DYNAMIC ANALYSIS OF REACTOR COOLANT SYSTEMS UNDER LOCA CONDITIONS

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SUMMARY

This paper presents a rigorous analysis of a pressurized water reactor coolant system subjected to a postulated break in the primary coolant piping. The procedures described include structural modeling and analytical techniques for a non-linear time history dynamic analysis of a three dimensional coupled model of the reactor coolant system including details of the reactor internals, pressure vessel, supports and piping. The dynamic analysis is performed to determine the response of the reactor coolant system supports to the simultaneous effects of pipe break thrust and external and internal horizontal and vertical asymmetric pressure loads applied to the reactor vessel and internals as a consequence of the postulated pipe rupture.

Condensed structural models are created from highly detailed representations of each component by maintaining response frequency characteristics and interface response compatibility. The location, type and size of the pipe breaks have been determined by stress survey and mechanistic break area criteria. Examples of internal and external forcing functions on the vessel and internals, as well as their separate and combined effects on the reactor vessel supports are illustrated.

The pressurized water reactor internals, including the fuel and supporting structures, are suspended from the closure flange region of the reactor vessel and surrounded by a cylindrical "core support barrel" (CSB). The vessel in turn is surrounded by the biological shield wall. The CSB and reactor vessel are essentially concentric cylinders throughout the length of the CSB.

Upon postulation of a rupture in a primary coolant pipe, several rapidly occuring events cause internal and external transient loads to act upon the reactor vessel. For a reactor vessel inlet break, decompression takes place in the annulus between the CSB and vessel on the side of the vessel nearest the pipe break before pressure on the opposite side changes. The difference in pressure across the CSB induces loads in opposite directions on the CSB and the reactor vessel. Simultaneously, as fluid escapes through the break, the annulus between the reactor vessel and biological shield wall becomes asymmetrically pressurized. The difference in pressure across the reactor vessel causes additional horizontal and vertical loads on the vessel. In addition, the vessel is loaded by the effects of initial tension release and blowdown thrust on the broken pipe. For a reactor vessel outlet break, the internal loads are caused by vertical flow and asymmetric axial decompression of the vessel.

The hydraulic loads applied internal to the vessel are determined by use of qualified thermo hydraulic analysis codes such as those of WHAM, FLASH or RELAP series. These loads, which are provided as input to the analysis presented here, are a function only of the type and size of the reactor coolant system. The external cavity pressure loads are determined by similar analyses but are in part dependent on the configuration of the biological shield wall.

The dynamic analysis of the mathematical models subjected to these loads is performed using the STRUDL code to define the characteristics of the structure and the CE program DAGS to calculate the non-linear time history response. The non-linearities in the model arise from the presence of gaps in a variety of orientations at reactor vessel supports, between reactor internals components and at vessel-to-internals interfaces. Where necessary, the geometric non-linearities include the plastic action of energy absorbing material. These effects, plus the phenomenon of hydrodynamic coupling between concentric components, are rigorously accounted for in the DAGS analysis of the combined system. A single analysis of the three dimensional system simultaneously loaded in three directions is performed for each defined break.

Results of these analyses include maximum vessel motions and maximum values and time histories of vessel support reactions.

1. Introduction

This paper presents a rigorous analysis of a pressurized water reactor coolant system to determine support loads due to a postulated break in the primary coolant piping. The procedures described include structural modeling and analytical techniques for a non-linear time history dynamic structural analysis of a three-dimensional model of the reactor coolant system including coupled details of the reactor internals, pressure vessel, supports and piping. The dynamic analysis is performed to determine the reactions of the reactor vessel supports to the simultaneous effects of pipe thrust of external and internal horizontal and vertical asymmetric pressure loads applied to the reactor vessel and internals as a consequence of the postulated pipe rupture.

2. System Description

The reactor internals, including the fuel and supporting structures, are suspended from the closure head flange region of the reactor vessel and are surrounded by the cylindrical "core support barrel" (CSB) as shown in Figure 1. The vessel in turn is surrounded by the biological shield wall, which forms the reactor vessel cavity.

The reactor internals are designed to support the reactor core fuel assemblies, guide the control element assemblies (CEA), absorb and transmit the dynamic loads and other loads to the reactor vessel flange, provide flow paths for the reactor coolant, and provide guides for in-core instrumentation. Detailed description of the internals components can be found in reference [1]. The major structures, however, are the core support barrel, fuel assemblies, upper guide structure, lower support structure and core shroud. Interfaces with the reactor vessel are at the vessel flange, where the core support barrel and upper guide structure flanges are supported and held down, at the core barrel snubbers and at the core barrel outlet nozzle juncture. The CSB and reactor vessel are essentially concentric cylinders throughout the length of the CSB.

Details of the reactor vessel support arrangement are shown in Figure 2. The vessel is supported on four vertical columns located under the vessel inlet nozzles. A pad welded to each inlet nozzle provides the surface to which the column is bolted. This pad is designed to act as a radial key and to mate with the support structure. Low friction bearings and tangential gaps allow free growth during thermal expansion, while resisting horizontal vessel motion during earthquakes and following a postulated loss of coolant accident (LOCA). At the bottom of each column is a baseplate bolted to the supporting structure. The baseplate acts as a keyway for a horizontal key welded to the vessel lower head. Low friction bearings and tangential gaps are provided to allow radial motion of the key while resisting motions resulting from earthquake or LOCA. In addition, non-linear energy absorbing material is used at the keyway interface to limit the shear load applied to the vessel lower head during a LOCA.

3. Background

Upon postulation of a break in a primary coolant pipe, several rapidly occurring events cause internal and external transient loads to act upon the reactor vessel. For a reactor vessel inlet pipe break, asymmetric pressure changes take place in the annulus between the CSB and the vessel. Decompression occurs on the side of the vessel annulus nearest the pipe break before pressure on the opposite side changes. The momentary difference in pressure across the CSB induces lateral loads in opposite directions on the CSB and the reactor vessel. Vertical loads are also applied to the internals and to the vessel due to vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, as fluid escapes through the break, the annulus between the reactor vessel and biological shield wall becomes asymmetrically pressurized, resulting in a difference in pressure across the reactor vessel causing additional horizontal and vertical external loads on the vessel. In addition, the vessel is loaded by the effects of initial tension release and blowdown thrust at the pipe break.

The loads occur simultaneously and are shown in Figure 3. For a reactor vessel outlet break, the same type of loadings as shown in Figure 3 occur but the internal loads are more predominately vertical due to more rapid decompression of the upper plenum. For each postulated break the time history of reactor vessel support reactions due to the complete set of simultaneous horizontal and vertical loads are calculated.

4. Models

Condensed structural models of the RCS and reactor internals were created from highly detailed representations of each component by maintaining response characteristics and interface response compatibility. The internals model and the RCS model were dynamically condensed* and coupled to form the final model used for the analysis of the vessel supports (See Figures 4 and 5). The condensed model of the internals was used only to represent the effects of the reactor internals on the reactor vessel support reactions and not to analyze the internals themselves. The adequacy of the reduced model of the internals for this purpose was verified by analysis of models of a coupled system having a detailed internals model and simplified representation of the vessel and supports. Figure 6 shows good agreement between the reactor vessel support reactions using the detailed and the reduced internals models.

The three-dimensional non-linear reactor vessel support interfaces described in System Description Section are included in the model shown in Figures 4 and 5. The condensed coupled model contains non-linear interfaces at the core barrel lower supports, the vessel outlet, above and below the fuel assemblies and at the vessel flange where vertical non-linearities occur due to the preload holddown mechanisms for the core barrel and upper guide structure.

The details of the coupled model are summarized in Table I. The model is

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* Masses were lumped to provide a reasonable number of dynamic degrees of freedom while the static or stiffness description of the structure is maintained in high detail.

three-dimensional and has 788 total static degrees of freedom and 61 mass degrees of freedom.

The physical definition of the structure was given as input to the STRUDL [2] computer code that generated the condensed stiffness matrix used in the dynamic analysis. Hydrodynamic effects, including both virtual mass and annular effects, were accounted for in the coupling between the RV and CSB and between the CSB and the core shroud. The resulting mass matrix is non-diagonal because the inclusion of the annular hydrodynamic effects ties the vessel and internals with what can be visualized as "acceleration springs.[3]"

Reactor vessel support stiffnesses of the building structures are included in the model.

5. Forcing Functions

The model shown in Figures 4 and 5 and described in Table I was subjected to the loads that result from a postulated rupture at the reactor vessel nozzle. The size of the breaks considered have been determined based on a dynamic analysis of the reactor coolant piping including motion limiting devices as described in reference [4]. For the reactor inlet guillotine break, the location and geometry of the pump supports and pipe stop determines the break size of 350 in^2. The reactor outlet break is limited to 100 in^2 by the steam generator supports and the inherent stiffness of the hot leg without pipe stops.

For each postulated break, a thermo-hydraulic analysis was performed according to the procedures and models developed in reference [1]. The resulting pressure and flow parameters were used to calculate three-dimensional time history forcing functions acting on the reactor vessel and internals at each dynamic degree of freedom shown in Table I.

The reactor cavity pressure forces are determined by a multi-node subcompartment pressurization analysis using as input the mass energy release data from the above described thermo-hydraulic analysis. Reactor cavity configurations differ from plant to plant but a typical nodalization for an inlet break is shown in Figure 7. One important aspect of the cavity configuration is the presence of the neutron shield plug in the cavity annulus which can greatly affect the character of blowdown flow in certain regions of the cavity. The system presented herein assumed the presence of an annular neutron shield plug immediately below the reactor vessel nozzles which effectively cuts off flow to the lower cavity. This produces a higher pressure acting on a smaller surface area of the vessel but, more importantly, produces an overturning effect on the vessel because the center of pressure on the vessel is above the elevation of the horizontal supports.

6. Nature of the Loads

Figure 8 shows the oscillatory nature of the horizontal loading internal to the vessel and the relative non-oscillatory characteristics of the external forces. The internal forces result from the rapid subcooled decompression waves and reflections inside the vessel, while the cavity pressure is more gradually applied and is always in the same direction. The pipe break thrust force consists of the effects of pipe tension release and hydraulic thrust as outlined in reference [4].

Figure 9 shows plots of individually applied forces in the vertical direction. It can be seen that the vertical force on the vessel and the vertical force on the internals are out of phase throughout the time span of interest, but the magnitudes are different from the beginning. This net load on the vessel is caused by the fact that the inertia of the fluid and resistance of flow provided by the internals structure allows a time lag between pressure differences and flow response. The phasing of loads shown in Figure 9 combined with the overturning effect of the internal and external horizontal loads makes estimation of the maximum vertical support reactions difficult without the complete combined analysis.

7. Response Analysis

The dynamic analysis of the response of the system is performed using the DAGS and FORCE [5] codes to determine the maximum reactions at the vessel support locations shown in Figure 10. The DAGS program performs a closed form solution to the coupled equations of motion rigorously accounting for changes to the stiffness matrix of the structure because of opening and closing of elastic or plastic gaps. The FORCE program uses the displacements of the structure generated by DAGS to produce a time history of the reaction loads at any internal or external point in the structure.

8. Results

Results of maximum reactor vessel support reactions are given in Table II for a 350 in^2 inlet break and a 100 in^2 RV outlet break. It can be seen from Table II that the 350 in^2 inlet break produces the controlling values for all support force and moment components.

8.1 Figure 11 is a simultaneous plot of the reaction at each of the four upper horizontal supports. It can be seen that there are actually two sets of reactions. One set (Supports 1 and 3) is oriented "facing" the load (30^o from perpendicular to the axis of the broken pipe) while the other set (Supports 2 and 4) is oriented parallel to the direction of the loading. Consequently, there is a set of "major" reactions (1 and 3) and a significantly lower set of "minor" reactions (2 and 4). This force relationship tends to provide relief to the support designer because it is the set of one major and one minor reaction force, which is generally used to size each support structure. The reactions calculated individually within each set are almost identical indicating very little torsional motion of the vessel even though the two sets of supports are gapped and not orthogonal.

8.2 Figure 12 shows the vessel upper horizontal support reaction both with and without the internal loads. It can be seen from Figure 12 that the horizontal effects of internal load is essentially superimposed on the external load effect to produce the total effect. Inside the vessel, interface gaps are found to open and close many times. It can be seen from Figure 12 that the total horizontal effect of all loads on the system causes the vessel to traverse the support gaps, impact the supports and remain in contact oscillating about a deflected, closed gap position. The internal loads can be seen to contribute 40% of the total reaction, an amount which is significant but can be designed for.

8.3 Table III is a summary of the maximum axial load in the vertical support column and its time of occurrence for analyses of the system using each forcing function separately and for the analysis where all loads are applied simultaneously. Figure 13 shows plots of the individual vertical reactions for each forcing function. It can be seen from the plots and the corresponding maxima in Table III that the magnitude of the maximum reaction for each individual forcing function is a significant percentage of the actual combined maximum, indicating that each of the forces produce significant results and no one of which can be neglected. Also, the maximum reaction is not caused by any one load but by the shifting phase of each relative to the others. This makes estimation of the combined effects difficult without the combined analysis. Also shown in the table is a comparison of the actual maximum reaction determined by the simultaneous loading and the summation of the individual maxima from separate loadings. The results indicate that because of the oscillatory type of forcing functions, phasing is very important vertically and individual consideration of the loads would be overly conservative.

8.4 The analysis of the 350 in^2 inlet break was performed both with and without hydrodynamic effects. Figure 14 shows the displacement response of a point on the CSB for the 350 in^2 inlet break, both with and without hydrodynamic effects. Figure 15 shows the response of a point on the vessel and Figure 16 shows the resulting support loads. As would be expected, the inclusion of hydrodynamic effects greatly reduces the frequency and magnitudes of core barrel response but has a lesser effect on the more massive reactor vessel. The overall effect on the plot of vessel support reactions is that the high frequency "wrinkles" are smoothed out, the frequency envelope is slightly lowered and the magnitude of the response is significantly reduced.

8.5 Table IV shows the results of analyses performed using, in turn, a large and small gap at the reactor vessel upper horizontal support. Analyses were also done using external and internal loads separately for each gap configuration. Results show that the vessel support reaction due to cavity pressure and pipe break thrust is increased by a large gap, while the effects of the internal hydraulic forces are maximized for a small gap.

The cavity pressure and thrust loads applied directly to the vessel accelerate the vessel through the larger support gap causing a larger impact load than that for a small gap. The internal hydraulic loads however are applied to both the vessel and internals, in opposite directions. For a small support gap, the internal loads on the vessel are transmitted directly to the support through the stiff load path of the vessel, while the more flexible internals take time to deform and traverse internal gaps before load is transmitted to the supports. For the larger support gap the internals loads have time to "catch up" to counteract the vessel load before the vessel support gap is closed.

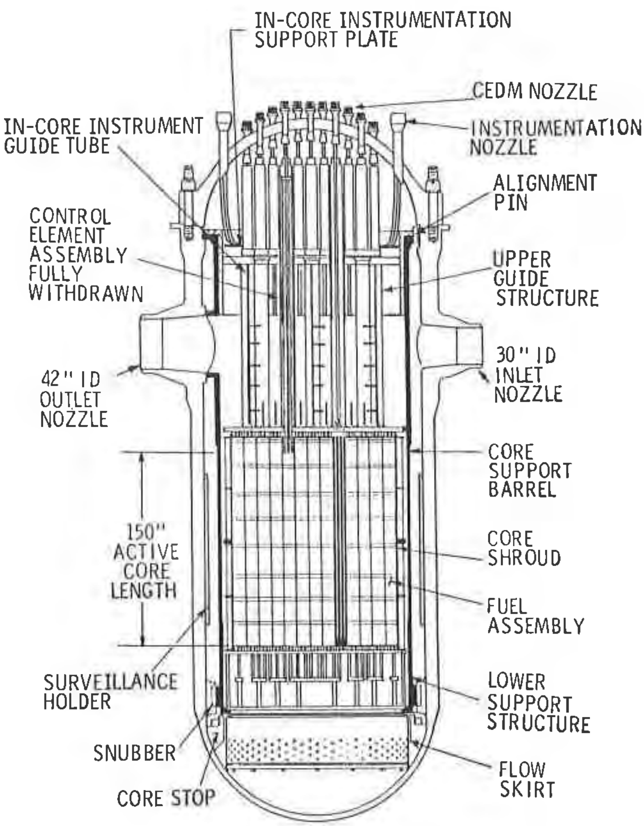
Because of these opposing effects the total effect due to the inclusion of the asymmetric internal hydraulic loads cannot be readily measured by separate analysis. The result of the particular combined analysis shown here is that the larger reaction is produced with the small support gap although the net difference is only 7.2 percent of the support reaction. The result also indicates that as in the case of vertical reactions superposition of the horizontal effects of separate loading conditions can be quite conservative.

9. Conclusions

9.1 The study of the complete effects of LOCA induced loads on the pressurized water reactor vessel supports requires a relatively sophisticated analysis of a non-linear structure subjected to complex three-dimensional spacial loading in order to accurately predict response.

9.2 Loadings inside the vessel oscillate rapidly but resulting vessel support reactions are relatively smooth because of hydrodynamic effects and the filtering effect of the large mass of the vessel. External loadings are rapid but more uniform and predictable.

9.3 Internal asymmetric loads contribute less than 50 percent of the total load on the vessel upper horizontal supports.



1. Reactor vessel arrangement

TABLE I

SUMMARY OF COUPLED MODEL

COMPONENT	MASS POINT NUMBER	DEGREES OF FREEDOM
REACTOR VESSEL	9918 9913	XYZ
	9909 9905 9914 9999	XZ
CORE SUPPORT BARREL	18 9 13	XYZ
	5 14 16	XZ
LOWER SUPPORT STRUCTURE	2	XYZ
FUEL ASSEMBLY	20 24 28	XYZ
	22 26	XZ
CORE SHROUD	6 10	XYZ XZ
UPPER GUIDE STRUCTURE	38 32 29	XYZ
STEAM GENERATORS	500 3500	X

HYDRODYNAMIC COUPLING	ACTIVE DIRECTIONS
9918-18 9999-16 9909-9 9905-5 9914-14 9913-13	XZ

GAP LOCATIONS	DEGREES OF FREEDOM	INITIAL CONDITION
9918-18 9918-38 18-38 28-29 20-2	Y	CLOSED WITH COMPRESSIVE PRELOAD
9914-14 9999-16	XZ	OPEN, GAP = ±

TABLE II

RESULTS OF LOCA ANALYSIS OF REACTOR VESSEL SUPPORTS

MAXIMUM LOCA ABSOLUTE REACTOR VESSEL SUPPORT LOADS

Accident Force Component (Fig. 14)	350 in^2 Inlet Nozzle Guillotine Force - KIPS Moment - Inch - KIPS	100 in^2 Outlet Nozzle Guillotine Force - KIPS Moment - Inch - KIPS
H	5123	2719
Fa	42	12
Fb	2761	914
Fc	84	51
Ma	6449	3872
Mb	2893	1540
Mc	5144	1419

TABLE III

MAXIMUM VERTICAL REACTIONS IN ONE REACTOR VESSEL SUPPORT COLUMN FOR VARIOUS LOADS RESULTING FROM A 350 IN^2 INLET NOZZLE PIPE BREAK

CASE	LOADING	F_{MAX} (KIPS)	TIME (SECONDS)
1	Internal vertical loads on vessel	1800	0.055
2	Vertical loads on internals	1850	0.123
3	Internal horizontal loads (internals and vessel)	1450	0.160
4	External horizontal loads	1000	0.040
5	(Cases 2, 3, 4, 5)	6100	-
6	Individual reactions Time of combined max. at t = 0.091 sec.	2345	0.091
7	Combined internal and external loads	2760	0.091

TABLE IV

MAXIMUM UPPER HORIZONTAL SUPPORT REACTIONS

Loading	Maximum Upper Horizontal Support Reaction (KIPS) Gap at Upper Horizontal Support (In.)	
	0.001	0.050
Internal hydraulic forces only	3038	1297
External cavity pressure forces & pipe thrust only	3200	3638
Combined analysis	5123	4755

References

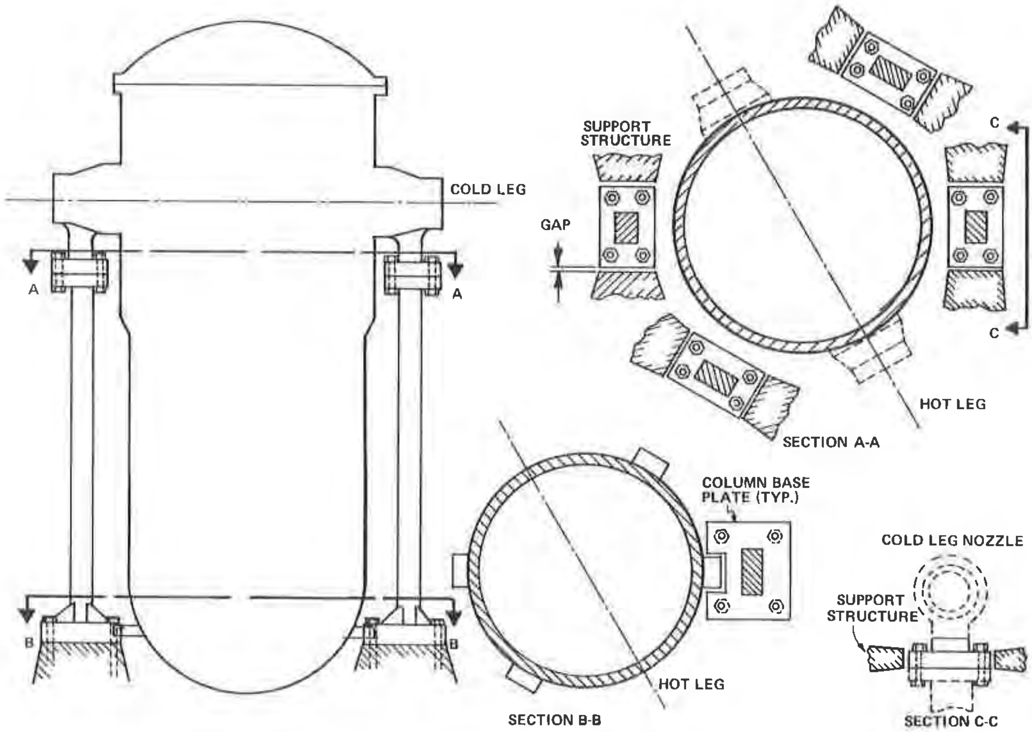
[1] CENPD-42, "Combustion Engineering Topical Report on Dynamic Analysis of Reactor Vessel Internals Under LOCA Conditions."

[2] ICES-STRUDL II, The Structural Design Language, Engineering Users Manual, First Edition, November 1968, MIT.

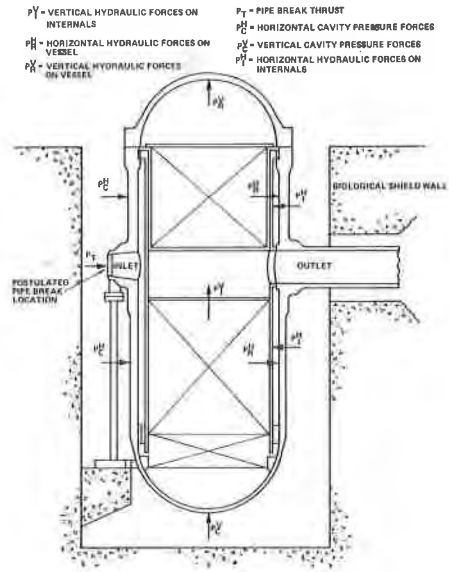
[3] FRITZ, R.J., and KISS, E., "The Vibration Response of a Cantilevered Cylinder Surrounded by an Annular Fluid," KAPL-M-6539, Knolls Atomic Power Laboratory, Schenectady, N.Y.

[4] CENPD-168 Rev. 1, "Combustion Engineering Topical Report on Design Basis Pipe Breaks."

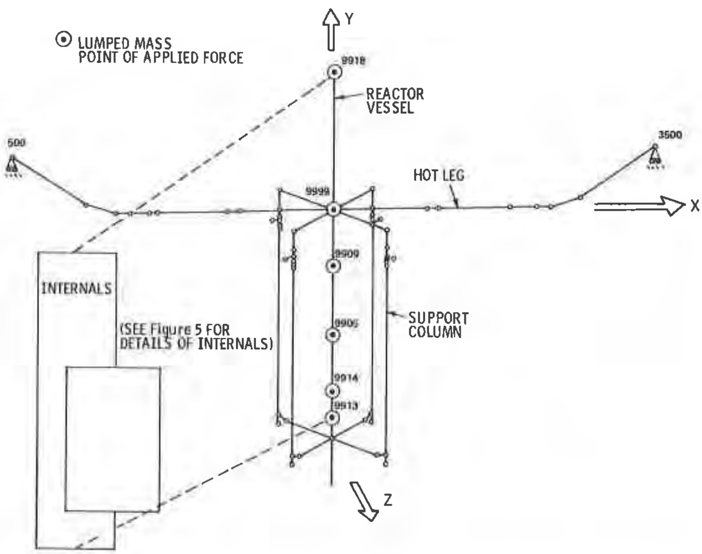
[5] LIEN, J.S.Y., KASSAWARA, R.P., and SMITH, H.B., "Dynamic Analysis of Piecewise Linear Structures," ASCE Second Specialty Conference, New Orleans, La., 1975.



2 Reactor vessel LOCA supports

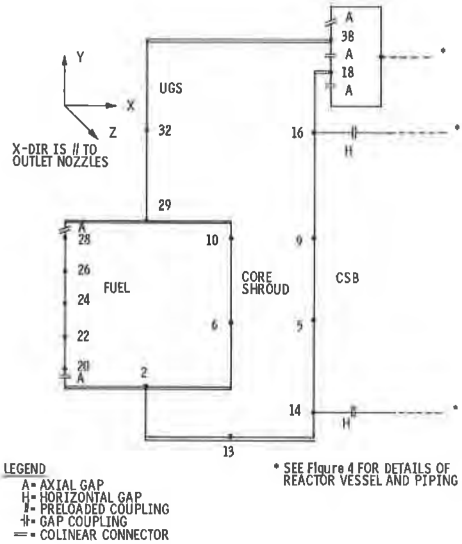


3. Reactor vessel forces following LOCA

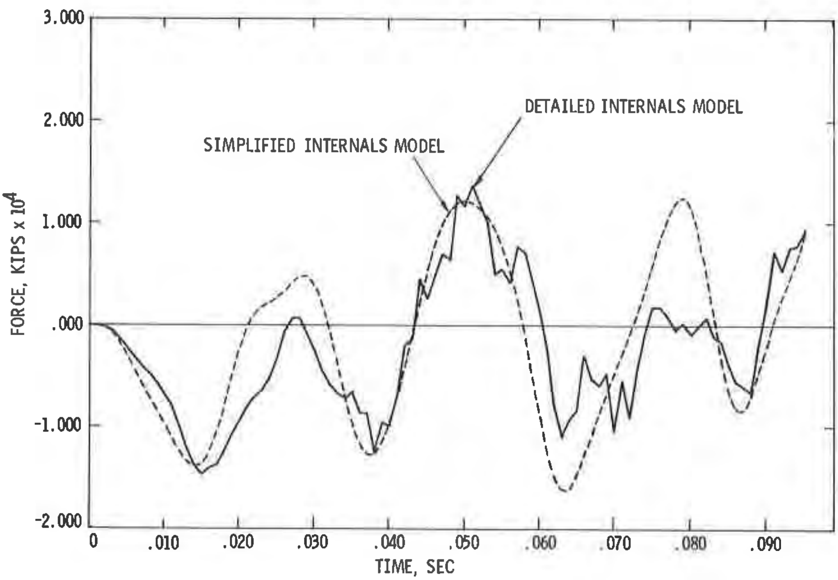


4. Reduced model of reactor coolant system

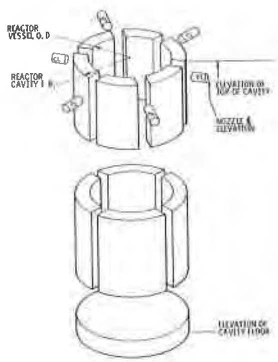
F 4/1*



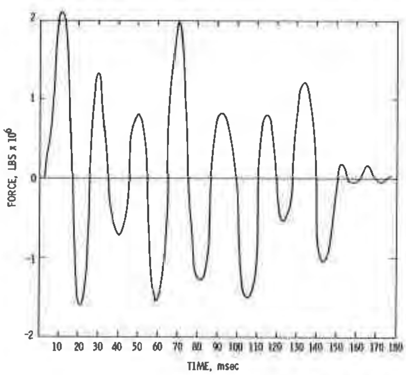
5. Reduced model of reactor internals



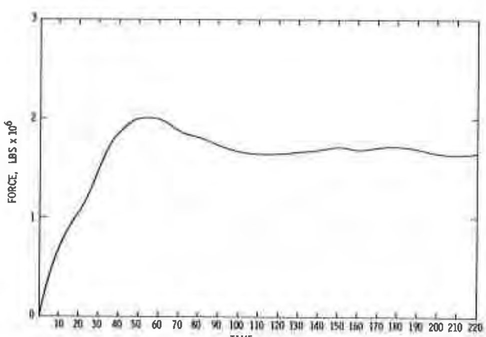
Reactor vessel support reactions - detailed and simplified internals models



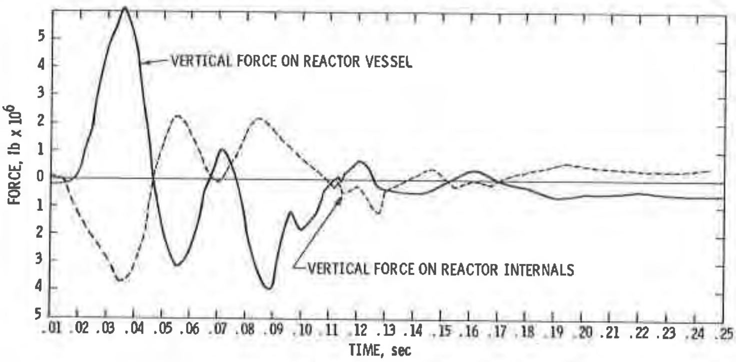
7. Typical reactor cavity subcompartment nodalization



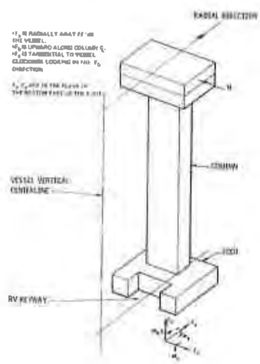
8a. Core support barrel horizontal forcing function



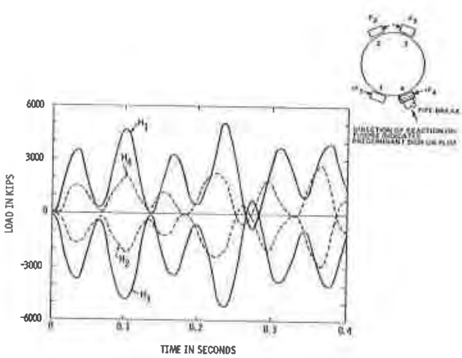
8b. Cavity pressure forcing function



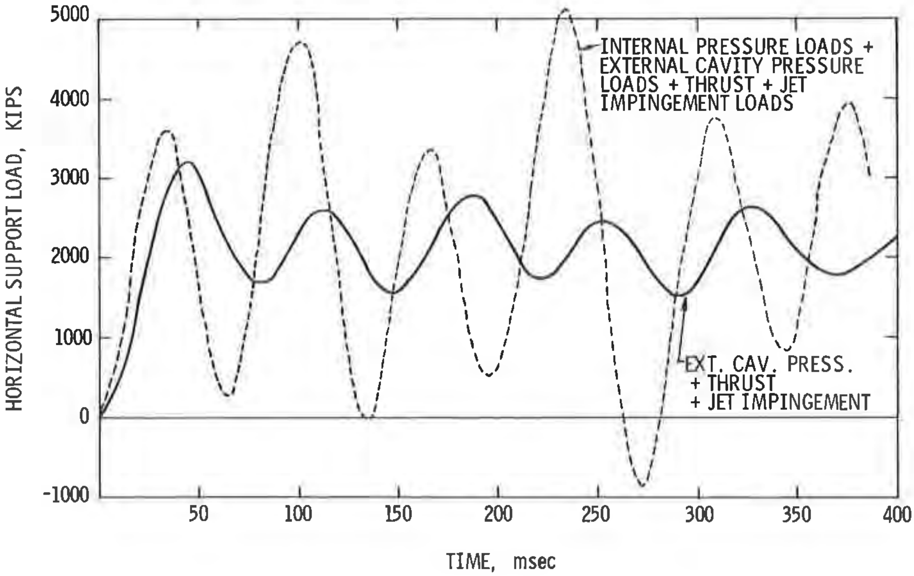
9. Vertical forcing functions on reactor vessel



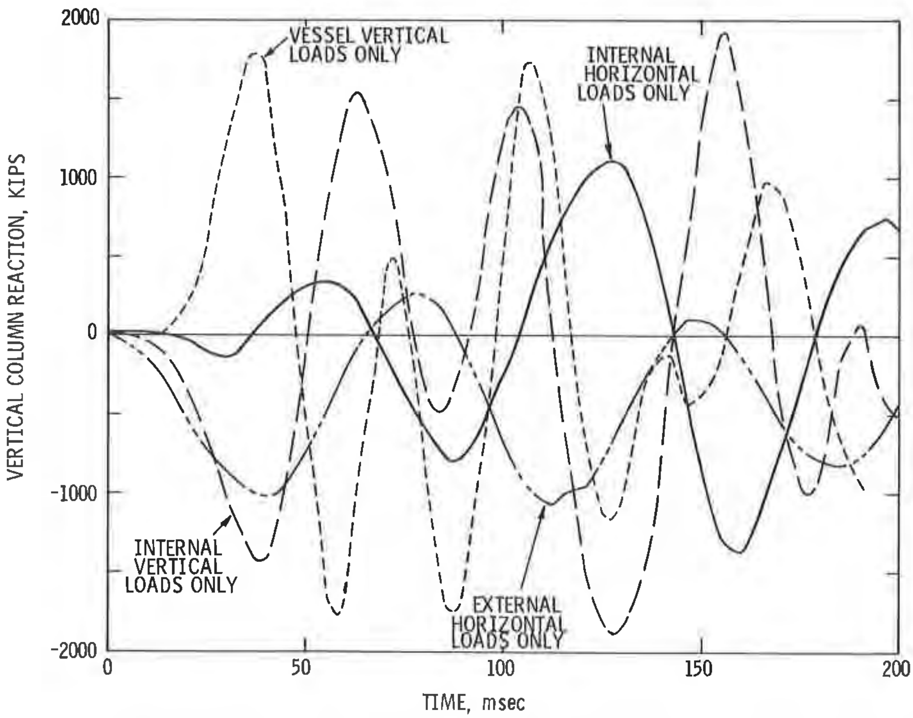
10. Reactor vessel support reaction components



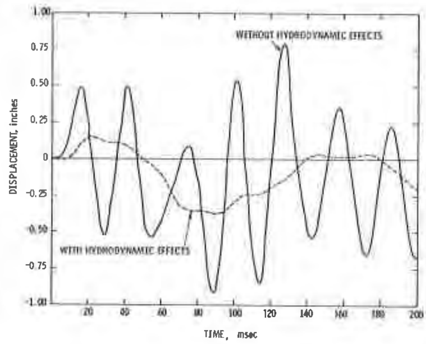
11. RV horizontal support load time history 350 sq. in. cold leg guillotine rupture



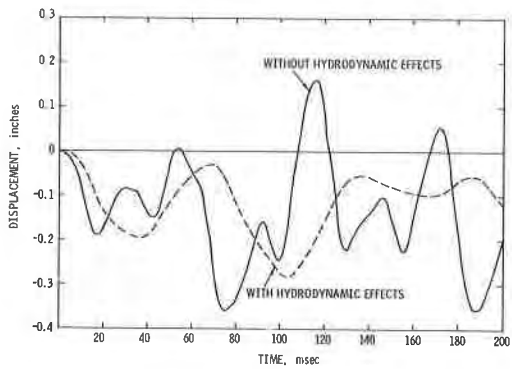
12. 350 sq. in. cold leg guillotine at RV nozzle maximum RV upper horizontal support load versus time



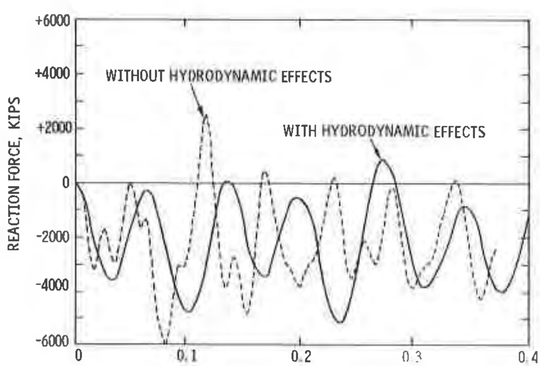
13. Reactor vessel column reactions 350 sq. in. cold leg guillotine rupture



14. Core support barrel response to 350 square inch inlet break



15. Reactor vessel response to 350 square inch inlet break



16. Effects of hydrodynamics on reactor vessel support reactions