REACTOR COOLANT SYSTEM ASYMMETRIC LOADS EVALUATION PLAN

Prepared by

COMBUSTION ENGINEERING, Inc.
NUCLEAR POWER SYSTEMS DIVISION

POWER
SYSTEMS
COMBUSTION ENGINEERING, INC.

REACTOR COOLANT SYSTEM
ASYMMETRIC LOADS
EVALUATION PLAN

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For

BALTIMORE GAS AND ELECTRIC COMPANY
CONSUMERS POWER COMPANY
NORTHEAST UTILITIES
OMAHA PUBLIC POWER DISTRICT

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1.0 INTRODUCTION

This document describes a program and methods of analysis for conducting an evaluation of reactor coolant system components and supports when subjected to loads resulting from postulated pipe ruptures.

The reactor coolant system of the plants for which this evaluation is proposed has been evaluated for the effects of pipe rupture prior to obtaining an operating license. The methods employed in that evaluation, although less sophisticated than those presented here, provided assurance that safe shutdown following a postulated pipe rupture could be accomplished.

In May of 1975, it was determined by one applicant that asymmetric loading resulting from a postulated pipe rupture could have a significant effect on reactor vessel supports. Investigation of those supports and of other reactor coolant system components and supports utilizing the existing criteria for postulating pipe breaks and the existing methods of analysis led to further questions concerning the adequacy of the components and supports to withstand the effects of postulated pipe ruptures.

This document describes a proposed evaluation of the reactor pressure vessel, fuel assemblies (including grid structures), control rod drives, emergency core cooling piping attached to the primary coolant pipe, primary coolant piping, reactor vessel, steam generator, pump supports, and reactor internals when subjected to the effects of thrust, subcompartment pressures, and reactor vessel asymmetric pressures following design basis pipe ruptures in the reactor coolant system. The portion of the plan describing the evaluation of the biological shield wall will be presented at a later date.

The proposed evaluation employs methods and codes previously approved by NRC or those that are currently under review.

2.0 NONENCLATURE AND ABBREVIATIONS

Major Components of the Reactor Coolant System	RCS
Reactor Coolant Pump	RCP
Reactor Vessel	RV
Steam Generator	SG
Control Element Drive Mechanism	CEDM
Emergency Core Cooling System	ECCS
Core Support Barrel	CSB
Upper Guide Structure	UGS

3.0 REFERENCES AND CODES

3.1 Design Basis Pipe Breaks for the Combustion Engineering Two Loop Reactor Coolant System CENPD-168A, Combustion Engineering Inc., June, 1977

3.2 ICES-STRUDLII, The Structural Design Language, Engineering Users Manual, First Edition, Massachusetts Institute of Technology, November, 1968

3.3 SAPIV, A Structural Analysis Program for Static and Dynamic Response of Linear Systems, University of California, Berkeley, June, 1973

3.4 Design Basis Pipe Breaks for the Combustion Engineering Two Loop Reactor Coolant System CENPD-168A, Appendix A-5, Combustion Engineering Inc., June, 1977

3.5 "Description of Loss-of-Coolant Calculational Procedures", CENPD-26, Combustion Engineering Inc., August, 1971

3.6 Standard Review Plan 6.2.1.2, "Subcompartment Analysis", February, 1975

3.7 CESSAR, "Combustion Engineering Standard Safety Analysis Report", Section 6.2.1.1-4 approved December 31, 1975

3.8 "Reactor Plant Subcompartment Analysis", CENPD-141 Revision 2, March 1978

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

3.10 MARC-CDC, Non-Linear Finite Element Analysis Program, Control Data Corp., Minneapolis, Minn. 1976

3.11 "Structural Analysis of Fuel Assemblies for Combined Seismic and Loss of Coolant Accident Loadings", CENPD-178P, August, 1976

3.12 "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss of Coolant Accident Conditions with Application of Analysis to CE 800 Mwe Class Reactors", CENPD-42, 1971

3.13 "CESHOCK - A Computer Code to Solve the Dynamic Response of Lumped Mass Systems", Described and Verified in above Reference 3.12.

3.14 "SAMMSOR - A Finite Element Program to Determine the Stiffness and Mass Matrices of Shells of Revolution", Described and verified in above Reference 3.12.

3.15 "DYNASOR - A Finite Element Program for the Dynamic Non-Linear Analysis of Shells of Revolution", Described and verified in above Reference 3.12.

3.16 "LOAD - A Computer Code to Calculate Dynamic Axial LOCA Loads Using the Control Volume Formulation", Calculation SP80-STA-25, 5/15/78

3.17 "ASHSD - A Dynamic Stress Analysis Code of Axisymmetric Structures Under Arbitrary Loading", Described and verified in above Reference 3.12.

3.18 "RUMBLE - A Computer Code to Compute Fuel Bundle Stresses Based on Deflected Shapes", Described in above Reference 3.11.

3.19 "DDIFF Code Topical Report", CENPD-141, April 30, 1974

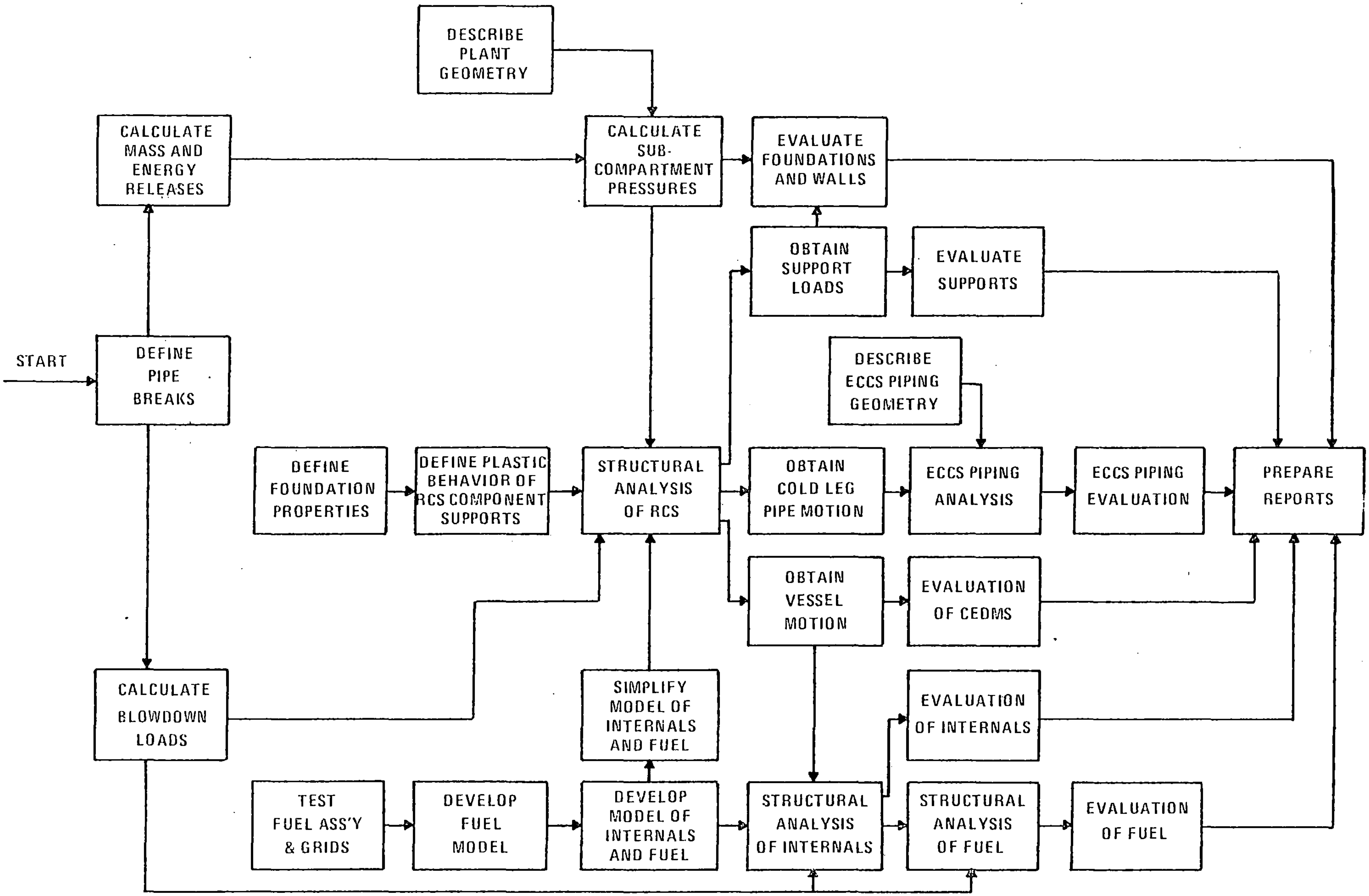
3.20 RELAP4/MOD5, ANCR-NUREG-1335, A Comprehensive Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems.

4.0 DESCRIPTION OF EVALUATION

4.1 ANALYTICAL SEQUENCING

The evaluation of RCS components and supports for the effects of pipe rupture requires consideration of a number of interrelated and dependent analyses. The relationship between them is shown in the proposed analytical sequence, Figure 4.1.1.

Subsequent sections describe each analytical step and the relationship of that step to its predecessor and successor.



ANALYTICAL SEQUENCING FOR EVALUATION OF RCS COMPONENTS AND SUPPORTS

FIGURE 4.1.1

4.2 PIPE BREAKS

4.2.1 DESIGN BASIS

Guillotine ruptures will be postulated to occur at the following locations:

a) Reactor Vessel Hot Leg Nozzle
b) Reactor Vessel Cold Leg Nozzle
c) Steam Generator Outlet Nozzle
d) Steam Generator Inlet Nozzle

4.2.2 METHOD OF ANALYSIS

For locations where relief from a full area break is anticipated, or where break opening times in excess of 10 milliseconds are likely, the flow area and break opening time will be developed using dynamic structural analysis methods descussed in Reference 3.1. There will be a dynamic non-linear time history analysis performed for each postulated rupture which will result in pipe end deflection time histories, flow area time histories, and maximum flow area.

4.2.3 MODELING

To determine the circumferential break characteristics a three-dimensional model of the reactor coolant system (RCS) containing reduced models of the reactor vessel (RV), steam generator (SG) and reactor coolant pumps (RCP) will be constructed using lumped mass parameters as discussed in Reference 3.1 (see Figure 4.2.1). This model will fully represent total system mass and stiffness as well as provide details of vessel and component supports. The basic model will consider intact pipes. For each circumferential rupture location at which a flow area is to be developed, the basic model will be modified. The modifications will provide complete pipe discontinuity at the specified location and greater pipe mass detail at both sides of the rupture. (see Figure 4.2.2).

4.2.4 FORCING FUNCTIONS

Three dimensional time history forcing functions discussed and developed in Reference 3.1 will be applied to each side of each circumferential break analyzed.

MODEL JOINTS	DESCRIPTION
221, 211, 3221, 3211	SG SHEAR KEYS— RESTRAINT ⊥ HOT LEG ONLY
250, 3250	SG SNUBBERS— RESTRAINT // HOT LEG ONLY
70, 3070	LOWER SG SUPPORTS— STOPS // HOT LEG AWAY FROM RV; RESTRAINTS ⊥ HOT LEG AND VERTICAL
1900, 2900, 3900	RV SLIDING SUPPORTS— RADIAL MOVEMENT; TANGENTIAL AND DOWNWARD RESTRAINTS

(TYPICAL)

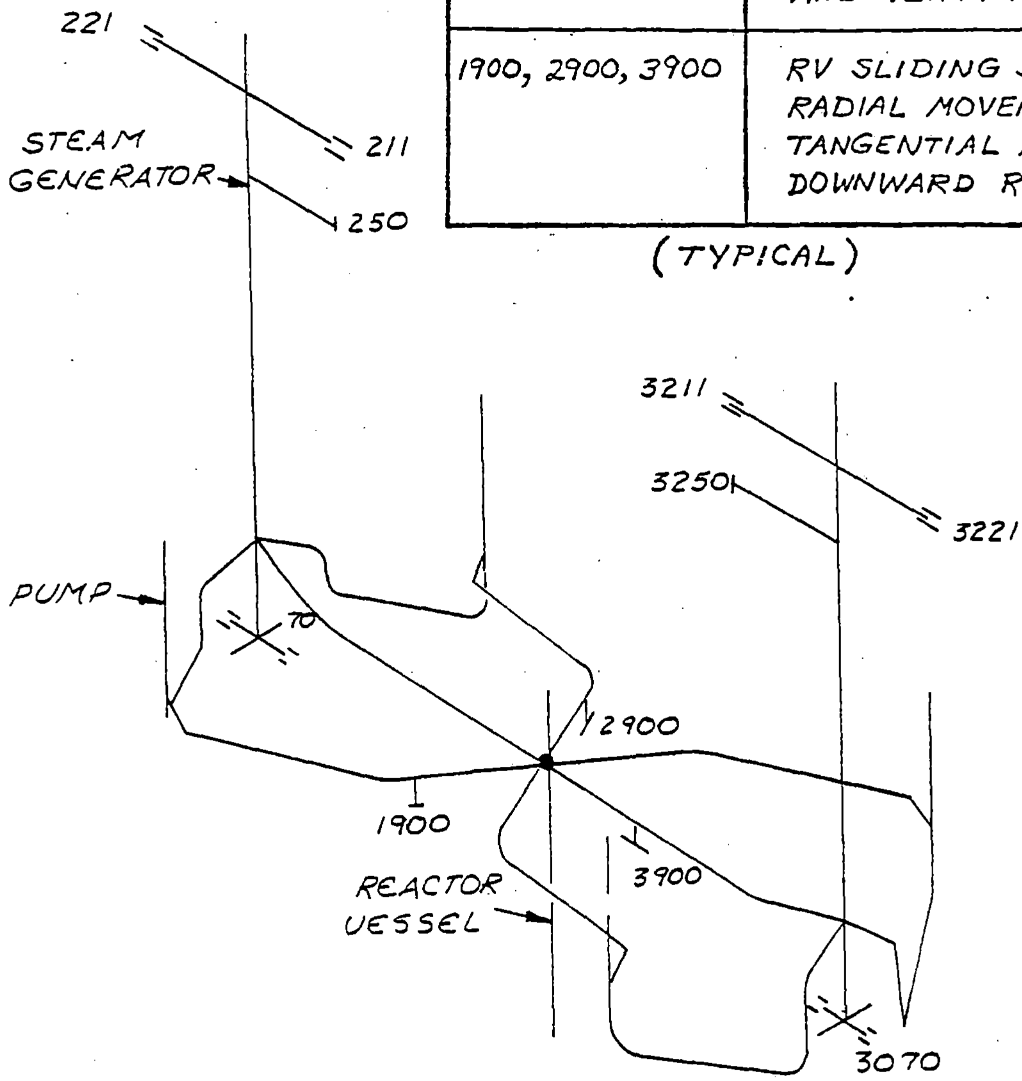


FIGURE- 42.1 REACTOR COOLANT SYSTEM MODEL FOR PIPE BREAK ANALYSIS - INTACT LOOPS

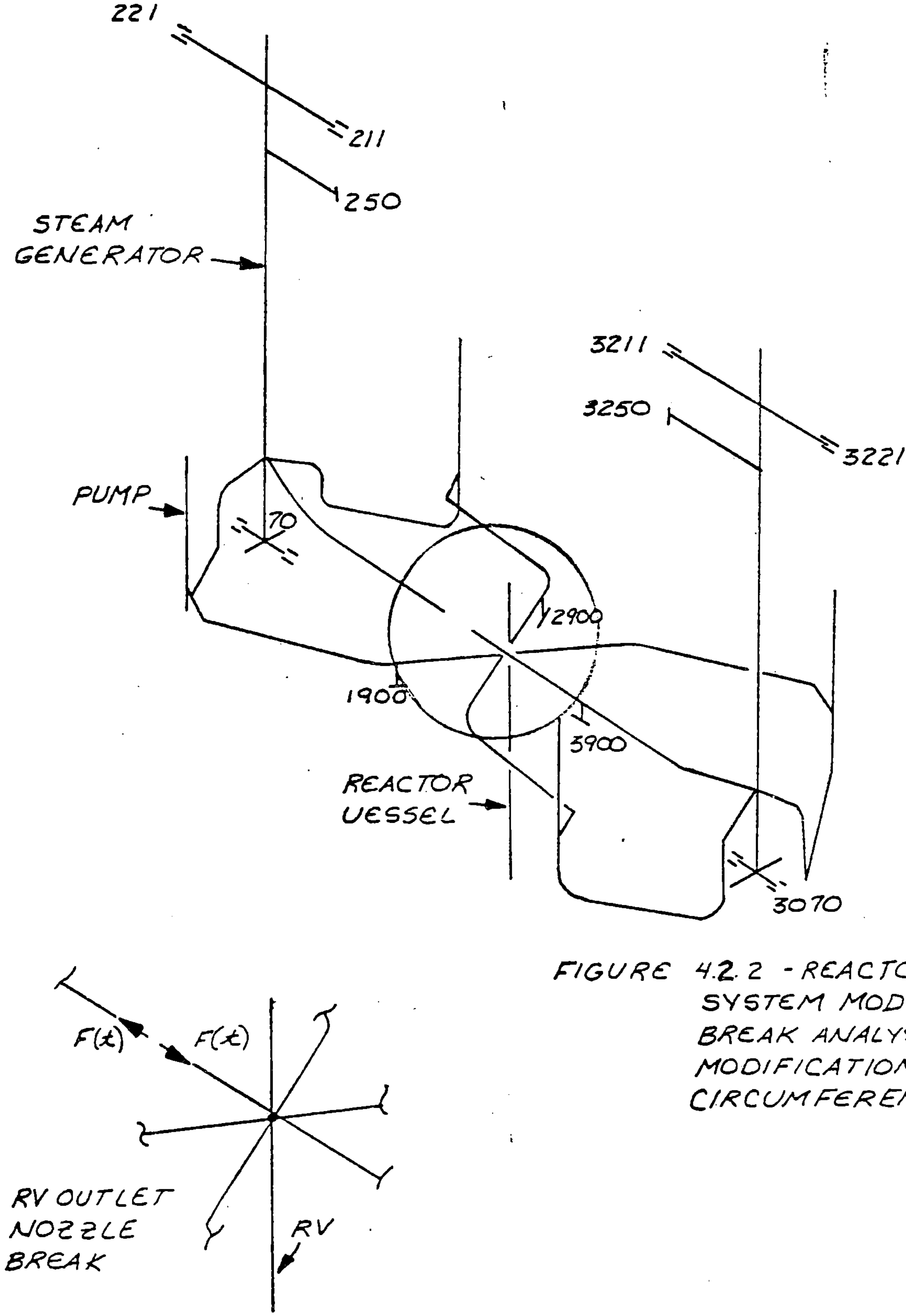


FIGURE 4.2.2 - REACTOR COOLANT SYSTEM MODEL FOR PIPE BREAK ANALYSIS - TYPICAL MODIFICATION FOR A CIRCUMFERENTIAL BREAK

4.2.5 COMPUTER CODES

The physical definition of the RCS for each postulated circumferential rupture will be supplied in the STRUDL (Reference 3.2) computer code or the SAP IV (Reference 3.3) computer code to generate the appropriate condensed stiffness matrix. The condensed matrix, masses, definition of gapped interfaces, damping and the appropriate three-dimensional force will be supplied to the DAGS (Reference 3.4) computer code which will generate severed pipe end deflection time histories which in turn will be used to calculate flow area time histories. The use of these codes for pipe break analysis is described in Reference 3.1.

4.2.6 RESULTANT PIPE BREAKS

Design basis pipe breaks for analysis of the RCS will be postulated. The break flow areas and opening times computed by this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific analyses will be performed. The resulting flow areas and opening times for ruptures will be supplied as input to the blowdown loads analysis described in Section 4.4 and to the subcompartment pressurization analysis described in Section 4.3.

4.3 SUBCOMPARTMENT ANALYSIS

4.3.1 DESIGN BASES

Subcompartment pressures in the reactor cavity and steam generator compartment resulting from dispersion of fluid emanating from design basis pipe breaks will be calculated. Methods for determination of characteristics of design basis pipe breaks are discussed in Section 4.2. The calculated subcompartment pressures constitute one of the forcing functions employed in the evaluation of compartment structural design and component structural design. Definitions of design basis pipe breaks will be furnished as part of the structural evaluation.

4.3.2 DESIGN FEATURES

The reactor cavity and steam generator compartment will be subdivided into nodes to reflect physical plant characteristics with respect to components, structures, piping and other major obstructions. Drawings showing plant characteristics and resultant nodalization will be provided. Tabulations of the nodal net-free volumes, flow path areas, L/A ratios and geometric and friction flow loss factors will also be provided.

Where applicable, the treatment of movable obstructions to vent flow will be discussed. The space occupied by piping and component insulation will be deducted in determining volumes and vent areas.

4.3.3 DESIGN EVALUATION

4.3.3.1 Method for Mass and Energy Releases

The modified CEFLASH-4 computer program will be used to compute the blowdown release rates. The CEFLASH-4 program is described in Reference 3.5 and its acceptability is stated in Reference 3.6. The modification to this CEFLASH-4 code is the incorporation of a critical flow correlation subroutine which provides a best estimate of the blowdown rates. This is the same critical flow subroutine as discussed in Reference 3.7. The Henry/Fauske critical flow correlation is used for subcooled and low quality fluid conditions and the Moody critical flow correlation for the remainder of the saturation regime. A flow multiplier of 0.7 will be used throughout. (See Appendix A)

Reactor coolant system nodalization shown in Figure 4.3.1 will be used.

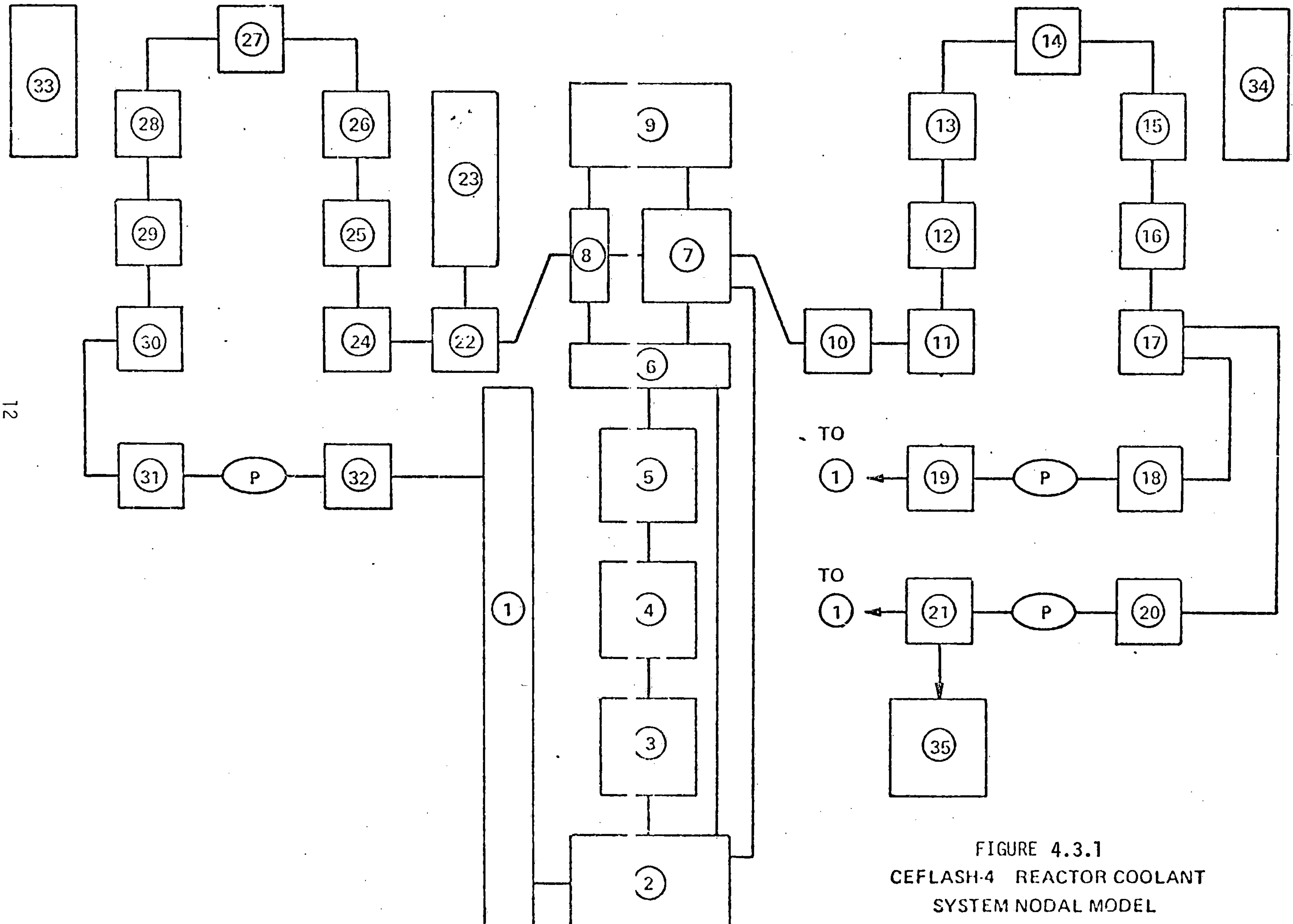


FIGURE 4.3.1
CEFLASH-4 REACTOR COOLANT
SYSTEM NODAL MODEL
(LEGEND FOLLOWS)

Legend for Figure 4.3.1

1. Reactor vessel downcomer.

2. Reactor vessel lower plenum.

3-5. Reactor core.

6. Fuel alignment plate region.

7. Reactor vessel exit plenum.

8. CEA shrouds.

9. Reactor vessel upper head.

10. Reactor outlet pipe.

11. Steam generator inlet plenum.

12-16. Steam generator tubes.

17. Steam generator outlet plenum.

18. Pump suction pipe.

19. Pump discharge pipe.

20. Pump suction pipe.

21. Pump discharge pipe.

22. Reactor outlet pipe.

23. Pressurizer.

24. Steam generator inlet plenum.

25-29. Steam generator tubes.

30. Steam generator outlet plenum.

31. Pump suction pipe (2).

32. Pump discharge pipe (2).

33. Steam generator secondary side.

34. Steam generator secondary side.

35. Containment.

When a circumferential pipe rupture is postulated, two nodes are used to represent the severed pipe.

4.3.3.2 Results for Mass and Energy Releases

Blowdown release rates will be generated for each pipe break postulated in the reactor cavity and steam generator compartment. A table of blowdown mass flow rate and energy release rate as a function of time will be provided for each break.

4.3.3.3 Application of Mass and Energy Release Results

The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific verifications and/or analyses will be performed.

4.3.3.4 Method for Subcompartment Pressure Analysis

The DDIFF-1 MOD 7 (Reference 3.8) computer program will be used to perform the subcompartment pressure analysis except that the generic reactor cavity analysis results will be obtained from RELAP4 (Reference 3.20). A compartment multi-node, space-time pressure response analysis will be performed.

Compartment nodal models will be developed following a detailed review of the geometric features. Significant spatial variations in pressure that might exist because of geometric influences will be precluded. Advantage will be taken on nodalization sensitivity studies previously performed. Guidance from results reported in SAR's and in Reference 3.19 will be utilized. No additional nodalization sensitivity studies will be made.

4.3.3.5 Reactor Cavity Analysis

The reactor cavity will be modelled to obtain the pipe break blowdown spatial pressure-time history response to determine the differential pressures on the cavity structure and reactor vessel. Postulated ruptures in the reactor vessel inlet and outlet pipes will be evaluated. No margin will be added to the calculated pressures for the design evaluations. Pressurization analysis of regions distant to the break location is not required.

Graphs of pressure versus time for a representative number of nodes will be provided. The maximum calculated structural differential pressure for each node and time of its occurrence will be tabulated. A discussion will be made as to whether the differential pressure is uniformly applied to the compartment structure or spatially varied.

The loading on the reactor vessel due to reactor cavity pressurization will be determined by multiplying each individual nodal pressure transient by the corresponding projected area of the component within the nodal boundaries. The resultant transient load due to each individual nodal pressure will be applied at the center of pressure of the node on the component, oriented in space along the normal to the projected area of the component. All such loads acting on the component due to a given postulated pipe break are applied simultaneously at the separate centers of nodal pressure on a time basis. Projected nodal areas, the direction cosines of the resultant loads, the locations of load applied on the reactor vessel for computation of loads due to the pressure transients, and the coordinate system will be provided. Pressure response transients in the nodes interfacing with the reactor vessel will be given.

4.3.3.6 Steam Generator Compartment Analysis

The steam generator compartment will be modelled to obtain the blowdown spatial pressure-time history response to determine the differential pressures on the steam generator. Postulated ruptures in the steam generator inlet and outlet pipes will be evaluated. No margin will be added to the calculated pressures. Evaluation of steam generator compartment walls is not required.

Graphs of pressure versus time for a representative number of nodes will be provided. Maximum calculated differential pressure across the steam generator, as well as time of occurrence will be provided.

4.3.3.7 Application of Subcompartment Pressure Analysis

The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific verifications and/or analyses will be performed.

4.4 BLOWDOWN LOADS

The term "blowdown loads" is used to designate the thermodynamic and hydrodynamic induced forcing functions that occur throughout the primary system during a postulated loss of coolant accident. These forcing functions consist of the space-time distribution of fluid pressures, flow rates, and densities.

The transient pressures act directly on the adjacent structures. In addition, changes in the flow rates and fluid densities result in transient drag forces which also act on adjacent structures.

4.4.1 PRESSURE LOADS

The transient pressure, flow rate and density distributions will be computed by the CEFLASH-4B computer code according to the methods documented in Ref. 3.9. These calculations are valid for both the subcooled and saturated portions of the decompression.

The CEFLASH-4B computer code is based on a node-flow path concept in which control volumes (nodes) are connected in any desired manner by flow areas (flow paths). A complex node-flow path network is used to model the primary reactor coolant system (RCS). The CEFLASH-4B modeling procedure has been compared to a large scale experimental blowdown test with excellent agreement.

The laws of conservation of mass, energy and momentum along with a representation of the equation of state are solved simultaneously by CEFLASH-4B. The hyraulic transient of the reactor is coupled to the thermal response of the core by analytically solving the one dimensional radial heat condition equation in each core node.

The self-initialization option of the CEFLASH-4B code is used to establish pre-blowdown steady state conditions in the RCS through the use of specified input quantities.

The blowdown loads model uses a non-equilibrium critical flow correlation for computing the subcooled and saturated fluid discharge through the break.

The pertinent results of the CEFLASH-4B calculation will be used for structural response analyses.

There are several conservatisms in the formulation and use of the CEFLASH-4B code which contribute to conservative results for the blowdown loads. These do not necessarily translate into equivalent conservatisms for the structural response analyses which employ these blowdown loads as a portion of their input.

These conservatisms are:

1. All PWR walls and plates are rigid. Thus, the interaction between the fluid and the structure is not accounted for in CEFLASH-4B. An accounting for this phenomena could reduce the blowdown loads, but not the structural responses, by approximately 30%.

2. The acceleration of the primary fluid through the break is instantaneous. Thus, the outflow rate is up to the critical value as soon as any break area is exposed. In actuality, the inertia of the fluid, at the break, will cause a slight delay in the occurrence of full critical flow. This effort is not expected to be large.

The present method of applying the CEFLASH-4B code to the analysis of the primary system blowdown loads is documented in topical report CENPD-252-P.

The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific verifications and/or analyses will be performed.

4.4.2 DRAG LOADS

A break in the primary coolant system will result in large local pressure differences across various reactor vessel internal components and an acceleration of the local fluid velocity in various regions. The acceleration of the local fluid velocity as well as density changes can result in higher component drag loads than occur during steady state reactor operation.

4.4.2.1 Core Drag Loads

The core drag loads are considered to be dependent on the local fluid Reynolds number. The wall shear forces and spacer grid losses are evaluated from Reynolds number dependent functions obtained from experimental data. The total drag force on the fuel assemblies is obtained by summing the contributing portions from each segment represented in the CEFLASH-4B model of the core region.

4.4.2.2 CEA Shroud Loads

During normal operation the reactor coolant flows axially through the core into the upper guide structure. Within the upper guide structure the coolant flow changes direction so that it exits radially through the hot leg nozzles. During a LOCA (both hot and cold leg breaks) the transverse flow of the coolant across the CEA shrouds gives rise to loads which induce deflections in these shrouds. In addition, the total transverse load on all of the

CEA shrouds acts to produce a net lateral load on the upper guide strucutre.

The transverse drag forces will be determined from existing flow model experiments. The measured experimental forces will be scaled to represent the actual forces on the upper guide structure using the transient flow rate and density information computed by the CEFLASH-4B code.

The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific variations and/or analyses will be performed.

4.5 REACTOR VESSEL, RC PIPE, AND RCS SUPPORTS

4.5.1 DESIGN BASIS

The analyses described in this section will be performed to evaluate the response of major components of the reactor coolant system (RCS) to the forces associated with pipe rupture and normal operating conditions.

The forces associated with the postulated pipe breaks include pipe thrust forces at the break location, resultant subcompartment differential pressurization forces, and internal asymmetric hydraulic forces acting on the vessel and internals.

4.5.2 METHOD OF ANALYSIS

Dynamic analysis of the RCS will be performed using lumped parameter models which include details of the reactor vessel (RV) and internals, steam generator (SG) and internals, reactor coolant pumps (RCP) and interconnecting reactor coolant piping.

The pipe break thrust forces, asymmetric subcompartment pressurization forces, and asymmetric reactor internal hydraulic forces will be applied as simultaneous time-history forcing functions.

A non-linear time history dynamic analysis of a three-dimensional mathematical model of the reactor coolant system will be performed for each design basis pipe break to demonstrate the adequacy of the reactor vessel and steam generator supports and to generate time history motions for further subsystem analysis.

4.5.3 ELASTIC PLASTIC ANALYSIS OF SUPPORTS

An elastic plastic analysis of each component support region will be performed for two purposes. The first is to determine the non-linear load displacement relationship for inclusion in the RCS model described in Paragraph 4.5.4. The second purpose is for evaluation of the acceptability of the computed loads for each component support region. This evaluation is discussed in Paragraph 4.5.8.

4.5.3.1 Detailed Finite Element Models

The three-dimensional elastic plastic analysis of the component support regions will be performed with the MARC non-linear general purpose finite element program (Ref. 3.10). The finite element models used in the analysis must be sufficiently detailed to provide an accurate load displacement curve and instability load but simple enough to provide a reasonably

efficient solution with the MARC program. A typical model of a reactor vessel support on an inlet nozzle is shown in two views in Figures 4.5.1 and 4.5.2. The model provides for the determination of the load displacement relationships not only at the support but at the nozzle/vessel intersection and at the nozzle/pipe intersection.

Models with a similar degree of detail will be developed for each different RCS support region. The material properties used in the analyses will be determined based on laboratory tests already performed.

4.5.3.2 Load Displacement Relationships

Each support region model will be loaded statically in the direction of load expected during the RCS structural analysis. The load will be increased until the deformation increases without bound. The overall behavior of the region will be determined for input to the RCS structural analysis as a non-linear support stiffness.

4.5.3.3 Ultimate Load Analysis

The detailed stresses and strains for each loading up to ultimate load are obtained from the analysis described in Paragraph 4.5.3.2. This information as well as the instability loads (the loads at which deformation increases without bound) are stored for later use in evaluating the effect of the loads computed by the RCS structural analysis.

4.5.4 MODELS

Condensed structural models of the major components of the RCS and component internals will be developed from detailed representations of each component by incorporating response characteristics and maintaining interface response compatibility. For each analysis, a model of the RCS including reactor vessel, steam generator, reactor coolant pumps, and interconnecting piping will be employed. The model will contain a high degree of structural detail for all components, with concentration of mass detail depending on the component to be evaluated. However, a mass representation of other RCS components will be included for each model.

Load deflection characteristics of the supports and foundations of RCS components as determined by the procedures described in 4.5.3 will be included in the models.

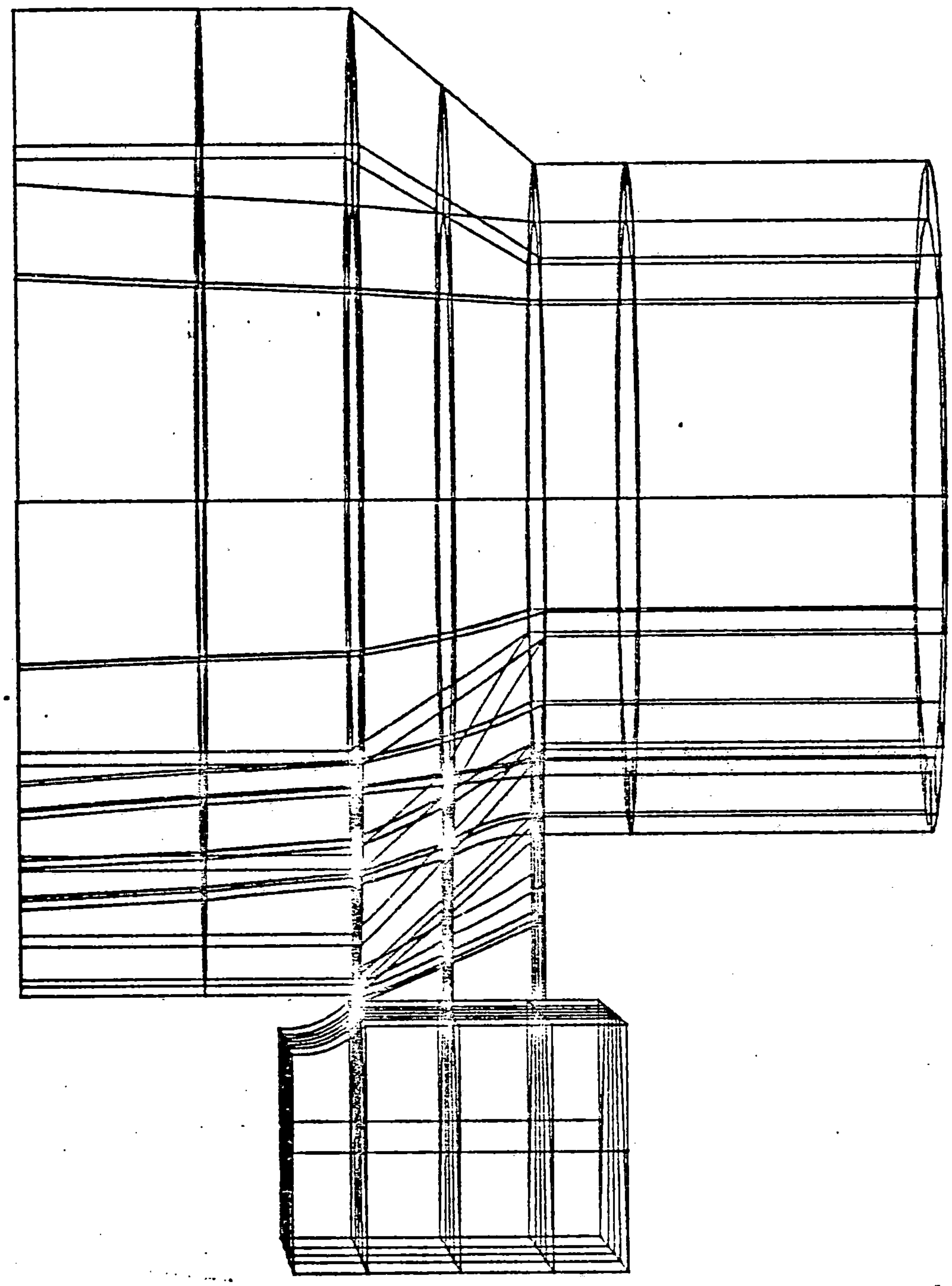


FIGURE 4.5.1. FINITE ELEMENT MODEL OF REACTOR VESSEL SUPPORT ON INLET NOZZLE SIDE VIEW

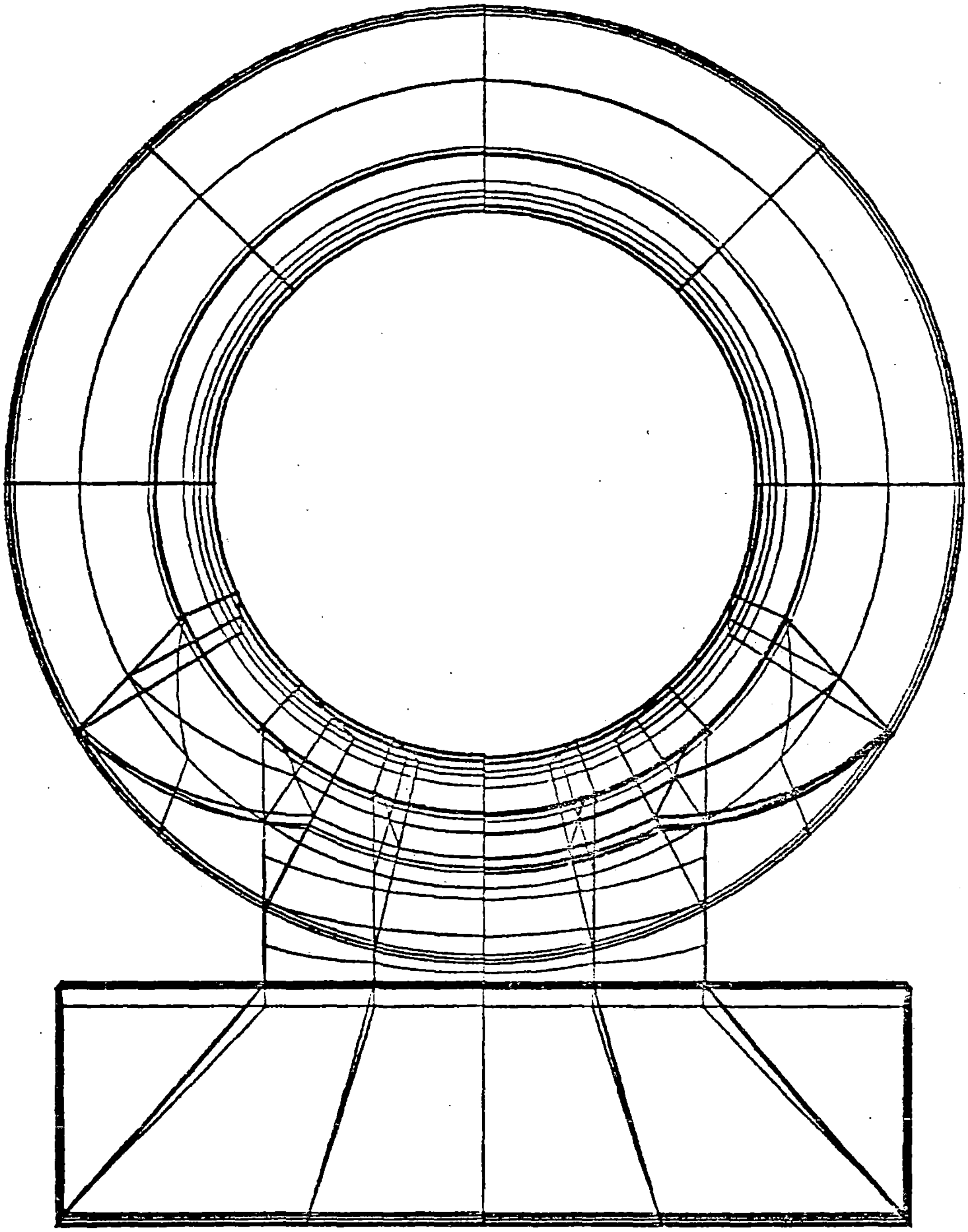


FIGURE 4.5.2 FINITE ELEMENT MODEL OF REACTOR VESSEL SUPPORT ON INLET NOZZLE. AXIAL VIEW

4.5.4.1 Models for Reactor Vessel Analysis

For analysis of the motions and support reactions of the RV under pipe rupture conditions, models of the RCS will be developed according to the methods described above. A simplified model of the RCS will be used in evaluating a representative model of the reactor internals as discussed in Section 4.6. The purpose of this model of the internals will be to represent the effects of the reactor internals on the reactor vessel support reactions.

The internals model will be coupled with the model of the reactor vessel which will in turn be incorporated as part of the model of the RCS. Mass detail will be concentrated at the RV and condensed mass representations of the SG and RCP's will be included.

Hydrodynamic effects, including both virtual mass and annular effects, will be accounted for in the coupling between the RV and core support barrel (CSB) and between the CSB and core shroud. The resulting mass matrix will be non-diagonal because of the inclusion of the hydrodynamic coupling effects.

The reactor vessel and all internal components will be moduled as colinear elements and gaps will be incorporated at internal and support interfaces wherever required. Figure 4.5.3 shows an example of the gapped interface at the RV horizontal support.

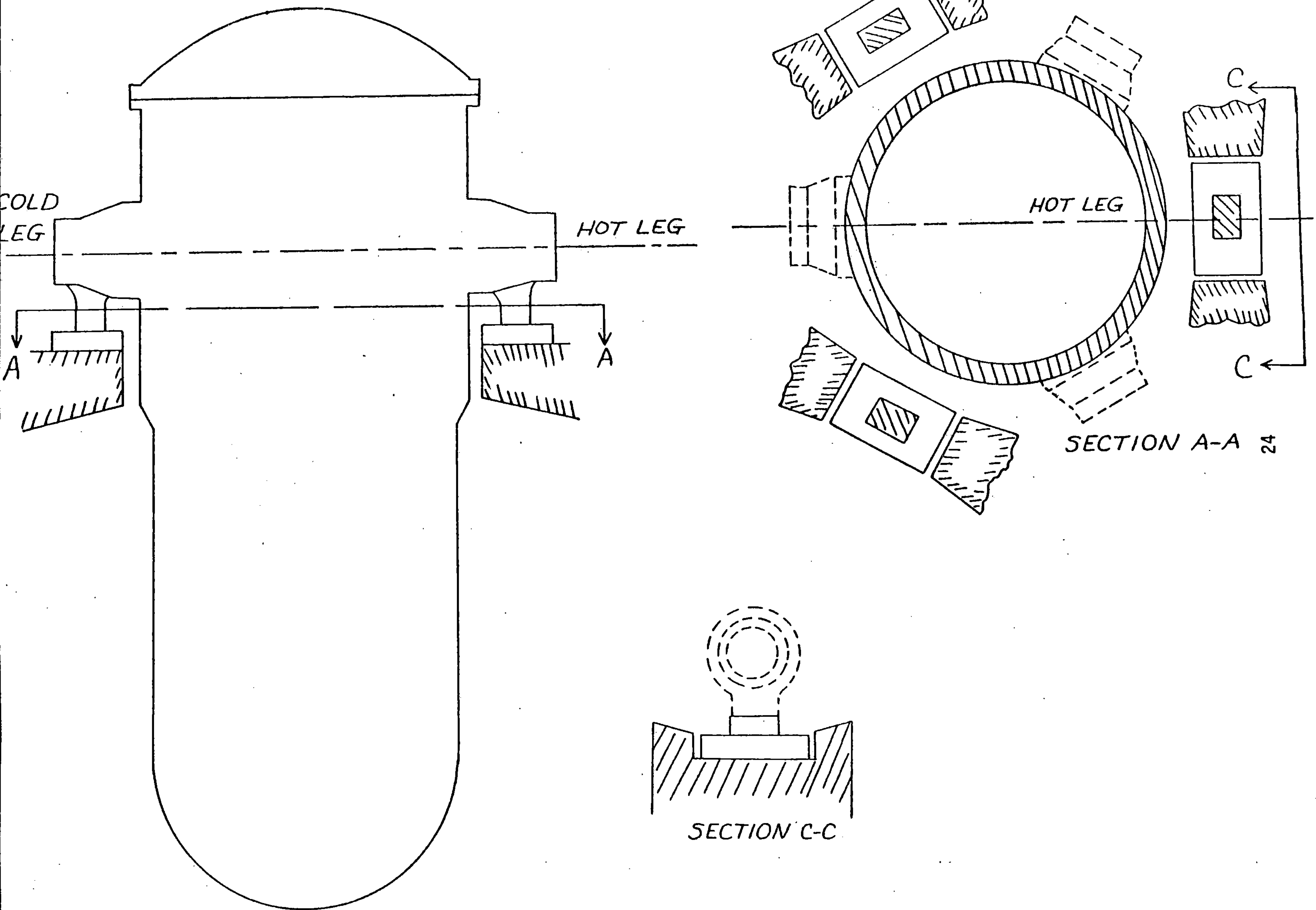
The model will be modified for each of the postulated pipe ruptures.

4.5.4.2 Models for Steam Generator Support Analysis

The detailed SG models will include SG internals, including a gross tube bundle elements and tube bundle to shell interface details. The SG and all internals will be modeled as colinear elements and gaps will be presented at internal and support interfaces as necessary.

Mass detail will be concentrated at the SG and condensed models of the RV and RCP's and will be included. The model will be modified to reflect each postulated rupture.

FIGURE 4.5.3 - REACTOR VESSEL LOCA SUPPORTS



4.5.5 FORCING FUNCTIONS

4.5.5.1 Reactor Vessel Support Analysis

The operating times and flow areas of postulated breaks will be determined by methods described in Section 4.2.

For each postulated break at the reactor vessel nozzles, a thermal-hydraulic analysis will be performed according to the procedures and models discussed in Section 4.4. The resulting pressure and flow parameters will be used to to calculate three-dimensional time history forcing functions acting on the reactor vessel and internals at appropriate locations over the length of the vessel and core support barrel. The asymmetric sub-compartment pressures in the reactor vessel cavity will be calculated by methods described in Section 4.3. Resulting pressures will be used to calculate the asymmetric cavity pressure forces acting on the vessel exterior. Pipe tension release or thrust force at the postulated break will be included to calculate the total three-dimensional time history forces applied to the exterior of the vessel. A schematic view of the simultaneous system of forces on the vessel is shown on Figure 4.5.4.

4.5.5.2 Steam Generator Support Analysis

For each design basis pipe break in the steam generator compartment, a subcompartment pressurization analysis will be performed using the procedures and models discussed in Section 4.2. Resulting pressures will be used to calculate asymmetric forces on the SG. These and the pipe tension release or thrust force at the postulated break will be used to calculate the three-dimensional time history forces applied to the exterior of the SG.

4.5.6 COMPUTER CODES

The physical definition of the RCS for each postulated rupture will be represented in the STRUDL computer code (Ref. 3.2) or the SAP IV computer code (Ref. 3.3) to generate the appropriate condensed stiffness matrix for the structure. The condensed matrix, internal and external mass and definition of the gapped interfaces, appropriate three-dimensional time history forces, and appropriate hydrodynamic coupling effects will be supplied in DAGS computer code (Ref. 3.4) which will generate support loads and time history motions by performing a non-linear time history analysis.

P_I^V = VERTICAL HYDRAULIC FORCES ON INTERNALS

P_R^H = HORIZONTAL HYDRAULIC FORCES ON VESSEL

P_R^V = VERTICAL HYDRAULIC FORCES ON VESSEL

P_T = PIPE BREAK THRUST

P_C^H = HORIZONTAL CAVITY PRESSURE FORCES

P_C^V = VERTICAL CAVITY PRESSURE FORCES

P_I^H = HORIZONTAL HYDRAULIC FORCES ON INTERNALS

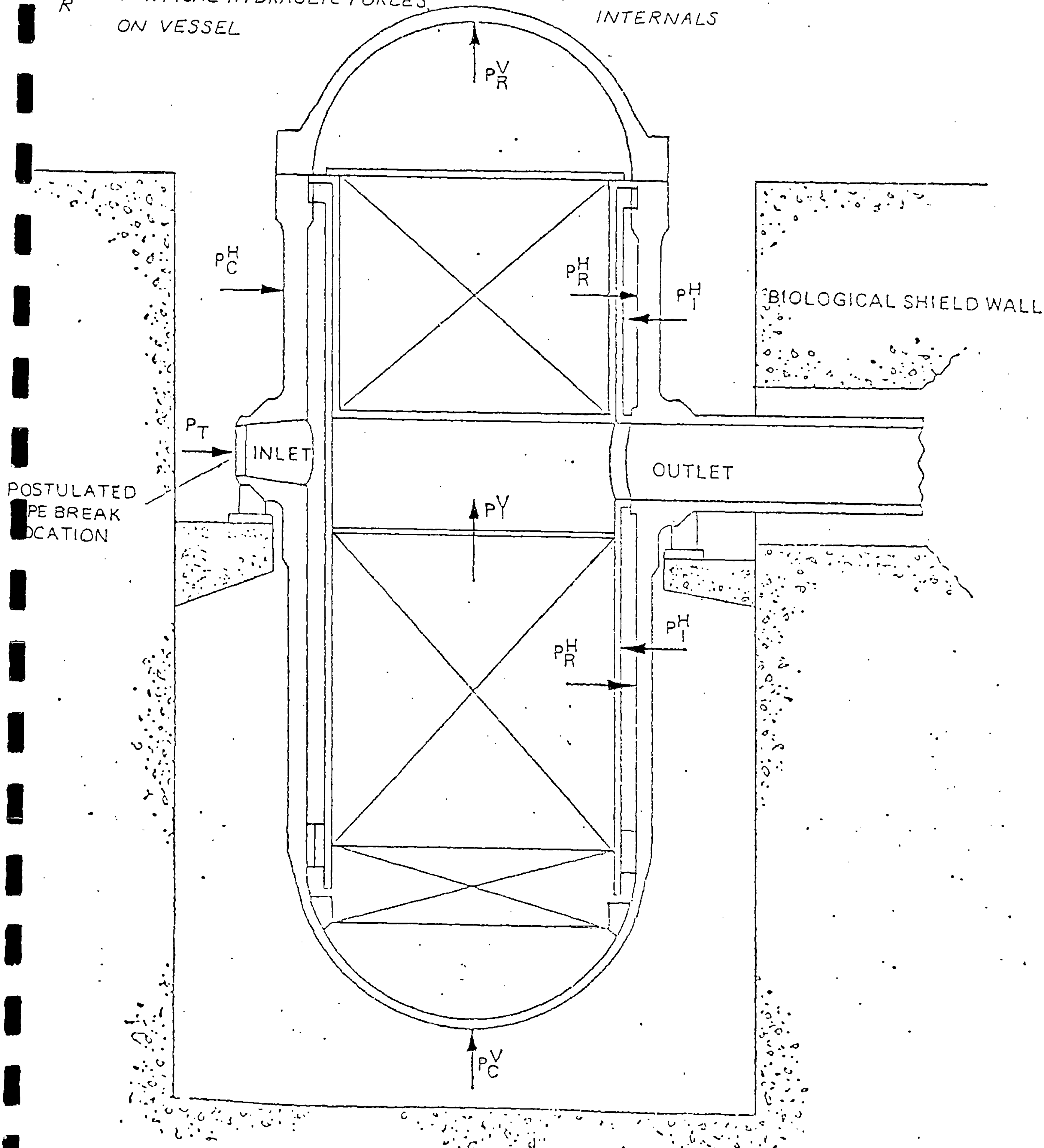


Figure 4.5.4 — REACTOR VESSEL FORCES FOLLOWING LOCA

4.5.7 RESULTS OF ANALYSIS

The results of the analyses described above include time histories of component maximum support loads and deflections for support evaluation and time histories of motion of components and piping for subsystem analysis of vessel internals (Section 4.6), CEDM (Section 4.8), and ECCS piping (Section 4.9).

The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific verifications and/or analyses will be performed.

4.5.8 EVALUATION OF COMPONENTS AND SUPPORTS

4.5.8.1 Acceptance Criteria

The loads resulting from the RCS Structural Analysis will be evaluated by comparison to the elastic-plastic analysis results. The objective of the evaluation will be to apply the criterion of the ASME Boiler and Pressure Vessel Code Section III, Divisionl, Appendix F, Article F1324. This criterion states that violation of the pressure boundary will not occur if the applied loads do not exceed 70% of the plastic instability load. An alternate criterion based on strain limits in which a suitably conservative strain limit must be justified may also be employed.

The reactor vessel (RV) is expected to be most highly stressed at the intersection of the RV shell and the nozzles. The acceptance criteria described above apply in this region.

The reactor coolant pipe is expected to be most highly stressed near the supports and at component nozzles. The acceptance criteria described above apply in these regions.

The acceptance criteria described above apply directly to all RCS supports.

4.5.8.2 Evaluation

The maximum load experienced by the nozzle regions of the reactor vessel for each design basis pipe break in the reactor cavity will be evaluated. The results of the RCS structural analysis will be compared to the results of the elastic-plastic analyses of Section 4.5.3.

The integrity of the reactor vessel will be evaluated by comparing the computed elastic-plastic behavior to the instability load or to strain limits according the the acceptance criteria of Paragraph 4.5.8.1.

The maximum load experienced by the support and nozzle regions of the reactor coolant piping will be evaluated for each design basis pipe break. The evaluation process is similar to that discussed above. The integrity of the reactor coolant piping will be evaluated by comparing the computed elastic-plastic behavior to the instability load or the strain limits according to the acceptance criteria of Paragraph 4.5.8.1.

The maximum load or deflection experienced for each RCS support will be evlauated for each design basis pipe break in the steam generators compartment. The evaluation process will be similar to that discussed above. The integrity of the reactor coolant system supports will be evaluated by comparing the computed elastic-plastic behavior to the instability load or to strain limits according to the acceptance criteria of Paragraph 4.5.8.1.

4.6 REACTOR VESSEL INTERNALS

4.6.1 DESCRIPTION OF INTERNALS

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

A representative set of reactor internals is shown in Figure 4.6.1 and described in the following:

CORE SUPPORT ASSEMBLY

The major support member of the reactor internals is the core support assembly. This assembled structure consists of the core shroud. The major material for the assembly is Type 304 stainless steel.

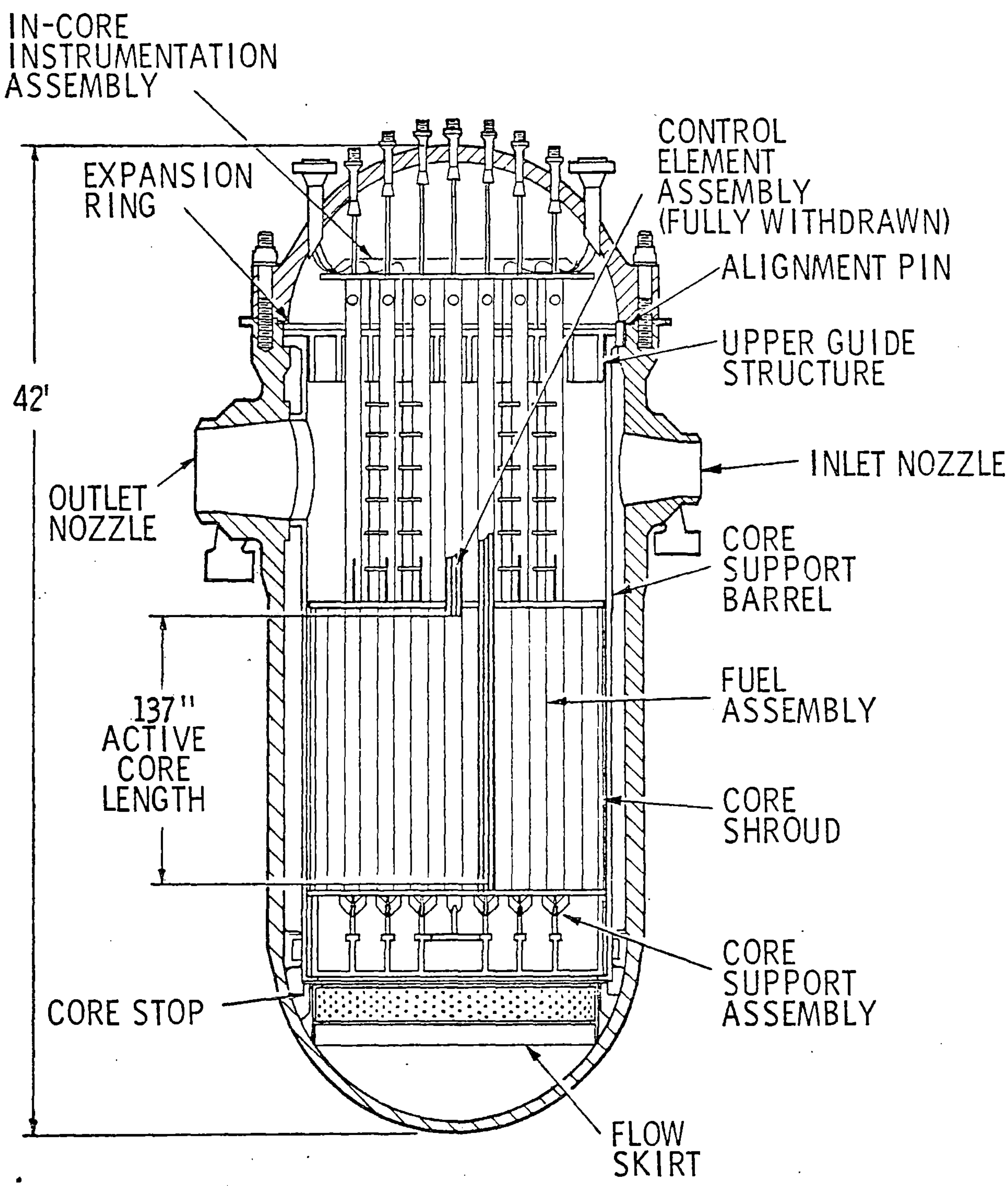
The core support assembly is supported at its upper end by the upper flange of the core support barrel which rests on a ledge in the reactor vessel flange.

The lower flange of the core support barrel supports and positions the lower support structure. The lower support structure provides support for the core by means of a core support plate supported by columns resting on beam assemblies. The core support plate provides support and orientation for the fuel assemblies. The core shroud which provides lateral support for the fuel assemblies is also supported by the core support plate. The lower end of the core support barrel is restrained radially by six core barrel-to-pressure vessel snubbers.

The core support barrel is a right circular cylinder suspended by a 4-inch thick flange 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 flanges are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

Since the core support barrel is long and is supported only 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 double lugs around the circumference and are the grooves

REACTOR VERTICAL ARRANGEMENT



of a "tongue-and-groove" assembly; the pressure vessel lugs are the tongues. Minimizing the clearance between the two mating pieces limits the amplitude of any vibration. During assembly, as the internals are lowered into the vessel, the pressure vessel tongues engage the core support grooves in an axial direction. With this design, the internals may be viewed as a beam with supports at the furthest extremities.

Radial and axial expansion of the core support barrel are accomodated, but lateral movement of the core support barrel is restricted by this design. The pressure vessel tongues have bolted, lock welded Inconel X shims and the core support barrel grooves are hardfaced with Stellite to minimize wear.

The core support plate is a Type 304 stainless steel plate into which the necessary flow distributor holes for the fuel assemblies have been machined. Fuel assembly locating pins (four for each assembly) are shrunk-fit into this plate. Columns and support beams are located between this plate and the bottom of the core support barrel in order to provide support for this plate and transmit the core load to the bottom flange of the core support barrel.

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. A gap is maintained between the core shroud outer perimeter and the core support barrel in order to provide some coolant flow upward between the core shroud and core support barrel, thereby minimizing thermal stresses in the core shroud and eliminating stagnant pockets.

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

The upper guide structure assembly consists of the upper support structure, control element shrouds, a fuel assembly alignment plate and an expansion compensating ring. The upper guide structure assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, prevents fuel assemblies from being lifted out of position during a severe accident condition and protects the CEA's from the effect of coolant crossflow in the upper plenum. The upper guide structure is handled as one unit during installation and refueling.

The upper end of the assembly is a structure consisting of a flange welded to a cylindrical section which supports the

upper ends of the shrouds. The periphery of the flange contains four accurately machined and located alignment keyways, equally spaced at 90 degree intervals, which engage the core barrel alignment keys. The reactor vessel closure head flange is slotted to engage the upper ends of the alignment keys in the core barrel. This system of keys and slots provides an accurate means of aligning the core with the closure head. The grid aligns and supports the upper end of the CEA shrouds. The shrouds extend between the upper end of the structure and the fuel assembly alignment plate.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate align the fuel assemblies. The fuel assembly alignment plate also has four equally spaced slots on its outer edge which engage with Stellite hard-faced pins protruding from the core shroud to limit lateral motion of the upper guide structure assembly during operation. The fuel alignment plate bears the upward force of the fuel assembly holddown devices. This force is transmitted from the alignment plate through the CEA shrouds to the upper guide structure support plate and thence to the expansion compensating ring.

The expansion compensating ring bears on the flange at the top of the assembly to accommodate axial differential thermal expansion between the core barrel flange, upper guide structure flange and pressure vessel flange support edge and head flange recess.

The upper guide structure assembly also supports the incore instrumentation thimble support frame and guide tubes. These tubes are conduits which protect the incore instrumentation and guide them during removal and insertion operations.

4.6.2 INTERNALS ANALYSIS MODELS

The postulated pipe breaks in the reactor cavity result in horizontal and vertical forcing functions which cause the internals to respond to both beam and shell modes. This section describes the structural models to be used for determining these responses.

4.6.2.1 Detailed Nonlinear Internals Models

Detailed structural mathematical models of the reactor internals will be developed based on the geometrical design. These models will be constructed in terms of lumped masses connected by beam or bar elements, and will include nonlinear effects such as impacting and friction. The models will be developed for input to the CESHOCK

code which solves the differential equations of motion for lumped parameter models by a direct step-by-step numerical integration procedure. The model definitions will employ the procedures established in Combustion Engineering Topical Reports, CENPD-42 and CENPD-178P, and in addition will include hydrodynamic coupling effects and a detailed representation of the core support barrel/upper guide structure/reactor vessel interfaces. Separate models will be formulated for the horizontal and vertical directions to more efficiently account for structural and response differences in those directions.

4.6.2.1.1 Horizontal Model

The models for the horizontal directions will be developed in terms of lumped masses connected by beam elements. The stiffness values for the beam elements will generally be evaluated using beam characteristics equations. In complex areas such as the core support barrel flanges, upper guide structure flange and lower support structure grid beams, the stiffnesses will be derived from finite element analyses. The lumped-mass weights will be based upon the mass distribution of the internals structures. Local masses such as plates and snubber blocks will be included at appropriate nodes. The effect of the surrounding water on the dynamics of the internals for horizontal motion will be accounted for by hydrodynamically coupling the components separated by a narrow annulus - the vessel, core barrel, and core shroud. The clearance between the core support barrel and the reactor vessel snubbers as well as the clearance between the core shroud, guide lugs and the fuel alignment plate will be simulated by nonlinear springs which account for the loads generated when impacting occurs. There will be a representation of the core in the internals models which provides appropriate inertial and impact feedback effects on the internals response.

4.6.2.1.2 Vertical Model

The vertical model stiffness values will generally be calculated using bar characteristic equations. In complex geometry regions, such as flange and grid beam areas, the stiffnesses will be derived from finite element analyses. Nonlinear

couplings will be included between components to account for structural interactions such as those between the fuel and core support plate, and between the core support barrel and upper guide structure upper flanges. Preloads, which are caused by the combined action of applied external forces, dead weights, and holddowns will also be included. Friction elements will be used to simulate the coupling between the fuel rods and spacer grids.

4.6.2.2 Reduced Internals Models

A reduced model of the reactor vessel internals will be developed for incorporation into the reactor coolant system model. The detailed nonlinear horizontal and vertical internals (plus core) models will be condensed and combined into a three-dimensional model compatible with the reactor coolant system model and the computer programs through which the latter model is analyzed. The purpose for this reduced internals model is to account for the effects on the reactor vessel support motion and the structural loading interaction between the internals and the vessel. The reduced internals model will be developed so as to produce reactor vessel support motions and loadings equivalent to those produced by the detailed internals models.

4.6.2.3 Core Support Barrel Shell Models

To evaluate the core support barrel under pipe rupture conditions, separate finite-element models will be formulated for hot and cold leg breaks. For the cold leg break, a dynamic response analysis will be performed because of the asymmetric nature of the applied loading. In contrast, the hot leg breaks result in uniform, axially varying compressive loads on the barrel and the responses will be determined with a buckling or instability analysis.

4.6.2.3.1 Shell Vibration Model

A cold leg break will cause pressure transients on the core support barrel which vary both circumferentially and axially. To evaluate the shell response of the CSB to these pressure transients, the ASHSD finite element computer code will be used. The CSB is modeled as a series of short shell elements joined at their nodal point circles. The length of the elements in the model will be made to be a fraction of the shell attenuation length. At areas of structural discontinuity, where rapid changes in the stress pattern will occur, the nodal points will be more closely spaced.

4.6.2.3.2 Shell Dynamic Stability Model

The buckling potential of the core support barrel for the hot leg break condition is evaluated using the finite element computer code, SAMMSOR-DYNASOR. The hot leg break causes a net external radial pressure on the core support barrel which could result in significant inward radial deformations and dynamic buckling.

The CSB will be modeled as a series of shell elements and will include representations of the upper and lower CSB flanges. The stiffness and mass matrices for the shell will be generated utilizing the SAMMSOR part of the code. The equatings of motion of the shell will be solved in DYNASOR using the Houbolt numerical procedure.

The initial imperfections to be applied to the CSB will be based on actual measurements of the CSB. These imperfections will be applied to the CSB in the analysis by means of pseudoloads for each circumferential harmonic considered. The actual pressure transient loading generated by the outlet break is assumed to be uniform circumferentially and to vary linearly in the longitudinal direction.

4.6.3 INTERNALS RESPONSE ANALYSIS

The dynamic responses of the reactor internals to the postulated pipe breaks in the reactor cavity will be determined utilizing the models described in the previous sections. Horizontal and vertical analyses will be performed for both hot and cold leg breaks to determine the lateral and axial beam mode responses of the internals to the simultaneous internal fluid forces and vessel motion excitation. Shell response analyses of the core support barrels will also be performed to obtain the total (beam and shell mode) response of these components.

4.6.3.1 Inlet (Cold Leg) Break Analysis

The postulated cold leg break results in simultaneous vertical and horizontal beam and shell mode excitation. The responses to these excitations will be calculated separately as described below. The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary plant specific verifications and/or analyses will be performed.

4.6.3.1.1 Horizontal Response

The dynamic time history responses of the reactor internals to the horizontal loads resulting from the cold leg break will be determined with the CESHOCK code. The input to this analysis is the core support barrel force time history and the vessel motion time history determined from the reactor coolant systems analysis. The core support barrel force time history will be obtained by representing the asymmetric pressure distribution time history as a Fourier expansion. The two terms (sin θ,cos θ) which excite the beam mode of vibration will then be integrated over the core support barrel and transformed into nodal force time histories.

The results from the analysis will consist of time dependent member forces, and nodal displacements, velocities and accelerations. The load response will be compared to previously calculated acceptable loads. The core plate motion time history will be used as input to the detailed model of a row of fuel assemblies.

4.6.3.1.2 Vertical Response

The vertical excitation of the internals will be calculated by the LOAD computer code, using a control volume method of analysis. In this method, the reactor internals are subsectioned and enclosed within volumes of fluid. The momentum equation is then applied to each volume, and a resultant force is calculated, which is distributed over the structural nodes within the volume. This method takes into consideration pressure, fluid friction, momentum changes, and gravitational forces acting on each volume. The resulting load time histories are in a form consistent for CESHOCK code input.

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

4.6.3.1.3 Shell Vibration Response

The asymmetric pressure transient caused by a cold leg break will be represented as a Fourier expansion at either 4 or 6 axial locations. Between these locations, the pressure is assumed to vary linearly. Thus,

a complete spatial load distribution on the barrel, which is compatible with the ASHSD computer program is derived. In ASHSD, each load harmonic is applied separately and the barrel response is calculated. The results for each harmonic will be then summed to obtain the total response. The ASHSD output will consist of nodal point displacements and resultant shell forces as functions of time.

4.6.3.2 Outlet (Hot Leg) Break Analysis

The analyses to be performed for the hot leg break, horizontal and vertical nonlinear beam response and core support barrel shell response, are described in the following subsections. The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary plant specific verifications and/or analyses will be performed.

4.6.3.2.1 Horizontal Response

The dynamic time history responses of the reactor internals to the horizontal loads resulting from the hot leg break will be determined with the CESHOCK code. The input to this analysis will be the CEA shroud crossflow load time histories and the vessel motion time history determined from the reactor coolant system analysis. The force time histories applied to the shroud mass points will be determined directly from the blowdown pressure time history and will include the drag force and forces due to the pressure differential on the shrouds.

The results from the analysis will consist of time dependent member forces, and nodal displacements, velocities and accelerations. The load response will be compared to previously calculated acceptable loads. The core plate motion time history will be used as input to the detailed model of a row of fuel assemblies.

4.6.3.2.2 Vertical Response

The method used in Section 4.6.3.1.2 will be applicable to both hot and cold leg breaks.

4.5.3.2.3 Shell Dynamic Stability Response

The response of the core support barrel to the inward radial pressure pulse caused by

a hot leg break will be determined with the DYNASOR code. The input to this analysis is the radial imperfection measured in the constructed barrel and the dynamically varying radial pressure distribution. The circumferentially uniform pressure loading will be computed from the blowdown pressure time history.

4.6.4 ACCEPTANCE CRITERIA FOR INTERNALS

The function of the internals following a postulated pipe rupture is to maintain the internals in a coolable configuration.

This is satisfied if either of the following criteria is met:

a. The stress limits are within the limits of the ASME Code, Section III, Division I, Appendix F.

b. The component deflections are limited so that the core is held in place and adequate core cooling is preserved.

4.6.5 EVALUATION OF INTERNALS

The maximum loads resulting from the postulated pipe ruptures will be determined utilizing the lateral and vertical postulated pipe ruptures time-dependent loadings in the structural analysis. These loads will be compared to previously determined acceptable loads on the internals. If required, the deformations of the internals will be compared to allowable deformations to demonstrate adequate core cooling. The results of this generic analysis will be related to specific plants as applicable in either full or scaled form. Where necessary, plant specific verifications and/or analyses will be performed.

4.7 FUEL

4.7.1 FUEL DESIGN DESCRIPTION

The fuel assemblies consist of square arrays of fuel rods, with each fuel rod being a zircaloy tube containing the fuel in the form of cylindrical UO_2 pellets. The spacing of the fuel rod array is maintained by spacer grids arranged at intervals along the fuel rods.

The axial positioning of the spacer grids is maintained by welding them to the zircaloy guide tubes which are the axial structural members of the assembly.

The guide rods, in turn, are bolted to the upper and lower end fittings of the assembly, these end fittings serving as the structural interface between the fuel assembly and the reactor internals, as well as providing for axial restraint of the fuel rods.

4.7.2 FUEL TESTING

Structural testing of fuel assemblies and fuel assembly components will support the detailed internals and fuel analyses. The tests described below will provide data on structural characteristics to be used as input to the dynamic analyses and for evaluation of the fuel assemblies.

4.7.2.1 Static Load Deflection Tests

A lateral load deflection test will provide data for development of a dynamic horizontal fuel assembly model with the correct values of natural frequency and stiffness. The full size fuel assembly will be supported by simulated reactor end conditions and a lateral load will be applied at the most central spacer grid. Load deflection characteristics will be determined as well as the hysteretic response of the fuel.

Ancillary to this test will be measurement of the rotational stiffness of the guide tube - spacer grid joint. This data will be used in the RUMBLE code for detailed fuel assembly stress analysis.

4.7.2.2 Fuel Bundle Impact Tests

A lateral fuel bundle impact test will provide data on spacer grid impact stiffness, fuel assembly coefficient of restitution, and spacer grid fluid-structure interaction. A full size fuel assembly will be supported by simulated reactor end conditions and laterally deflected at the most central spacer grid. The bundle will be released and allowed to impact a simulated core shroud section. This test will be performed in both air and water environments.

4.7.2.3 Spacer Grid Impact Tests

Spacer grid impact tests will provide data on spacer grid dynamic characteristics and the ultimate structural capabilities of the grid. The tests will be conducted in two (2) phases; a pendulum impact test and a drop impact test.

In the pendulum test a regidly mounted grid will be impacted by a swinging pendulum. This will simulate grid behavior in a through-grid loading such as is experienced when a bundle is impacted simultaneously from both sides. In the drop test, a fuel assembly section (one spacer grid with its accompanying proportionate mass of fuel rods) will be dropped onto a simulated core shroud. This will simulate grid behavior in a one-side loading such as is experienced when a bundle impacts the core shroud. The drop test will be performed in air and water environments.

4.7.3 FUEL ANALYSIS MODEL

Detailed structural lumped mass-spring models of the fuel assemblies will be developed based on correlation with static and dynamic test results. These models will be analyzed for both hot and cold leg breaks with the CESHOCK code. The resulting component loads and displacements will be used to determine fuel assembly stresses and spacer grid impact loads. A beam column model of a fuel assembly will also be developed for use in an analysis of simultaneous compressional and lateral loading.

4.7.3.1 Horizontal Model

Horizontal fuel assembly models will be developed from correlation with static stiffness, forced vibration, and pluck tests of a full size assembly and impact tests of spacer grids. This model will be used to represent the entire core in the detailed internals model and a single row of fuel assemblies in the fuel analysis model. The detailed fuel model will have nonlinear elements to represent the impact between adjacent fuel assemblies and the core shroud. In the dynamic analyses, fuel alignment plate and core support plate motions obtained from the detailed internals models with all pipe rupture loads and vessel motion considered, will be applied to the fuel models.

4.7.3.2 Vertical Model

A vertical fuel assembly model will be developed from correlation with static stiffness and drop tests and will be also included in the detailed internals model. The model will include nonlinear couplings to represent the end fittings and it will be subjected to LOCA and drag loadings. Friction elements will also be used to represent stick-slip motion between the fuel rods and spacer grids.

4.7.3.3 Dynamic Beam-Column Model

A detailed mathematical model of the fuel assembly will be developed based on correlation with static and dynamic test results. To accurately model lateral and axial structural properties, correlation with static stiffness, forced vibration, pluck lateral impact, will be made. The beam-column fuel assembly model will be used in a dynamic response/stability analysis of simultaneous axial and lateral loading. A detailed description of modeling techniques is found in Reference 3.11.

4.7.4 FUEL RESPONSE ANALYSIS

The detailed core model consisting of a complete row of fuel assemblies across the core will be analyzed for both hot and cold leg breaks. These analyses will utilize the CESHOCK code and the results will include the component load and displacement responses which will be used for the stress evaluation of the fuel assembly and the spacer grid impact loads. The reactor will be analyzed with CE 14x14 fuel. The analysis will be extended to include other fuel

in the event of substitution or changes in the array. The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary plant specific verifications and/or analyses will be performed.

4.7.4.1 Inlet Break Analysis

Vertical and lateral response analyses for a cold leg break will be performed using the detailed fuel model. Input to these analyses will be the vertical and lateral time histories of the fuel alignment plate and core support plate which will be obtained from the detailed internals model analyses with the reactor vessel motion included. Load and displacement responses from the fuel analysis models will be used in the detailed stress evaluation of the fuel and spacer grid impact loads.

4.7.4.2 Outlet Break Analysis

Vertical and lateral response analyses for a hot leg break will also be performed using the detailed fuel model. In this anlaysis, the model will be subjected to core support plate and fuel alignment plate motions obtained from the detailed internals models with reactor vessel motion included. The resulting responses of the fuel will be used to compute stresses and spacer grid impact loads.

4.7.4.3 Dynamic Beam-Column Analysis

The dynamic beam-column analysis will determine additional bending stress and stability of the fuel assembly due to concurrent lateral and axial loading. The response will include interaction with other structural components in the horizontal and vertical directions.

Initial conditions will correspond to the time at which the most severe combination of lateral deflection and axial compression exist in the fuel assembly as ascertained in 4.7.4.1 and 4.7.4.2. At this point in time the simultaneous horizontal and vertical loading histories (as obtained in 4.7.4.1 and 4.7.4.2) will be applied to the fuel alignment and core support plate nodes at the ends of the fuel assembly model. In addition the lateral loading histories of the nodes representing the adjacent fuel assembly and/or core shroud will be applied to account for structural interaction. The analysis will utilize the ANSYS code and the basic procedures established in Ref. 3.11.

4.7.5 FAULTED CONDITION CRITERIA FOR FUEL ASSEMBLY EVALUATION

The basic functional requirement which must be satisfied by the fuel assembly is that the structural components of the assembly (end fittings, guide tubes and grids) must be capable of maintaining the fuel rods in a coolable array when subjected to the mechanical loads predicted to result from the pipe rupture.

4.7.6 EVALUATION OF FUEL

As discussed in Section 4.7.3, the pipe rupture response analysis model produces results in the form of spacer grid impact load, axial and lateral loads on the fuel end fittings, and lateral deflected shapes for the fuel assembly. The spacer grids are the primary component in assuring the fuel rods remain in a coolable array. The capability of the grids to perform this function will be determined by comparing the predicted lateral impact load to the lateral load capability of the spacer grid, which is determined by test.

Previous analyses have shown that other fuel assembly components, e.g., the guide tubes, were acceptable. A comparison will be made between the postulated pipe rupture loads determined for this asymmetric load evaluation with the results of previous analyses to demonstrate acceptability. More detailed stress analyses will be performed if required. The analyses would be performed in the following manner.

Axial and lateral end fitting loads are applied (analytically) to the appropriate portions of the upper and lower end fittings, and the resultant stresses are calculated using relatively basic strength of materials methods. The comparatively simple geometry of these components does not require complex analysis methods to produce sufficiently accurate results.

Lateral deflection shapes are input directly into a computer model (RUMBLE, described in Reference 3.11) which then makes use of these shapes to calculate the resultant stress intensities in the axial structural members of the fuel assembly and in the fuel rods. Since the fuel assembly response to the postulated pipe rupture is a dynamic phenomenon, with an essentially infinite number of axial shapes which could be analyzed, it is necessary to be selective in establishing which shapes are likely to correspond to maximum stress conditions. This is accomplished by evaluating all shapes which correspond to either maximum deflection or maximum apparent curvature at each of the axial modes in the pipe rupture response model.

The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific evaluations and/or analyses will be performed.

The final evaluation criterion is the capability of the fuel assembly to maintain the fuel rods in a coolable array.

4.8 CONTROL ELEMENT DRIVE MECHANISMS (CEDMs)

4.8.1 DESIGN BASIS

The capability of the control element drive mechanisms (CEDMs) to withstand the effects of design basis pipe breaks will be evaluated by analysis. The effect of each break is experienced by the CEDM through pressure changes and motion of the reactor coolant system structural analysis described in Section 4.5.

4.8.2 METHODS OF ANALYSIS

The CEDMs will be analyzed by traditional dynamic elastic analysis and evaluated according to appropriate elastic stress limits for faulted conditions. If these limits are not satisfied, a detailed elastic plastic analysis will be performed.

4.8.2.1 Dynamic Analysis

An elastic dynamic analysis of the CEDMs will be performed using a framed structure analysis technique and the appropriate reactor vessel motion history. A sufficient number of CEDMs at various locations on the head will be analyzed in order to determine the most severely loaded one. This analysis will be performed by either the STRUDL II (Reference 3.2) or the SAP IV (Reference 3.3) program.

4.8.2.2 Detailed Analysis

If the stresses computed by the elastic dynamic analysis exceed the appropriate limits, an elastic plastic dynamic finite element analysis of the most severely loaded CEDM will be performed. This analysis will be performed by the MARC (Reference 3.10) computer program.

The material properties for stainless steel and inconel used in the analysis will be determined from laboratory tests already performed.

4.8.3 RESULTS OF ANALYSIS

The results of the elastic dynamic analyses will be stresses at the critical regions of the CEDMs as a function of time during the LOCA event. The results of each detailed elastic plastic analysis will be stress, strain and deformation history of the most severely loaded CEDM for each design basis pipe break in the reactor cavity. The loadings on the vessel head to shaft weld and the contact region at

the outside surface of the head will also be included. The results of these generic analyses will be related to specific plants in either full or scaled form. Where necessary, plant specific verifications or analyses will be performed.

4.8.4 EVALUATION OF CONTROL ELEMENT DRIVE MECHANISMS

4.8.4.1 Acceptance Criteria

The CEDMs are not required to operate for safe shutdown after a loss of coolant event resulting from the design basis pipe breaks. However the evaluations will be performed to show that integrity of the CEDM pressure boundary is maintained, i.e., that a leakage does not occur. The integrity of the pressure boundary is assumed if the applied loads do not exceed 70% of the plastic instability load. An alternate criterion based on strain limits in which a suitably conservative strain limit must be justified may also be employed.

4.8.4.2 Evaluation

The elastic dynamic stress results will be compared to the elastic faulted limits of the ASME Code.

In the event that the stress limits are not satisfied, the elastic plastic analysis of Paragraph 4.8.2.2 will be performed. The results of this analysis will be compared to the instability analysis of Paragraph 4.5.3.3.

The integrity of the CEDMs will be evaluated by comparing the computed elastic plastic behavior to the instability load according to the acceptance criteria.

4.9 EMERGENCY CORE COOLING SYSTEM (ECCS) PIPING

4.9.1 DESIGN BASIS

The capability of the emergency core cooling system (ECCS) piping that is attached to the intact primary coolant piping to withstand the effects of the design basis pipe breaks will be evaluated by analysis. The effect of each break is experienced by the ECCS piping through pressure changes and motion of the ECCS nozzles on the discharge leg piping. The motions of the nozzles will be computed by the reactor coolant system structural analysis described in Section 4.5.

4.9.2 METHODS OF ANALYSIS

The ECCS piping will be analyzed by traditional dynamic elastic analysis and evaluated according to appropriate elastic stress limits for ASME Level B conditions. If these limits are not satisfied, a detailed elastic plastic analysis to demonstrate functionability of the piping will be performed.

4.9.2.1 Dynamic Analysis

An elastic dynamic analysis of the ECCS piping will be performed using lumped parameter models and the appropriate ECCS nozzle motion history. The physical definition of the piping will be represented in STRUDL (Reference 3.2) or SAP IV (Reference 3.3) to generate the appropriate condensed stiffness matrix for the structure. The condensed matrix, mass, and definition of gapped supports, if any, will be supplied in the DAGS (Reference 3.4) program. The DAGS program will determine the motion history of the ECCS pipe by performing a non-linear time history analysis.

4.9.2.2 Detailed Analysis

If the stresses computed by the elastic dynamic analysis exceed the prescribed limits an elastic plastic finite element analysis will be performed for each region where significant plasticity is expected. This analysis will be similar to the instability analyses discussed in Paragraph 4.5.3 and will serve two purposes. First the analysis will determine a load displacement curve for the plastic region to be incorporated into the dynamic analysis, and second it will provide details of the deformation necessary to evaluate the functionability of the plastic region. The elastic plastic analysis will be performed by the MARC (Reference 3.10) computer program. The material properties

for the piping material used in the analysis will be determined from the extensive collection of piping data available.

4.9.3 RESULTS OF ANALYSIS

The results of elastic dynamic analyses will be stresses at the critical sections of the ECCS piping as a function of time during the postulated pipe ruptures event. The results of the dynamic analysis which considers the plastic behavior of some regions will be the load-displacement time history during the event. This load-displacement history relates to the detailed model for the corresponding local pipe deformation at the plastic regions. The results of this generic analysis will be related to specific plants in either full or scaled form. Where necessary, plant specific verifications or analyses will be performed.

4.9.4 EVALUATION OF ECCS PIPING

4.9.4.1 Acceptance Criteria

The integrity and functionability of the ECCS piping must be demonstrated. Integrity and functionability are assured if the Level B (upset condition) limits of the ASME Boiler and Pressure Vessel Code Section III, Division 1, are not exceeded. If the Level B limits are exceeded, then Level D or faulted limits may be used to demonstrate the integrity is maintained. Functionability may be assured by demonstrating that the deformations of the piping are acceptable.

4.9.4.2 Evaluation

The elastic dynamic stress results will be compared to the Level B stress limits of the ASME Code.

In the event that the stress limits are not satisfied, the elastic plastic analysis of Paragraph 4.9.2.2 will be performed. The result of the rerunning of the dynamic analysis considering plasticity will be deformation history of the pipe cross sections at the plastic regions.

The integrity of the piping will be evaluated by comparing the computed load to the instability load. The functionability will be evaluated by comparing the pipe section deformation to the deformation required for significant flow restriction.

4.10 BIOLOGICAL SHIELD WALL

(Later)

APPENDIX A

SELECTION OF A FLOW MULTIPLIER FOR MASS AND ENERGY RELEASE CALCULATIONS

Recent blowdown experiments performed by various tests (References A.1, A.2 and A.3) have indicated that use of a combination critical flow correlation predicting the blowdown mass release rates is required. This is due to differences in the flow process when the fluid stagnation conditions are subcooled and saturated. All of these test data demonstrate the influence of some degree of non-equilibrium between the phases for subcooled and very low quality fluid conditions. But for the remainder of the saturated regime only equilibrium state prevails.

The combined Henry/Fauske and Moody correlation described in Reference 3.7 reflects these influences by the assumptions used in its derivation. However, the test data mentioned above has shown that the mass flow rate from the vessel through a short length of pipe is over-estimated by the combined Henry/Fauske and Moody correlation throughout the whole blowdown period.

For the actual reactor system following a postulated pipe rupture this over-estimation in the subcooled and low quality saturated blowdown may be amplified since the existence of upstream geometry may enhance the phase mass and heat transfer, and bubble formation processes. Upon reaching the throat, the phases are closer to equilibrium than existed in experiments which had no upstream geometry. In the saturated blowdown, a much less appreciable slip value, expected in the reactor system, may result in lower flow rate than the prediction of Moody's theory. Furthermore, blowdown tests performed by Sozzi and Sutherland (Reference A.4) have revealed that critical flow rate decreases with increased throat diameter regardless of flow regime. Non-ideal nozzle shapes of ruptured geometry along with a larger break area in the postulated ruptures will result in further decreasing the blowdown rate.

Comparisons of pressure vessel fluid pressure data from LOFT-Test L1-2 (Reference A.2) with CEFLASH-4 show that CEFLASH-4 pressures during the early phase of blowdown agree well with LOFT measurements when the Henry/Fauske correlation in conjunction with a 0.7 flow reducing multiplier was used. Reported values of the Moody flow multiplier for a large number of published saturated blowdown experiments are summarized in Reference A.5. The Moody multiplier has always been found to be less than unity. The fact that it is approximately equal to 0.7 is probably due to the over-estimation of phase slip ratio.

Based on CEFLASH-4 verification against available data and realistic assessment of break flow rates in the reactor system, the combined Henry/Fauske and Moody correlation with a flow multiplier of 0.7 provides a reasonable prediction of critical flow rate from subcooled and saturated fluid stagnation state.

REFERENCES:

A.1 Hall, D.G., "A Study of Critical Flow Prediction for Semi-scale MOD-1 Loss-of-Coolant Accident Experiments," Tree-Nureg-1006, December, 1976.

A.2 Robinson, H.C., "Experiment Data Report for LOFT Non-Nuclear Test L1-2," Tree-Nureg-1026, January, 1977.

A.3 Hutcherson, M.N., "Contribution to the Theory of Two-Phase Blow-Down Phenomenon," ANL/RAS 75-42, November, 1975.

A.4 Sozzi, G.L., and Sutherland, W.A., "Critical Flow of Saturated and Subcooled Water at High Pressure," G.E. Report NEDO-13418, July, 1975.

A.5 Ardron, K.H. and Furness, R.A., "A Study of the Critical Flow Models Used in Reactor Blowdown Analysis," Nuclear Engineering and Design 39, 1976, P 257-266.

APPENDIX B

ASSESSMENT OF COMBINING LOCA AND SEISMIC LOADS FOR EVALUATION INVOLVING ASYMMETRIC LOADINGS

It is planned to not combine seismic loads with LOCA loads for the Asymmetric LOCA load evaluation. The basis for not combining these loads for this unique study is as follows:

a. Likelihood of a Simultaneous LOCA and Seismic Event

The likelihood of either a LOCA or a seismic event is very low. For example, in the NRC letter to PWR owners dated January 25, 1978, NRC estimated a LOCA probability in the range of 10^{-4} to 10^{-6} per reactor year. We consider that the NRC value of 10^{-6} is conservative for instantaneous failure of the large pipes at the particular locations which could cause asymmetric loads. The probability of a safe shutdown earthquake (SSE) is also low, for example on the order of 10^{-3} per reactor year or less. Since the asymmetric LOCA loads persist at high levels for less than one second, and since high seismic loads can be assumed to persist less than about 100 seconds, simultaneous seismic and LOCA events are of concern only during a 100 second time interval in any single year. Accordingly, as shown in Table B1, it is estimated that the probability of having a simultaneous asymmetric loads type LOCA and seismic event can reasonably be estimated at less than 3 X 10^{-15} per reactor year.

As indicated in the NRC letter of January 25, 1978, NRC has excluded events with a probability of 10^{-6} to 10^{-7} per reactor year from consideration as design basis events. Since the probability of a simultaneous LOCA and SSE can be estimated at less than 3 X 10^{-15} for the asymmetric LOCA load case, it is considered that combining seismic and LOCA loads is not warranted for this particular evaluation.

b. Likelihood of a Seismic Event Initiating a LOCA

For the particular pipe breaks involved in this unique study, the total pipe stresses under seismic conditions are no higher than the stresses in these pipes under normal plant transients. Accordingly, it is considered that the probability of a seismic event causing a LOCA of the type necessary to produce asymmetric LOCA loads is negligible, i.e., less than 10^{-6}.

TABLE B1

Event	Probability/Reactor-Year
1. LOCA which results in large Asymmetric Loads	$<10^{-6}$
2. SSE	$<10^{-3}$
3. LOCA and SSE in same 100 second period	$<10^{-6} \times 10^{-3} \times \frac{100 \text{ seconds}}{3.2 \times 10^{7} \text{ seconds/year}}$
	$= <3 \times 10^{-15}$ per reactor-year