the actual motion experienced during a seismic event consists of a single maximum or peak motion, and some number of cycles of lesser magnitude. The total or cumulative fatigue effect of all cycles of different magnitude will result in an equivalent cumulative usage factor. The equivalent cumulative usage factor can also be specified in terms of a finite number of cycles of the maximum or peak motion. Based on this consideration, Seismic Category I subsystems, components, and equipment are designed for total of 200 full-load cycles about a mean value of zero and with an amplitude equal to the maximum response produced during the entire OBE event.

3.7.3.3 Procedure Used For Modeling

Modeling of reactor internals, core, and control element drive mechanisms is described in Section 3.7.3.14. Modeling procedures used for analysis of NSSS vendor supplied auxiliary components are given in Section 3.9.3. 6

3.7.3.4 Basis for Selection of Forcing Frequencies

The basis for acceptability of the seismic design of equipment and subsystems is that the stresses and deformations produced by vibratory motion of the postulated seismic events, in combination with other coincident loadings, be within the limits established by applicable codes and standards in Section 3.9.3 of CESSAR. 6

Within practical limitations, the seismic design is accomplished in a manner to maintain the resonant frequencies well above the range which is significantly excited by the forcing frequencies. If the stresses and deformations resulting from analysis of the preliminary design exceed the established acceptable limits the stiffness of the restraint and supports system is modified as required to maintain the fundamental frequencies of equipment and subsystems sufficiently removed from the resonant range and, thereby, maintain the seismic response within the loads given in the component design specifications. The subsystem supports design is 6 sufficiently adaptable that, dependent on the quantitative change in frequency required and the subsystem involved, modifications can be made either by changing the stiffness of existing support assembly components or by adding additional support system restraints to the subsystems or components whose response otherwise exceeds the established limits.

If, during the analysis of the preliminary design, frequencies of the reactor coolant system were found to be in the range of resonance with those of the building, the supports for each of the components could be modified to increase their natural frequencies.

Specifically, the fundamental frequencies of the reactor vessel can be increased in both horizontal directions by the welding of a set of keys to the RV to further restrain lateral motion or rotation of the vessel. The keys would be laterally restrained by a structure supported by the primary shield wall.

The RCP moves in all three directions when seismically excited in any one direction. The fundamental frequency of the RCP can be raised by relocating the snubber from the top of the motor mount to the top of the motor. The orientation of the snubber would remain unchanged.

The SG frequency can be raised in the direction parallel to the axis of the RV outlet piping by the addition of a second set of snubbers and levers and in the direction perpendicular to the axis of the RV outlet piping by an additional set of keys above the original set.

See Section 3.9.3.1.3.1 for auxiliary components.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

The equivalent static load method is limited to analysis of components which can be realistically represented as single-degree-of-freedom systems or by simple beam or frame type models. This method involves the multiplication 6
of the total weight of the equipment or component member by the specific seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components that can be adequately characterized as a single-degree-of-freedom system are considered to have a modal participation factor of one. Seismic acceleration coefficients for simple multi-degree of 6

freedom systems, which may be in the resonance region of the amplified response spectrum curves, are increased by 50% to account conservatively for the increased modal participation.

3.7.3.6 Three Components of Earthquake Motion

Procedures for considering the effects of three components of earthquake motion in determining the seismic response of NSSS vendor supplied seismic subsystems and components are in accordance with Regulatory Guide 1.92.

Section 3.7.3.14 discusses the procedures used in the analysis of reactor internals, fuel assemblies, and control element drive mechanisms. Procedures for considering the effects of three components of earthquake motion for auxiliary components are provided in Section 3.9.3.1. 6

3.7.3.7 Combination of Modal Response

See Section 3.7.2.7.

3.7.3.8 Analytical Procedures for Piping

The interconnecting piping of the major components of the reactor coolant system is included in the seismic system analysis described in Section 3.7.2. 6

3.7.3.9 Multiply Supported Equipment Components With Distinct Inputs

The criteria and procedures used for seismic analysis of the multiply supported major components of the reactor coolant system are described in Section 3.7.2; analysis methods used for the reactor internals and fuel assemblies are given in Section 3.7.3.14.

Other seismic subsystems supported at two or more locations are analyzed using an upper bound envelope of all individual support response spectra to calculate maximum inertial responses. Responses due to relative support displacements, imposed on the supported subsystem in the most unfavorable combination, are then combined with the responses due to inertial effects by the absolute sum method. 6

3.7.3.10 Use of Constant Vertical Static Factors

See Section 3.7.2.10.

3.7.3.11 Torsional Effects of Eccentric Masses

See Section 3.7.2.11.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

See Applicant's SAR.

3.7.3.13 Interaction of Other Piping With Category I Piping

See Applicant's SAR. There are no CESSAR interfaces with non-Category I piping. 6

3.7.3.14 Seismic Analysis of Reactor Internals, Core and CEDMs

3.7.3.14.1 Reactor Internals and Core

The seismic analyses of the reactor internals and core consists of two phases. In the first phase, linear lumped-parameter models are formulated, natural frequencies and mode shapes for the models are determined, and the response is obtained utilizing the modal analysis response spectrum method. The response spectra used are based upon the acceleration of the reactor vessel flange. The response spectrum analysis is used to obtain preliminary design seismic loads and displacements in the vertical and horizontal directions.

Amendment No. 6
November 20, 1981

In the second phase, because the relative displacements between the core and core shroud and between the core-support barrel and pressure-vessel snubbers are sufficiently large to close the gaps that exist between these components, a nonlinear horizontal time history analysis is performed. 6 The horizontal nonlinear analysis is divided into two parts. In the first part, the internals and core are analyzed to obtain the internals response and the proper dynamic input for the reactor core model. In the second part, the core plate motion from the first part is applied to a more detailed nonlinear model of the reactor core. The input excitation to the internals model is the response time-history of the reactor vessel at the internals support determined from the RCS analysis. Coupling effects between the internals and reactor vessel are accounted for by including a simplified representation of the internals with the RCS model. This is discussed in subsection 3.7.2. When the linear vertical analysis indicates that the response of the core may be sufficiently large to cause it to lift off the core plate, a vertical nonlinear analysis of the internals is also performed. If this method is used a statement will be provided in the Applicant's SAR that a nonlinear analysis was performed and that the results were acceptable. 6

In these analyses, two horizontal components and one vertical component of the seismic excitation are considered and the maximum responses for the three components are combined by the method of square root of the sum of the squares.

Closely spaced modes are considered in accordance with Regulatory Guide 1.92. 6

3.7.3.14.1.1 Mathematical Models

Equivalent multimass mathematical models are developed to represent the reactor internals and core. The linear mathematical models of the internals are constructed in terms of lumped masses and elastic-beam elements. At appropriate locations within the internals and core, points (nodes) are chosen to lump the weights of the structure. A sketch of the internals and core showing the relative node locations for the horizontal model is presented in Figure 3.7.3-1. Figures 3.7.3-2 and 3.7.3-3 show the idealized linear horizontal and linear vertical models. The criterion for choosing the number and location of mass concentration is to provide for accurate representation of the dynamically significant modes of vibration of each of the internals components. Between the nodes properties are calculated for moments of inertia, cross-section areas, effective shear areas, and lengths. Separate horizontal and vertical models of the internals and core are formulated to more efficiently account for structural differences in these directions. In the horizontal nonlinear lumped mass representation of the internals and core, shown in Figure 3.7.3-4, gap and spring elements are used to represent contact between the fuel and core shroud. Lumped-mass nodes in the core are positioned to coincide with fuel-spacer grid locations. To simulate the nonlinear motion of the fuel, nonlinear spring couplings are used to connect corresponding nodes to the fuel assemblies and core shroud. Incorporated into these nonlinear springs is the spacer grid impact stiffness derived from test results. The core is modeled by subdividing it into fuel assembly groupings and choosing stiffness values to adequately characterize its beam response and contacting under dynamic loading.

Amendment No. 6
November 20, 1981

The horizontal nonlinear reactor core model consisting of one row of 17
individual fuel assemblies is depicted in Figure 3.7.3-5. In this model
each fuel assembly is represented with mass points located at spacer grid 6
locations. To simulate the gaps in the core, nonlinear spring couplings
are used to connect corresponding nodes on adjacent fuel assemblies and
core shroud. The impact stiffness and impact damping (coefficient of
restitution) parameters for the gap elements are derived from the impact
tests which are described in Section 4.2. The spacer grid impact representation
used for the analysis is capable of representing two types of fuel assembly
impact situations. In the first type, only one side of the spacer grid is
loaded. This type of impact occurs when the peripheral fuel assembly hits
the core shroud, or when two fuel assemblies strike one another. The second 6
type of impact loading occurs typically when the fuel assemblies pile up on
one side of the core. In this case, the spacer grids are subjected to a
through-grid compressive loading.

The fuel assemblies in the coupled core/internals model and the detailed
core model are modeled with beam elements to represent the horizontal
stiffness between mass points and rotational springs at each end to simulate
the end fixity existing at the top and bottom of the core. The value used
for fuel horizontal stiffness and end fixity are based upon a parametric
study in which analytic predictions are correlated with fuel assembly
static and dynamic test data. Fuel assembly structural damping as a function
of vibrational amplitude was derived from fuel assembly forced vibration
and pluck tests defined in Section 4.2. The damping values used in the
seismic analysis of the reactor internals are in accordance with the values
in Table 3.7.2-1. 6

The vertical nonlinear model incorporates nonlinear spring couplings to
account for the nonlinear behavior of the internals in the vertical direction.
The vertical nonlinear model is shown in Figure 3.7.3-8. 6

Additional salient details of the internals and core models are discussed
in the following paragraphs.

A. Hydrodynamic Effects

It has been shown both analytically and experimentally[9] that imersion of a body in a dense-fluid medium lowers its natural frequency and significantly alters its vibratory response as compared to that in air. The effect is more pronounced where the confining boundaries of the fluid are in clos proximity to the vibrating body as in the case for the reactor internals. The method of accounting for the effects of a surrounding fluid on a vibrating system has been to ascribe to the system additional or "hydrodynamic mass".

The hydrodynamic mass of an immersed system is a function of the dimensions of the real mass and the space between the real mass and confining boundary.

Hydrodynamic mass effects for moving cylinders in a water annulus are discussed in Reference 9 and 10. The results of these references are applied to the internals structures to obtain the total (structural plus hydrodynamic) mass matrix that is then used in the evaluation of the natural frequencies and mode shapes.

B. Core Support Barrel

The core support barrel is modeled as a beam with shear deformation. It has been shown that the use of beam theory for cylindrical shells gives sufficiently accurate results when shear deformation is included[11][12].

C. Fuel Assemblies

The fuel assemblies are modeled as uniform beams with rotational springs at each end to represent the proper end condition. The member properties for the beam elements representing the fuel assemblies are derived from the results of experimental tests of the fuel-assembly load deflection characteristics and fundamental natural frequency.

D. Support-Barrel Flanges

To obtain accurate lateral and vertical stiffness of the upper and lower core-support barrel flanges and the upper guide structure support barrel upper flange, finite-element analyses of these regions are performed. As shown in Figure 3.7.3-6 these areas are modeled with quadrilateral and triangular ring elements. Unit deflections and rotations are applied in the lateral and axial directions, and the resulting reaction forces are calculated. These results are then used to derive the equivalent member properties for the flanges.

E. Upper Guide Structure

For the horizontal model, the upper guide structure including CEA shrouds, connecting plates and tie rods are modeled as contilever beams. A separate member is modeled to account for the connection between the tie rods and the upper guide structure support plate.

F. Lower Support Structure

To obtain vertical stiffnesses for the lower support structure grid beams and cylinder, a finite element analysis is performed. A top view of the finite element model is shown in Figure 3.7.3-7. Displacements due to vertical (out-of-plane) loads applied at the beam junctions are calculated through the use of a computer program (13). Average stiffness values based ont these results yield an equivalent member cross-section area for the vertical model.

3.7.3.14.1.2 Analytical Techniques.

Natural Frequencies and Mode Shapes

The mass- and beam-element properties of the models are utilized in a computer program to obtain the natural frequencies and mode shapes. This computer code is described in Section 3.9.1.2.2.7. The program utilizes the stiffness-matrix method of structural analysis. The natural frequencies and mode shapes are extracted from the system of equations:

$$(\underline{K} - W_n^2 \underline{M})\ \phi_n = 0 \qquad (12)$$

where:

$\underline{K}$ = model stiffness matrix

$\underline{M}$ = model mass matrix

W_n = natural circular frequency for the n^{th} mode

ϕ_n = normal mode shape matrix for n^{th} mode

The mass matrix, $\underline{M}$, includes the hydrodynamic and structural masses.

B. Response Calculations Methods

1. Response Spectra Method

The response spectrum analysis is performed using the modal extraction data and the following relationships for each mode:

a. Nodal Accelerations

$$\ddot{X}_{in} = \gamma_n A_n \phi_{in} \qquad (13)$$

where:

$\ddot{X}_{in}$ = absolute acceleration at node "i" for node "n"

γ_n = modal participation factor

A_n = modal acceleration from response spectrum

ϕ_{in} = mode shape factor at node "i" for node "n"

b. Nodal Displacement

$$Y_{in} = \frac{\ddot{X}_{in}}{W_n^{\ 2}} \qquad (14)$$

where:

Y_{in} = displacement at node "i" for mode "n" relative to base

W_n = natural circular frequency for n^{th} mode

c. Member Forces and Moments

$$F_n = \frac{(\gamma_n A_n)}{W_n^2} \overline{F}_n \qquad (15)$$

where:

F_n = actual member force for mode "n"

$\overline{F}_n$ = modal member force for mode "n"

The effect of the fluid environment is accounted for by defining the modal participation as follows: 6

$$\gamma_n = \frac{\sum_{j=1}^{M} W_{si} \phi_{in}}{\sum_{i=1}^{M} \sum_{j=1}^{M} \phi_{in} W_{ij} \phi_{jn}} \qquad (16)$$

where:

W_{si} = structural weight of node "i"

W_{ij} = structural + hydrodynamic weight terms

M = number of masses

The SRSS method is normally used to combine the modal responses. Where modal frequencies are closely spaced, the responses of these modes are combined by the sum of their absolute values. The modal damping factors are obtained by the method of "mass mode weighting", which gives:

$$B_n = \frac{\Sigma M_{\underline{i}} \quad \phi_{in} \quad B_{\underline{i}}}{\Sigma M_{\underline{i}} \quad \phi_{in}} \qquad (17)$$

where:

B_n = modal damping factor

M_i = structural mass of mass node "i"

ϕ_{in} = absolute value of the mode shape at mass node "i"

B_i = damping associated with mass point "i"

Nonlinear Analysis

The nonlinear seismic response and impact forces for the internals and fuel are determined using the CESHOCK computer program (refer Section 3.9.1.2). The computer program provides the numerical solution to transient dynamic problems by step-by-step integration of the differential equations of motion. The input excitation for the model is the time-history accelogram of the reactor vessel.

Input to the CESHOCK computer program consists of initial conditions, nodal lumped masses, linear-spring coefficients, mass moments of inertia, nonlinear spring curves, and the acceleration time-histories. The output from the CESHOCK computer program consists of displacements, translational and angular accelerations, impact forces, shears, and moments.

3.7.3.14.1.3 Results

The nonlinear response loads for the internals, including impacting if any exist, are determined for the vertical and horizontal directions. Loads for the fuel are determined in a separate reactor core nonlinear analysis. The results are determined for the Safe Shutdown Earthquake (SSE) and the Operational Basis Earthquake (OBE).

3.7.3.14.2 Control Element Drive Mechanisms (CEDM)

The pressure-retaining components of the CEDM are designed to the appropriate stress criteria of ASME Code Section III for all loadings specified. The structural integrity of the CEDM when subjected to seismic loadings is verified by combination of test and analysis. Methods of modal dynamic analysis employing response spectrum techniques or time history analysis are supported with experimentally obtained information.

3.7.3.14.2.1 Input Excitation Data

For the dynamic analyses, a response spectra or time history definition of the excitation at the base of the CEDM nozzle is obtained from the seismic analysis of the RCS. The excitation is applied simultaneously in three mutually perpendicular directions (2 horizontal and 1 vertical).

3.7.3.14.2.2 Analysis

A dynamic analysis of the mathematical structural model is performed utilizing one or more of the computer programs discussed in Section 3.9.1.2.

3.7.3.14.2.3 Tests

A functional test ultilizing a minimum drop weight is performed to verify that drop characteristics meet the input design requirements. Results from this test are compared to the calculated CEDM deflections under seismic loading for the individual site. Verification of the proper function is thus established based on both analytical and test results.

3.7.3.15 Analysis Procedure for Damping

The procedures used to account for damping in the analysis of the reactor internals and core are given in Section 3.7.3.14. Uniform modal damping factors are used in the analysis of other NSSS vendor supplied seismic subsystems. 6

REFERENCES FOR SECTION 3.7.3.14

1. Newmark, N.M. and Hall, W.J., "Seismic Design Criteria for Nuclear Reactor Facilities," Proceedings, 3rd World Conference on Earthquake Engineering Vol. 2, Santiago, Chile, 1969.

2. Jennings, P.C., Housner, G.W., and Tsai, N.C., "Simulated Earthquake Motions," Earthquake Engineering Research Laboratory, California Institute of Technology, April 1968.

3. Tsai, N.C., "Spectrum-Compatible Motions for Design Purposes," Journal of Engineering Mechanics Division, ASCE, Vol. 98 EM2, April 1972.

4. McNeill, R.L., "Machine Foundations: The State-of-the-Art," Proceedings of Specialty Sessions 2, Seventh International Conference on Soil Mechanics and Foundation Engineering, Mexico City, Mexico, August 1969.

5. "Seismic Analyses of Structures and Equipment for Nuclear Power Plants, "BC-TOP-4A, Revision 3, Bechtel Power Corporation, San Francisco, California, November 1974.

6. "Methods of Direct Application of Element Damping, San Onofre Units 2 and 3," Bechtel Power Corporation, Los Angeles office, January 1972.

7. Przemieniecki, J.S., "Theory of Matrix Structural Analysis," Chapter 13, Mc-Graw-Hill Book Company, New York, New York, 1968.

8. ICES STRUDL-II, "The Structural Design Language Engineers Users Manual," MIT Press, Cambridge, Mass., 1968.

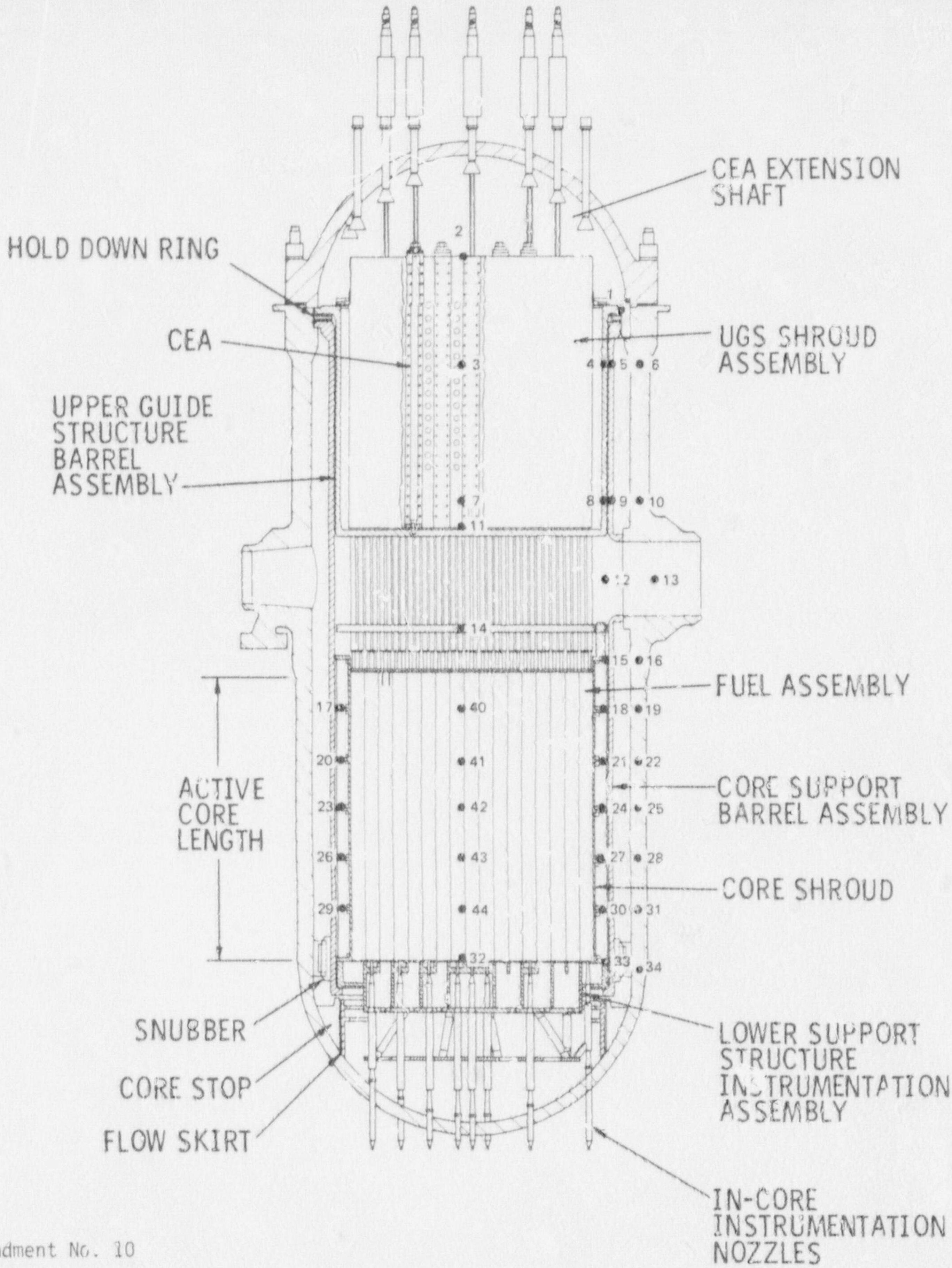
9. Fritz, R.J., "The Effect of Liquids on the Dynamic Motions of Immersed Solids", Journal of Engineering for Industry, Paper No. 71-VIB-100.

10. McDonald, C.K., "Seismic Analysis of Vertical PVMPs Enclosed in Liquid Filled Containers", ASME Paper No. 75-PVP-56.

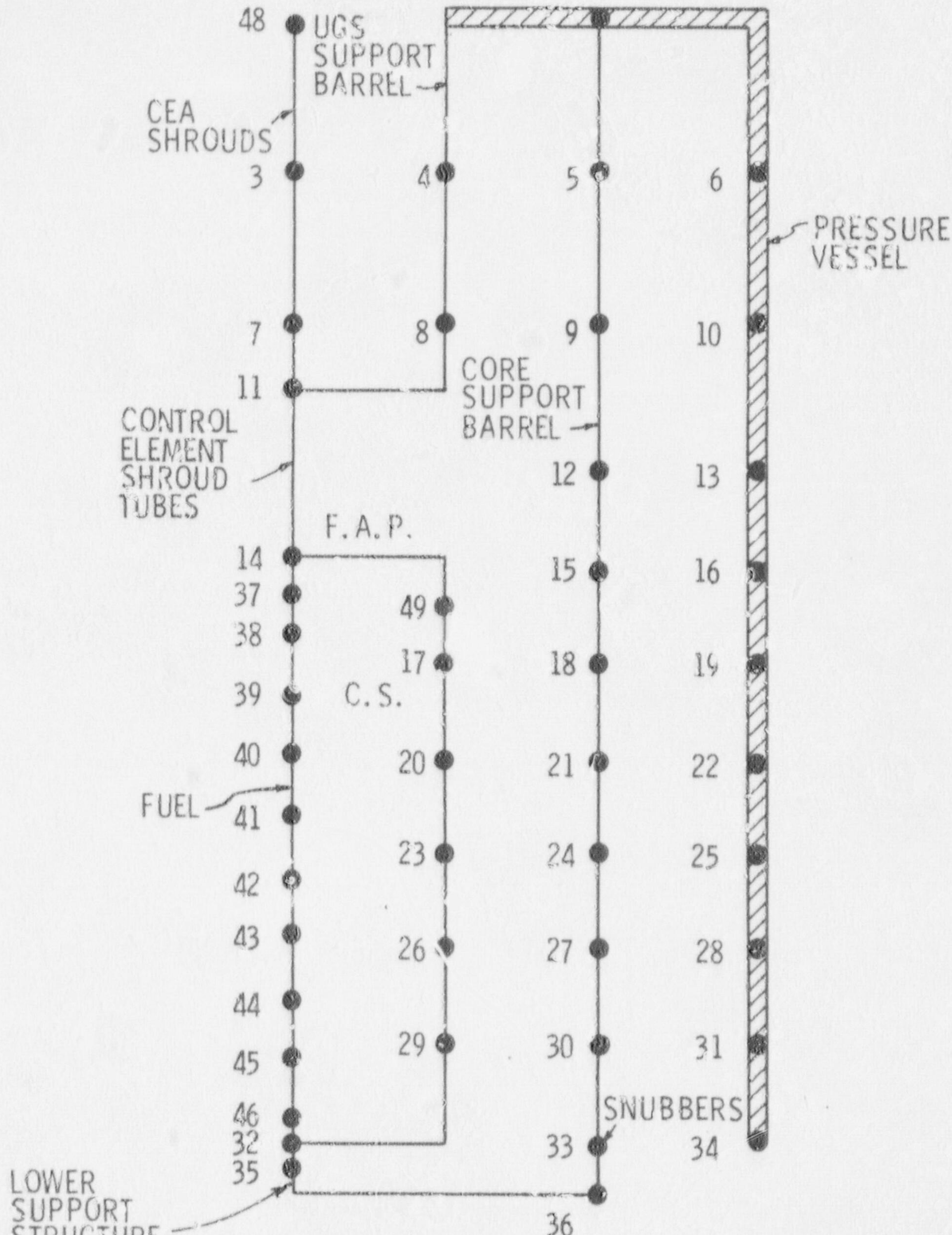
11. Pahl, P.J., "Modal Response on Containment Structures," Seismic Design for Nuclear Power Plants, MIT Press, Cambridge, Mass.

12. Forsberg, K., "Axisymmetrical and Beam-Type Vibrations of These Cylindrical Shells," AJAA Journal, Volume 7, February 1969.

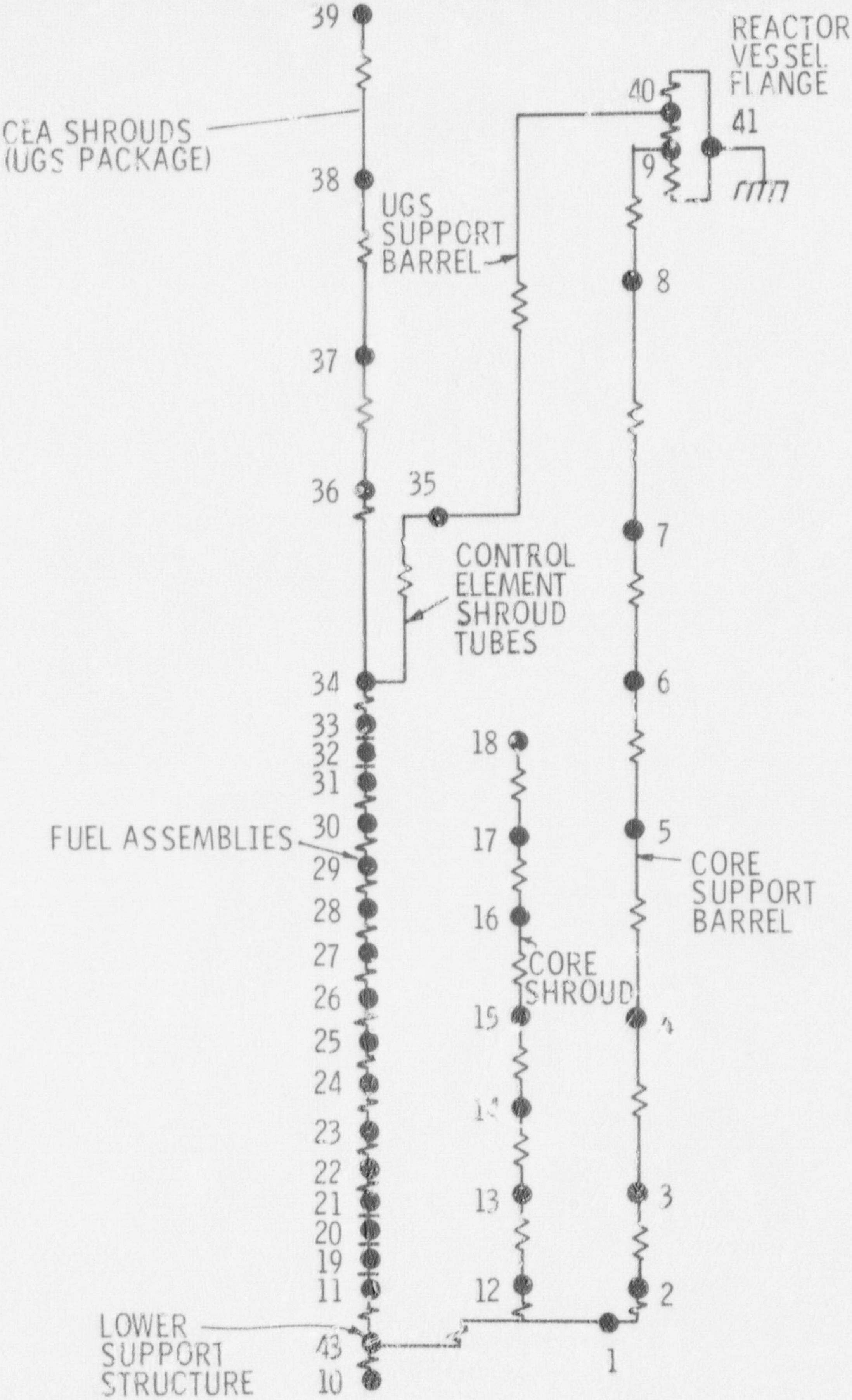
13. MRI/STARDYNE - Static and Dynamic Structural Analysis System: User Information Manual, Control Data Corporation, June 1, 1970.



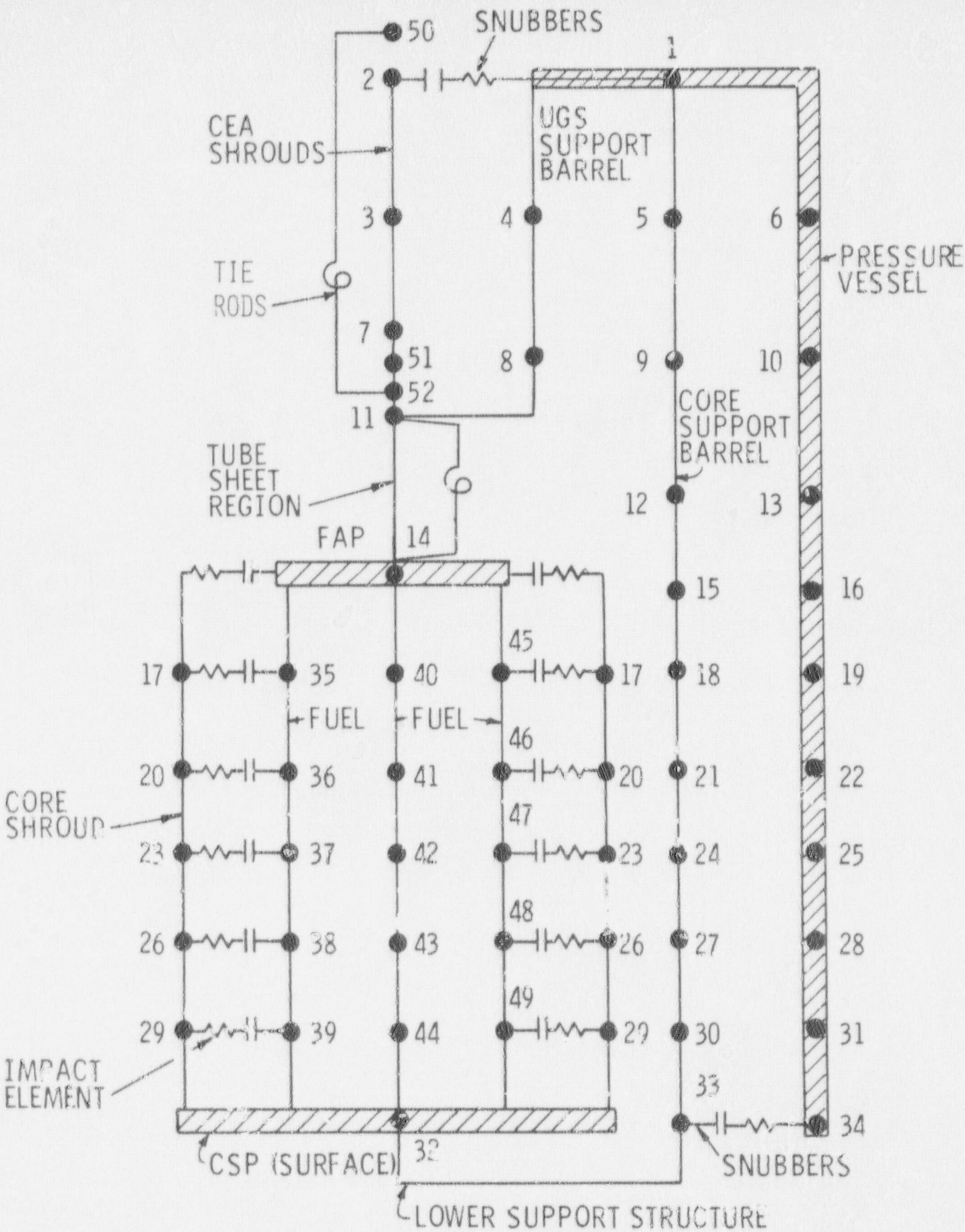
C-E SYSTEM 80	REACTOR INTERNALS HORIZONTAL SEISMIC MODEL	Figure 3.7.3-1



C-E SYSTEM 80	REACTOR INTERNALS LINEAR HORIZONTAL SEISMIC MODEL	Figure 3.7.3-2

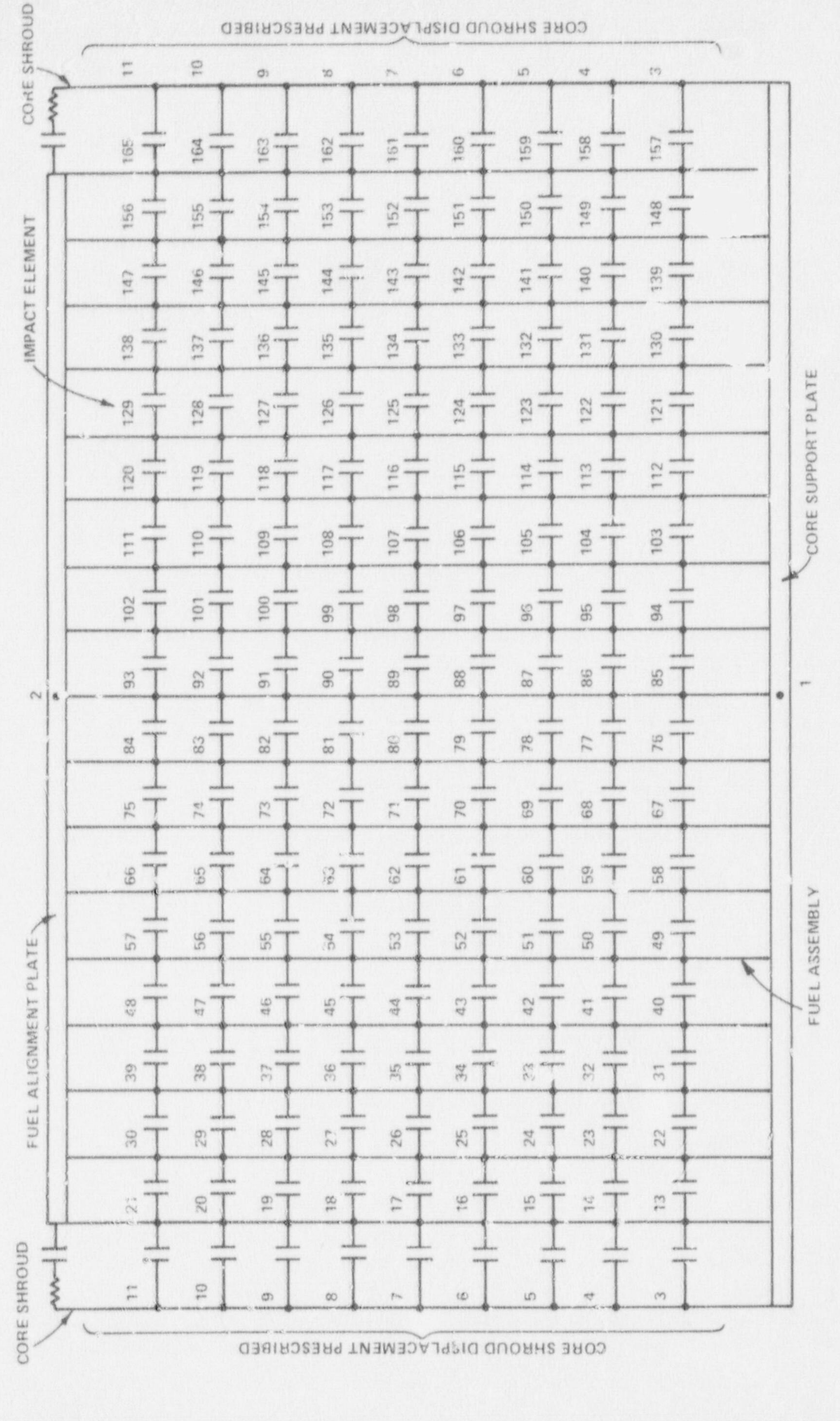


C - E SYSTEM 80	REACTOR INTERNALS LINEAR VERTICAL SEISMIC MODEL	Figure 3.7.3-3



Amendment No. 10
June 28, 1985

C-E SYSTEM 80	REACTOR INTERNALS NONLINEAR HORIZONTAL SEISMIC MODEL	Figure 3.7.3-4



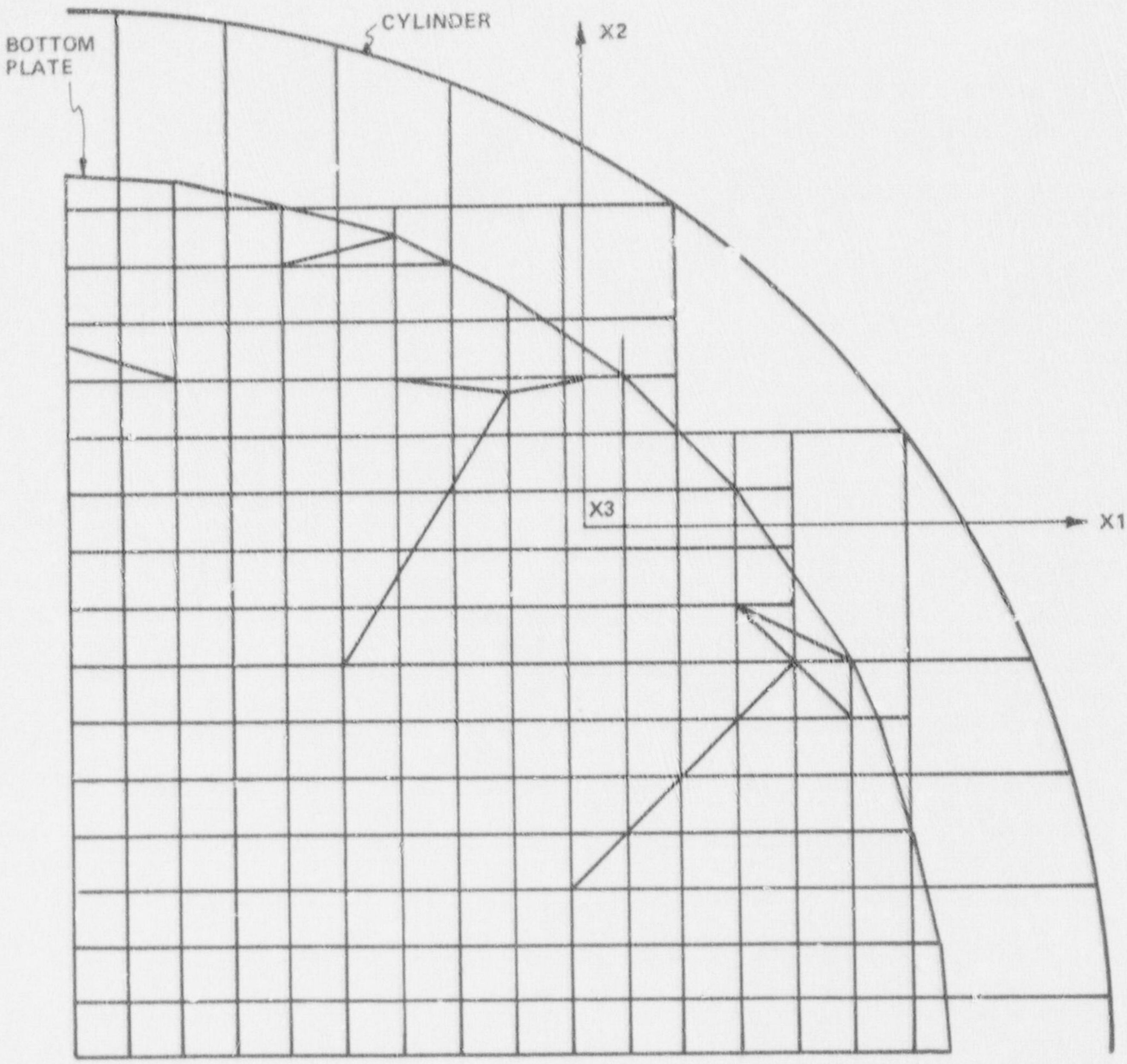
C - E SYSTEM 80	SYSTEM 80 CORE SEISMIC MODEL ONE ROW OF 17 FUEL ASSEMBLIES	Figure 3.7.3-5

C - E SYSTEM 80

CORE-SUPPORT BARREL UPPER FLANGE FINITE-ELEMENT MODEL

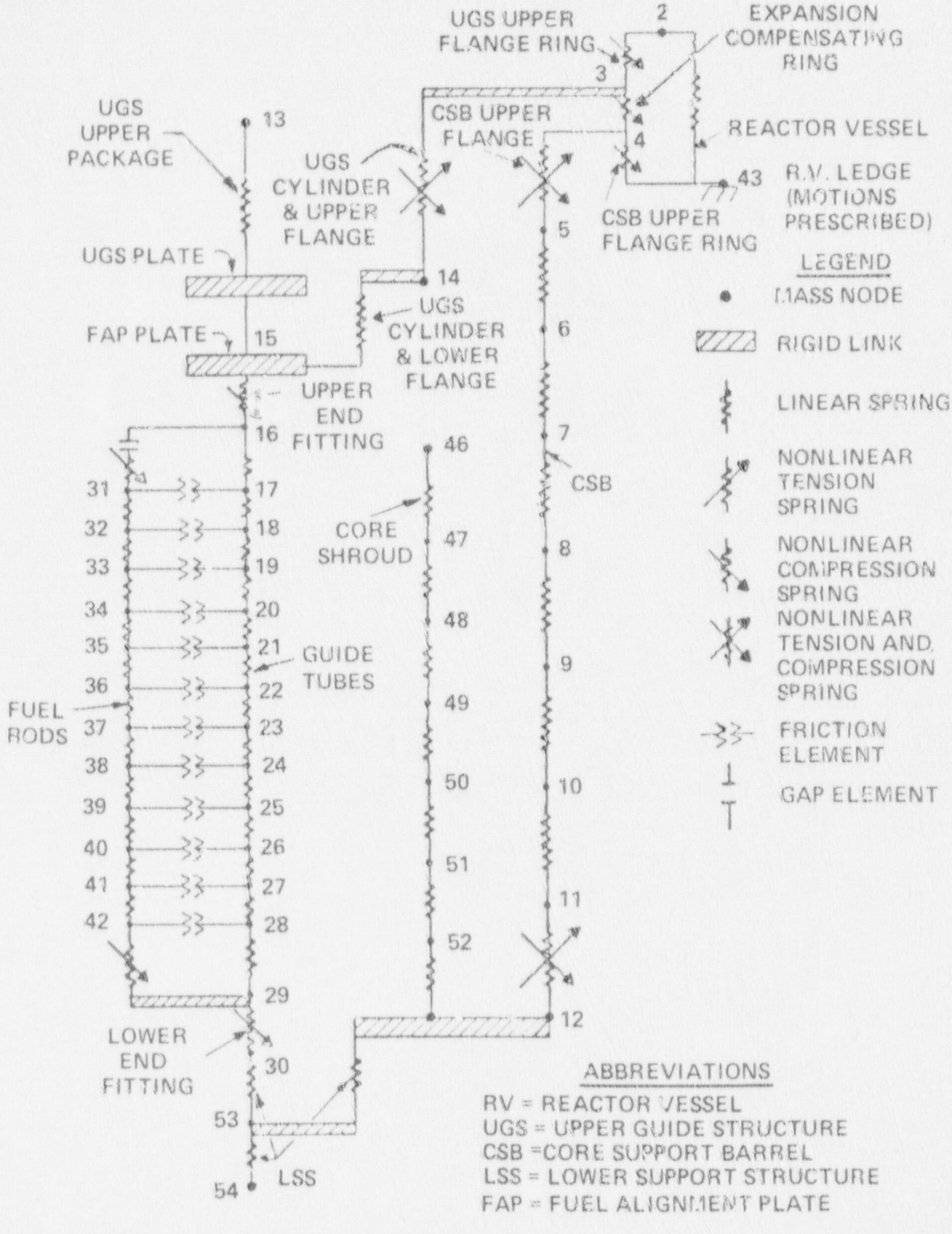
Figure 3.7.3-6

LOWER SUPPORT STRUCTURE TOP VIEW



PROJECTION ON X1-X2 PLANE

C - E SYSTEM 80	LOWER SUPPORT STRUCTURE FINITE ELEMENT MODEL	Figure 3.7.3-7



Amendment No. 6
November 20, 1981

C-E SYSTEM 80	REACTOR INTERNALS NONLINEAR VERTICAL SEISMIC MODEL	Figure 3.7.3-8

3.7.4 SEISMIC INSTRUMENTATION PROGRAM

See Applicant's SAR.

THIS PAGE INTENTIONALLY BLANK.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

See Applicant's SAR.

3.8.2 STEEL CONTAINMENT SYSTEMS

See Applicant's SAR.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

3.8.3.1 Description of the Internal Structures

Description of steel or concrete internal structures provided to support, shield or otherwise interface with the CESSAR scope will be provided in the Applicant's SAR.

3.8.3.2 Applicable Codes, Standards, and Specifications

See Applicant's SAR for criteria used in the design of internal structures.

3.8.3.3 Loads and Load Combinations

Loads at all structural interfaces with the CESSAR licensing scope of equipment are provided to the Applicant by C-E for use in the design of internal structures. The loads and loading combinations provided for use in the design of internal structures are the same as specified for design for the mating equipment, discussed in Section 3.9.1.4.

For details of the design of internal structures considering the equipment interface loads, see the Applicant's SAR.

3.8.3.4 Design and Analysis Procedures

For procedures used in the design of internal structures see the Applicant's SAR.

For procedures used in the design and analysis of reactor coolant system linear supports, see Section 3.9.1.4.

3.8.3.5 Structural Acceptance Criteria

See Section 3.9.1.4 for equipment in the CESSAR scope.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

See Applicant's SAR.

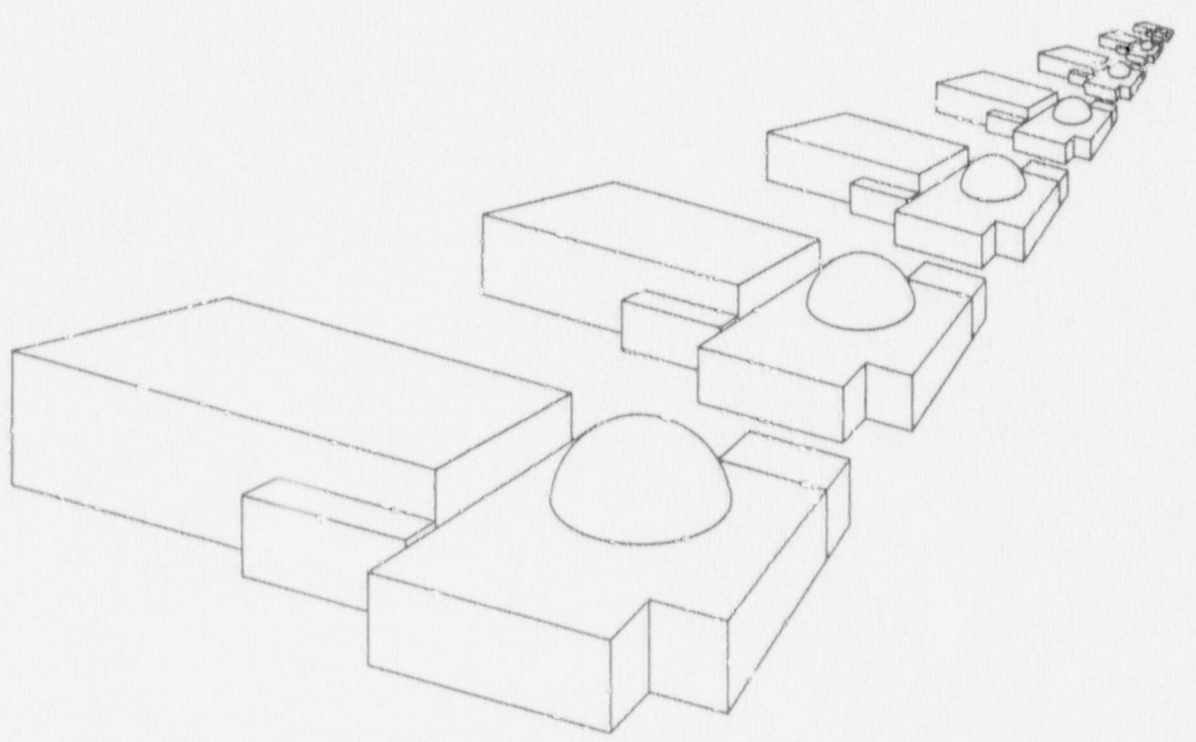
3.8.5 FOUNDATIONS

See Applicant's SAR.

THIS PAGE INTENTIONALLY BLANK.

SYSTEM 80+™

STANDARD DESIGN



CESSAR DESIGN CERTIFICATION

Volume 3

COMBUSTION ENGINEERING

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

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

Pressure and temperature fluctuations resulting from the normal, upset, emergency and faulted transients are computed by means of computer simulations of the reactor coolant system, pressurizer, and steam generators. Design transients are detailed in the equipment specifications. The component designer then uses the specification curves as the basis for design and fatigue analysis.

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

The fatigue analysis is based upon a series of dynamic events depicted in the respective component specifications. Associated with each dynamic event is a mechanical, thermal-hydraulic transient presentation along with an assumed number of occurrences for the event. The presentation is generally simple and straightforward, since it is meant to envelope the actual plant response. The intent is to present material for purposes of design.

Similarly, the characterization of a given dynamic event with a specific name is unimportant. Any plant dynamic occurrence with consequences which fall within the envelopes associated with one of these dynamic events is by definition represented by that dynamic event. The fundamental concept is to ensure that the consequences of the normal and upset conditions which are expected to occur in the power plant are enveloped by one or more of the dynamic event portrayals in the component specifications. The number of occurrences selected for each dynamic event is conservative, so that in the aggregate, a 60-year useful life will be provided by this design process. E

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

The principal design bases of the reactor coolant system (RCS) and reactor internals structures are given in Sections 5.2 and 3.9.5, respectively.

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

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

Pressure and thermal stress variations associated with the design transients are considered in the design of supports, valves, and piping within the reactor coolant pressure boundary (RCPB).

In addition to the design transients listed above and included in the fatigue analysis, the loadings produced by the OBE and SSE were also applied in the design of components and support structures of the RCS. The OBE and SSE are classified as upset and faulted condition events respectively. For the number of cycles pertaining to the OBE, refer to Section 3.7.3.2. E

3.9.1.2 Computer Programs Used in Stress Analyses

3.9.1.2.1 Reactor Coolant System

The following paragraphs provide a summary of the applicable computer programs used in the structural analyses for ASME Code Class 1 systems, components, and supports in the CESSAR-DC scope. The summaries include individual descriptions and applicability data. The computer codes employed in these analyses have been verified in conformance with design control methods, consistent with the quality assurance program described in Chapter 17.

3.9.1.2.1.1 MDC STRUDL

E

The MDC STRUDL computer program provides the ability to specify characteristics of framed structure and three-dimensional solid structure problems, perform static and dynamic analyses, and reduce and combine results.

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

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

The program is used to define the dynamic characteristics of the structural models used in the dynamic seismic analyses of the reactor coolant system components. The natural frequencies and mode shapes of the structural models and the influence coefficients which relate member end forces and moments and support reactions to unit displacements are calculated. The influence coefficients are calculated for each dynamic degree-of-freedom of each mass point and for each degree-of-freedom of each support point.

The program can perform either time-history analysis or spectrum analysis using the modal super position technique. Support reactions, member loads and joint acceleration are computed by back substituting from the modal coordinates to physical coordinates through the applicable transformation matrice and then combining modal contributions from each individual mode included in the response analysis. E

MDC STRUDL is a program which is in the public domain and has had sufficient use to justify its applicability and validity. Extensive verification of the C-E version has been performed to supplement the public documentation. The version of the program in use at C-E was developed by the McDonnell Automation Company/Engineering Computer International and is run on the IBM computer system. MDC STRUDL is described in more detail in Reference 1. E

3.9.1.2.1.2 C-E MARC

The C-E MARC program is a general purpose nonlinear finite element program with structural and heat transfer capabilities. It is described in detail in Reference 2.

C-E MARC is used for stress analysis of regions of vessels, piping or supports which may deform plastically under prescribed loadings. It is also used for elastic analyses of complex geometries where the graphics capability enables a well defined solution. The thermal capabilities of C-E MARC are used for complex geometries where simplification of input and graphical output are preferred.

C-E MARC is the C-E modified version of the MARC program, which is in the public domain and has had sufficient use to justify its applicability and validity. Extensive verification of the C-E version has been performed to supplement the public documentation. E

3.9.1.2.1.3 JEST

JEST is a proprietary computer code developed for evaluation of nuclear piping systems with hypothesized flaws. The code performs fracture mechanics related calculations such as J-Integral parameter and crack mouth opening displacements used in leak-before-break and stability evaluations. Input to the E code consists of the geometry, material properties, the flaw size and the loading conditions. The code uses elastic-plastic estimation scheme type solutions developed by Electric Power Research Institute (EPRI) and General Electric (GE). Several EPRI/GE analytical solutions are available in the code. The program is used to automate calculations that would have been performed manually.

3.9.1.2.1.4 SUPERPIPE

SUPERPIPE is a linear finite element program for the static and dynamic analysis of piping systems. These systems may include such components as bends, elbows, tees, reducers, socket or butt welds, flexible couplings, and flanges, with the appropriate flexibility factors and stress indices accounted for. Support types may include rigid, spring, constant-force, snubber, anchor, or user-specified, and may have any desired orientation.

Analyses performed include thermal, weight, applied load, frequency and mode shape, response spectrum, and time-history. Following the static and dynamic analysis phase, the program performs a complete ASME B&PV Code, Section III Class 1 stress check, combining analysis results in may manner specified by the user to create the appropriate loading cases applicable for each of the ASME code stress equations. The user also supplies the number of occurrences of each steady-state and transient load state, with which the program performs a complete fatigue damage calculation.

SUPERPIPE, which is in the public domain, was developed by the Impell Corporation and is described in detail in Reference 20.

3.9.1.2.1.5 DFORCE

The computer code program DFORCE calculates the internal forces and moments at designated locations in a piecewise linear structural system, at each time step, due to the time history of relative displacements of the system mass points and boundary points. The program also selects the maximum value of each component of force or moment at each designated location, and the times at which they occur, over the entire duration of the specified dynamic event. The program forms appropriate linear combinations of the relative displacements at each time step and performs a complete loads analysis of the deformed shape of the structure at each time step over the entire duration of the specified dynamic event.

The program is used to calculate the time dependent reactions in structural models subjected to dynamic excitation which are analyzed by the CEDAGS program.

To demonstrate the validity of the DFORCE program, results for test cases were obtained and shown to be substantially identical to those obtained for an equivalent analysis using the public domain program MDC STRUDL.

3.9.1.2.1.6 SG LINK

SG LINK determines steam generator and snubber stroke and building interface boundaries for the steam generator snubber lever system. The program verifies the kinematics of the snubber lever linkage systems based on input motions of the steam generator lug and detailed snubber lever system geometry. E

3.9.1.2.1.7 CEDAGS

The computer program CEDAGS (C-E Dynamic Analysis of Gapped Structure) performs a piecewise linear direct integration solution of the coupled equations of motion of a three dimensional structure which may have clearances or gaps between the structure and any of its supports or restraints (boundary gaps) or between points within the structure (internal gaps). The contacted boundary points may be oriented in any selected direction and may respond rigidly, elastically, or plastically. The structure may be subjected to applied dynamic loads or boundary motions.

The CEDAGS program is used to calculate the dynamic response of piecewise linear structural systems subjected to time varying load forcing functions resulting from postulated pipe break conditions. E

To demonstrate the applicability and validity of the CEDAGS program, the solutions to an extensive series of tests problems were obtained and shown to be substantially identical to results obtained by hand calculations or alternate computer solutions.

3.9.1.2.1.8 CE177, Head Penetration Reinforcement Program E

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

This program is used to perform preliminary sizing and reinforcement calculations for hemispherical heads in the reactor vessel. Program was verified by comparisons of program results and hand calculated solutions of classical problems.

3.9.1.2.1.9 CE102, Flange Fatigue Program E

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

This program is used to perform the fatigue analysis of the reactor vessel closure head and vessel flange assembly. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.10 CE105, Nozzle Fatigue Program E

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

This program is used to perform the fatigue analysis of reactor vessel nozzles and steam generator feedwater nozzle. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.11 CEC26, Edge Coefficients Program E

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

This program is used to perform the fatigue analysis of reactor vessel wall transition. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.12 CE124, Generalized 4 x 4 Program E

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

This program is used to perform fatigue analysis of reactor vessel bottom head juncture. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.13 SEC 11

The SEC 11 program automates the flaw evaluation method of ASME B&PV, Section XI, Appendix A. This program performs the crack growth analyses and assesses the margin against critical crack size according to the criteria in Appendix A. The program has been verified by direct comparison of program results and hand calculations. The program is used for leak-before-break type analyses. E

3.9.1.2.1.14 ANSYS

ANSYS is a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis. This program is in the public domain. Additional descriptive information on this code is provided in Section 3.9.1.2.2.2.

This program is used for numerous applications for all components in the areas of structural, fatigue, thermal and eigenvalue analysis. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.15 CE301, The Structural Analysis for Partial Penetration Nozzles, Heater Tube Plug Welds, and the Water Level Boundary of the Pressurizer Shell Program E

This program computes various analytical parameters, primary plus secondary stresses and stress intensities, peak stresses and stress intensities, and the cyclic fatigue analysis with usage factors at cuts of interest. This program is utilized to satisfy the requirements of Section III, of the ASME B&PV Code.

This program is used in the fatigue analysis of partial penetration nozzles in the pressurizer and piping. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.16 CE223, Primary Structure Interaction Program E

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

This program is used in the fatigue analysis of the steam generator primary structure. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.17 CE362, Tube-To-Tubesheet Weld Program E

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

This program is used in the fatigue analysis of steam generator tube-to-tubesheet weld. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.18 CE286, Support Skirt Loading Program E

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

This program is used in the structural analysis of steam generator support skirt. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.19 CE210, Principal Stress Program E

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

This program is used in the fatigue analysis of steam generator components. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.20 CE211, Nozzle Load Resolution Program E

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

This program is used in the fatigue analysis of steam generator nozzles. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.21 KINI2100 Program E

A general purpose finite difference heat transfer program. This program is used for steady-state and transient thermal analysis.

This program is used in numerous thermal relaxation analysis for all components. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.22 CEFLASH-4A

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

This program is used in a steam line break accident structural analysis. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.1.23 CRIBE

E

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

This program is used for determining steam generator performance. Program was verified by comparisons of program results and hand-calculated solutions of classical problems.

3.9.1.2.2 Code Class CS Internals, Fuel and CEDMs

E

The following computer programs are used in the static and dynamic analyses of reactor internals, fuel, and CEDMs.

3.9.1.2.2.1 MRI/STARDYNE

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

$$[K] \cdot \{\delta\} = \{P\} \quad (1)$$

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

where:

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

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

$[m]$ = the mass matrix

ω = the natural frequencies

$\{q\}$ = the normal modes

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

The finite elements used to date in C-E analyses are the elastic beam, plate and ground support spring members. The assumptions governing their use are as follows: small deformation, linear-elastic behavior, plane sections remain plane, no coupling of axial, torque and bending, geometric and elastic properties constant along length of element.

Further description is provided in Reference 4.

The MRI/STARDYNE code is used in the analysis of reactor internals. The program is used to obtain the mode shapes, frequencies and response of the internals to predescribed static and dynamic loading. The structural components are modeled with beam and plate elements. Ground support spring elements are used, at times, to represent the effects of surrounding structures. The geometric and elastic properties of these elements are calculated such that they are dynamically equivalent to the original structures. The response analysis is then conducted using both modal response spectra and modal time history techniques. Both methods are compatible with the program.

The program is also used to perform a static finite element analysis of the lower support structure to determine its structural stiffness.

MRI/STARDYNE is in the public domain and has had sufficient use to justify its applicability and validity. Extensive verification of the C-E version has been performed to supplement the public documentation.

3.9.1.2.2.2 ANSYS

ANSYS is a general purpose nonlinear finite element program with structural and heat transfer capabilities. It is described in Reference 5.

ANSYS is used to perform detailed stress analyses of the fuel assembly due to combined lateral and vertical dynamic loads resulting from postulated seismic and loss-of-coolant-accident conditions.

Static finite element analyses of reactor internal structures, such as flanges, expansion compensating ring and core shroud, are performed with ANSYS to determine vertical and lateral stiffnesses and thermal stresses.

ANSYS is a proprietary code in the public domain. The developers, Swanson Analysis Systems, Incorporated have published an ANSYS verification manual with numerous examples of its usage.

3.9.1.2.2.3 ASHSD

The ASHSD program uses a finite element technique for the dynamic analysis of complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The three-dimensional axisymmetric continum is represented as an axisymmetric thin shell. The axisymmetric shell is discretized as a series of frustums of cones.

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

The assumptions governing the axisymmetric thin shell finite element representation of the structure are those consistent with linear orthotropic thin elastic shell theory. Further description is provided in Reference 6.

ASHSD is used to obtain the dynamic response of the core support barrel under normal operating conditions and due to a LOCA. An axisymmetric thin shell model of the structure is developed. The spatial Fourier series components of the time varying normal operating hydraulic pressure or LOCA loads are applied to the modeled structure. The program yields the dynamic shell and beam mode response of the structural system.

ASHSD has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results. The details of these comparisons may be found in References 6 and 7.

3.9.1.2.2.4 CESHOCK

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

The output from the code consists of minimum and maximum values of translational and angular accelerations, forces, shears, and moments for the problem time range. In addition, the above quantities are presented for all printout times requested. Plots can also be obtained for displacements, velocities and accelerations as desired. Further description is provided in Reference 8.

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

Lateral and vertical lumped-mass and spring models of the internals are formulated. Various types of springs (linear, compression only, tension only, or nonlinear springs) are used to represent the structural components. Thus, judicious use of load-deflection characteristics enables effects of components impacting to be predicted. Transient loading appropriate to the horizontal and vertical directions is applied at mass points and a dynamic response (displacements and internals forces) is obtained.

CESHOCK has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted analytical results via an independent computer code. The details of these comparisons may be found in References 7 and 8.

3.9.1.2.2.5 SAMMSOR/DYNASOR

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

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

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

The equations of motion of the shell are solved in DYNASOR using Houbolt's numerical procedure with the nonlinear terms being moved to the right-hand side of the equilibrium equations and treated as generalized pseudo-loads. The displacements and stress resultants can be determined for both symmetrical and asymmetrical loading conditions. Asymmetrical dynamic buckling can be investigated using this program. Solutions can be obtained for highly nonlinear problems utilizing as many as five circumferential Fourier harmonics. Further description is provided in References 9 and 10.

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

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

SAMMSOR/DYNASOR has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations, accepted experimental test or analytical results, and results obtained with a similar independently written program in the public domain. The details of these comparisons may be found in Reference 7.

3.9.1.2.2.6 MODSK

MODSK is a C-E computer program which solves for the natural frequencies and mode shapes of a structural system. The natural frequencies and mode shapes are extracted from the system of equations:

$$(K-W_n^2 M)\ \phi_n = 0$$

where:

K	=	model stiffness matrix
M	=	model mass matrix
W_n	=	natural circular frequency for the n^{th} mode
ϕ_n	=	normal mode shape matrix for the n^{th} mode

The solution to the general eigenvalue problem is obtained using the dual Jacobi rotation method.

The MODSK code is used in the analyses of reactor internals to obtain frequencies and mode shapes, and damping parameters. The results of these analyses are incorporated into overall reactor vessel internals models, which calculates dynamic response due to seismic and LOCA conditions.

The MODSK program was developed by C-E and is used on the CDC 7600 computer. To demonstrate the validity of the MODSK program, results from lateral and vertical test problems were obtained and shown to be substantially identical to those obtained from an equivalent analysis using the public domain program ANSYS (Refer to Section 3.9.1.2.2.2).

3.9.1.2.2.7 SAPIV

The SAPIV computer code is a structural analysis program capable of analyzing two and three-dimensional linear complex structures subjected to any arbitrary static and dynamic loading or base acceleration. The analysis technique is based on the finite element displacement method. The structure to be analyzed can be represented using bars, beams, plates, membranes and three-dimensional finite elements.

Structural stiffness and load vectors are assembled from the element matrices which are derived assuming various displacement functions within each element whereas lumped mass matrices are used to represent inertia characteristics of the structure. In the static analysis, the assembled equations of equilibrium are solved by using a linear equation solver. Dynamic analysis capabilities include modal analysis, modal superposition and direct integration methods of computing dynamic response and response spectrum techniques.

SAPIV has been applied to the eigenvalue and response spectra analyses of spent fuel storage racks and lifing rig structures.

The SAPIV code is used in the computation of dynamic response of control element drive mechanisms under mechanical and seismic loads. Both modal analysis and response spectrum capabilities of the code are used to find the natural frequencies and mode shapes and the dynamic loads in CEDM components.

SAPIV is in the public domain and has had sufficient use to justify its applicability and validity. Extensive verification of the C-E version has been performed to supplement the public documentation.

3.9.1.2.2.8 CEFLASH-4B

The CEFLASH-4B computer code (Reference 14) predicts the reactor pressure vessel pressure and flow distribution during the subcooled and saturated portion of the blowdown period of a Loss-of-Coolant-Accident (LOCA). The equations for conservation of mass, energy and momentum along with a representation of the equation of state are solved simultaneously in a node and flow path network representation of the primary reactor coolant system.

CEFLASH-4B provides transient pressures, flow rates and densities throughout the primary system following a postulated pipe break in the reactor coolant system.

The CEFLASH-4B computer code is a modified version of the CEFLASH-4A code (References 15 through 17). The CEFLASH-4A computer code has been approved by the NRC (References 18 and 19). The capability of CEFLASH-4B to predict experimental blowdown data is presented in Reference 14.

3.9.1.2.2.9 LOAD

LOAD calculates the applied forces of the axial internals model which is contained within water control volumes using results from the CEFLASH-4B blowdown loads analysis as input. The fluid momentum equation is applied to each volume and a resultant force is calculated. Each force is then apportioned to the various structural nodes contained within the volume. Use of the fluid momentum equation takes into account pressure forces, fluid friction, water weight, and momentum changes within each volume. The resultant forces are combined with the reactor vessel motions obtained from the reactor coolant system analysis before the structural responses are determined. The LOAD code has been verified by demonstrating that its solutions are substantially identical to those obtained from hand calculations.

3.9.1.2.3 Non-NSSS Structures and Components

The following computer programs are used in the analyses of Non-NSSS structures and components.

(LATER)

3.9.1.3 <u>Experimental Stress Analyses</u>

Requirements for experimental stress analysis have not been imposed on any equipment in the CESSAR-DC scope. E

3.9.1.4 <u>Considerations for the Evaluation of the Faulted Condition</u>

3.9.1.4.1 Seismic Category I RCS Items

The major components of the reactor coolant system (RCS) are designed to withstand the forces associated with the design basis pipe breaks discussed in Section 3.6, in combination with the forces associated with the Safe Shutdown Earthquake and normal operating conditions. For structural evaluation, the design basis pipe breaks are those breaks for which leak-before-break cannot be demonstrated. Since the dynamic effects of breaks in piping systems listed in Section 3.6.2.2.1 are eliminated by leak-before-break, the pipe break loads analysis procedure considers only those branch line pipe breaks not eliminated by leak-before-break.

See Section 3.9.3 for discussion of loading combinations.

Analyses are performed to generate component loads and motions due to the forces associated with branch line pipe breaks. the analyses account for the reactor vessel and supports, major connected piping and components and the reactor internals. The results of the analyses include loads on major component supports and RCS piping loads.

The analyses performed for branch line breaks use MDC STRUDL code (see Section 3.9.1.2.1.1).

The resultant component and support reactions are specified, in combination with the appropriate normal operating and seismic reactions, for design verification by the methods discussed below and in Section 3.9.3.

The system or subsystem analysis used to establish, or confirm, loads which are specified for the design of components and supports is performed on an elastic basis.

When an elastic system analysis is employed to establish the loads which act on components and supports, elastic stress analysis methods are also used in the design calculations to evaluate the effects of the loads on the components and supports. In particular, inelastic methods such as plastic instability and limit analysis methods, as defined in Section III of the ASME Code, are not used in conjunction with an elastic system analysis. Figure 3.9-1. The RCS and its supports, which are analyzed using elastic methods, are shown in diagram form in Figure 3.9-1.

Inelastic methods of analysis are used in cases where it is deemed desirable and appropriate to permit significant local inelastic response. In these cases, if any, the system or subsystem analysis performed to establish the loads which act on components and component supports are modified to include the inelastic strain compatibility in the local regions of the components and component supports at which significant local inelastic response is permitted.

Inelastic methods defined in Section III of the ASME Code as plastic instability or limit analysis methods are not used.

3.9.1.4.1.1 **Reactor Internals and CEDMs**

See Sections 3.7.3.14 and 3.9.2.5.

3.9.1.4.1.2 **Non-Code Items**

The components not covered by the ASME Code but which are related to plant safety include:

A. Internal Structures (Class IS).

B. Fuel.

C. Control element drive mechanisms (CEDMs).

D. Control element assemblies (CEAs).

Each of these components is designed in accordance with specific criteria to ensure their operability as it relates to safety. The fuel assembly and control element assembly design is discussed in Section 4.2. The non-code components of the control element drive mechanisms (CEDMs) are proven by testing as described in Section 3.9.4.4. E

3.9.1.4.2 **Seismic Category I Non-NSSS Items**

The analytical method for evaluating the faulted condition uses a linear elastic model as described in Section 3.7.3. The ASME Section III allowable stress limits will be met for faulted loads, including the safe shutdown earthquake and system transient loads described in Section 3.9.1. For any exceptions to the above, such as the pipe break analysis described in Section 3.6.2, maximum allowable strain limits from accepted standards will be satisfied.

3.9.2 **DYNAMIC SYSTEM ANALYSIS AND TESTING**

3.9.2.1 **Piping Vibrations, Thermal Expansion, and Dynamic Effects**

Safety-related piping systems were designed in accordance with the ASME B&PV Code, Section III. The preoperational test program for the Class 1, 2 and 3 piping systems will simulate actual operating modes to demonstrate that the appurtenances comprising these systems will meet functional design requirements and that piping vibrations are within acceptable levels.

3.9.2.1.1 Steady-State Vibration

Essential systems and systems with the potential to experience significant vibration will be monitored for steady-state vibration. The piping will be monitored during normal operating and test modes along with operating modes expected to result in the most severe vibration. The piping will be visually inspected and vibration movements will be taken using portable instrumentation at locations where the vibration is judged to be the most severe. When necessary, the piping will be instrumented and monitored remotely.

The measured piping displacements will be compared with allowable displacement limits that are based on the allowable amplitudes, S_a, given below.

S_a = 7,690 psi for carbon steel with UTS $\leq$ 80 kips/in^2. This represents the alternating stress intensity at 10^{11} cycles and is extrapolated from Figure I-9.1 of Appendix I of ASME Code, Section III. S_a for stainless steel is equal to the alternating stress intensity at 10^{11} cycles taken from Figure I-9.2 of Appendix I of ASME Code, Section III. E

If the measured piping displacements exceed allowable limits, one or more of the following actions will be taken so that the vibration can be qualified.

A. Analyses will be performed to show that the measured displacements are acceptable.

B. Additional testing will be performed to show that the peak stresses due to the vibration are acceptable.

C. The source of the excessive vibrations will be eliminated.

D. The pipe supporting arrangement will be modified to reduce the vibration to acceptable levels.

3.9.2.1.2 Transient Vibration

Vibration monitoring will be completed for systems expected to experience significant transients. The piping will be instrumented to measure the system response during the transient events.

The measured response will be compared with analytically predicted values from the piping stress report. If the predicted values are exceeded, the measured response will be shown to be acceptable by additional analyses or testing; or the source of the transient will be eliminated or modified to reduce the transient loadings or nodifications to the pipe supporting arrangement will be made to reduce the system response to acceptable levels.

3.9.2.1.3 Thermal Expansion

Safety-related systems that are expected to experience significant thermal movements will be monitored for thermal expansion. A preheatup walkdown will be performed so that locations of potential thermal interferences can be identified and appropriate corrective action taken prior to heatup. One complete thermal cycle, i.e., cold position to hot position to cold position, will be monitored. The piping and components will be visually inspected and piping displacements will be monitored at predetermined locations. The measurement locations will be based on the locations of snubbers, hangers, and expected large displacements. When necessary, the piping will be instrumented and monitored remotely.

Acceptable limits of pipe displacement, based on analytically predicted movements from the piping stress reports, will be determined prior to testing. The measured displacements will be compared to the acceptance limits to determine whether the piping systems are free to expand as expected. If the measured displacements are not within the acceptance limits, then analyses will be performed or corrective action will be taken, as appropriate, to ensure that pipe stress and support and equipment allowables are not exceeded.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

3.9.2.2.1 Nuclear Steam Supply System

The operability of all active safety-related mechanical equipment related to the NSSS is demonstrated by analysis and/or testing. The methods and procedures used and the results of tests and analyses that confirm implementation of the design criteria for safety-related mechanical equipment, including supports, are provided in Section 3.9.3.2.

3.9.2.2.2 Non-NSSS Items

The following dynamic testing procedures are used for Seismic Category I mechanical equipment and equipment supports.

3.9.2.2.2.1 Seismic Testing and Analysis

The ability of equipment to perform its Seismic Category I functions during and after an earthquake is demonstrated by tests and/or analysis. The selection of testing and/or analysis for a particular piece of equipment is based on practical considerations. When practical, the Seismic Category I operations are activated and tested during the vibratory testing. When this is not practical, these operations are simulated by a combination of tests and analysis.

3.9.2.2.2.2 Seismic Analysis

Equipment that is large, simple (e.g., panels, pumps and valves), and/or consumes large amounts of power is usually qualified by an analysis to show that the loads, stresses, and deflections are less than the values which give assurance of proper operation. Analysis is also used to show that there are no natural frequencies below the frequency range of a test facility.

3.9.2.2.2.3 Basis for Test Input Motion

When equipment is qualified by test, the response spectrum or the time history at the point of attachment to the supporting structure is the basis for determining the test input motion. E

3.9.2.2.2.4 Random Vibration Input

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

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

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

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

3.9.2.2.2.5 Input Motion

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

3.9.2.2.2.6 **Fixture Design**

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

E

3.9.2.2.2.7 **Equipment Testing**

Equipment testing is based on prototype basis. Similarity between the equipment being tested and the installed equipment is assured. This is usually done by the vendor who supplies the equipment.

3.9.2.3 **Dynamic System Analysis Methods for Reactor Vessel Core Support and Internal Structures**

3.9.2.3.1 **Introduction**

The flow-induced vibration of the reactor internals components during normal operation can be characterized as a forced response to both deterministic (periodic and transient) and random pressure fluctuations in the coolant. Methods have been developed to predict the various components of the hydraulic forcing function and the response of the reactor internals to such excitation.

This analytical methodology is summarized in Figure 3.9-2. The method separates the response calculations into two groups in accordance with the physical nature of the loading i.e., deterministic or random. Methods for developing the deterministic component of the hydraulic forcing function are discussed in Section 3.9.2.3.2, while those relating to the random component are discussed in Section 3.9.2.3.3. Where complex flow path configurations or wide variations in pressure distribution are involved, the hydraulic forcing functions are formulated using a test-analysis combination method utilizing data obtained from plant tests and/or scaled model tests.

The response of the reactor vessel core support and internal structures (to include Core Support Barrel Assembly, Upper Guide Structure Assembly and Lower Support Structure Assembly) to the normal operating hydraulic loads are calculated by finite element techniques. The mathematical models used in these response analyses are described in Section 3.9.2.3.4. The methods used in calculating the structural responses are discussed in Section 3.9.2.3.5.

3.9.2.3.2 Periodic Forcing Function

3.9.2.3.2.1 Core Support Barrel Assembly

An analysis based on an idealized hydrodynamic model is employed to obtain the relationship between reactor coolant pump pulsations in the inlet ducts and the periodic pressure fluctuations on the core support barrel. A detailed description of this model and subsequent solution are given in References 21 through 27. The model represents the annulus of coolant between the core support barrel and the reactor vessel. In deriving the governing hydrodynamic differential equation for the above model, the fluid is taken to be compressible and inviscide. Linearized versions of the equations of motion and continuity are used. The excitation on the hydraulic model is harmonic with the frequencies of excitation corresponding to pump rotational speeds and blade passing frequencies. The result of the hydraulic analysis is a system of equations which define the forced response, natural frequencies and natural modes of the hydrodynamic model. The forced response equations define the spatial distributions of pressure on the core support barrel system as a function of time.

3.9.2.3.2.2 Upper Guide Structure

The dynamic force on the upper guide structure assembly is due to flow induced forces on the tube bank. The periodic components of these forces are caused by pressure pulsations at harmonics of the pump rotor and blade passing frequencies, and vortex shedding due to crossflow over the tubes.

A series of tests on full size tubes at reactor pressure and temperature indicated no evidence of periodic vortex shedding at the Reynolds Number and turbulence levels expected in the tube bank (Reference 28). Thus, the only significant periodic force is that due to pump pulsations. Data from this same test series was utilized to determine the magnitude of these pulsations at the pump rotor, twice the rotor, blade passing, and twice blade passing frequencies.

3.9.2.3.2.3 Lower Support Structure Assembly

The ICI nozzles and the skewed beam supports for the ICI support plate are excited by periodic and/or random, flow induced forces.

The periodic component of this loading is due to pump related pressure fluctuations and vortex shedding due to crossflow. High turbulence intensity caused by jetting through the flow skirt makes it unlikely that regular vortex shedding will occur (References 29 and 30). If it were assumed to occur, the maximum shedding frequency would be well below the lowest structural frequency for both the ICI support nozzles and skewed beams. The magnitude and frequency of this periodic force are accounted for based on data in the literature for crossflow over both vertical (References 31 and 32) and skewed (Reference 33) isolated tubes.

Derivation of pump frequency related loads is accomplished by assuming that these periodic pressure variations are propagated undiminished through the flow skirt from the lower portion of the core barrel - reactor vessel annulus. The magnitude of these pulsations is based on a combination of analytical predictions, based on Reference 21, and data from previous precritical programs (References 23 and 24).

3.9.2.3.3 Random Forcing Function

3.9.2.3.3.1 Core Support Barrel Assembly

The random hydraulic forcing function is developed by analytical and experimental methods. An analytical expression is developed to define the turbulent pressure fluctuation for fully developed flow (Reference 34). This expression is modified, based upon the result of scale model testing (References 35 and 36), to account for the fact that flow in the downcomer is not fully developed. Based upon tests results, an expression is developed to define the spatial dependency of the turbulent pressure fluctuations. In addition, experimentally adjusted analytical expressions are developed to define the peak value of the pressure spectral density associated with the turbulence and the maximum area of coherence, in terms of the boundary layer displacement, across which the random pressure fluctuations are in phase (References 25, 26 and 27). The transient behavior of the random fluctuations during loop startup and shutdown is assumed to be identical to that of the periodic excitations.

3.9.2.3.3.2 Upper Guide Structure

Results of the full size tube tests (Reference 28) showed that at normal operating conditions the shroud tubes are excited by upstream and wake produced turbulent buffeting (References 28, 37 and 38). The forcing function for this type of loading can be represented as a band limited white noise power spectrum (Reference 28). The magnitude of this spectrum is computed based on data from these tests. The resultant velocity dependent force is combined with static drag loads to compute the amplitude response and stress levels.

3.9.2.3.3.3 Lower Support Structure Assembly

The ICI nozzles and ICI support plate support beams are both subject to turbulent buffeting by the flow skirt jets. The outermost ICI nozzles and beams receive full impact of the jets before the jets decay due to fluid entrainment and the presence of inner tube rows. The force spectrum of these jets is assumed to be represented as wide band white noise. The magnitude of this spectrum is based on data in the literature for impingement of turbulent jets (Reference 39 and 40). This velocity dependent magnitude is applied to each tube, assuming no change in jet characteristics, between the outermost and inner tubes. The approach velocity for each tube is calculated from an analytical expression based on experimental data on the velocity distribution in the lower portion of the reactor vessel-core barrel annulus and the flow skirt.

3.9.2.3.4 Mathematical Models

A finite element analysis is performed on each of the reactor internals components using mathematical models. These models are designed to provide the most efficient analysis under the most significant loading condition to which each structure is exposed. The core support barrel assembly is modeled as a shell using the ASHSD computer code (Reference 6) (Figure 3.9-3). The structure is fixed at the upper flange to determine the beam modes and frequencies. The shell modes and frequencies are found by considering the upper flange fixed and the lower flange pinned. These analyses include hydrodynamic mass effects. All significant mode shapes and frequencies are used in combination to perform the normal operating deterministic response analysis. A simplified finite element model of the barrel assembly is generated on the STARDYNE computer code (Reference 4) for use in the random response analysis.

The control element shroud tubes in the upper guide structure assembly are modeled as beams supported at the ends by plate elements. The end plates are in turn supported by spring elements which represent the stiffness of additional surrounding structure. A typical model of this configuration is shown in Figure 3.9-4. The STARDYNE computer code (Reference 4) is employed to allow the same models to be utilized for modal analysis as well as deterministic and random response analysis.

The lower support structure assembly is modeled in several ways. Beam and plate elements are assembled in a comparatively coarse mesh to model the entire Instrument Nozzle Assembly (Figure 3.9-5). This representation of the structure is used on the STARDYNE computer code (Reference 4) to determine the modes, frequencies and response actions of the assembly as a system.

The reaction points in this model are taken at the bottom plate level of the LSS Assembly. Typical ICI nozzles (Figure 3.9-6) and Skewed Beams (Figure 3.9-7) are modeled as fine mesh beam elements reacted at the support points by spring elements representing the surrounding structure flexibility. These component models are used on the STARDYNE computer code (Reference 4) to provide the individual structural modes, frequencies and responses within the system. The results of both individual and system analysis are combined to provide the total response.

3.9.2.3.5 Response Analysis

3.9.2.3.5.1 Deterministic Response

The normal mode method (Reference 41) is used to obtain the structural response of the reactor internals to the deterministic forcing functions developed in Section 3.9.2.3.2. The method is applied to the appropriate finite element models described in Section 3.9.2.3.4. Generalized masses based on mode shapes and the mass matrices from the finite element computer programs are calculated for each component's modes of vibration. Modal force participation factors are based on the mode shapes and the predicted periodic forcing functions are calculated for each mode and forcing function. The generalized coordinate response for each mode is then obtained through solution of the corresponding set of independent second order single-degree of freedom equations. Utilizing displacement and stress mode shapes from the finite element computer programs, the modal responses of the reactor internals are obtained by means of the appropriate coordinate transformations. Response to any specific forcing function is obtained through summation of the component modes for that forcing function.

3.9.2.3.5.2 Random Response

The normal mode method (Reference 41) is used to obtain the structural response of the reactor internals subjected to random forcing functions. The random forcing functions are assumed to be of both the band limited and wide band white noise varieties as described in Section 3.9.2.3.3. Experimental and analytical expressions are used to define the force power spectral density associated with flow related turbulence and jet impact. The appropriate mathematical models described in Section 3.9.2.3.4 are used in the STARDYNE computer code (Reference 4). This code computes the response RMS displacements, loads and stresses in a multi-degree-of-freedom linear elastic structural model subjected to stationary random dynamic loadings, such as those described in Section 3.9.2.3.3.

The largest response of the Core Support Barrel is expected to be in the "beam" mode. The simplified finite element model of this structure, described in Section 3.9.2.3.4, is used to compute these displacements.

The Upper Guide Structure and Lower Support Structure will not respond to random excitation as complete assemblies but rather will experience local disturbances of individual components within the assemblies. The modal analyses from the finite element models of these components, (Figures 3.9-4, 3.9-5 and 3.9-7) already used for deterministic analysis, are once again utilized to determine the random responses via the normal mode procedure.

3.9.2.4 Comprehensive Vibration Assessment Program (CVAP)

In accordance with Regulatory Guide 1.20 (Reference 44), a CVAP is developed for System 80+. System 80+ is designated as non-prototype Category I, per Regulatory Guide 1.20, with Palo Verde Unit 1, a Combustion Engineering System 80 Reactor as the valid prototype (Reference 44). Palo Verde Unit 1 and System 80+ design are substantially the same with regard to arrangement design, size and operating conditions.

The CVAP for System 80+ design will consist of an Analysis and Inspection Program. The Analysis Program will consist of dynamic analyses which will be documented in an ASME Design Stress Report. In addition, flow loads and structural responses for System 80+ will be compared with System 80 to confirm System 80+ design is as non-prototype Category 1 reactors. E

The Inspection Program will consist of a pre-hot functional and a post-hot functional inspection of the reactor internals. The duration of the hot functional testing will be established to insure that 10E+7 cycles of vibration will have occurred before the post-hot functional inspection. A detailed inspection of major load bearing surfaces, contact surfaces, welds, and maximum stress locations identified in the Analysis Program will be performed. Photographic documentation will be taken of all observations made during the pre- and post-hot functional inspections. A comparison will be made of the structures to verify that no loss in structural integrity due to flow induced vibration has occurred.

The Analysis Program and Inspection Program will together confirm the adequacy of the analysis prediction techniques and the structural integrity of System 80+ design according to the guidance of Regulatory Guide 1.20.

3.9.2.5 Dynamic System Analysis of the Reactor and CEDMs Under Faulted Conditions

Dynamic analyses are performed to determine blowdown loads and structural responses of the reactor internals and fuel to postulated pipe break and SSE loadings and to verify the adequacy of their design.

Because of Leak-Before-Break arguments, all main RCS loop pipe breaks and all major primary branch line pipe breaks have been eliminated from consideration of dynamic effects. The reactor vessel motion and blowdown loads associated with small branch line breaks result in loads on the reactor internals which are calculated and determined to be neglible compared to the SSE loads. To obtain the total stress intensities for faulted conditions, the SSE loads are increased by a factor of 1.10 to account for the minor contribution of these pipe breaks to the combined stress. E

3.9.2.6 Correlation of Test and Analytical Results

Analytical predictions are compared with data obtained from the precritical program to ensure consistency.

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

E

ASME B&PV Code Section III Class 1, 2 and 3 Piping and Components are designed and constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code and Code Case(s).

In accordance with ASME Code, a specification is provided for piping supports which defines the jurisdictional boundary for the NF portion of the piping support.

For equipment component supports, such as those for pumps and vessels, the supports are generally furnished by the manufacturer along with the equipment. The supports are designed and classified by the vendors and meet either ASME Subsection NF, the rules for the class of the component being furnished, or AISC, as appropriate.

Reactor coolant loop piping and associated components and component supports are designed and analyzed by Combustion Engineering. Loading conditions, stress limits, design transients, and methods of analysis for ASME Code Class 1 reactor coolant loop piping and associated components and component supports are discussed in Section 3.9.3.1.

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

The loading combinations specified for the design ASME B&PV Section III Code Class 1 components, supports, and piping are categorized as normal, upset, emergency and faulted. The following specific loading combinations are specified for design:

A. The concurrent loadings associated with the Level-A (normal) plant conditions of dead weight, pressure and the thermal and expansion effects during startup, hot standby, power operation and normal shutdown to cold shutdown conditions.

B. The concurrent loadings associated with either the normal plant condition or the Level-B (upset) plant condition and the vibratory motion of the Operational Basis Earthquake (OBE).

C. The concurrent loadings associated with the Level-C (emergency) condition.

D. The concurrent loadings associated with the Level-A (normal) plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with the Level-D (faulted) system condition (postulated pipe rupture for branch line breaks not eliminated by leak before break analysis). The SSE and pipe rupture loadings are combined by the SRSS method or a more conservative method. E

The specific design transients specified for design are discussed in Section 3.9.1.1.

ASME B&PV Code Class 1, 2 and 3 piping and components of fluid systems are designed and constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code. Hydrostatic testing is performed per Section III. E

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems are presented in the sections which describe the systems.

Stress analysis was used to determine structural adequacy of pressure components under the operating conditions of normal, upset, emergency or faulted, as applicable.

Significant discontinuities were considered such as nozzles, flanges, etc. In addition to the design calculation required by the ASME B&PV Section III code, stress analysis was performed by methods outlined in the code appendices or by other methods by reference to analogous codes or other published literature.

3.9.3.1.1 ASME Code Class 1 Components and Supports

Design transients for ASME Code Class 1 components, supports and piping are discussed in Section 3.9.1.1. Loading combinations for ASME Code Class 1 components are described in Table 3.9-2. Stress limits for ASME Code Class 1 components, supports and piping are described in Table 3.9-3. The operating pressures of Code Class 1 active valves are limited to the pressures taken from the applicable primary pressure class pressure-temperature rating of the ASME Code, Section III, for the maximum temperature for the applicable condition. E

3.9.3.1.2 Core Support Structures (Class CS) and Internal Structures (Class IS) E

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

3.9.3.1.3 ASME Code Class 2 and 3 Components and Supports

Loading combinations applicable to Code Class 2 and 3 components and supports are described in Table 3.9-2. System operating conditions due to the design transients defined in Table 3.9-1, as well as any other auxiliary system specific conditions, are reviewed to determine the appropriate operating parameters to be used in the design of Code Class 2 and 3 components.

The design stress limits for each of the component's loading conditions are presented in Tables 3.9-5 through 3.9-9. Inelastic methods, as permitted by ASME Section III for Class 1 components, were not used for these components. E

3.9.3.1.3.1 Tanks, Heat Exchangers, and Filters

Pressure vessels supplied for the auxiliary systems are:

A. Shutdown Cooling Heat Exchanger.

B. Safety Injection Tanks. E

C. Containment Spray Heat Exchanger.

D. Containment Spray Mini-Flow Heat Exchanger.

E. Shutdown Cooling Mini-Flow Heat Exchanger.

Vessel assemblies, including supports, support attachment welds, and anchor bolts, are capable of withstanding specified horizontal and vertical seismic accelerations. The seismic accelerations are applied separately at the center of gravity acting in each of two orthogonal horizontal directions and either vertical direction. The stresses or reaction loads at a given point, due to the three separate analyses, are combined by the SRSS method to define a total seismic design condition. The design allowable nozzle forces and moments act in directions that yield the highest stress which combined with the seismic loads, as determined above, and other concurrent loads.

For Class 2 and 3 pressure retaining parts under the concurrent loadings of the OBE and normal operation Level-B (upset conditions), the primary membrane stress is less than 1.1S, and the primary membrane plus bending stress is less than 1.65S. No Level-C (emergency) condition that has been identified for the applicable components is more severe than the upset condition; therefore, no appropriate stress criteria are provided. Under the concurrent loadings of the normal operating condition and the SSE, the primary membrane stress is less than 2.0S, and the primary membrane plus bending stress is less than 2.4S (where S = Allowable value of ASME B&PV Code, Section III). E

Vessel components not subject to fluid pressure, such as supports, attachment welds, and anchor bolts, were designed to the stress criteria of ASME B&PV Code, Section III for the loading conditions defined above.

In cases where the natural frequency could not be increased to avoid amplification of the floor response of the postulated seismic input for a specific plant, the components are modeled as multi-mass systems, and their modal frequencies and maximum reactions are determined from the floor response spectra for the plant. The maximum damping values used are 2% for OBE and 3% for SSE. The design point reactions due to each modal loading are combined as the sum of the absolute values or by root sum square of the modal reactions, as appropriate per recommendation of Regulatory Guide 1.92. E

3.9.3.1.3.2 Valves

ASME Class 2 and 3 valves are designed by analysis to standard rules. For all loading conditions for active valves, the design pressure rating and Level-A stress limits are not exceeded. Loading combinations are in accordance with Table 3.9-2. Stress limits are in accordance with Note (a) of Table 3.9-3 for Class 1 active valves and Table 3.9-8 for non-active valves. E

3.9.3.1.3.3 Pumps

Pumps supplied for the Auxiliary Systems are:

A. Safety Injection (active) (Safeguard) Code Class 2.

B. Shutdown Cooling (active) (Safeguard) Code Class 2. E

C. Containment Spray (active) (Safeguard) Code Class 2.

The design rules and associated design stress limits applied in the design of ASME Code Class 2 and 3 pumps are in accordance with the ASME Code, Section III, Subsections NC and ND, respectively. The results are as described herein.

Stress limits for active pumps are shown in Table 3.9-1 and stress limits for non-active pumps are shown in Table 3.9-6. Loading combinations are in accordance with Table 3.9-2.

Pump assemblies, including supports, support attachment welds, and bolts, are capable of withstanding specified horizontal and vertical seismic accelerations. The seismic accelerations are applied separately at the center of gravity acting in each of two orthogonal horizontal directions and either vertical direction.

The stresses or reaction loads at a given point, due to the three separate analyses, are combined by the SRSS method to define a total seismic design condition. The design allowable nozzle forces and moments act in directions that yield the highest stress when combined with the seismic loads, as determined above, and other concurrent loads.

E

The stress criteria of the ASME Code, Section III are applied in the design of component supports to the same Code Class as the pressure boundary involved within the jurisdictional boundaries defined in the code for the loading conditions defined above. Those steel support structures which are considered to be an extension of the building structure, but supplied with the pump assembly (i.e., bedplates), are designed to the stress criteria of the AISC Manual of Steel Construction.

In addition, the Safeguard Pump assemblies are required to be capable of withstanding the following thermal transients:

A. Safety Injection and Containment Spray suction temperature increases from 40°F to 300°F in 10 seconds. After each temperature change the end point is assumed to hold until temperature equilibrium is attained. Temperature returns to 40°F in several days. This transient would be applied a minimum of 10 times during the design life of the pump. E

B. Shutdown cooling operation applied for 500 cycles as follows: E

   1. Suction temperature increases from 70°F to 350°F in about 1 minute.

2. Suction temperature decrease from 350°F to 70°F in several hours.

Note that the containment spray pumps will also experience a limited number of shutdown cooling transient cycles.

3.9.3.1.4 Piping and Piping Supports

3.9.3.1.4.1 ASME Code Class 1

A. Piping

For ASME Code Class 1 piping, the combinations of design loadings are categorized with respect to service levels, identified as Level A, Level B, Level C, or Level D, as shown in Tables 3.9-10 and 3.9-11. The design stress limits for each of the loading combinations are found in ASME B&PV Code, Section III, NB-3600.

B. Piping Supports

For pipe supports, the design loading combinations are presented in Tables 3.9-11 and 3.9-14. The design service stress limits for all loading service levels shall be consistent with ASME B&PV Section III, Subsection NF.

3.9.3.1.4.2 ASME Code Class 2 and 3

A. Piping

For ASME Code Class 2 and 3 piping the combinations of design and service loadings are categorized with respect to system service levels identified as Design, Level A, B, C and D as shown in Tables 3.9-12 and 3.9-13. The design stress limits for each of the loading combinations are found in ASME B&PV Code, Section III, NC/ND-3600. E

B. Piping Supports

For pipe supports, the design and service loading combinations are presented in Tables 3.9-11 and 3.9-14. The design stress limits for all loading service levels shall be consistent with ASME Section III, Subsection NF.

C. Functional Capability

To address the functional capability of piping, the criteria outlined in Texas Utilities letter TXX 3423 is used. These criteria have been reviewed and accepted by the Mechanical Engineering Branch of the NRC.

3.9.3.2 Pump and Valve Operability Assurance

3.9.3.2.1 Active ASME Code Class 2 and 3 Pumps and Class 1, 2 and 3 Valves Furnished with the NSSS

3.9.3.2.1.1 Operability Assurance Program

Active pumps and valves are defined as pumps and valves that must perform a mechanical motion in order to shut down the plant, maintain the plant in a safe shutdown condition, or mitigate the consequences of a postulated event. The operability (i.e., performance of this mechanical motion) of active components during and after exposure to design bases events is confirmed by: E

A. Designing each component to be capable of performing all safety functions during and following design bases events. The design specification includes applicable loading combinations, and conservative design limits for active components. The specification requires that the manufacturer demonstrate operability by analysis, by test, or by a combination of analysis and test. The results are independently reviewed by the NSSS Supplier considering the effects of postulated failure modes on operability. E

B. Analysis and/or test demonstrating the operability of each design under the most severe postulated loadings. Methods/results of operability demonstration programs are detailed in Sections 3.9.3.2.1.2 and 3.9.3.2.1.3. E

C. Inspection of each component to assure compliance of critical parameters with specifications and drawings. This inspection confirms that specified materials and processes were used, that wall thicknesses met code requirements, and that fits and finishes met the manufacturer's requirements based on design clearance requirements.

D. Shop testing of each component to verify "as-built" conditions, as defined in Sections 3.9.3.2.1.2 and 3.9.3.2.1.3.

E. Startup and periodic in-service testing in accordance with ASME Boiler and Pressure Vessel Code, Section XI to demonstrate that the active pumps and valves are in operating condition throughout the life of the plant.

NSSS active pumps are listed below with a brief description of active safety function of each. NSSS active valves are listed in Table 3.9-4.

Active Components	Active Safety Function
Safety injection pumps	Operate at flow rates to runout
Shutdown cooling pumps	Operate at design flow
Containment spray	Operate at design flow

E

3.9.3.2.1.2 Operability Assurance Program Results for Active Pumps

Operability of the Safety Injection, Shutdown Cooling and Containment Spray pumps under required conditions has been demonstrated by analyses of the assemblies and by analyses and tests of the motors.

For the safety injection, shutdown cooling and containment spray pumps, allowable stresses are not exceeded, clearances are acceptable and shaft and pedestal bolt deflections do not cause stresses to exceed the normal values.

Where necessary, lumped mass models are used with the computer programs to determine the natural frequencies and displacements. The models are conservative (i.e., simplifications tend to make them more flexible).

To verify "as-built" conditions the pumps are hydrostatically tested in accordance with the ASME B&PV Code, Section III to confirm acceptability of structural integrity of pressure retaining parts, tested for seal leakage, and tested for performance and NPSH characteristics in accordance with the Hydraulic Institute Standard to verify operation within specified parameters. The motors are Class IE and are tested in accordance with IEEE Standard 112A-1978 to verify operation within specified parameters. Additionally, IEEE Standard 323-1974, as endorsed by Regulatory Guide 1.89, and IEEE Standard 344-1975, as endorsed by Regulatory Guide 1.100, are applicable for motors to assure operability during and following design basis events. E

3.9.3.2.1.3 Operability Assurance Program for Active Valves

Safety-related active valves must perform their mechanical motion during or after design basis events. The qualification program E assures that these valves will operate during a seismic event. Qualification tests and/or analyses are conducted for all active valves.

Class 1, 2 and 3 valves are designed/analyzed according to the rules of the ASME Boiler and Pressure Vessel Code, Section III, Section NB-3500, NC-3500, and ND-3500 respectively.

Procurement specifications for safety-related active valves shall conform to the intent of Regulatory Guide 1.48 and will stipulate that the valve vendor shall submit either detailed calculations and/or test data to demonstrate operability when subjected to the specification loading and stress criteria (normal through faulted conditions). The decision to accept actual or prototype test data, or analysis for operability assurance is made during the normal design and procurement process. The decision to test is based on: E

A. Whether the component is amenable to analysis.

B. Whether proven analytical methods are available.

C. Whether applicable prototype test data is available.

If analysis or prototype test data is not sufficient, testing is conducted to qualify the component or to verify the analytical technique.

Where appropriate, valve stem deflection calculations are performed to determine deflections due to short term seismic and other applicable loadings. Deflections so determined are compared to allowable clearances. It must be noted that seismic events are of short duration; thus, contact (if it occurs) does not demonstrate that operability is adversely affected. Cases where contact occurs are reviewed on a case by case basis to determine acceptability.

The operability of active Code Class 1, 2 and 3 components is assured through an extensive program of design verification qualification testing and thorough surveillance of the manufacturing, assembly and shop testing of each active component. Each aspect of the design related to pressure boundary integrity and operability is either tested or verified by calculations. Procedures for testing are developed by component manufacturers and reviewed and approved before the tests are conducted. The design analyses of the component take into consideration environmental conditions including loadings developed from seismic, operational effects, and pipe loads. Where necessary and feasible, the conclusions of these analyses are confirmed by test. E

On all active valves, an analysis of the extended structure is also performed for static equivalent SSE loads supplied at the center of gravity of the extended structure. The maximum stress limits allowed in these analyses show that structural integrity is within the limits developed and accepted by the ASME Code.

The safety-related valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed:

A. Shell hydrostatic test to ASME Sections III requirements.

B. Backseat and main seat leakage tests.

C. Disc hydrostatic test.

D. Functional tests to verify that the valve will open and close within the specified time limits.

E. Operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation) according to IEEE Standard 382-1972, as endorsed by Regulatory Guide 1.73.

Cold hydro qualification tests, hot functional qualification tests, periodic in-service inspections, and periodic in-service operation are performed in situ to verify and assure the functional ability of the valves. These tests ensure the reliability of the valve for the design life of the plant. The valves are designed using either stress analyses or the pressure containing minimum wall thickness requirements.

All the active valves shall be designed to have a first natural frequency which is greater than 33 Hz. This is shown by suitable test or analysis.

In addition to the above, the following specific operability assurances are provided for the various type valves:

3.9.3.2.1.3.1 Pneumatically Operated Valves

Pneumatic operated valves are furnished by several vendors. Methods of operability demonstration are summarized below. E
Spring actuation of the valve is the required active safety function. Loss of electric power or supply air will result in venting of the actuator and return of the valve to the safe position. Each vendor provides their own method to demonstrate valve operability. The operability for these valves is demonstrated by analysis, test or by a combination of analysis

and test. The vendor considers concurrent loads including seismic, design pressure and pipe loads.

The three-way solenoid valve was qualified by test and analysis to IEEE Standard 382-1972, as endorsed by Regulatory Guide 1.73, IEEE Standard 323-1974 and IEEE Standard 344-1975. Testing included thermal aging, radiation aging, wear aging, vibration endurance, seismic event simulation, and loss-of-coolant-accident. All test results provided satisfactory evidence of air solenoid valve operability. E

Limit switches, used to determine valve position, were qualified by testing and analysis to IEEE Standard 323-1974, IEEE Standard 344-1975 and IEEE Standard 382-1972. Switches were successfully performance tested for aging simulation, wear aging, radiation exposure, seismic qualification, and design basis event environmental conditions. For valves outside of containment and utilizing EA-170 limit switches, the switches were seismically qualified to IEEE Standard 344-1975 and were tested to sustain radiation dosages up to 2×10^8 rads. E

3.9.3.2.1.3.2 Motor Operated Valves

Motor operated valves are qualified by analysis as a minimum as described above. The analysis for each valve assembly considers the effects of seismic loads, design pressure, and piping reaction forces to provide assurance of operability.

To provide full qualification of the motor operated valve actuator, environmental and seismic qualification tests were conducted to simulate the following conditions:

A. Inside Containment (LOCA).

B. Outside Containment.

C. Seismic Qualification.

D. Steam Line Break Accident.

Mid-size valve actuators were subjected to complete environmental qualification consisting of inside containment and outside containment. Each qualification exposed the actuator to thermal and mechanical aging, radiation aging, seismic aging, environmental transient profile test, and steam line break. For the steam line break test an actuator was subjected to a very high superheated temperature to demonstrate that the electrical components of the actuator never exceeded the saturated temperature corresponding to the ambient pressure for the short duration of the test. This short term test provided evidence E that the existing qualification envelopes the steam line break for superheated temperatures as high as approximately 492°F for a few minutes (see Section 3.11). E

The qualification of the mid-size valve actuator was used to generically qualify all sizes of mid-size valve actuator operators for the environmental test conditions in accordance with IEEE Standard 382-1972. All sizes are constructed of the same materials with components designed to equivalent stress levels, and to the same clearances and tolerances with the only difference being in physical size which varies corresponding to the differences in unit rating.

All the qualifications were conducted per IEEE Standard 382-1972 and meet the requirements of IEEE Standard 323-1974 and IEEE Standard 344-1975 as they apply to valve motor actuators. Further, since the actuators performed satisfactorily without maintenance throughout the various qualifications, the valve actuators are fully qualified for use in CE Nuclear Power Generating Plants.

3.9.3.2.1.3.3 Pressurizer Safety Valves

Pressurizer Safety valves are 6 x 8 valves. Operability has been successfully demonstrated by a combination of dynamic testing and analysis or by static testing. Operability was successfully demonstrated with a 6g seismic load by one vendor or with a 7.1g seismic load by another vendor. Dynamic testing has demonstrated that the natural frequency of both valves was greater than 33 Hz. A summary of the test programs follows:

A. Vendor A Safety Valves

   1. Natural Frequency Demonstration

      Vibration input was in a single, horizontal direction. It was established by previous experience that the horizontal direction was more significant than the vertical direction, and that there was no material difference between the various horizontal directions. The frequency of vibration was increased from 5 to 75 Hz at a rate of 1 octave per minute. Accelerometers were mounted on the valve assembly. The actual natural frequency under test conditions was 38 Hz.

   2. Operability Demonstration

      A series of tests demonstrated that the valve would fully open and reseat during and after a seismic acceleration. Vibration input ranged from 3 to 6g and

10 to 33 Hz. The tests were performed using saturated steam. In addition, analysis was used to establish the significance of nozzle loading. The results indicated that deformation was significantly less than the internal clearances. This loading was, therefore, neglected in the seismic operability tests.

B. Vendor B Safety Valves

1. Natural Frequency Demonstration

A resonance survey was performed along three orthogonal axes with one axis being the centerline of the outlet port. (Valve mounted on inlet port.) No resonant frequencies were detected in the range of 1-50 Hz on any axis.

2. Operability Demonstration

A series of tests demonstrated that the valve would fully open and reseat during and after applying the following loading combinations: Static seismic loads up to 7.1g were applied to the valve in the direction of least bending stiffness. In addition the maximum permissible piping loads were applied concurrently. The tests were performed using saturated steam. Valve operation was satisfactory.

C. EPRI Testing of Safety Valves

Pressurizer safety valves were tested in the EPRI Test Program under full pressure and full flow conditions. This testing has demonstrated that stable valve operation under these conditions is dependent upon the inlet pipe configuration, built up back pressure range and blowdown setting. Prior to valve shipment, the inlet pipe configuration and built up back pressure range for the specific plant will be examined by CE and the applicable valve vendor. If necessary, the valves will be adjusted to provide blowdown settings which will result in stable valve operation. These blowdown settings will be recommended by the vendor and approved by CE. These adjustments will be based on the results obtained in the EPRI Test Program. Required adjustments to the valve to assure operability will be documented in the site-specific SAR. E E

3.9.3.2.1.3.4 Check Valves

The check valves are characteristically simple in design and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve since the valve disc is designed to be isolated from the casing wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any casing distortions caused by nozzle load. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valve will also undergo:

A. Stress analysis, including the SSE loads.

B. In-shop hydrostatic tests.

C. In-shop seat leakage test.

D. Periodic in-situ valve exercising and inspection, to assure the functional ability of the valve.

3.9.3.2.2 Non-NSSS Active ASME Code Class 2 and 3 Pumps and Class 1, 2 and 3 Valves

3.9.3.2.2.1 Pumps

Safety-related active pumps are subjected to in-shop tests that include hydrostatic tests of casing to 150% of the design pressure, and performance tests to determine total development head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor characteristics. Vibration is monitored during the performance tests.

E

In addition to the required testing, the pumps are designed and supplied in accordance with the following specified criteria:

A. In order to ensure that the active pump will not be damaged during the seismic event, the pump manufacturer is required to demonstrate by test and/or analysis that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed. The natural frequency of the support is determined and used in conjunction with the plant seismic response spectra. The deflection determined from the static shaft analysis is compared to the allowable rotor clearances. The pump manufacturer is required to demonstrate the pump operability during and after the SSE.

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The static deflection analyses are performed using the adjusted accelerations.

B. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to ensure that unacceptable system misalignment cannot occur.

C. To complete the seismic qualification procedures, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation during the maximum seismic event in accordance with IEEE Standard 344-1975. If the testing option is chosen, sine-beat or sweep testing for the electrical equipment is justified by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible or that the sine-beat or sine-sweep is of sufficient magnitude to conservatively account for this effect:

1. The equipment response is basically due to one mode.

2. The sine-beat response spectrum in the region of significant response. E

3. The floor response spectrum consists of one dominant mode and has a narrow peak at this frequency.

The degree of coupling in the equipment, in general, determines if a single or multiaxis test is required. Multiaxis testing is required if there is considerable cross-coupling. If coupling is very light, then single-axis testing is justified or, if the degree of coupling can be determined, then single-axis testing can be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

From this, it is concluded that the safety-related pump/motor assemblies manufacturer show that it will not be damaged and will continue operating under SSE loadings and will perform their intended functions.

3.9.3.2.2.2 Valves

Safety-related active valves are subjected to the following tests:

A. Shell hydrostatic tests, in accordance with ASME B&PV Code, Section III requirements.

B. Backseat and main seat leakage tests.

C. Disc hydrostatic tests.

D. Functional tests that verify that the valve will open and close with the specified time limits when subjected to the design differential pressure.

E. Operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident, environment simulation) in accordance with IEEE Standards 323-1974, 344-1975, and 382-1972.

After installation, cold hydrostatic tests, hot functional tests, and periodic inservice operation are performed to verify and assure the functional ability of the valve. These tests enhance reliability of the valve for the design life of the plant.

The valves are designed using either stress analysis or standard design rules for minimum wall thickness requirements. On all active valves with extended topworks, an analysis is also performed for static equivalent OBE loads applied at the center of gravity of the extended structure. E

The maximum stress limits allowed in the analyses are those recommended by the ASME Code for the particular ASME Class of valve analyzed.

In addition to these tests and analyses, valves are tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The valve is mounted in a manner that represents typical valve installation. The valve unit includes the operator and all appurtenances normally attached to the valve appurtenances in service. The operability of the valve during SSE is demonstrated by satisfying the following criteria:

A. All the active valves with extended topworks are designed to have a first natural frequency greater than 33 Hz. This may be shown by test and/or analysis. Valves with a first natural frequency less than 33 Hz are discussed below.

B. While in the shop and installed in a suitable test rig, the extended topworks of the valve are subjected to a statically applied equivalent seismic load. The load is applied at the center of gravity of the operator in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static load tests.

C. The valve is then operated with the equivalent seismic static load applied (i.e., from the normal operating status to the faulted operating status). The valve must perform its safety-related function within the specified operating time limits. Three full-stroke operations are required.

D. Motor operators and other electrical appurtenances necessary for operation are qualified as operable during the SSE by IEEE Standard 344-1975, Seismic Qualification Standards, prior to their installation on the valve.

The piping designer supports the piping in such a way that the equivalent seismic static load accelerations are not exceeded at the valve inlet and outlet support points. If the frequency of the valve with topworks, by test or analysis, is less than 33 Hz, a dynamic analysis of the valve is performed to determine an equivalent acceleration that is to be applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and frequency content of the plant floor response spectra. The adjusted accelerations are determined using margins similar to that contained in the horizontal and vertical accelerations used for "rigid" valves. The adjusted accelerations are used in the static analysis, and valve operability is assured by the methods outlined in listings B to D above, using the modified acceleration input. E

The above testing program applies only to valves with overhanging structures (e.g., the operator). The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types (e.g., motor-operated gate valve) are tested. Specific valves are qualified by the tests, and the results are extended to qualify valves within a range of sizes. An analysis is conducted to prove the similarity between the tested valve and the installed ones.

Due to the simple characteristics of check valves and other compact valves, they are qualified by the following tests and analysis:

A. Stress analysis of the attached piping for SSE loads.
B. In-shop hydrostatic test.

C. In-shop seat leakage test.

D. Periodic valve exercise and inspection to assure the functional ability of the valve.

Using the methods described, safety-related active valves in the system are qualified for operability during a seismic event.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Safety valves and relief valves are analyzed in accordance with the ASME Section III Code.

The method of analysis for safety valves and relief valves suitably accounts for the time-history of loads acting immediately following a valve opening (i.e., first few milliseconds). The fluid-induced forcing functions are calculated for each safety valve and relief valve using one-dimensional equations for the conservation of mass, momentum, and energy. The calculated forcing functions are applied at locations along the associated piping where a change in fluid flow direction occurs. Application of these forcing functions to the associated piping model constitutes the dynamic time-history analysis. The dynamic response of the piping system is determined from the input forcing functions. Therefore, a dynamic amplification factor is inherently accounted for in the analysis. Alternatively, an equivalent static analysis may be used following the criteria given in Appendix II of the ANSI/ASME B31.1 Code. This appendix provides a methodology for calculating appropriate dynamic load factors. Where more than one safety relief valve is installed on the same piping run, the sequence of openings that induces the maximum stress will be considered. E

Snubbers or strut-type restraints are used as required. The stresses resulting from the loads produced by the sudden opening of a relief or safety valve are combined with stresses due to other pertinent loads and are shown to be within allowable limits of the ASME Section III Code. Also, the analyses show that the loads applied to the nozzles of the safety and relief valves do not exceed the maximum loads specified by the manufacturer.

3.9.3.4 Component Supports

Supports for ASME Section III Code Class 1, 2 and 3 components are specified for design in accordance with the loads and loading combinations discussed in Section 3.9.3.1 and presented in Table 3.9-2.

Component supports which are loaded during normal operation, seismic and following a pipe break (branch line breaks not eliminated by leak-before-break) are specified for design for loading combinations (A) through (D) of Section 3.9.3.1. Design stress limits applied in evaluating loading combinations (A), (B), and (C) of Section 3.9.3.1 are consistent with the ASME Code, Section III. The design stress limits applied in evaluating loading combination (D) of Section 3.9.3.1 are in accordance with the ASME B&PV Code, Section III. Loads in compression members are limited to 2/3 of the critical buckling load.

Where required, snubber supports are used as shock arrestors for safety-related systems and components. Snubbers are used as structural supports during a dynamic event such as an earthquake or a pipe break, but during normal operation act as passive devices which accommodate normal expansions and contractions of the systems without resistance. For System 80+, snubbers are minimized, to the extent practical, through the use of design optimization procedures.

Assurance of snubber operability is provided by incorporating analytical, design, installation, in-service, and verification criteria. The elements of snubber operability assurance for System 80+ include: E

A. Consideration of load cycles and travel that each snubber will experience during normal plant operating conditions.

B. Verification that the thermal growth rates of the system do not exceed the required lock-up velocity of the snubber.

C. Accurate characterization of snubber mechanical properties in the structural analysis of the snubber-supported system.

D. For engineered, large bore snubbers, issuance of a design specification to the snubber supplier, describing the required structural and mechanical performance of the snubber; verification that the specified design and fabrication requirements are met.

E. Verification that snubbers are properly installed and operable prior to plant operation, through visual inspection and through measurement of thermal movements of snubber-supported systems during start-up tests.

F. A snubber in-service inspection and testing program, which includes periodic maintenance and visual inspection, inspection following a transient event, a functional testing program, and repair or replacement of snubbers failing inspection or test acceptance criteria.

3.9.4 CONTROL ELEMENT DRIVE MECHANISMS

3.9.4.1 Descriptive Information of CEDM

The control element drive mechanism (CEDMs) are magnetic jack type drives used to vertically position and indicate the position of the control element assemblies (CEAs). Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its 153-inch stroke in response to operation signals.

The CEDM is designed to function during and after all normal plant transients. The CEA drop time for 90% insertion is 4.0 seconds maximum. The drop time is defined as the interval between the time power is removed from the CEDM coils to the time the CEA has reached 90% of its fully inserted position. The CEDM pressure boundary components have a design life of 60 years. The CEDM is designed to operate without maintenance for a minimum of 1-1/2 years and without replacing components for a minimum of 3 years. The CEDM is designed to function normally during and after being subjected to the Operating Basis Earthquake loads. The CEDM will allow for tripping of the CEA during and after a Safe Shutdown Earthquake. E E

The design and construction of the CEDM pressure housing fulfill the requirements of the ASME boiler and Pressure Vessel Code, Section III, for Class 1 vessels. The CEDM pressure housings are part of the reactor coolant pressure boundary, and they are designed to meet stress requirements consistent with those of the vessel. The pressure housings are capable of withstanding, throughout the design life, all normal operating loads, which include the steady-state and transient operating conditions specified for the vessel. Mechanical excitations are also defined and included as a normal operating load. The CEDM pressure housings are service rated at 2500 psi at 650°F. The loading combinations and stress limit categories are presented in Table 3.9-16 and are consistent with those defined in the ASME code.

THIS PAGE INTENTIONALLY BLANK

The design duty requirements for the CEDM is a total cumulative CEA travel of 100,000 feet operation without loss of function.

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

3.9.4.1.1 Control Element Drive Mechanism Design Description

The CEDMs are mounted on nozzles on top of the reactor vessel closure head. The CEDMs consist of the upper and lower CEDM pressure housings, motor assembly, coil stack assembly, reed switch assemblies, and extension shaft assembly. The CEDM is shown in Figure 3.9-8. The drive power is supplied by the coil stack assembly, which is positioned around the CEDM housing. Two position indicating reed switch assemblies are supported by the upper pressure housing shroud, which encloses the upper pressure housing assembly.

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

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

The axial position of a CEA in the core is indicated by three independent readout systems. One counts the CEDM steps electronically, and the other two consist of magnetically actuated reed switches located at regular intervals along the CEDM. These systems are designed to indicate CEA position to within ± 2-1/2 inches of the true location. This accuracy requirement is based on ensuring that the axial alignment between CEAs/PLCEAs is maintained within acceptable limits. E

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

3.9.4.1.1.1 CEDM Pressure Housing

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

3.9.4.1.1.2 Motor Assembly

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

Both upper latches and lower latches are used to perform the stepping of the CEA and by proper sequencing perform a load transfer function and to minimize latch and extension shaft wear. The upper latch also performs the holding when CEA motion is not required. Engagement of the extension shaft occurs when the appropriate set of magnetic coils is energized. This moves sliding magnets which cam a two-bar linkage moving the latches inward. The upper latches move vertically 7/16 inches while the lower latches move vertically 3/8 inches to perform both the load transfer and stepping action. Total CEA motion per cycle is 3/4 inches.

3.9.4.1.1.3 Coil Stack Assembly

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

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

3.9.4.1.1.4 **Reed Switch Assembly**

Two reed switch assemblies provide separate means for transmitting CEA position indication. Reed switches and voltage divider networks are used to provide two independent output voltages proportional to the CEA position. The reed switch assemblies are positioned so as to utilize the permanent magnet in the top of the extension shaft. The permanent magnet actuates the reed switches as it passed by them. The reed switch assemblies are provided with accessible electrical connectors at the top of the upper pressure housing.

3.9.4.1.1.5 **Extension Shaft Assembly**

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

The drive shaft is a long tube made of Type 304 stainless steel. It is threaded and pinned to the extension shaft. The drive shaft has circumferential notches in 3/4 inch increments along the shaft to provide the means of engagement to the control element drive mechanism.

The magnet assembly, located in the top of the extension shaft assembly, consists of a housing, magnet and plug. The magnet is made of two cylindrical alnico -5 magnets. This magnet assembly is used to actuate the reed switch position indication and is contained in a housing which is plugged at the bottom of the housing.

3.9.4.1.2 **Description of the CEDM Motor Operation**

Withdrawal or insertion of the CEA is accomplished by programming current to the various coils. There are three programmed conditions for each coil (i.e., high voltage for initial gap closure, low voltage for maintaining the gap closed and zero voltage to allow opening of the gap).

3.9.4.1.2.1 **Operating Sequence for the Double Stepping Mechanism**

The initial condition is the hold mode. In this condition, the upper latch coil is energized at low voltage.

A. Withdrawal (Ref. Figure 3.9-8)

1. The upper lift coil is energized causing the 7/16" upper lift gap to close lifting the CEA.

2. Low current is supplied to hold the CEA in the withdrawn position.

3. The lower latch coil is energized causing the lower latches to engage the drive shaft with 1/32-inch clearance.

4. The upper lift coil is deenergized allowing the upper latches to drop 7/16 inches and the drive shaft to lower 1/32 inches placing the load on the lower latches.

5. The upper latch coil is deenergized disengaging the upper latches.

6. The lower lift coil is energized lifting the drive shaft 3/8 inches.

7. The upper latch coil is energized engaging the upper latches in the drive shaft with 1/32-inch clearance.

8. The lower lift coil is deenergized allowing the lower latches to drop 3/8 inches and causing the drive shaft to drop 1/32 inches applying the load on the upper latches.

9. The lower latch coil is deenergized disengaging the lower latches from the drive shaft.

B. Insertion

1. The lower latch coil is energized causing the lower latches to engage the drive shaft.

2. The lower lift coil is energized lifting the lower latches 3/8 inches and lifting the drive shaft 1/32 inches thus applying the load to the lower latches.

3. The upper latch coil is deenergized causing the upper latches to disengage the drive shaft.

4. The upper lift coil is energized moving the deenergized upper latch assembly up 7/16 inches. E

5. The upper latch coil is energized engaging the latches with clearance.

6. The lower lift coil is deenergized allowing the lower latch to drop with the drive shaft. The drive shaft will move down 3/8 inch, stopping on the upper latch assembly, which is energized and in its up position. E

7. The lower latch coil is deenergized disengaging the lower latches.

8. The upper lift coil is deenergized lowering the upper latch assembly with the drive shaft 3/8 inch. E

3.9.4.2 Applicable CEDM Design Specifications

The pressure boundary components are designed and fabricated in accordance with the requirements for Class 1 vessels per the applicable Edition and Addenda of Section III of the ASME Boiler and Pressure Vessel Code. The pressure boundary material complies with the requirements of Section III and IX of the ASME Boiler and Pressure Vessel Code and Code Case N4-11. E

The adequacy of the design of the non-pressure boundary components have been verified by prototype accelerated life testing as discussed in Section 3.9.4.4.

The reed switch position transmitter assembly of the CEDM is designed to comply with IEEE 323-1974, standard for "Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations," and IEEE 344-1975, "Recommended Practice Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations." The electrical components are external to the pressure boundary and are non-pressurized.

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

3.9.4.3 Design loads, Stress Limits and Allowable Deformations

The CEDM stress analyses consider the following loads:

A. Reactor coolant pressure and temperature

B. Reactor operating transient conditions

C. Dynamic stresses produced by seismic loading

D. Dynamic stresses produced by mechanical excitations

E. Loads produced by the operation and tripping of the mechanism

The methods used to demonstrate that the CEDMs operate properly under seismic conditions are presented in Section 3.7.3.14.

The design and fabrication of the CEDM pressure boundary components fulfills the requirements of the ASME Code, Section III, for Class I vessels. The pressure housings are capable of withstanding throughout the design life all the steady state and transient operating conditions specified in Table 3.9-16.

The adequacy of the design of the CEDM pressure boundary and non-pressure boundary components has been verified by prototype accelerated life testing as discussed in Section 3.9.4.4.

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

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

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

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

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

3.9.4.4 CEDM Performance Assurance Program

3.9.4.4.1 CEDM Testing

3.9.4.4.1.1 Prototype Accelerated Life Tests

The System 80+ CEDM is similar to and based on existing magnetic jack mechanisms presently in use on operating reactors such as Maine Yankee (Docket No. 50-309) and Calvert Cliffs (Docket 50-317), the 150-inch core reactors such as Arkansas Nuclear One Unit 2 (Docket No. 50-368) and San Onofre Units 2 & 3 (Docket No. 50-361/362), and is the same as the System 80 CEDM presently in use at Palo Verde (Docket Nos. 50-528, 529).

The significant differences between the System 80+ drives and pre-System 80 CEDMs are:

E

A. The elimination of the pulldown coil.

B. The use of the lift coils to perform both a load transfer function and stepping action.

The elimination of the pulldown coil required installation of a coil spring to ensure positive resetting of the latch assemblies. In addition, the drive shaft was modified by placing the teeth on 3/4-inch pitch in place of the 3/8-inch spacing of previous drive shafts to allow load transfer and stepping with the same coil. The safety release mechanism uses the same materials and clearances as on all previous magnetic jack mechanisms. The following describes accelerated life tests on both a pre-System 80 mechanism as well as on a prototype System 80 CEDM. Both programs provide design verification for the System 80+ CEDM. E

A pre-System 80 prototype CEDM was subjected to an accelerated life test accumulating a minimum of 157,000 feet of travel on all CEDM components. In addition, the latch guide tube bearings in the motor assembly saw an additional 50,000 feet of operation.

The prototype mechanism was installed on a test facility which was operated at a nominal temperature of 600°F and 2250 psi. After 50,000 feet of operation lifting 230 pounds at 40 inches per minute, the motor was removed from the test motor housing and the bearing surfaces inspected. During this inspection it was found that excessive wear existed on the upper gripper magnet and upper gripper housing bearings.

The gripper housing magnet bearing configuration was revised and replacement parts with this revision were incorporated into the prototype mechanism. This configuration was reinstalled into the test facility and the mechanism operated as before for an additional 157,000 feet of travel. The replacement parts showed a wear of only .001 inches while the latch guide tube bearings had a total wear of 0.012 inches. The mechanism at disassembly was still operational with no abnormalities. This test constituted operation equivalent to 1.5 to 2.0 times the design duty requirements of the mechanism.

A prototype System 80 CEDM was assembled and installed in a test loop, where the accelerated wear test was conducted at 615°F and 2250 psi. The total weight attached to the CEDM was 450 pounds and this was moved at a nominal speed of 30 inches per minute. A total of 34,000 feet of travel was then completed without difficulty. Included in that test footage were 300 full-height gravity scrams.

The mechanism motor was removed from the test facility and disassembled for inspection. The latch guide tube bearings showed a maximum diametral wear of 0.003 inches with negligible wear on the gripper housing to gripper magnetic bearings. Alignment tabs, which maintain orientation of the gripper with the latch guide tube, showed extensive wear but had not caused mechanism malfunctions. These alignment tabs have been replaced in the production units with an improved design.

Upon completion of the accelerated wear test, 300 full height light weight drops were completed utilizing a 75-pound test weight. The maximum CEA drop time to 90% insertion was 2.93 seconds which met the 4.0 second criterion. All release times were less than the 0.3 seconds with normal releases completed in less than 0.200 seconds.

3.9.4.4.1.2 First Production Test

A qualification test program was completed on the first production C-E magnetic jack CEDM. A similar test program was invoked for the System 80 CEDMs. During the course of this program, over 4000 feet of travel was accumulated and 30 full height gravity drops were made without mechanism malfunction or measurable wear on operating parts. The program included the following:

A. Operation at 40 in./min lifting 230 pounds (dry) at ambient temperature and 2300 psig pressure for 800 feet.

B. Six full-height 230 pounds dry weight gravity drops at ambient temperature.

C. Operation at simulated reactor operating condition at 40 in/min lifting 230-pound for 1700 feet.

D. Six full-height drops at simulated reactor operating conditions with 230 pounds of weight.

E. An operational test at ambient temperature and 2300 psig pressure, lifting 335 pounds for 500 feet. E

F. Six full-height drops of the 335 pound weight.

G. Operation at simulated reactor conditions for 1700 feet at 20 in/min, lifting 335 pounds.

H. Operation at ambient temperature and 2300 psig for 1100 feet and 20 full-height drops with an attached dry weight of 130 pounds.

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

3.9.4.4.1.3 **Operating Experience at the Palo Verde Nuclear Generating Station**

The System 80+ CEDMs are identical to those in operation at PVNGS. That experience has shown that the CEDMS operate without malfunction and without any measurable wear. E

3.9.5 **REACTOR VESSEL CORE SUPPORT AND INTERNALS STRUCTURES**

3.9.5.1 <u>Design Arrangements</u>

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

3.9.5.1.1 **Core Support Structure**

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

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

3.9.5.1.1.1 **Core Support Barrel**

The core support barrel is a right circular cylinder including a heavy external ring flange at the top end and an internal ring flange at the lower end. The core support barrel is supported from a ledge on the pressure vessel. The core support barrel, in turn, supports the lower support structure upon which the fuel assemblies rest. Press-fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head, and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide alignment between these components in the vessel flange region. The core support barrel assembly is shown in Figure 3.9-10.

The upper section of the barrel contains two outlet nozzles that interface with internal projections on the vessel nozzles to minimize leakage of coolant form inlet to outlet. Since the weight of the core support barrel is supported at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally-spaced lugs around the circumference of the barrel and act as a tongue-and-groove assembly with the mating lugs on the pressure vessel. Minimizing the clearance between the two mating pieces limits the amplitude of vibration. During assembly, as the internals are lowered into the pressure vessel, the pressure vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted. The pressure vessel lugs have bolted, captured Inconel X shims. The core support barrel lug mating surfaces are hardfaced with Stellite to minimize wear. The shims are machined during initial installation to provide minimum clearance. The snubber assembly is shown in Figure 3.9-11.

3.9.5.1.1.2 Lower Support Structure and Instrument Nozzle Assembly

The lower support structure and ICI nozzle assembly position and support the fuel assemblies, core shroud, and ICI nozzles. The structure is a welded assembly consisting of a short cylinder, support beams, a bottom plate, ICI nozzles, and and ICI nozzle support plate. The lowest support structure is made up of a short cylindrical section enclosing an assemblage of grid beams arranged in egg-crate fashion. The outer ends of these beams are welded to the cylinder. Fuel assembly locating pins are attached to the beams. The bottoms of the parallel beams in one direction are welded to an array of plates which contain flow holes to provide proper flow distribution. These plates also provide support for the ICI nozzles and, through support columns, the ICI nozzle support plate. The cylinder guides the main coolant flow and limits the core shroud bypass flow by means of holes located near the base of the cylinder. The ICI nozzle support plate provides lateral support for the nozzles. This plate is provided with flow holes for the requisite flow distribution. The lower support structure and ICI nozzle assembly is shown in Figure 3.9-12.

3.9.5.1.1.3 Core Shroud

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The shroud consists of a welded vertical assembly of plates designed to channel the coolant through the core. Circumferential rings and a top and bottom end plate provide lateral support. The rings are attached to the vertical plates by means of welded ribs which extend the full length of the core shroud. A small gap is provided between the core shroud outer perimeter and the core support barrel in order to provide upward coolant flow in the annulus, thereby minimizing thermal stresses in the core shroud. The core shroud is shown in Figure 3.9-13. Four hardfaced alignment lugs, spaced 90 degrees apart, protrude vertically from the top of the core shroud and engage in corresponding hardfaced slots in the upper guide structure fuel alignment plate to ensure proper alignment between the upper guide structure assembly, core shroud, and lower support structure.

3.9.5.1.2 Upper Guide Structure Assembly

The Upper Guide Structure Assembly (UGS) assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the control element spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during a severe accident condition and protects the control elements from the effects of coolant cross flow in the upper plenum. The UGS assembly is handled as one unit during installation and refueling.

The UGS assembly consists of the UGS support barrel assembly and the CEA shroud assembly (Figure 3.9-14). The UGS support barrel assembly consists of UGS support barrel fuel alignment plate, UGS base plate and control element shroud tubes. The UGS support barrel consists of a right circular cylinder welded to a ring flange at the upper end and to a circular plate (UGS base plate) at the lower end. The flange, which is the supporting member for the entire UGS assembly, seats on its upper side against the pressure vessel head during operation. The lower side of the flange is supported by the holddown ring, which seats on the core support barrel upper flange. The UGS flange and the holddown ring engage the core support barrel alignment keys by means of four accurately machined and located keyways equally spaced at 90 degree intervals. This system of keys and slots provides an accurate means of aligning the core with the closure head and thereby with the CEA drive mechanisms. The fuel alignment plate is positioned below the UGS base plate by cylindrical control element shroud tubes. These tubes are attached to the UGS base plate and the fuel alignment plate by rolling the tubes into the plates and welding. The fuel alignment plate is designed to align the lower ends of the control element shroud tubes which in turn locate the upper ends of the fuel assemblies. The fuel alignment plate also has four equally spaced slots on its outer edge which engage with Stellite hardfaced lugs protruding from the core shroud to provide alignment. The control element shroud tubes bear the upward force on the fuel assembly holddown devices. This force is transmitted from the alignment plate through the control element shroud tubes to the UGS barrel base plate.

The CEA shroud assembly limits cross flow and provides separation of the CEA assemblies. The assembly consists of an assemblage of large vertical tubes connected by vertical plates in a grid pattern. The shroud assembly is mounted on the UGS base plate and is held in position by eight tie rod tube assemblies which are threaded into the UGS base plate at their lower end. The tie rods are bolted against plates located at the top of the CEA shroud assembly and are pretensioned. The tubes and connecting plates are furnished with multiple holes to permit hydraulic communication. Guides for the CEA extension shafts are provided by the guide structure support system (GSSS). E

The holddown ring provides axial force on the flanges of the upper guide structure assembly and the core support structure in order to prevent movement of the structures under hydraulic forces. The holddown ring is designed to accommodate the differential thermal expansion between the pressure vessel and the internals in the vessel ledge region.

3.9.5.1.3 Flow Skirt

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

3.9.5.1.4 In-Core Instrumentation Support System

The complete in-core neutron flux monitoring system includes self-powered in-core detector assemblies, supporting structures and guide paths, an external movable detector drive system and an amplifier system to process detector signals. The self-powered in-core detector assemblies and the amplifier system are described in Section 7.7. The external movable detector drive system and the instrumentation supporting structures and guide paths are described in this section and shown in Figure 3.9-15.

The support system begins outside the pressure vessel, penetrates the bottom of the vessel boundary and terminates in the upper end of the fuel assembly. Each in-core instrument is guided over its full length by the external guidance conduit, the pressure vessel nozzles, the lower support structure ICI nozzles and the instrument guide tube of the fuel assembly. Figure 3.9-12 shows the in-core instrument support structure. The in-core instrumentation support system routes the instruments so that detectors are located in selected fuel assemblies throughout the core. An equal instrument length exists for all locations. The guide tube routing outside the reactor vessel is a simple 180° bend to the seal table. The pressure boundaries for the individual instruments are at the out-of-reactor seal table, where the external electrical connections to the in-core instruments are made (Figure 3.9-15).

The in-core instrument assemblies contain a movable detector guide tube to allow insertion of a miniature movable flux detector. The assemblies have an integral seal plug which forms a seal at the instrument seal table and through which the signal cables and movable guide tube pass. Static O-ring seals are used to seal against operating pressure.

The movable detector drive system consists of two drive machines, two transfer machines, two drive cables with detectors and the interconnecting tubing. Because the two halves of the system are identical with only several connections between them (leak detection and gas purge), only half of the system is described below.

A fission chamber is used as the movable flux detection device. The detector signal cable is wound with an edgewise helical steel wrap to form the drive cable. This cable construction allows a hobbed wheel in the drive machine to drive the cable in either direction. The drive machine consists of a cable reel, a drive motor, gear reducer, hobbed drive wheel and a shaft position encoder. The detector may be positioned from the control room by use of the plant computer or a separate control box.

The detector may be shifted from any location to any other location in less than eight minutes. The detectors are shifted by the transfer machine which is mounted above the seal table. The machine consists of a geared drive motor, multiple position Geneva positioning mechanism, inlet and outlet tubes and miscellaneous limit and interlock switches. External commands control the motor to position the mechanism so that the inlet path is lined up with the correct outlet path. The transfer machine also has connections for inert gas blanketing and for guide tube leak detection. The gas connection allows an inert gas supply to blanket the transfer machine and movable detector guide tubes during machine operation.

The leak detector alarm system is a float switch mounted in a chamber which is fed from both transfer machines. Any leak which might occur in a movable detector guide tube flows to the transfer machine and then to the transfer machine sump, which exits to the leak detector. A solenoid valve past the leak detector allows remote drainage of the leak detector sensing line.

3.9.5.2 Design Loading Conditions

The following loading conditions are considered in the design of the core support and internals structures. E

A. Normal operating temperature differences

B. Normal operating pressure differences

C. Flow loads

D. Weights, reactions and superimposed loads

E. Vibration loads

F. Shock loads (including operating basis and safe shutdown earthquakes)

G. Anticipated transient loadings not requiring forced shutdown

H. Handling loads (not combined with other loads above) E

3.9.5.3 Design Loading Categories

The design loading conditions are categorized as follows:

3.9.5.3.1 Level A and Level B Service Loadings E

This category includes the combinations of design loadings consisting of normal operating temperature and pressure differences, loads due to flow, weights, reactions, superimposed loads, vibration, shock loads including operating basis earthquake, and transient loads not requiring shutdown.

3.9.5.3.2 Level D Service Loadings

The following loading combination shall be considered as Level D Service Loadings.

A. Normal Operation Loads

B. Either the Design Basis Pipe Break (DBPB), or Main Steam/Feed Water Pipe Break (MS/FWPB), or Loss of Coolant Accident (LOCA) Loads

C. Safe Shutdown Earthquake (SSE) Loads

The DBPB is defined as a postulated pipe break that results in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system (i.e. less than 150 GPM).

LOCA is defined as the loss of reactor coolant at a rate in excess of the reactor coolant normal makeup rate, from breaks in the reactor coolant pressure boundary inside primary containment up to, and including, a break equivalent in size to the largest remaining primary branch line not eliminated by leak before break (LBB) criteria.

3.9.5.4 Design Bases for Reactor Internals

The stress limits to which the reactor internals are designed are listed in Table 3.9-17.

No Level C condition has been identified for the applicable components. Therefore, no stress criteria are provided.

The operating categories and stress limits are defined in the applicable section of the Section III of the ASME Boiler and Pressure Vessel Code.

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

A. Under Level A and Level B service loadings, the core will be held in place and deflections will be limited so that the CEAs can be inserted under their own weight as the only driving force.

B. Under service loading combinations other than Level A and B service loadings that require CEA insertability, deflections are limited so that the core will be held in place, adequate

E

core cooling is preserved, and all CEAs can be inserted. Those deflections that would influence CEA movement are limited to less than 80% of the deflections required to prevent CEA insertion.

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

Location	Allowable Deflection
Fuel lower end fitting, lower support structure	2.600 inches (Disengagement)
Fuel upper end fitting, upper guide structure	1.216 inches (Disengagement)
CEA Shroud (lateral)	0.209 inches (CEA Insertion)

E

THIS PAGE INTENTIONALLY BLANK

In the design of critical reactor vessel internals components which are subject to fatigue, the stress analysis is performed utilizing the design fatigue curve of Figure I-9-2 of Section III of the ASME Boiler and Pressure Vessel Code. A cumulative usage factor of less than one is used as the limiting criterion.

As indicated in the preceding sections, the stress and fatigue limits for reactor internals components are obtained from the ASME Code. Allowable deformation limits are established as 80% of the loss-of-function deflection limits. These limits provide adequate safety factors assuring that so long as calculated stresses, usage factors, or deformations do not exceed these limits, the design is conservative.

3.9.6 IN-SERVICE TESTING OF PUMPS AND VALVES

The in-service testing program for Code Class 1, 2 and 3 pumps and valves will be developed in accordance with the requirements of Section XI of the ASME B&PV Code. This program will be implemented to assess operational readiness during preservice and in-service inspection.

3.9.6.1 In-service Testing of Pumps

In-service testing of pumps is limited to those Code Class 2 and 3 pumps which are required to perform a specific function in shutting down a reactor or in mitigating the consequences of an accident, and that are provided with an emergency power source. The required hydraulic and mechanical parameters will be measured by the methods and with frequency prescribed in Subsection IWP of ASME Section XI. The pump test plan and schedule are included in the technical specifications. E

3.9.6.2 In-service Testing of Valves

Code Class 1, 2 and 3 valves will be categorized in accordance with Subarticle IWV-2100 of ASME B&PV Code Section XI. Valves will be tested to the requirements of Subsection IWV for each valve category. The testing plan will not include those Code Class 1, 2 and 3 valves which are exempt from testing in accordance with Subarticle IWV-1200 of Section XI. The valve test procedure and schedule are included in the technical specifications.

REFERENCES FOR SECTION 3.9

1. MDC STRUDL, McDonnell Douglas Corp., St. Louis Mo.

2. "MARC-CDC User Information Manual," Volume I and Volume III, MARC Analysis Corp. and Control Data Corp. Minneapolis, Minn., 1976.

3. "LION, Temperature Distributions for Arbitrary Shapes and Complicated Boundary Conditions," Schmid, J. R., et al, KAPL-M-6532 (EC-57), July 27, 1966.

4. MRI/STARDYNE-Static and Dynamic Structural Analysis System: User Information Model, Control Data Corporation, June 1, 1970.

5. DeSalvo, G. P. and Swanson, J. A., ANSYS-Engineering Analysis System Swanson Analysis Systems, Inc., Elizabeth, Pa., 1972.

6. "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," Ghosh, S. and Wilson, E., Dept. No. EERC 69-10, University of California, Berkley, September 1969.

7. Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions with Application of Analysis to C-E 800 Mwe Class Reactors," Combustion Engineering, Inc., CENPD-42, August 1972 (Proprietary). E

8. "SHOCK, A Computer Code for Solving Lumped-Mass Dynamic Systems," Gabrielson, V. K., SCL-DR-65-34, January 1966.

9. "SAMMSOR II - A Finite Element Program to Determine Stiffness and Mass Matrices of Shells-of-Revolution," Tillerson, J. R. and Haisler, W. E., Texas A&M University, TEES-RPT-70-18, October 1970.

10. "DYNASOR II - A Finite Element Program for the Dynamic Non-Linear Analysis of Shells-of-Revolution," Tillerson, J. R. and Haisler, W. E., Texas A&M University, TEES-RPT-70-19, October 1970.

11. "Early Blowdown Analysis for Loss of Fluid Test Facility," Fabic, S., Kaiser Engineers, Report No. 65-28-RA, June 1965, Revised April 1967.

12. "FLASH-4 A Fully Implicit Fortran IV Program for the Digital Simulation of Transients in a Reactor Plant," Porsching, T. A., et al, WAPD-TM840 March 1969.

13. "Description of Loss-of-Coolant Calculational Procedures," Combustion Engineering, Inc., CENPD-26, August 1971 (Proprietary).

14. "Method for the Analysis of Blowdown Induced Forces in a Reactor Vessel," Combustion Engineering, Inc., CENPD-252-P, December, 1977 (proprietary).

15. "CEFLASH-4A: A Fortran-IV-Digital Computer Program for Reactor Blowdown Analysis," Combustion Engineering, Inc., CENPD-133P, August, 1974 (proprietary).

16. "CEFLASH-4A: A Fortran-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," Combustion Engineering, Inc., CENPD-133P, Supplement 2, February, 1975 (proprietary).

17. Scherer, A. E., Licensing Manager, (C-E), Letter to D. F. Ross, Assistant Director of Reactor Safety Division of Systems Safety, LD-76-026, March, 1976 (proprietary).

18. Parr, O. D., Chief Light Water Reactor Project Brack 1-3, Division of Reactor Licensing (NRC), Letter to F. M. Stern, Vice President of Projects (C-E), June, 1975.

19. Kneil, K., Chief Light Water Reactors Brack No. 2, Letter to A. E. Scherer, Licensing Manager (C-E), August, 1976, (Staff Evaluation of CENPD-213). E

20. SUPERPIPE - A Computer Program for Structural Analysis and Code Compliance Verification of Piping System, Impell Corporation, Walnut Creek, California.

21. "Theory of Pump Induced Pulsating Coolant Pressure in PWRs," Penzes, L. E., 2nd Int. Conf. on Structural Mechanics in Reactor Technology, Vol. II, Part E-F.

22. "Forced Vibration of a Shell Inside a Narrow Water Annulus," Horvay, G., Bowers, G., Nuclear Engr. Design V34, 1975.

23. Main Yankee Precritical Vibration Monitoring Program, Final Report, CENPD-93, February, 1973.

24. Omaha Pre-critical Vibration Monitoring Program, Final Report, CEN-8(0), May, 1974.

25. Calvert Cliffs Analysis of Flow-Induced Structural Response, CEN-4(B).

26. Comparison of Calvert Cliffs, Main Yankee and Ft. Calhoun Design Parameters and Flow-Induced Structural Response, CENPD-115 Supp. 1, April, 1974.

27. Comparison of ANO-2, Maine Yankee and Ft. Calhoun Reactor Internals Design Parameters and Flow-Induced Structural Response, CEN-8(A)-P Supp. 1, May 1975.

28. "Response of a Tube Bank to Turbulent Crossflow Induced Excitation," Lubin, B. T., ASME-77-DET-142.

29. "Pressure Distribution on Circular Cylinder at Critical Reynolds Numbers," Batham, J. P., Jl Fluid. Mech., v57, pt 2, 1973.

30. "Supercritical Reynold Number Simulation for Two Dimensional Flow Over Circular Cylinders," Szecheny, I. E., Jl. Fluid Mech, v70, pt 3, 1975.

31. "Properties of Karman Vortex Streets," Chen, Y. N., Sulzer Technical Review, 1972.

32. "Fluctuating Lift Forces of the Karman Vortex Streets on Single Circular Cylinders and in Tube Bundles," Chen, Y. N., Part 1 (ASME-71-VIBR11), Part 2 (ASME-71-VIBR-12), Part e (ASME-71-VIBR-13). E

33. "Vortex Excited Oscillations of Yawed Circular Cylinders," King, R., ASME 76-WA/FE-16.

34. Mark, W. D., Chandiramani, K. L., Karnopp, R. L., Reactor Internals Vibration Study, BBN Report 151273, Oct. 1967.

35. Final Report on Studies of Flow in a 0.248 Scale Model of the Omaha PWR, H. L. Crawford and L. J. Flanigan, August, 1970, Battelle Memorial Institute.

36. Final Report On Studies of Flow in a 1/5-Scale Model of the Palisades PWR, CEND-358, L. A. Schultz, D. A. Trayser, L. J. Flanigan, Battelle Memorial Institute, April, 1969.

37. Owen, P. R., "Buffeting Excitation of Boiler Tube Vibration," Jl. of Mechanical Science, Vol. 7, N. 4, 1965.

38. Davies, P.O.A.L., "An Experimental Investigation of the Unsteady Pressure Forces on a Circular Cylinder in a Turbulent Cross Flow," Bruun, H. H., Jl. of Sound & Vibration, V40, N.4, 1975.

39. "A Study of Free Jet Impingement," Coleman, D. D., et al., Part 2, Jl. Fluid Mech. 1971 V.45, Part 3, pg. 477-512.

40. "Spectra of Turbulence in a Round Jet," Gibson, M. M., Je. Fluid Mechanics V15, 1963.

41. Harty, W. C., Rubinstein, M. F., Dynamics of Structures, Prentice-Hall, 1964.

42. "Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading," Report No. EERC 68-10, S. Ghosh & E. L. Wilson, Sept. 1969, Univ. of Calif.

43. "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," USNRC Regulatory Guide 1.20 Rev. 2, May, 1976.

44. "A Comprehensive Vibration Assessment Program for Palo Verde Nuclear Generating Station Unit 1 (System 80 Prototype)," Combustion Engineering, Inc., CEN-263, Rev. 1 January, 1985 (Proprietary).

45. "Structural Analysis of Fuel Assemblies for Seismic and Loss-of-Coolant Accident Loading," Combustion Engineering, Inc., CENPD-178, Revision 1, August 1981. E

TABLE 3.9-1

(Sheet 1 of 3)

TRANSIENTS USED IN STRESS ANALYSIS OF CODE CLASS 1 COMPONENTS

Normal Conditions

Occurrence	Conditions
Heatup and cooldown cycles	500 heatup and cooldown cycles during the design life of the components in the system. The rate of heating and cooling is 100°F/hr between 70°F and 565°F except for the pressurizer which has a rate of 200°F/hr between 70°F and 653°F. The heatup and cooldown rate of the system is administratively limited to assure that these limits will not be exceeded. This condition is based on a normal plant cycle of one heatup and cooldown per month rounded up to the next highest hundred.
Power changes	15,000 power change cycles over the range of 15% to 100% of full load at a rate of 5% of full load per minute either increasing or decreasing.
Normal cyclic variations	10^6 step changes of ±100 psi and ±10°F (±20°F for surge line) when at operating conditions. This condition is selected based on 1 million-cycles approximating an infinite number of cycles so that the limiting stress is the endurance limit. Grouped together in these cycles are: pressure variations associated with fluctuation in pressurizer pressure between the setpoint for actuation of the backup heaters and the opening of the spray valves; temperature variations associated with the CEA controller deadband; and 2,000 step power changes of ±10% of full load assuming 1 cycle per week for 50 weeks of the year.

TABLE 3.9-1 (Cont'd)

(Sheet 2 of 3)

TRANSIENTS USED IN STRESS ANALYSIS OF CODE CLASS 1 COMPONENTS

Upset Conditions

Occurrence	Conditions
Reactor trip, turbine trip, loss of reactor coolant flow	480 cycles are used to envelope all anticipated upset transients, (one occurrence per month for the life of the plant) which includes any combination of reactor trips, equipment malfunctions, or a total loss of reactor coolant flow. For design purposes, conservative temperature/pressure time histories are provided in the design specification for each Class 1 component, which reflects its unique response during these events. Further thermal transient information is specified for the nozzles on these components, when they experience additional transients due to changing flow conditions.
OBE condition	See Section 3.7.3.2 for the procedures used to determine the number of earthquake cycles during the seismic event.

Faulted Condition

1. The concurrent loading produced by normal operation at full power, plus the design basis earthquake, plus loss-of-coolant-accident (pipe rupture) are used to determine the faulted plant loading condition.

2. Loss of Secondary Pressure: One cycle of a postulated loss of secondary pressure due to a complete double ended severance of one steam generator or feedwater nozzle, but not simultaneously. These are not considered credible events in forming the design basis of the reactor coolant system. However, they are included to demonstrate that the reactor coolant system components will not fail structurally in the unlikely event that one of these events occur.

TABLE 3.9-1 (Cont'd)

(Sheet 3 of 3)

TRANSIENTS USED IN STRESS ANALYSIS OF CODE CLASS 1 COMPONENTS

Test Condition

Occurrence	Conditions
Primary system hydrostatic	10 primary side cycles from 15 psi to 3,125 psi at a temperature between 120°F to 400°F. These cycles are based on one initial hydrostatic test plus a major repair every 4 years for 36 years which includes equipment failure and normal plant cycles. The secondary side of the steam generator is at atmospheric pressure during this test.
Primary system leak	200 cycles from 15 psi to 2250 psi at a temperature between 120°F to 400°F. These cycles are based on a normal plant maintenance operation involving 5 shutdowns per year for 60 years. E

TABLE 3.9-2

LOADING COMBINATIONS ASME CODE CLASS 1, 2, AND 3 COMPONENTS

Condition	Design Loading[(a)] Combination
Design	PD
Level A (Normal)[(b)]	PO+DW
Level B (Upset)[(b)]	PO+DW+OBE
Level C (Emergency)	PO+DW+DE
Level D (Faulted)	PO+DW+SSE+DF

E

a) Legend:

PD	=	Design pressure
PO	=	Operating pressure
DW	=	Dead weight
OBE	=	Operating Basis earthquake
SSE	=	Safe shutdown earthquake
DE	=	Dynamic system loadings associated with the emergency condition
DF	=	Dynamic system loadings associated with a postulated pipe rupture for branch line breaks not eliminated by leak before break analysis.

E

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

TABLE 3.9-3

STRESS LIMITS FOR ASME CODE CLASS 1
COMPONENTS, PIPING, AND COMPONENT SUPPORTS

	COMPONENT AND PIPING STRESS LIMITS (a)	COMPONENT SUPPORT STRESS LIMITS (c)
Design	NB-3221, NB-3231 and NB-3652	NF-3221 or NF-3321, and NF-3225
Level A (Normal)	NB-3222, NB-3232 and NB-3653	NF-3221 or NF-3321, and NF-3225
Level B (Upset)	NB-3223, NB-3233 and NB-3654	NF-3221 or NF-3321, and NF-3225
Level C (Emergency)	NB-3224, NB-3234 and NB-3655	NF-3221 or NF-3321, and NF-3225
Level D (Faulted)	NB-3225, NB-3235 and NB-3656	NF-3221 or NF-3321, and NF-3225

E

NOTES:

a. Stress limits listed are used as required by ASME Section III, and applicable addenda for all components except active components. Active components are designed to the stress limits of NB-3221 and NB-3231 for Design Conditions and the stress limits of NB-3222 and NB-3232 for all other conditions for active components.

b. For faulted condition loadings, bolts in the load path connecting two members of an NF support for Class 1 components are designed in accordance with Appendix XVII of the ASME Code for friction type connections with tensile stresses limited to the lesser of 0.7 Su or Sy.

c. Stress limits used are as required by ASME Section III and modified by Regulatory Guide 1.124 and 1.130. Component standard supports may be designed to the limits of NF-3280. E

TABLE 3.9-4

(Sheet 1 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
SI 197	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 196	Safety Injection Sys. (Operate)	Swing Check	1	None
CS 547	Containment Spray Sys. (Operate)	Swing Check	2	None
CS 546	Containment Spray Sys. (Operate)	Swing Check	2	None
SD 769	Shutdown Cooling Suction Relief (Operate)	Relief	2	None
SD 768	Shutdown Cooling Suction Relief (Operate)	Relief	2	None
SI 239	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 244	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 238	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 246	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 237	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 245	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 236	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 247	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 317	Safety Injection Sys. (Operate)	Globe	2	Motor
SI 319	Safety Injection Sys. (Close)	Globe	1	Pneumatic
SI 318	Safety Injection Sys. (Close)	Globe	1	Pneumatic
SI 165	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 169	Safety Injection Sys. (Operate)	Swing Check	1	None
SD 766	Shudown Cooling Sys. Check (Operate)	Swing Check	2	None

E

TABLE 3.9-4 (Cont'd)

(Sheet 2 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
SD 767	Shutdown Cooling Sys. Check (Operate)	Swing Check	2	None
SI 307	SIT Fill/Drain (Clos·'	Globe	2	Pneumatic
SI 166	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 170	Safety Injection Sys. (Operate)	Swing Check	1	None
SI 323	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 322	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 321	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 320	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 339	Safety Injection Tank Fill Valve (Close)	Globe	2	Pneumatic
SI 327	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 343	Safety Injection Tank Isolation (Operate)	Gate	1	Motor
SI 314	Safety Injection Isolation (Operate)	Globe	2	Motor
SI 315	Safety Injection Isolation (Operate)	Globe	2	Motor
SI 347	Leakage Return to RWT (Close)	Globe	1	Pneumatic
SI 338	Safety Injection Tank Fill Valve (Close)	Globe	2	Pneumatic
SI 326	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 342	Safety Injection Tank Isolation (Operate)	Gate	1	Motor
SI 316	Hot Leg Injection Isolation (Operate)	Globe	2	Motor
SI 346	Leakage Return to RWT (Close)	Globe	1	Pneumatic
SI 344	Safety Injection Tank Fill Valve (Close)	Globe	1	Pneumatic

E

TABLE 3.9-4 (Cont'd)

(Sheet 3 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
SI 325	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 341	Safety Injection Tank Isolation (Operate)	Gate	1	Motor
SI 310	Safety Injection Throttle (Operate)	Globe	2	Motor
SI 311	Safety Injection Throttle (Operate)	Globe	2	Motor
SI 313	Safety Injection Valve (Operate)	Globe	2	Motor
SI 345	Safety Injection Sys. (Close)	Globe	1	Pneumatic
SI 336	Safety Injection Tank Fill Valve (Close)	Globe	2	Pneumatic
SI 324	Safety Injection Tank Vent (Operate)	Globe	2	Solenoid
SI 312	Safety Injection Sys. Valve (Operate)	Globe	2	Motor
SD 654	Shutdown Cooling Sys. Isolation (Operate)	Globe	2	Motor
SD 655	Shutdown Cooling Sys. Isolation (Operate)	Globe	2	Motor
SD 673	Shutdown Cooling Suction (Operate)	Gate	1	Motor
SD 672	Shutdown Cooling Suction (Operate)	Gate	1	Motor
SD 671	Shutdown Cooling Suction (Operate)	Gate	1	Motor
SD 670	Shutdown Cooling Suction (Operate)	Gate	1	Motor
SD 659	Shutdown Cooling Suction (Operate)	Gate	2	Motor
SD 658	Shutdown Cooling Suction (Operate)	Gate	2	Motor
SD 656	Shutdown Cooling Sys. (Operate)	Globe	2	Motor
SD 657	Shutdown Cooling Sys. (Operate)	Globe	2	Motor
CS 503	Containment Spray Sys. (Operate)	Swing Check	2	None

E

TABLE 3.9-4 (Cont'd)

(Sheet 4 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
CS 502	Containment Spray Sys. (Operate)	Swing Check	2	None
SI 139	Safety Injection Sys. (Operate)	Swing Check	2	None
SI 140	Safety Injection Sys. (Operate)	Swing Check	2	None
SI 141	Safety Injection Sys. (Operate)	Swing Check	2	None
SI 142	Safety Injection Sys. (Operate)	Swing Check	2	None
SD 731	Shutdown Cooling Sys. (Operate)	Swing Check	2	None
SD 730	Shutdown Cooling Sys. (Operate)	Swing Check	2	None
CS 522	Containment Spray Sys. (Operate)	Swing Check	2	None
CS 523	Containment Spray Sys. (Operate)	Swing Check	2	None
SI 304	Hot Leg Injection (Operate)	Gate	2	Motor
SI 305	Hot Leg Injection (Operate)	Gate	2	Motor
CS 600	Containment Spray Isolation Valve (Operate)	Gate	2	Motor
CS 601	Containment Spray Isolation Valve (Operate)	Gate	2	Motor
SD 652	SDCHX Throttle (Operate)	Butterfly	2	Motor
SD 653	SDCHX Throttle (Operate)	Butterfly	2	Motor
SD 650	SDCHX Bypass (Operate)	Globe	2	Motor
SD 651	SDCHX Bypass (Operate)	Globe	2	Motor
SI 199	Safety Injection Sys. (Operate)	Check	1	None
SI 198	Safety Injection Sys. (Operate)	Check	1	None

E

TABLE 3.9-4 (Cont'd)

(Sheet 5 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
SI 340	Safety Injection Sys. (Operate)	Check	1	None
CH 205	Auxiliary Spray (Close)	Globe	1	Solenoid
CH 208	Charging Line (Close)	Globe	1	Solenoid
CH 209	Charging Line Bypass (Close)	Globe	1	Manual
CH 241	Seal Injection Flow Control	Globe	2	Pneumatic Diaphragm
CH-242	Seal Injection Flow Control	Globe	2	Pneumatic Diaphragm
CH 243	Seal Injection Flow Control	Globe	2	Pneumatic Diaphragm
CH 244	Seal Injection Flow Control	Globe	2	Pneumatic Diaphragm
CH 255	Seal Inj. Containment Isolation (Operate)	Globe	2	Motor
CH 303	Charging Line Isolation Check (Close)	Check	2	None
CH 304	SDC Purification Isolation (Close)	Check	2	None
CH 307	SDC Purification Contain. Isolation (Close)	Gate	2	Manual
CH 431	Auxiliary Spray Check (Close)	Lift Check	1	None
CH 433	Charging Line Check (Close)	Lift Check	1	None
CH 447	Auxiliary Spray Check (Close)	Check	1	None
CH 448	Charging Line Check (Close)	Check	1	None
CH 494	RMW Supply Line to RDT Check (Close)	Lift Check	2	None
CH 505	RCP Controlled Bleed-Off (Close)	Globe	2	Pneumatic
CH 506	Containment Isolation (Close)	Globe	2	Pneumatic
CH 515	Letdown Isolation Valve (Close)	Globe	1	Pneumatic
CH 516	Letdown Line Isolation Valve (Close)	Globe	1	Pneumatic

E

TABLE 3.9-4 (Cont'd)

(Sheet 6 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
CH 560	RDT Suction Isolation (Close)	Globe	2	Pneumatic
CH 561	RDT Suction Isolation (Close)	Globe	2	Pneumatic
CH 580	RMW Supply Isolation to RDT Iso. (Close)	Globe	2	Pneumatic
CH 787	Seal Injection Check (Operate)	Lift Check	1	None
CH 802	Seal Injection Check (Operate)	Lift Check	1	None
CH 517	Letdown Line Isolation (Close)	Globe	2	Pneumatic
CH 523	Containment Isolation (Close)	Globe	2	Pneumatic
CH 524	Charging Line Isolation (Close)	Globe	2	Motor
CH 520	Containment Isolation (Close)	Globe	2	Pneumatic
CH 807	Seal Injection Check (Operate)	Lift Check	1	None
CH 812	Seal Injection Check (Operate)	Lift Check	1	None
CH 835	Seal Injection Check (Operate)	Lift Check	2	None
CH 866	Seal Injection Check (Operate)	Lift Check	1	None
CH 867	Seal Injection Check (Operate)	Lift Check	1	None
CH 868	Seal Injection Check (Operate)	Lift Check	1	None
CH-869	Seal Injection Check (Operate)	Lift Check	1	None
RC 200	RCS (Operate)	Safety	1	None
RC 201	RCS (Operate)	Safety	1	None
RC 202	RCS (Operate)	Safety	1	None
RC 203	RCS (Operate)	Safety	1	None
RC 244	RCS (Operate)	Check	1	None
SD 1	Safety Depressurization System	Gate	1	Motor
SD 2	Safety Depressurization System	Globe/Angle	1	Motor

E

TABLE 3.9-4 (Cont'd)

(Sheet 7 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
SD 3	Safety Depressurization System	Gate	1	Motor
SD 4	Safety Depressurization System	Globe/Angle	1	Motor
RV 101	Safety Depressurization System	Globe	2	Solenoid
RV 102	Safety Depressurization System	Globe	2	Solenoid
RV 103	Safety Depressurization System	Globe	2	Solenoid
RV 104	Safety Depressurization System	Globe	2	Solenoid
RC 105	Safety Depressurization System	Globe	1	Solenoid
RC 106	Safety Depressurization System	Globe	1	Solenoid
RC 107	Safety Depressurization System	Globe	1	Solenoid
RC 108	Safety Depressurization System	Globe	1	Solenoid

E

NOTES:

1. (Operate) is defined as valve being capable of both opening and closing.

2. (Close) is defined as valve being capable of moving to or maintaining a closed position.

3. (Open is defined as valve being capable of moving to or maintaining an open position.

TABLE 3.9-5

STRESS CRITERIA FOR SAFETY-RELATED ASME CLASS 2 AND CLASS 3 VESSELS

Service Level	Stress Limits*
Design and Level A	$\sigma_m \leq 1.0$ S
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S
Level B	$\sigma_m \leq 1.1$ S
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S
Level C	$\sigma_m \leq 1.5$ S
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.80$ S
Level D	$\sigma_m \leq 2.0$ S
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S

* Stress limits are taken from ASME III, Subsections NC and ND (Table 3321-2).

E

TABLE 3.9-6

STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS AND PUMP SUPPORTS

Service Level	Stress Limits*	P_{max}**
Design and Level A	$\sigma_m \le 1.0\ S$	1.0
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \le 1.5\ S$	
Level B	$\sigma_m \le 1.1\ S$	1.1
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \le 1.65\ S$	
Level C	$\sigma_m \le 1.5\ S$	1.2
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \le 1.80\ S$	
Level D	$\sigma_m \le 2.0\ S$	1.5
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \le 2.4\ S$	

* Stress limits are taken from ASME III, Subsections NC and ND (Table 3416-1).

** The maximum pressure shall not exceed the tabulated factors listed under P_{max} times the design pressure.

E

TABLE 3.9-7

DESIGN CRITERIA FOR ACTIVE PUMPS AND PUMP SUPPORTS

Service Level	Stress Limits*
Design and Level A	ASME B&PV Section III, Article NC-3400 and ND-3400
Level B	$\sigma_m \leq 1.0\ S$ $\sigma_m + \sigma_b \leq 1.5\ S$
Level C	$\sigma_m \leq 1.2\ S$ $\sigma_m + \sigma_b \leq 1.65\ S$
Level D	$\sigma_m \leq 1.2\ S$ $\sigma_m + \sigma_b \leq 1.8\ S$

* The stess limits specified for active pumps are more restrictive than the ASME B&PV Section III limits. For Service Level D (membrane plus bending), stresses may exceed 1.8 S but must remain below the material yield stress. In such cases, a deflection analysis is performed to assure that the maximum displacements are within the deflection limits which will not impair the operability of the equipment. E

TABLE 3.9-8

(Sheet 1 of 2)

STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 INACTIVE NSSS AND INACTIVE BOP VALVES

Service Level	Stress Limits (Notes 1-4, 6)	P_{max} (Note 5)
Design and Level A	$\sigma_m \leq 1.0$ S	1.0
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5$ S	
Level B	$\sigma_m \leq 1.1$ S	1.1
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65$ S	
Level C	$\sigma_m \leq 1.5$ S	1.2
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80$ S	
Level D	$\sigma_m \leq 2.0$ S	1.5
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4$ S	

E

NOTES: 1. Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied: (1) the section modulus and areas of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110% of those for the piping connected (or joined) to the valve body inlet and outlet nozzles; and (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above shall be multiplied by the ratio of S_{pipe}/S_{valve}. If unable to comply with this requirement, the design by analysis procedure of NB3545.2 is an acceptable alternate method.

TABLE 3.9-8 (Cont'd)

(Sheet 2 of 2)

STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2
AND CLASS 3 INACTIVE NSSS
AND INACTIVE BOP VALVES

NOTES:

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this table are not applicable to valve stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet.

5. The maximum pressure resulting from Service Levels B, C, or D shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied. E

6. Stress limits are taken from ASME III, Subsections NC and ND, (Table 3521-1).

TABLE 3.9-9

BOP DESIGN CRITERIA FOR ACTIVE VALVES

Service Level	Design Criteria
Level A	ASME Section III Article NC-3500 and ND-3500
Level B	$\sigma_m \leq 1.0\ S$ $\sigma_m + \sigma_b \leq 1.5\ S$
Level C	$\sigma_m \leq 1.2\ S$ $\sigma_m + \sigma_b \leq 1.65\ S$
Level D	$\sigma_m \leq 1.2\ S$ $\sigma_m + \sigma_b \leq 1.8\ S$

Notes 1 through 5 of Table 3.9-8 also apply to this table. E

TABLE 3.9-10

LOADING COMBINATIONS FOR ASME SECTION III CLASS 1 PIPING

Service Level	Loading Combination
Design	Design Pressure, Design Temperature, Deadweight
Level A	Level A Transients, Deadweight
Level B	Level B Transients, Deadweight, Operating Basis Earthquake
Level C	Level C Transients, Deadweight
Level D	Level D Transients, Deadweight, Safe Shutdown Earthquake

E

NOTE: The dynamic loads are combined by the square root of the sum of the squares.

TABLE 3.9-11

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY VALVE PIPING AND SUPPORTS ASME CLASS 1 PORTION

Service Level	Load Combination*
Design	Design Pressure, Weight
Level A	Level A Transients, Weight
Level B	Level B Transients, Weight, OBE, VT**
Level C	Level C Transients, Weight, VT**
Level D	Level D Transients, Weight, SSE, VT**

* Dynamic loads are combined by the square root of the sum of the squares (SRSS). E

** Valve thrust loads (VT) are loads resulting from the rapid acceleration or deceleration of a water mass, noncondensible gases, or both.

TABLE 3.9-12

LOADING COMBINATIONS FOR ASME SECTION III
CLASSES 2 AND 3 PIPING

Service Level	Loading Combination
Level A	Design Pressure, Design Temperature, Deadweight
Level B	Level B Transients Deadweight, Operating Basis Earthquake
Level C	Level C Transients Pressure, Deadweight
Level D	Level D Transients Deadweight, Safe Shutdown Earthquake, or Safe Shutdown Earthquake and Pipe Rupture Loads

NOTE: Dynamic loads are combined by the square root of the sum of the squares (SRSS).

E

TABLF 3.9-13

LOAD COMBINATIONS FOR SAFETY VALVE PIPING ASME CLASS 2 AND 3 PIPING

Service Level	Load Combination*
Design	Design Pressure, Weight
Level A	Level A Transients, Weight
Level B	Level B Transients, Weight, OBE, VT**
Level C	Level C Transients, Weight, VT**
Level D	Level D Transients, Weight, SSE, VT**

* Dynamic loads are combined by the square root of the sum of the squares (SRSS). E

** Valve thrust loads (VT) are loads resulting from the rapid acceleration or deceleration of a water mass, noncondensible gases, or both.

TABLE 3.9-14

DESIGN LOADING COMBINATIONS FOR ASME CODE, CLASSES 1, 2, AND 3 PIPING SUPPORTS

Service Level	Loading Combination
Level A and Design	DW
Level B	DW + OBE + RV DW + OBE + DU
Testing	DW + DT
Level C	DW + SSE + DE
Level D	DW + SSE + DF

E

Legend:

DW - Piping deadweight
OBE - Operating Basis Earthquake
SSE - Safe Shutdown Earthquake
DT - Loads associated with testing
RV - Relief Valve
DU - Other transient dynamic events associated with the upset plant condition
DE - Dynamic events defined as emergency condition
DF - Dynamic events defined as a faulted condition

NOTE: Dynamic loads are combined by the square root of the sum of the squares (SRSS).

TABLE 3.9-15

STRESS LIMITS FOR CEDM PRESSURE HOUSINGS

Operating Condition	Stress Categories and Limits of Stress Intensities (a)
1. Level A and Level B: Normal Operating Loading plus Normal Operating & Upset Plant Transients plus Operating Basis Earthquake Forces.	Figures NB-3221-1 and 3222-1, including notes.
2. Level D: Normal Operating Loadings plus Faulted Plant Transients plus Safe Shutdown Earthquake Forces.	Article F-1000, Appendix F, Rules for Evaluation of Service Conditions Loading with Level D Service Limits.
3. Testing: Testing Plant Transients	Paragraph NB-3226

E

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

1. Level A and Level B: The CEDMs are designed to function normally during and after exposure to these conditions.

2. Level D: For SSE, the deflections of the CEDM pressure housing are limited to the elastic design limits of Article F-1330, Appendix F (defined above) so that the CEAs can be inserted after exposure to these conditions.

NOTE: a. References listed are taken from Section III of the ASME Boiler and Pressure Vessel Code.

TABLE 3.9-16

STRESS LIMITS FOR DESIGN AND SERVICE LOADS

Design Limits

The core support and internal structures shall be designed to meet the Design Limits defined in NG-3221 of ASME Boiler and Pressure Vessel Code Section III Subsection NG for Design Loadings.

Level A Service Limits

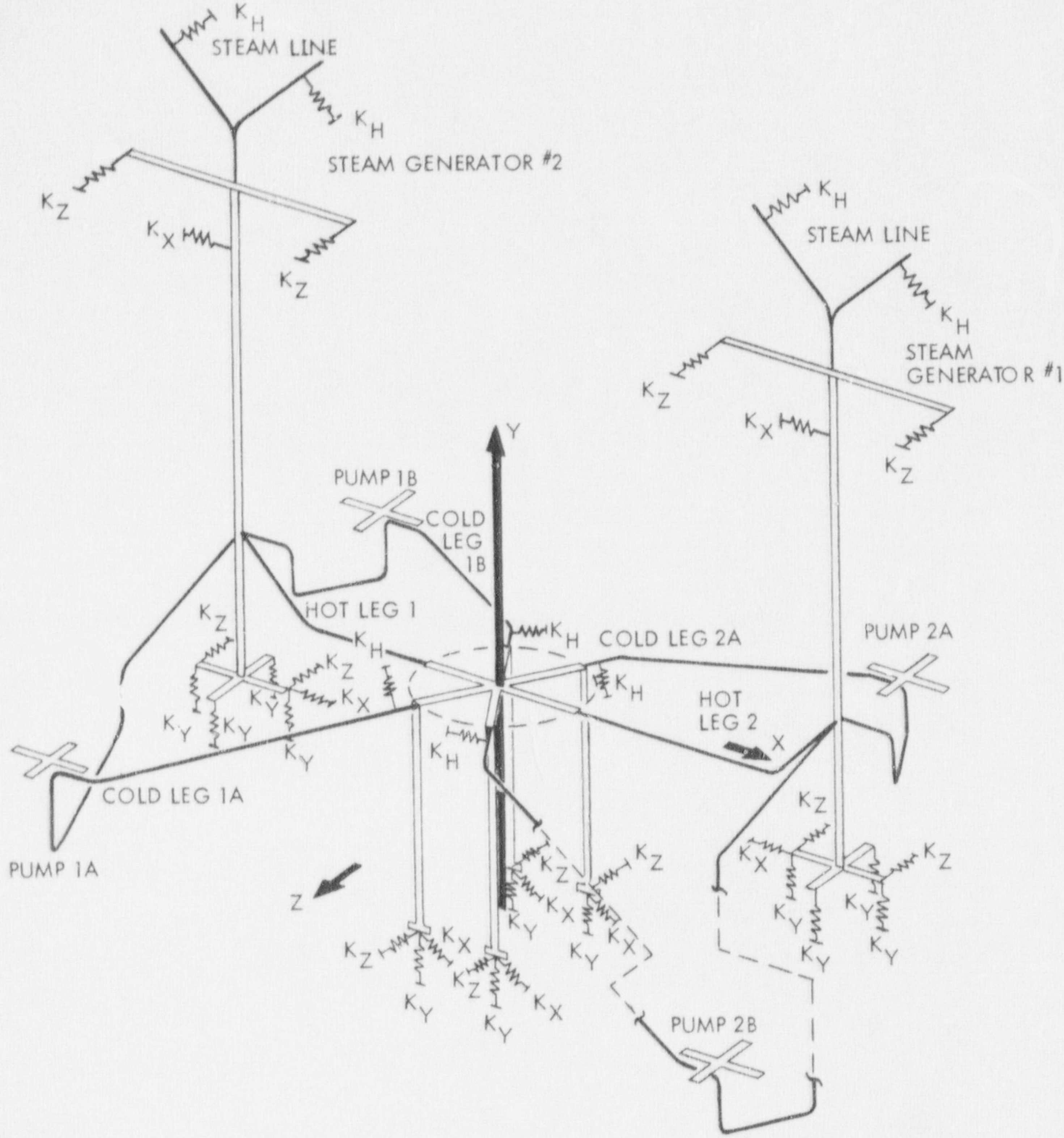
The core support and internal structures shall be designed to meet the Level A Service Limits defined in NG-3222 of IB1D for Level A Service Loadings.

Level B Service Limits

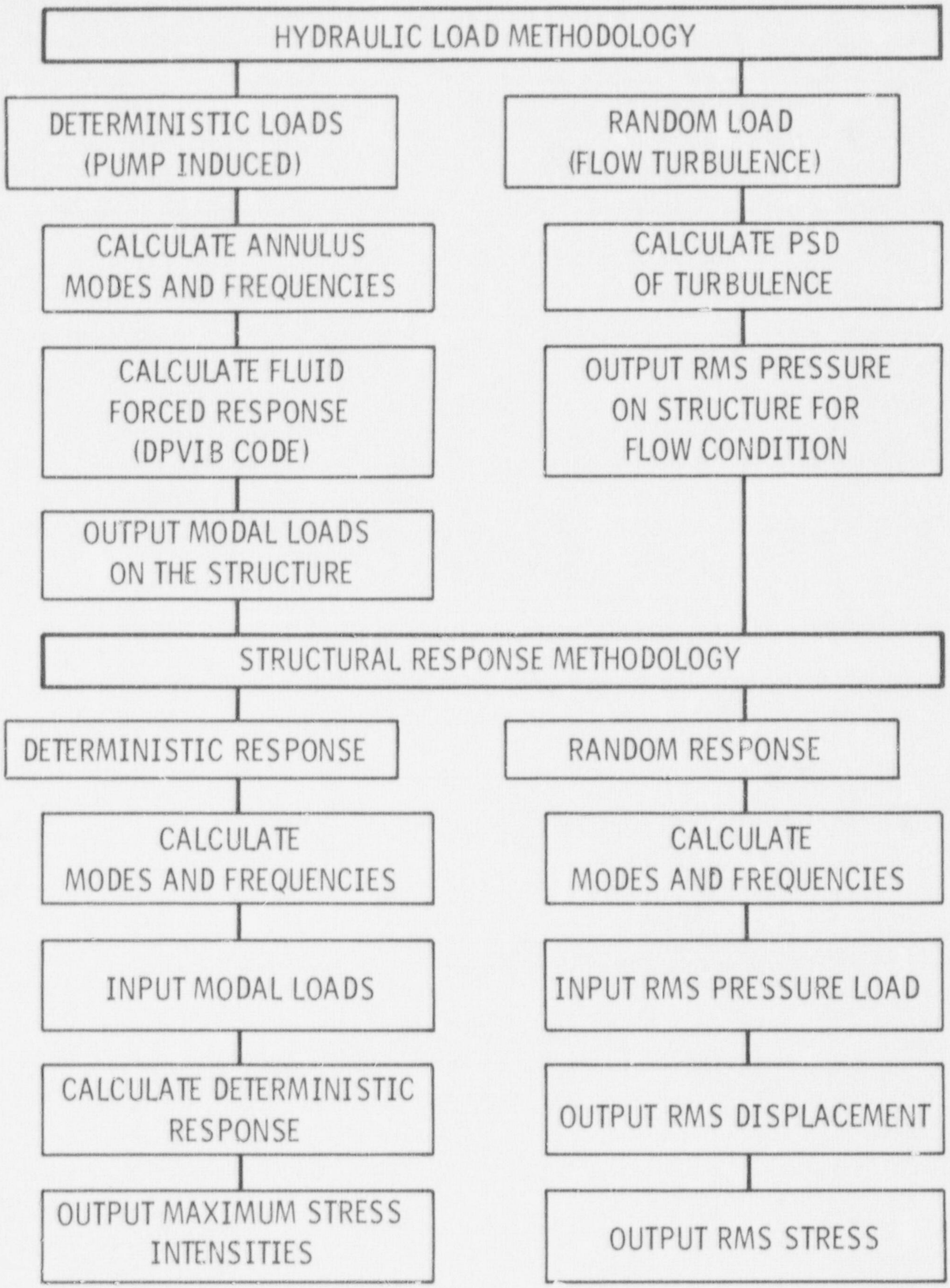
The core support and internal structures shall be designed to meet the Level B Service Limits defined in NG-3223 of IB1D for Level B Service Loadings. E

Level D Service Limits

The core support structures shall be designed to meet the Level D Service Limits defined in NG-3225 of IB1D for elastic system analysis of Appendix F of Reference 3.1.2 using Level D Service Loadings.



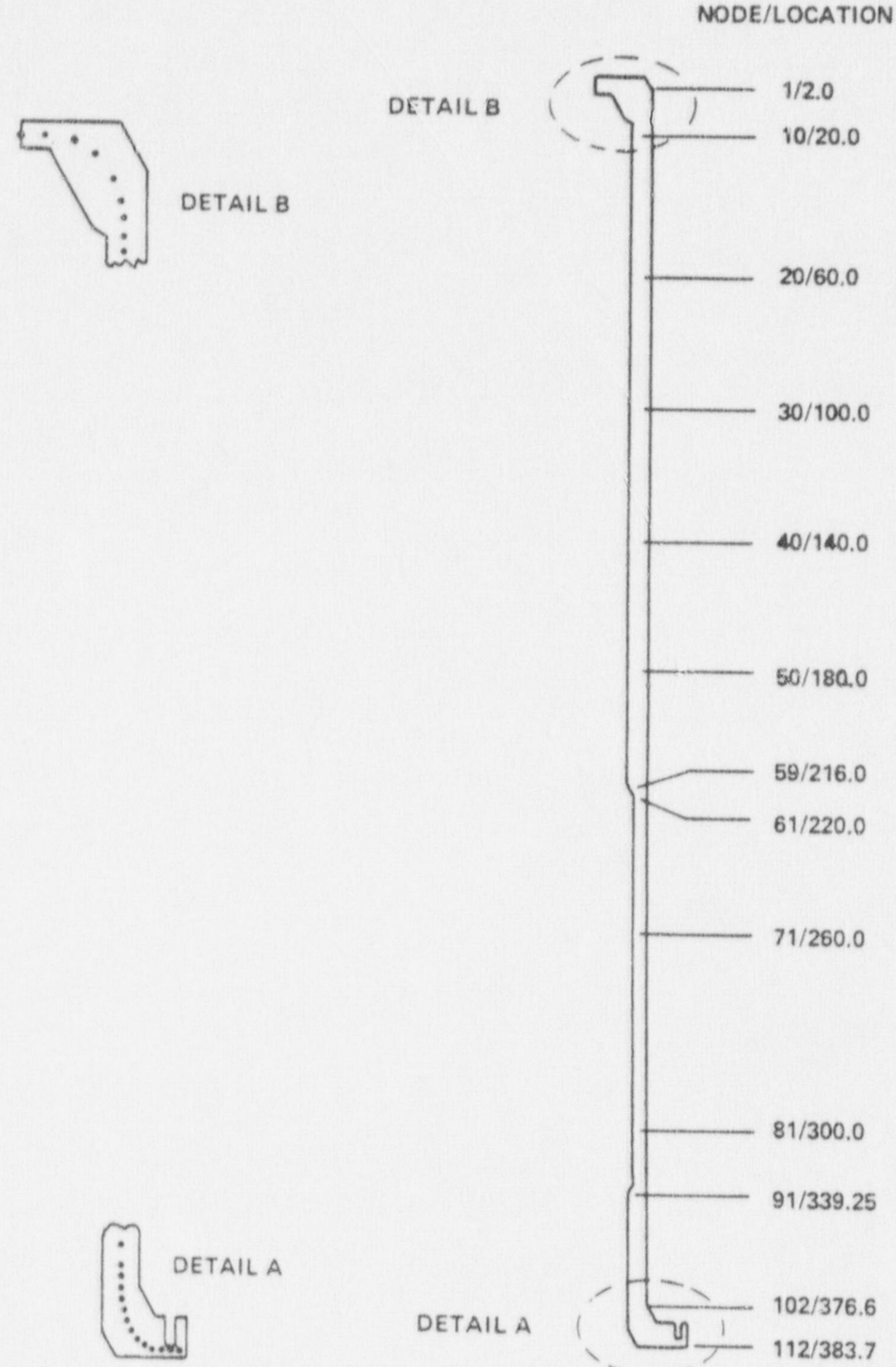
SYSTEM 80+™	REACTOR COOLANT SYSTEM SUPPORTS DIAGRAM	Figure 3.9-1



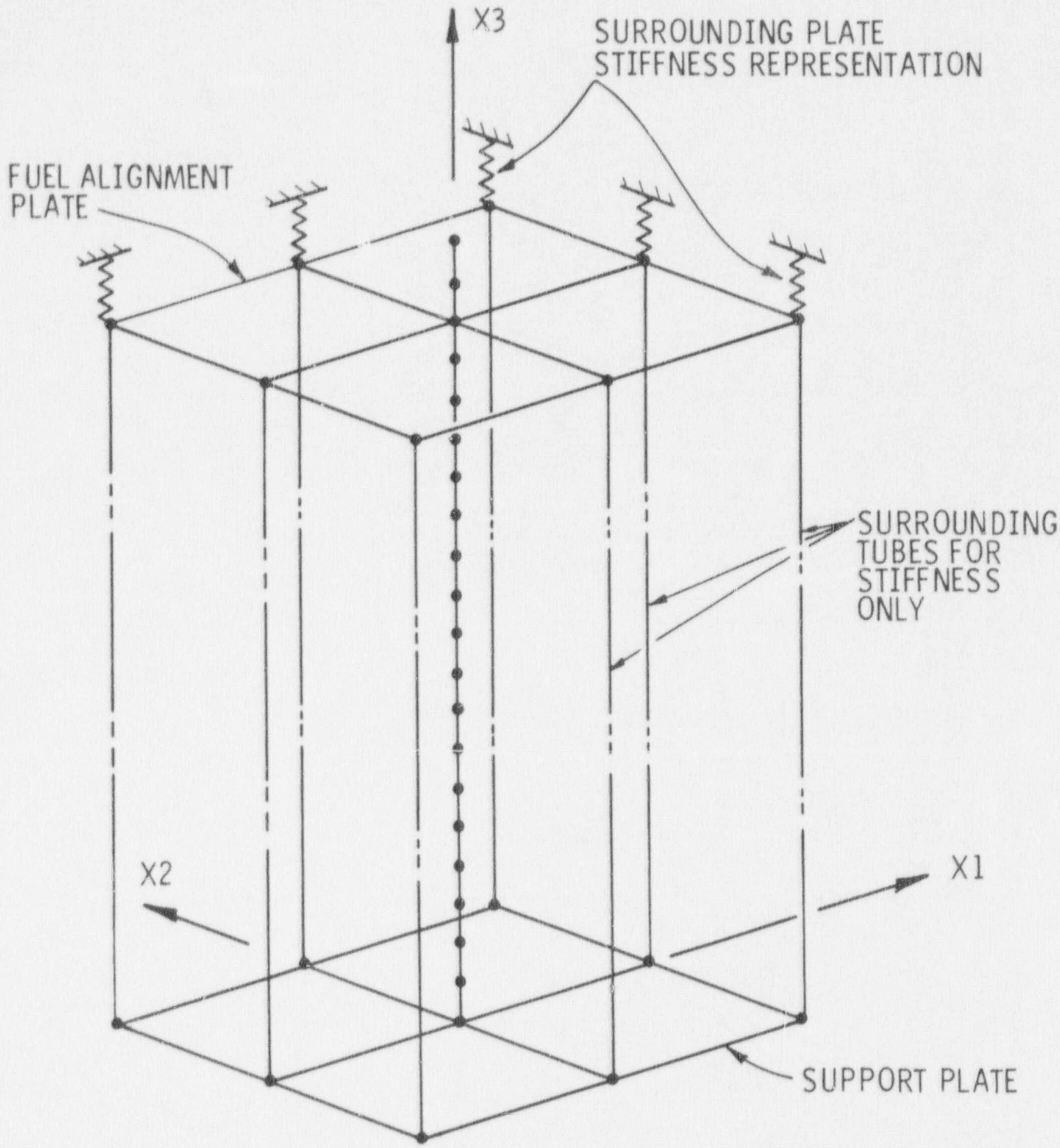
SYSTEM 80+™

SUMMARY OF ANALYTICAL METHODOLOGY

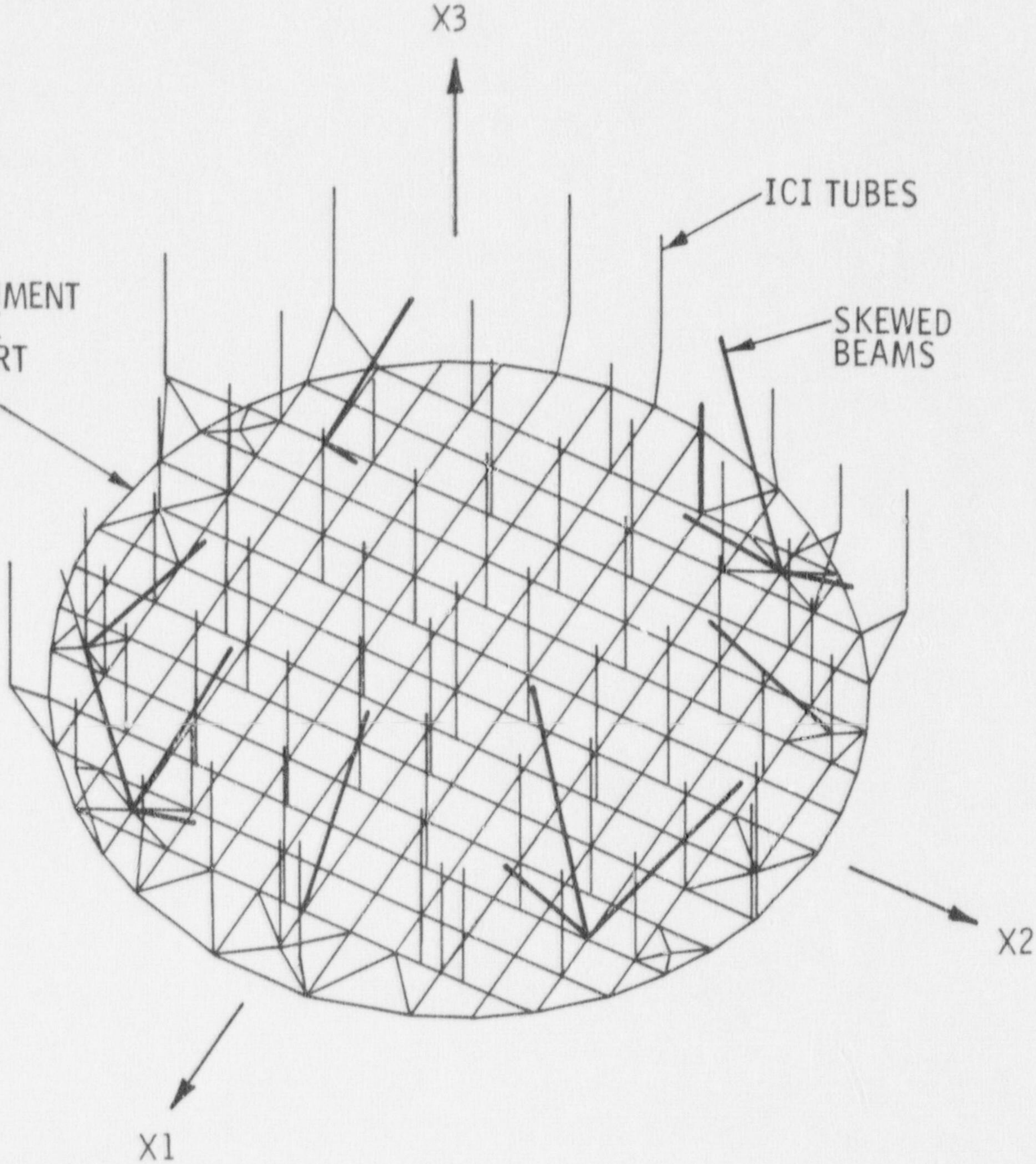
Figure 3.9-2



SYSTEM 80+™	ASHSD FINITE ELEMENT MODEL OF THE CSB SYSTEM	Figure 3.9-3



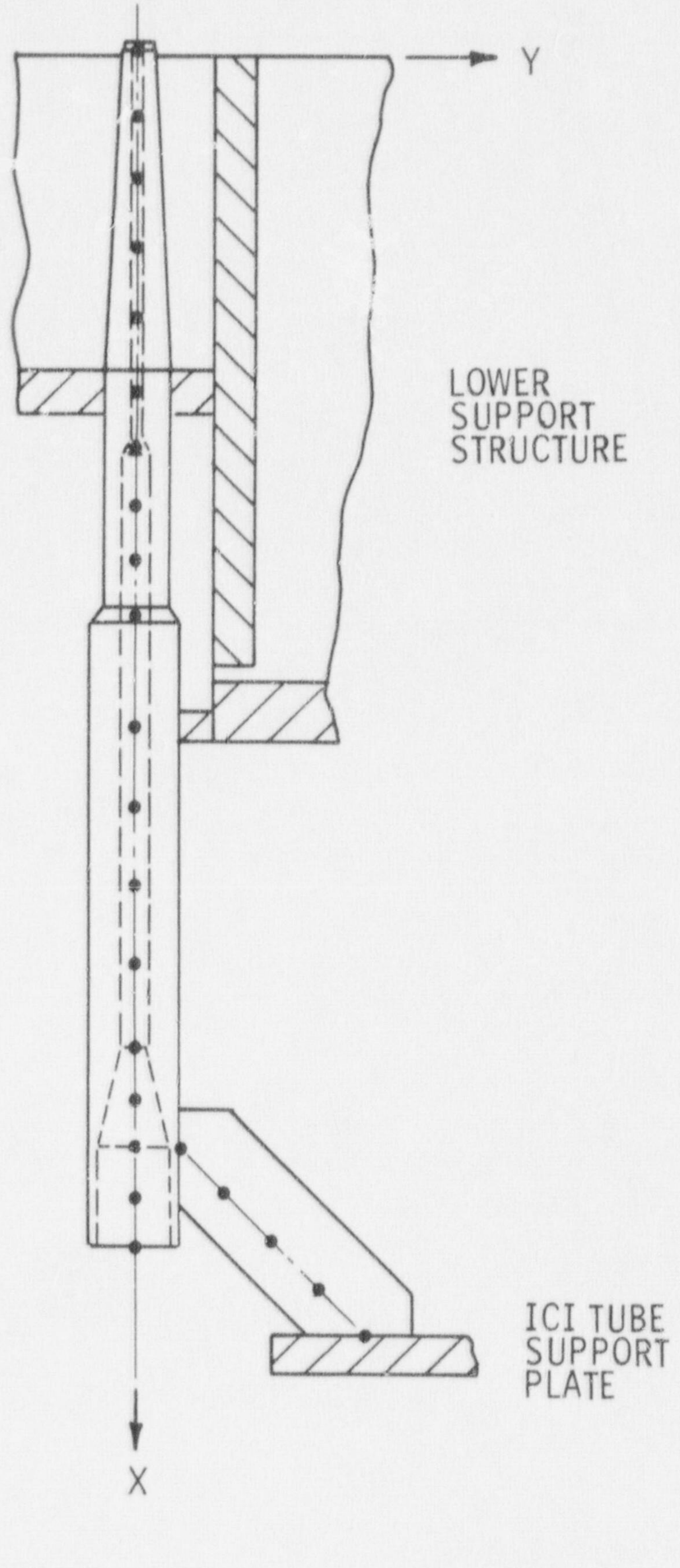
SYSTEM 80+™	CONTROL ELEMENT SHROUD TUBE FINITE ELEMENT MODEL	Figure 3.9-4



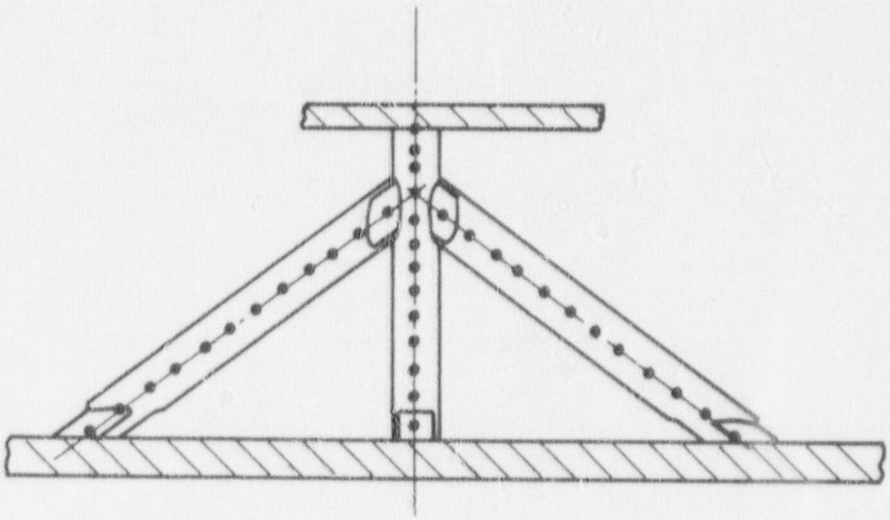
SYSTEM 80+™

LOWER SUPPORT STRUCTURE
INSTRUMENT NOZZLE ASSEMBLY
FINITE ELEMENT MODEL

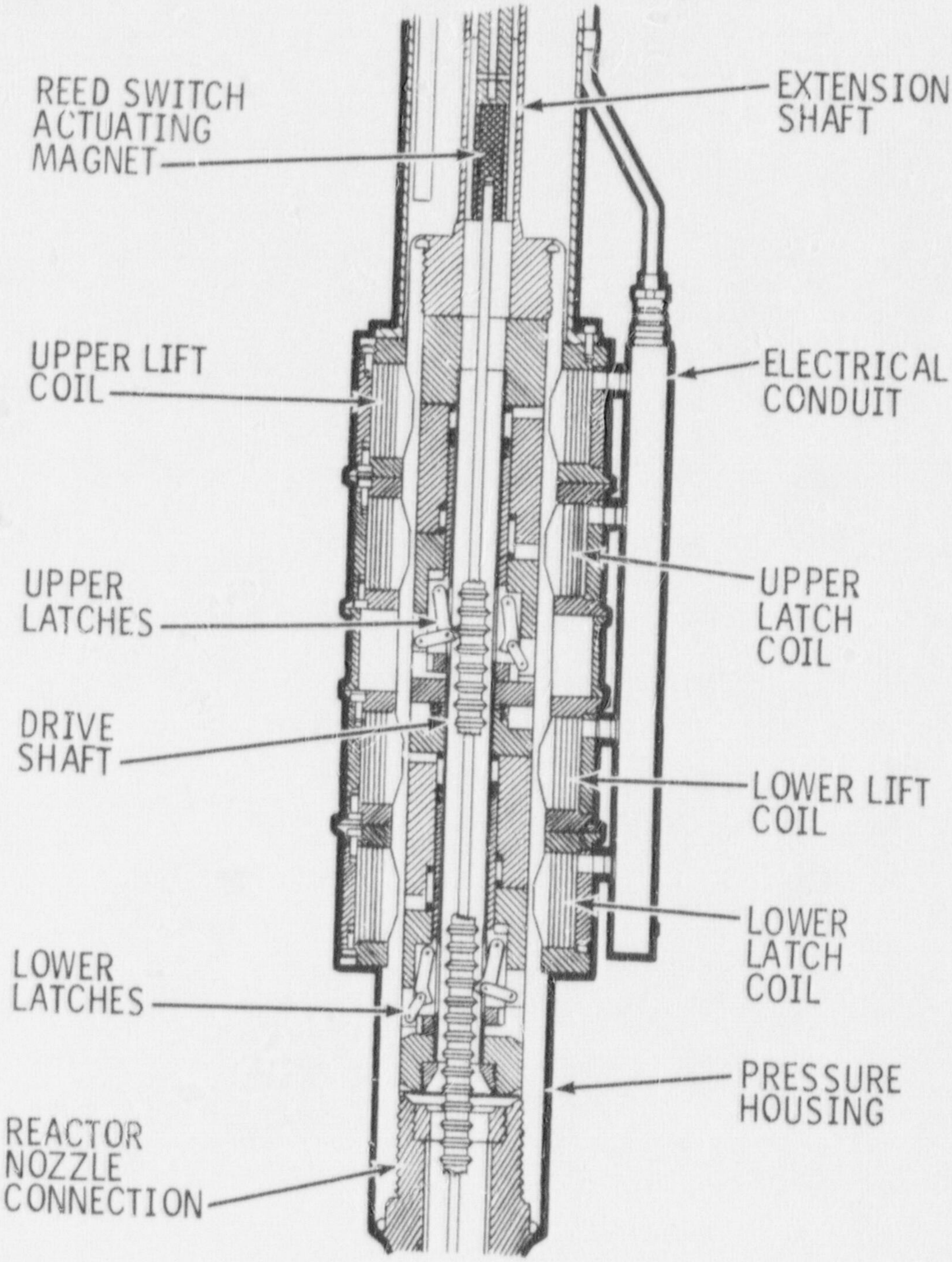
Figure 3.9-5



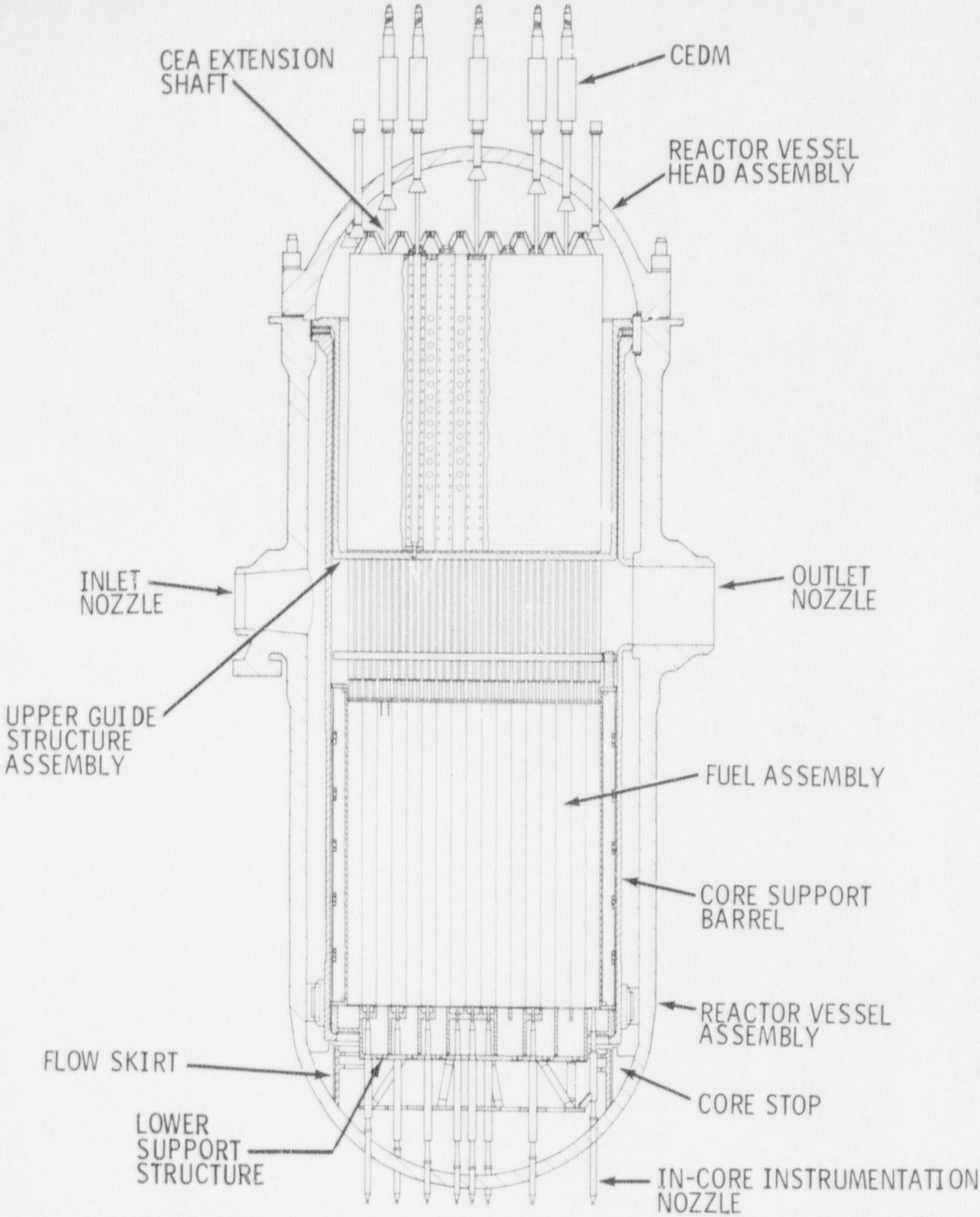
SYSTEM 80+™	ICI SUPPORT TUBE; OUTER POSITION FINITE ELEMENT MODEL	Figure 3.9-6



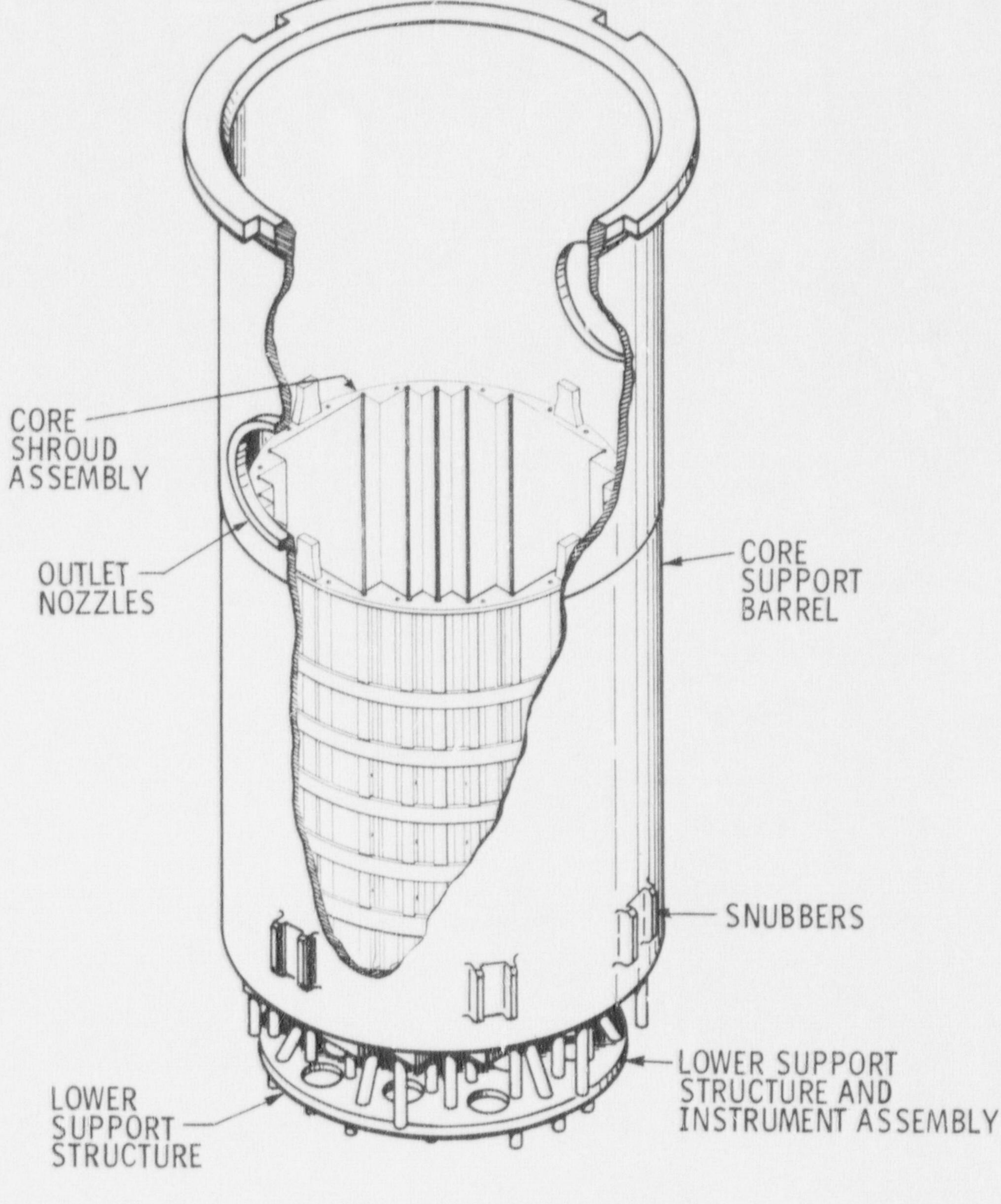
SYSTEM 80+™	SKEWED BEAM SUPPORT COLUMNS FINITE ELEMENT MODEL	Figure 3.9-7



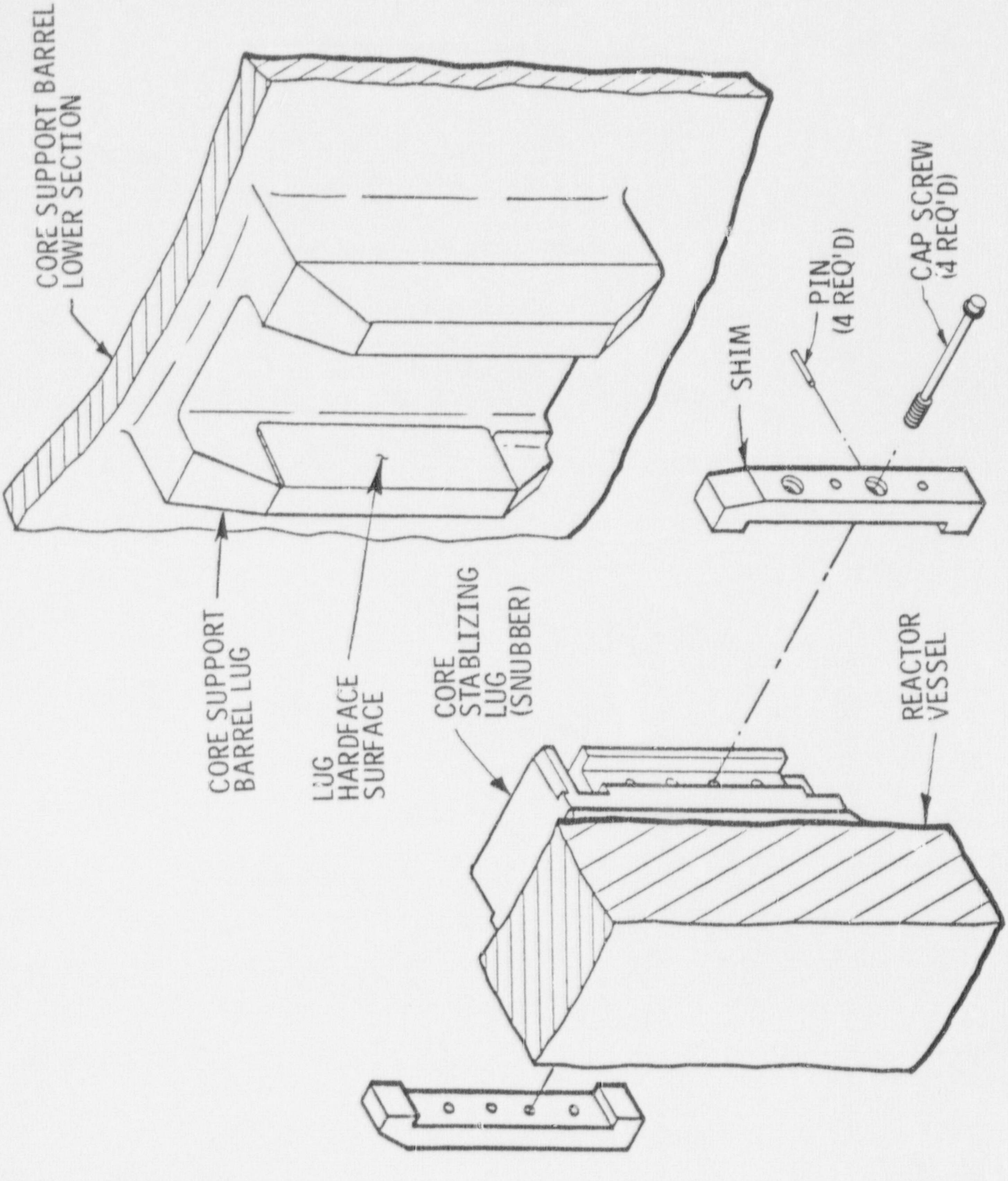
SYSTEM 80+™	CONTROL ELEMENT DRIVE MECHANISM (MAGNETIC JACK)	Figure 3.9-8



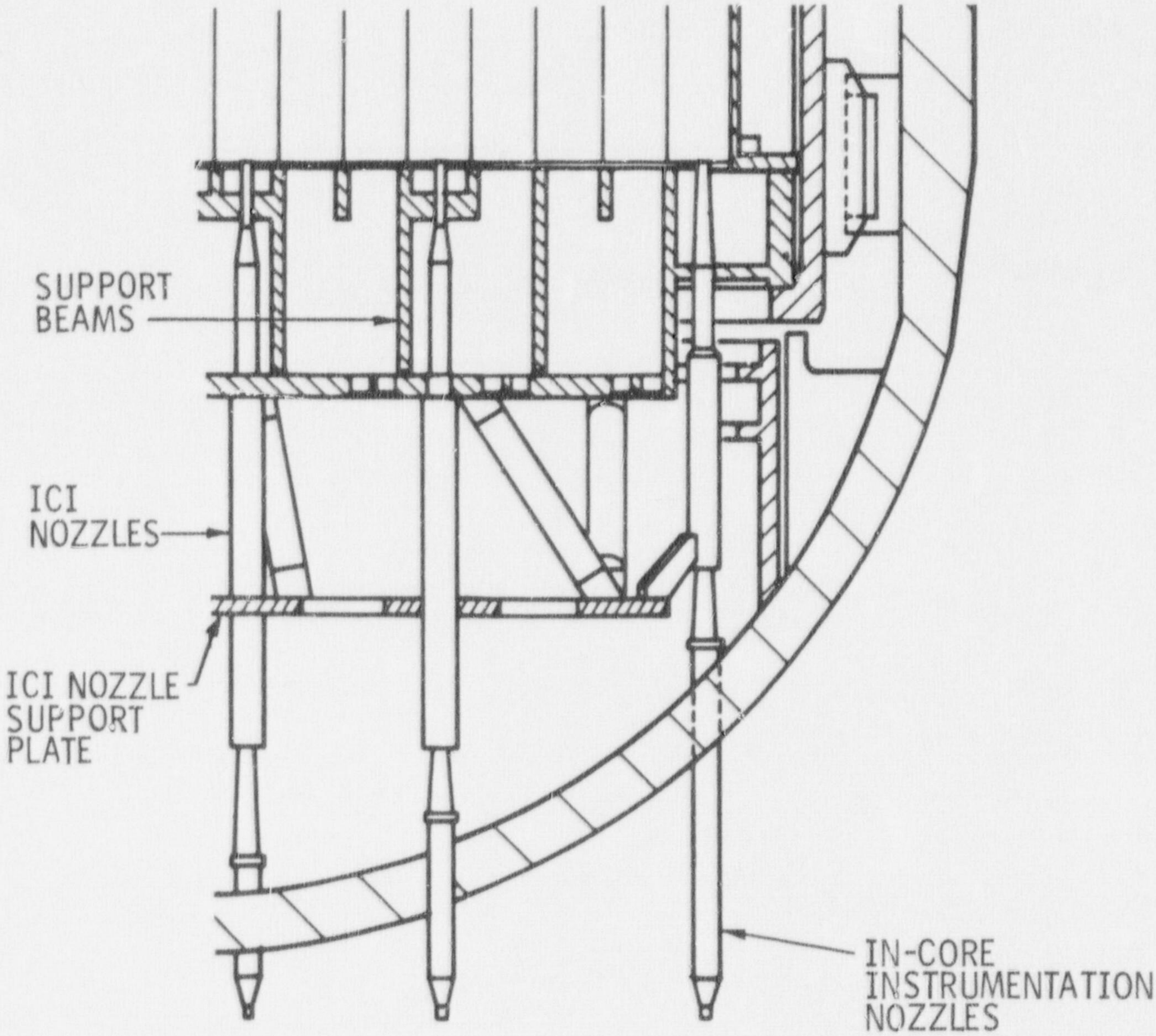
SYSTEM 80+™	REACTOR VERTICAL ARRANGEMENT	Figure 3.9-9



SYSTEM 80+™	CORE SUPPORT BARREL ASSEMBLY	Figure 3.9-10



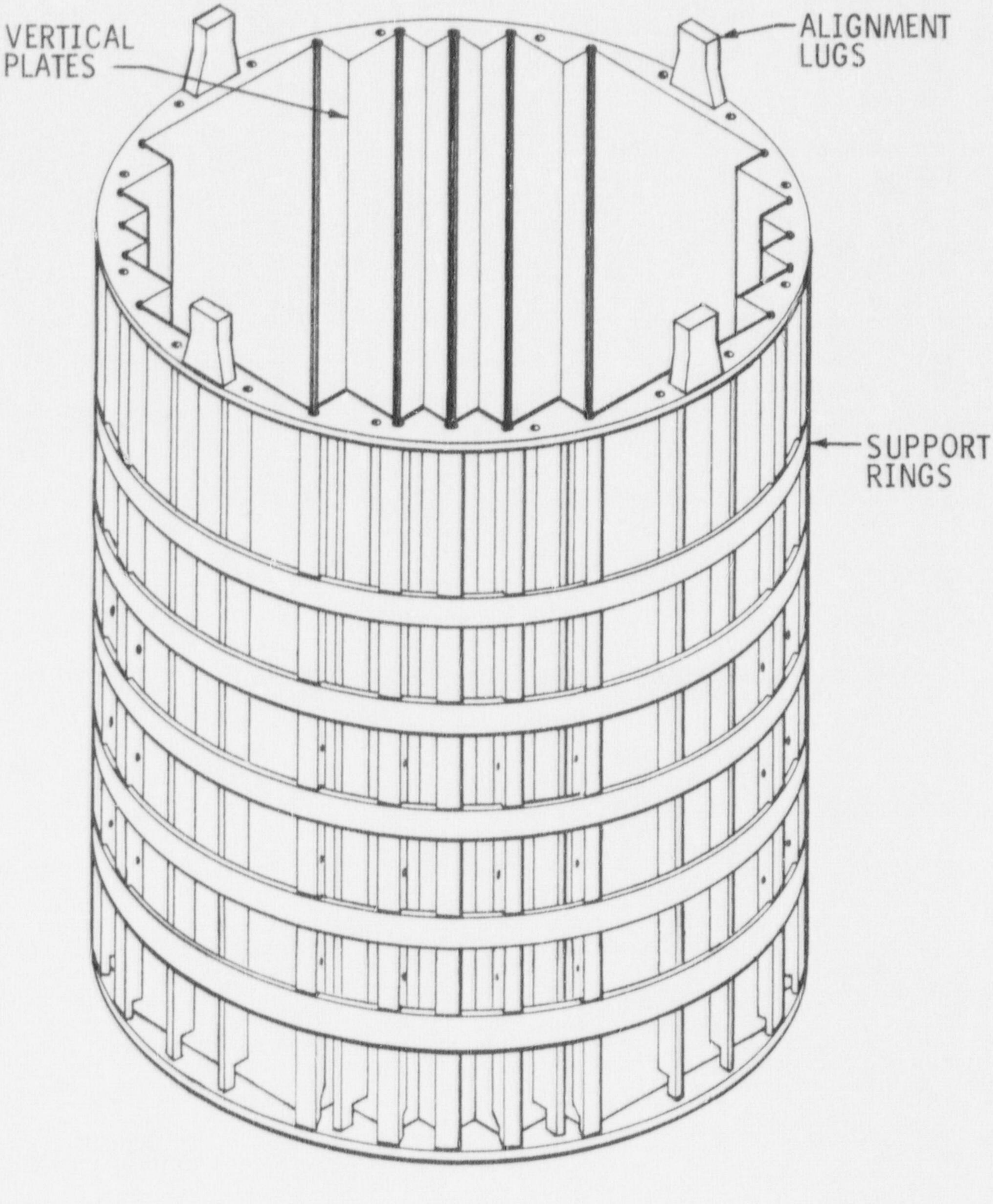
SYSTEM 80+™	REACTOR VESSEL CORE SUPPORT BARREL SNUBBER ASSEMBLY	Figure 3.9-11



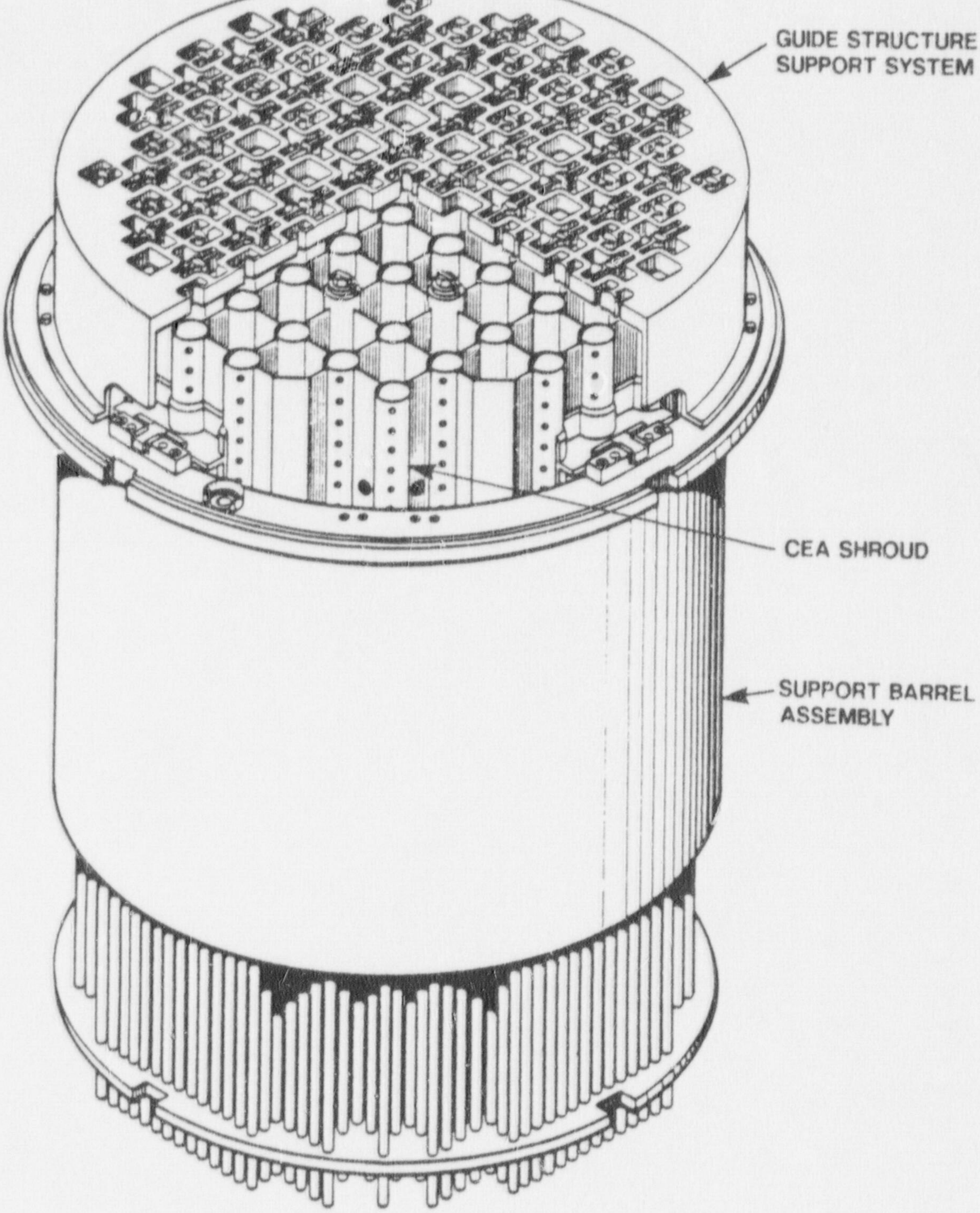
SYSTEM 80+™

IN-CORE INSTRUMENT SUPPORT STRUCTURE

Figure 3.9-12

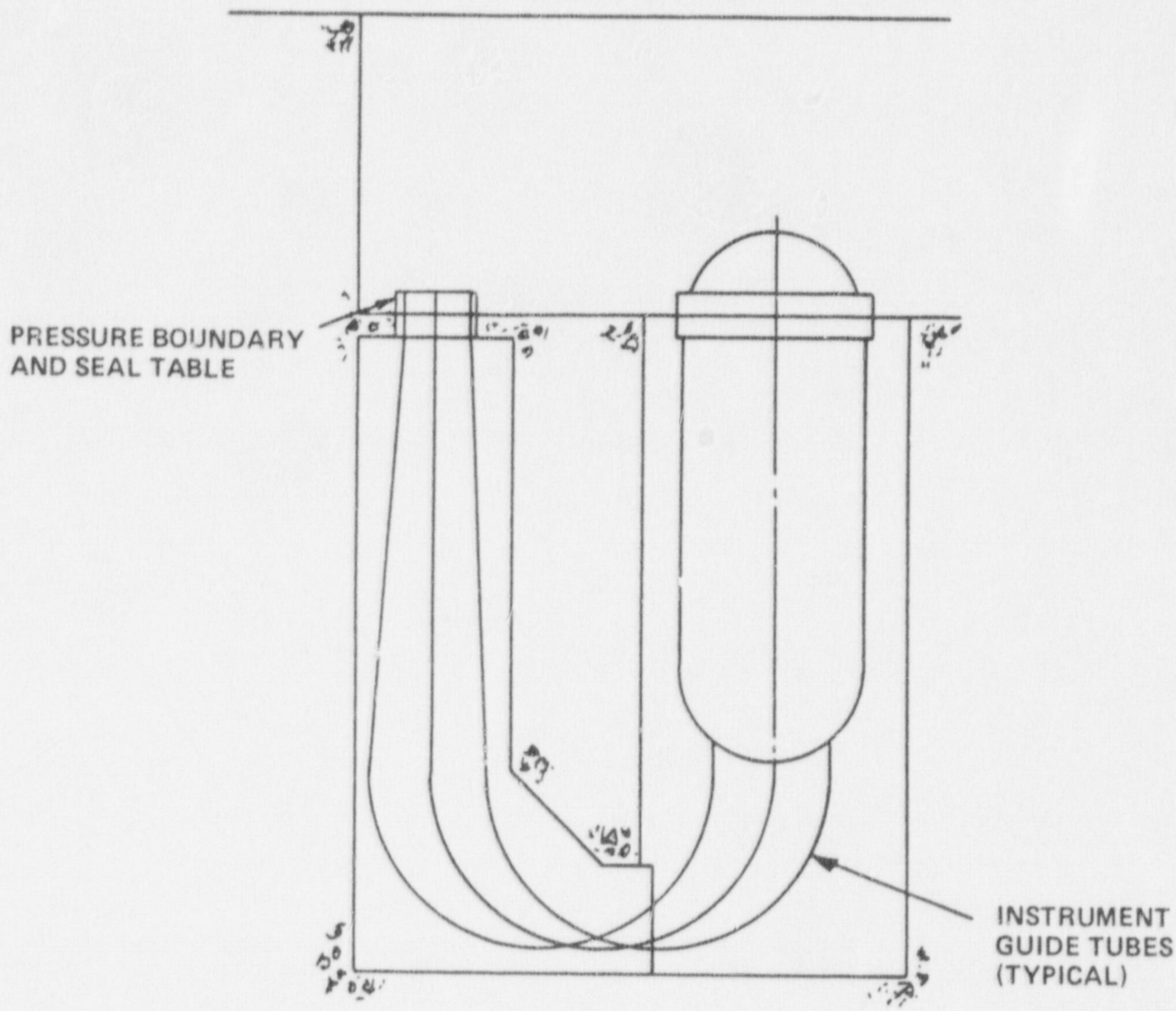


SYSTEM 80+™	CORE SHROUD ASSEMBLY	Figure 3.9-13



Amendment E
December 30, 1988

SYSTEM 80+™	UPPER GUIDE STRUCTURE ASSEMBLY	Figure 3.9-14



Amendment E
December 30, 1988

SYSTEM 80+™	IN-CORE INSTRUMENT SYSTEM	Figure 3.9-15

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section describes the seismic design criteria and analyses, tests, procedures, and acceptance criteria applied to Seismic Category I instrumentation and electrical equipment, except for valve and pump motors, and their supports. The information applicable to instrumentation and control equipment is contained in Combustion Engineering's Topical Report CENPD-182, "Seismic Qualification of C-E Instrumentation Equipment." Valve and pump motors are discussed in Section 3.9.2.2.

3.10.1 SEISMIC QUALIFICATION CRITERIA

Instrumentation and electrical equipment used for post-accident monitoring, the Reactor Protective System (RPS), the Engineered Safety Features Actuation System (ESFAS), the actuation devices for ESF system actuated components, and the emergency power system are designed to Seismic Category I requirements to ensure the ability to initiate required protective actions during, and following, a Safe Shutdown Earthquake (SSE); and, to supply power, following an SSE, to components required to mitigate the consequences of events which require safety system operation. D

3.10.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Seismic Category I instrumentation and electrical equipment required to perform a safety action during a seismic event, after a seismic event, or both are qualified (with appropriate documentation) in accordance with the requirements of the equipment specifications. These requirements are consistent with those of IEEE 344-1987, "Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations" and include the following: D

A. The seismic excitation for which the equipment must qualify will be determined based on location in the plant.

B. The equipment will be designed to perform its intended function during and after an earthquake of the intensity of the Safe Shutdown Earthquake. D

C. Analysis, testing or operating experience will be required to substantiate the adequacy of the design depending on the type of equipment under consideration and its intended safety function. D

D. The quality assurance program, as described in Chapter 17, illustrates the procedures used in assuring the implementation of the requirements. D

Seismic Category I instrumentation and electrical equipment requiring seismic qualification are listed in Section 3.11. The test program where used will provide the following:

E. A test program is required to confirm the functional operability of all Seismic Category I electrical and associated mechanical equipment and instrumentation during and after an earthquake of magnitude up to and including the SSE.

F. The characteristics of the required input motion shall be specified by one of the following:

  1. response spectrum

  2. power spectral density function

  3. time history

  Such characteristics, as derived for the structure or system seismic analysis, shall be representative of the input motion at the equipment mounting locations.

G. Equipment shall be tested in the operational condition. Operability shall be verified during and after the testing.

H. The actual input motion shall be characterized in the same manner as the required input motion, and the conservatism in amplitude and multi-frequency energy content up to approximately 33 Hz shall be demonstrated. D

I. Random vibration input motion shall be used. However, single frequency input, such as sine beats, may be utilized provided one of the following conditions are met:

  1. The characteristics of the required input motion indicate that the motion is dominated by one frequency (i.e., by structural filtering effects).

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

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

J. The input motion shall be applied to one vertical and one principal (or two orthogonal) horizontal axes simultaneously unless it can be demonstrated that the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions will be such that a purely rectilinear resultant input is avoided. The acceptable alternative is to have vertical and horizontal inputs in-phase, and then repeated with inputs 180 degrees out-of-phase. In addition, the test will be repeated with the equipment rotated 90 degrees horizontally. Biaxial and triaxial input motion may be utilized where practical. D

K. The fixture design shall meet the following requirements:

1. Simulate the actual service mounting.

2. Cause no dynamic coupling to the test item.

L. The in-situ application of vibratory devices to superimpose the seismic vibratory loadings on the complex active device for operability testing is acceptable when application is justifiable.

M. The test program may be based upon selectively testing a representative number of mechanical components according to type, load, level, size or other appropriate classification on a prototype basis. D

3.10.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

To ensure qualification for the required forces, acceleration requirements are included in equipment specifications as design parameters. Vendors will use this information as the basis for analysis or testing depending on the type, size, shape, or complexity of equipment to be qualified.

The equipment specification will include, as a minimum, the following seismic requirements:

A. The appropriate seismic excitation for which the equipment must qualify will be determined based on location in the plant.

B. The equipment is required to perform its intended function during and after a Safe Shutdown Earthquake.

C. Analysis, testing, past qualifications, or a combination of these is required to substantiate the adequacy of the design, depending on the type of equipment and its intended safety function. D

D. The quality assurance program used in assuring the implementation of the requirements of CENPD-182 are discussed in Chapter 17. D

The seismic qualification program, as described in CENPD-182, meets the specified requirements for Seismic Category I equipment.

E. Analyses or tests shall be performed for all supports of electrical and associated mechanical equipment and instrumentation to ensure their structural capability to withstand seismic excitation.

F. The analytical results will include the following:

1. The required input motions to the mounted equipment shall be obtained and characterized in the manner as stated in Section 3.10.2 item F.

2. The combined stresses of the support structures shall be within the allowable limits found in recognized mechanical handbooks.

G. Supports shall be tested with either equipment or dynamically equivalent models installed. If the equipment is not operating or not installed during the support test, the response at the equipment mounting locations shall be monitored and characterized in the manner as stated in Section 3.10.2 item F. In such a case, equipment shall be tested separately and the actual input to the equipment shall be more conservative in amplitude and frequency content than the monitored response.

H. The requirements of Section 3.10.2 items F, H, I, J, and K are applicable when tests are conducted on the equipment supports.

Specifically, cabinet and support test requirements will be conducted as follows:

The design seismic environment of equipment located within support structures (cabinets) will be determined by either test or analysis.

I. Testing will consist of one of the following procedures:

1. <u>Fully Operational Cabinet Test</u>

   The cabinet, fully loaded with equipment, will be tested in its operating state. During testing, a sample of safety-related functions will be monitored. This test will demonstrate both structural integrity and functional operability.

2. <u>Weighted Cabinet Test With Subsequent Equipment Tests</u>

   (a) The cabinet will be tested with simulated equipment in place of the actual equipment. The simulated equipment will be equal in mass, mass distribution, and mounting to the actual equipment such that the dynamic response of the weighted cabinet is equal to that of the fully loaded cabinet. During testing, the motions present at the equipment mounting points will be recorded. This test will demonstrate the cabinet structural integrity and determine the local seismic environment of the actual equipment.

   (b) The actual equipment will be independently tested to those motions determined by the weighted cabinet test. The equipment will be operational and all safety-related functions will be monitored during the test. This test will demonstrate functional operability of the equipment. D

3. <u>Equipment Test</u>

   Equipment which is not mounted in a cabinet will be tested or analyzed in its operating state in a configuration which simulates its intended mounting.

J. For structures which can be modeled, a dynamic analysis may be substituted for the weighted cabinet test to determine the motions at the enclosed equipment mounting points.

For both testing and analysis, the input motions to the cabinet shall be derived from the building motions at the cabinet's intended location. D

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The design criteria with respect to environmental effects on the electrical and mechanical equipment of the Reactor Protective System and the Engineered Safety Feature systems to ensure acceptable performance in all environments (normal and accident) depend upon equipment location and function. Such equipment is qualified to meet its performance requirements under the environmental and operating conditions in which it will be required to function and for the length of time for which its function is required. As far as practical, equipment for these systems is located outside the containment building or other areas where adverse environmental conditions could exist. Compatibility of mechanical and electrical equipment with environmental conditions is provided within the following design criteria:

A. For operation under normal conditions the systems are designed and qualified to remain functional after exposures within the following ranges of environmental conditions:

   1. Design temperatures maintained at the equipment location during normal operation by the ventilating and cooling system described in Section 9.4. Temperature ranges are given in Appendix 3.11A, Table 3.11A-1 through 3.11A-14.

   2. Relative humidity ranges are given in Appendix 3.11A, Table 3.11A-1 through 3.11A-14.

   3. Pressure ranges are given in Appendix 3.11A, Table 3.11A-1 through 3.11A-14.

   4. Maximum expected integrated radiation exposures for 60 years at the equipment location during normal operation are given in Appendix 3.11A, Table 3.11A-1 through 3.11A-14. | D

B. In addition to the normal operation environmental requirements given in listing A above, the mechanical and electrical components required to mitigate the consequences of a design basis event (DBE) or to attain a safe shutdown of the reactor are designed to remain functional after exposure to the environmental conditions anticipated following the specific DBE which they are intended to mitigate. Anticipated environmental conditions and requirements are listed below.

1. The temperature, pressure, and humidity ranges following the design basis accidents such as the loss of coolant accident (LOCA), the main steam line break (MSLB), control element assembly ejection, feedwater line break (FLB), or "worst case" combined (LOCA & MSLB) are indicated in Appendix 3.11A.

2. The time integrated post-accident radiation doses are indicated in Appendix 3.11A. Equipment will be designed for the types and levels of radiation associated with normal operation plus the radiation associated with the limiting design basis accident (DBA). If more than one type of radiation is significant, each type may be considered separately.

3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Apendix 3.11B lists and categorizes systems required to mitigate a DBE or to attain a safe shutdown. Specific equipment and components for each system are discussed in the appropriate section of the Safety Analysis Report as referenced in Appendix 3.11B. The major component categories, such as motor-operated valves, pump motors, instrumentation and pressure boundary equipment in each system, and the location of the components by area are also provided.

3.11.2 QUALIFICATION TESTS AND ANALYSES

Qualification tests and analyses performed in accordance with the methodologies defined in Reference 1 on NSSS instrumentation and electrical equipment (including pump and valve motors and electrical accessories) fulfill the requirements of IEEE Standard 323-1974, and "Category 1" of NUREG-0588. For mechanical equipment, environmental qualification is based on engineering, evaluation, and material selection where sufficiently reliable data are available.

3.11.2.1 Mechanical and Electrical Component Environmental Design and Qualification for Normal Operation

Equipment listed in Appendix 3.11B is required to be designed and qualified for 60 years of continuous operation in the temperature, pressure, humidity, and radiation environment that exists at the equipment location during normal operation, assuming proper routine preventive maintenance is performed, such as periodic replacement of seals and packing. System 80 equipment utilized in System 80+ and designed and qualified for a 40-year plant life will be requalified to the 60 year lifetime utilizing CENPD-255-A, Revision 3 on an equipment refurbishment or replacement period to be specified. D D

Appendix 3.11A provides the ranges of the design temperatures, pressures, and humidities, as well as the exposures to chemical spray and radiation for each area in which safety-related equipment listed in Appendix 3.11B is located.

3.11.2.2 Mechanical and Electrical Component Environmental Design and Qualification for Operation After a Design Basis Event

Equipment listed in Appendix 3.11B is designed to remain functional in the temperature, pressure, humidity, and chemical spray environment conditions that exist at the equipment location after the design basis LOCA. This equipment is also designed for the maximum calculated integrated radiation exposure after the design basis LOCA, as discussed in Section 3.11.5. The temperature, pressure, and humidity environment inside the containment after a LOCA is discussed in detail in Section 6.2.1.3. The containment spray characteristics are given in Section 6.2.2.1. The integrated post-accident radiation dose for those areas at which equipment is located is given in Appendix 3.11A. The temperature, pressure, and humidity environment inside the containment after a MSLB is discussed in detail in Section 6.2.1.4.

The requirements of the General Design Criteria, Appendix A to 10 CFR 50, are met as follows:

- Criterion 1 - Quality Standards and Records: refer to Section 3.1.1.
- Criterion 4 - Environmental and Missile Design Basis: refer to Section 3.1.4.
- Criterion 23 - Protection System Failure Modes: refer to Section 3.1.19.
- Criterion 50 - Containment Design Basis: refer to Sections 3.1.43 and 6.2.1.

The requirements of Quality Assurance Criterion III, Appendix B to 10 CFR 50, are met as discussed in the design and procurement quality assurance program (see Chapter 17).

The recommendations contained in the documents discussed below, listings A through D, and other applicable regulatory guides and standards have also been utilized.

A. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment."

B. Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants." A description of the tests and analysis by which active valves are qualified is provided in Section 3.9.2.2.

C. The qualification methods and documentation requirements of IEEE Standard 323-1983, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and "Category 1" of NUREG-0588, are discussed in Reference 1. D

D. Pressure boundary components inside the containment are designed for the appropriate temperature and pressure environment in accordance with the applicable code to which the component is constructed. Environmental Qualification testing is not considered necessary for such components.

AGING FOR HARSH AND NON-HARSH ENVIRONMENT EQUIPMENT

As stated in Reference 1, the aging portion of the qualification program is defined based upon whether or not equipment is located in a harsh or non-harsh environment. Equipment located in a harsh environment will undergo an aging analysis and an accelerated age conditioning program. Equipment located in a non-harsh environment will undergo an aging analysis that focuses on the identification of known aging mechanisms that significantly increase the equipment's susceptibility to its design basis event (seismic event only for non-harsh environments). If no known significant aging mechanisms are found, a surveillance/preventive maintenance (S/PM) program will be developed to monitor for degradation trends that suggest increasing seismic susceptibility. If an aging mechanism is found that is known to significantly increase the equipment's seismic susceptibility with time, that mechanism will be analyzed to determine whether an accelerated aging program or a periodic part replacement program is appropriate.

RADIATION FOR HARSH AND NON-HARSH ENVIRONMENT EQUIPMENT

Equipment is designed for the types and levels of radiation associated with normal operation plus the radiation associated with the limiting Design Basis Accident (DBA). These levels are defined in Appendix 3.11A.

Equipment which is exposed to radiation above 10^4 Rads will be irradiated to its anticipated Total Integrated Dose (TID) prior to type testing unless determined by analysis that radiation does not affect its ability to perform its required function. Where the application of the accident dose is planned during DBE testing, it need not be included during the aging process.

Equipment which will be exposed to radiation levles of 10^4 Rads or below is analyzed to determine whether low level radiation could impact its ability to perform its required function. Where analysis supported by partial type test data cannot demonstrate proper operation at the required radiation levels, type testing will be performed. Additionally, electronic equipment exposed to low level radiation will be addressed by an aging analysis which focuses on the identification of semiconductor (organic material) components that are considered to be age-sensitive in the design life. For electronic components that are age-sensitive a surveillance/preventive maintenance program will be developed. Reference 1 outlines this methodology. D

Mechanical and electrical equipment will be qualified to the typical radiation environments defined in Appendix 3.11A. If more than one type of radiation is significant, each type may be applied separately.

Gamma

Cobalt-60 is considered an acceptable gamma radiation source. Other sources may be found acceptable, and will be justified. Electrical equipment will be tested to typical gamma radiation levels defined in Appendix 3.11A.

Beta

Equipment exposed to beta radiation will be identified and an analysis will be performed to determine if the operability of the equipment is affected by beta radiation ionization and heating effects. Qualification is performed by test unless analysis demonstrates that the safety function will not be degraded by Beta exposure. Equipment will be tested and/or analyzed to the beta radiation levels defined in Appendix 3.11A. Where testing is recommended, a gamma equivalent radiation source will be used.

Neutron

Equipment exposed to neutron radiation will be identified and neutron radiation levels defined. When actual neutron dose qualification testing is not performed, an equivalent gamma radiation dose will be used for qualification testing to simulate neutron exposure. The basis for establishing an equivalent gamma radiation dose will be provided.

Paints/Radiation Effects

An analysis is performed addressing paint exposure to beta and gamma radiation. Qualification of painted equipment is verified by test if analysis indicates that the safety function of the equipment could be impaired by paint failure due to radiation.

Chemical Spray

After a postulated accident, such as the LOCA or MSLB, components located in the containment building may be exposed to a chemical spray. Equipment is environmentally tested to these conditions and performance requirements demonstrated during and after the test. The most severe spray composition is determined by single failure analysis of the spray system. Corrosion effects due to long term exposure will be addressed, as appropriate. D

Where qualification for chemical spray environment is required, the simulated spray will be initiated at the time shown in Appendix 3.11A.

Typical values of chemical spray composition, concentration and pH are defined in Appendix 3.11A, Tables 3.11A-1, 3.11A-2 and 3.11A-13.

3.11.3 QUALIFICATION TEST RESULTS

3.11.3.1 NPM Instrumentation and Electrical Equipment

Qualification testing and analyses of NPM instrumentation and electrical equipment are discussed in Reference 1.

3.11.3.2 NPM Mechanical Equipment

Qualification test results and analyses of NPM mechanical equipment are provided in Section 3.9.2.2 . D

3.11.4 CLASS 1E INSTRUMENTATION LOSS OF VENTILATION EFFECTS

Loss of ventilation is discussed in the site-specific SAR. Interface criteria are presented in Chapter 7. D

Class 1E equipment which is located in the control room or similar areas includes the following:

Plant Protection System (PPS) D

Main Control Panels

Process Instrument Cabinet

Other instrumentation, such as process transmitters and signal converters and the Reactor Trip Switchgear System circuit breakers, are located in the auxiliary building or containment building. Equipment in these areas is qualified for the maximum expected temperature, radiation, humidity, and pressure under which the equipment is expected to operate.

The following are the normal and abnormal environmental conditions for which Class 1E safety-related C-E equipment is qualified to operate according to the service location of the equipment and the expected environmental condition.

Appendix 3.11A, Tables 3.11A-1 through 3.11A-14 which define typical environmental conditions and associated environmental test profiles are defined in Figures 11A-6A through 3.11A-10.

3.11.5 CHEMICAL SPRAY, RADIATION, HUMIDITY, DUST, SUBMERGENCE, AND POWER SUPPLY VOLTAGE AND FREQUENCY VARIATION

3.11.5.1 Chemical Environment

Engineered Safety Feature systems are designed to perform their safety-related functions in the temperature, pressure, and humidity conditions described in Sections 3.11.1, 6.2 and 6.3. In addition, components of ESF systems inside the containment are designed to perform their safety-related functions in the presence of the existing chemical environment, resulting from the boric acid recirculated through the Safety Injection System (SIS) and Containment Spray System (CSS). The SIS is designed for both the maximum and long-term boric concentration and pH. These chemical environment conditions are given in Appendix 3.11A. D

3.11.5.2 Radiation Environment

The components in the Engineered Safety Feature and Reactor Protective Systems are designed to meet their performance requirements under the environmental and operating conditions in which they will be required to function and for the length of time for which their function is required. The components are designed to ensure acceptable performance under normal operational radiation exposure in addition to the single most adverse post-accident environment. The normal operational exposures are based on the design source terms provided in Sections 11.1 and 12.2. Radiation environments for those components for which the most adverse accident conditions are post-LOCA are based on the source term assumptions consistent with Regulatory Guides 1.4 and 1.7. Radiation environments for those components for which the most adverse accident condition is other than the LOCA (such as the main steam line break, feedwater line break or CEA ejection) are based on conservative estimates of the fuel assembly gas gap activities and maximum reactor coolant specific activities as discussed in Section 11.1.

3.11.5.3 Humidity

Equipment not subjected to steam environments during DBE testing will be environmentally tested to short term high humidity levels prior to operation and performance requirements demonstrated during and after the test. Equipment that is subjected to steam environments will be subjected to the appropriate test profiles in Appendix 3.11A.

3.11.5.4 Dust

Dust environments will be considered when establishing service conditions and qualification requirements. The potential effects of dust exposure will be evaluated relative to effects upon equipment safety function performance.

Where dust could have a degrading effect on equipment safety function performance, it will be addressed in the qualification program through the development of a maintenance program and/or an upgrading of equipment interface requirements.

3.11.5.5 Submergence

Equipment locations and operability requirements will be reviewed to establish whether or not specific equipment could be subject to submergence during its required operating time. Flood levels both inside and outside containment will be reviewed and potential impacts on equipment qualification appropriately addressed. Where operability during submergence is required, qualification will be demonstrated by type test and/or analysis supported by partial type test data.

3.11.5.6 Power Supply Voltage and Frequency Variation

Power supply voltage and frequency variation is addressed in several areas throughout the equipment design and verification process. During the design process, interface requirements dictate the acceptable range of power supply variation. Equipment specifications incorporate these interface requirements into the design to ensure acceptable operation within the defined range of power supply voltage and frequency variation. When equipment fabrication is completed, design verification tests are performed to demonstrate design adequacy.

REFERENCES

1. "Qualification of Combustion Engineering Class 1E Instrumentation," CENPD-255-A Revision 3, Combustion Engineering, Inc., October 1985. D

APPENDIX 3.11A

TYPICAL ENVIRONMENTAL CONDITIONS AND TEST PROFILES FOR STRUCTURES AND COMPONENTS

(LATER) D

APPENDIX 3.11B

IDENTIFICATION AND LOCATION

OF

MECHANICAL AND ELECTRICAL SAFETY-RELATED SYSTEMS AND COMPONENTS

(LATER) D