ON THE MODELING OF REACTOR INTERNALS IN THE HCDA ANALYSIS

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SUMMARY

The advent of large high-speed digital computers has made it possible to solve, numerically on the computer, the basic conservation equations of the continuum mechanics. It has become common practice to use computer programs to study numerically the response of LMFBR reactors under HCDA conditions. However, an LMFBR reactor is a very complex structure. No computer code, no matter how sophisticated it may be, can include all the details of the reactor in the mathematical model. Therefore, to perform a manageable HCDA analysis, the representation of the reactor must be simplified.

This paper deals with the modeling of the reactor internals in the HCDA analysis. Internals which must be included in the analysis are core blankets, shields, fission-gas plenum, core barrel and core support structure. This is because they are placed so close to the reactor core that the expansion of the reactor core is strongly influenced by the presence of these internals. The general approach to the mathematical model is to incorporate, to the extent possible, the actual geometries and the correct material properties.

In the HCDA analysis, the mechanism for producing mechanical damage to the primary containment is the expansion of the reactor core. Therefore, it is very important to model the core gas properly. The inertia of the core gas must be included. Four to six zones are needed for a proper representation of the core.

The core blankets and shields can also be represented by zones. If they are segmented, they must be modeled as solid materials with no circumferential strength. Modeling of the core blankets and shields as hydrodynamic materials can lead to underestimation of the slug impact loading on the reactor head as well as the upper vessel wall deformation. Sliding lines must be provided at both sides of the core blankets and shields. The fission-gas plenum can be treated as a group of mixed materials containing gases, liquids and solids.

Core barrels must be modeled as thin shells. If the core barrel is to be treated as continuum solid material, the thickness of the core barrel will have to be divided into at least three to five zones. This not only brings about a requirement for large number of zones, but also limits the time steps to very small values. Again, sliding lines must be provided at both sides of the core barrel.

The core support structure can be modeled either as a lattice of shell elements similar to the actual structure or as an elastic-plastic solid continuum material with overall bending stiffness equivalent to the actual structure. However, the connection between the core support and the reactor vessel must be modeled properly. It is shown that improper modeling of the core support connection can lead to unrealistic results on the pressure loadings in the reactor lower plenum. It is also shown that the coolant passage openings on the core support structure can be included in the mathematical model.

Finally, to demonstrate the validity of the mathematical model, the results of comparisons with experimental data are presented.

1. Introduction

The advent of large high-speed digital computers has made it possible to solve, numerically on the computer, the basic conservation equations of the continuum mechanics. It has become common practice to use computer programs to study numerically the response of liquid-metal fast breeder reactors (LMFBR) under hypothetical core-disruptive accident (HCDA) conditions. However, an LMFBR reactor is a very complex structure. No computer code, no matter how sophisticated it may be, can include all the details of the reactor in the mathematical model. Furthermore, it is also impractical to produce a mathematical model that will have the same complexity as the real reactor. Therefore, to perform a manageable HCDA analysis, the representation of the reactor must be simplified, so that: (1) reliable analysis of the model is still achieved by the computer code, (2) the results are on the conservative side, but are not too over-conservative, and (3) the model accurately describes the response of the reactor primary system under the HCDA loads.

This paper deals with the modeling of the reactor internals in the HCDA analysis: what should be included in the analysis; how they should be modeled and how they will affect the results of the HCDA analysis, if they have modeled differently. Internals which must be included in the analysis are core blankets, shields, fission-gas plenum, core barrel and core support structure. This is because they are placed so close to the reactor core that the expansion of the reactor core is strongly influenced by the response of these internals. Unless these internals are properly modeled and included in the analysis, the results may become erroneous and unconservative. Since in the HCDA analysis the mechanism for producing mechanical damage to the primary containment is the expansion of the core gas, the modeling of the reactor core is also discussed.

2. Modeling of the Reactor Internals

2.1 Reactor Core

The reactor core is represented by zones. Four to six zones for a reactor core are generally sufficient for an HCDA analysis using Lagrangian formulations. It is usually assumed that, at the beginning of the calculation, the vaporized core material resulting from the excursion is present in some or all the core zones. The expansion of the vaporized core material in the core zones is the mechanism for producing mechanical damage to the primary system. The inertia of the core gas must be properly included. In some cases, it is advantageous to assume that the vaporized core material and the other core materials such as molten fuel and cladding are a homogeneous mixture thus the vaporized core material is present in all of the core zones, and the expansion of the reactor core zones will be less distorted. The other core materials are considered to be nonexpansible material. They occupy the same volume as the core expands and they attribute only the inertia effect in the core expansion. If less than four zones are used for a reactor core, one should rezone the reactor core to a large number of zones at a later time after it has expanded to a larger volume. The Langrangian representation of a reactor core with few zones can not properly represent the behavior of the core gas in the stages of contraction and the subsequent re-expansion and re-contraction. This is illustrated in a sequence of core configurations shown in Fig. 1, where 1a is the initial reactor configuration in which the reactor core is represented by two Lagrangian zones due to the limitation of the reactor core size and the Lagrangian meshes used in the analysis; 1b shows the reactor core configuration at a certain time during the computation when the core has expanded to a sufficient size; 1c is the same core configuration after rezoning where the number of core zones has increased to 8; 1d shows the core configuration at the time of the maximum expansion; 1e shows the core configuration at the stage of contraction; and 1f is the core configuration at the same time instant as is in 1e, but the number of core zones remained to be two. As can be seen from Fig. 1, the two-zone core model does not simulate properly the behavior of the core gas at the stage of contraction, not to mention the subsequent stages of re-expansion and re-contraction.

2.2 Core Blankets

The blankets are also represented by zones. They should have the same configuration, mass, compressibility, and wave-propagation properties as the actual material, but they are considered to have no tensile strength in either the axial or circumferential directions and are represented by fluid material.

The modeling of the core blankets as fluid material is based upon the fact that they are loosely connected and cannot resist hoop stresses. Insofar as they have no hoop stresses, it is advantageous to ignore also their tensile strength in the axial and radial directions and to treat them as hydrodynamic materials, so that zone distortions induced by the expansion of the core zones adjacent to these internals will be reduced, which is essential in long-duration computer runs.

2.2 Shields

The shields can be represented by zones too. They should have the same configuration, mass, and other properties as close to those of the actual material as possible. The shields are made of segmented plates. Although they have no tensile strength in the circumferential direction, they exhibit considerable tensile strength in the radial direction. Therefore, they can not be represented by the hydrodynamic material. Modeling of the shields as hydrodynamic materials can lead to underestimation of the slug impact loading on the reactor head, as well as the upper vessel wall deformation. To minimize the zone distortions induced by the expansion of the core zones and by the hydrodynamic representation of the blankets,sliding lines must be provided at the inner surface of the shield materials. The mathematical formulations for shield materials with no circumferential strength are given in Reference [1].

Two calculations were performed for a simple reactor configuration which had a 0.5 cm thick vessel wall, a 0.127 cm thick core barrel, and a thick segmented radial shield as shown in Fig. 2 to illustrate the effect of radial shield on coolant slug impact, as well as deformations of core barrel and vessel wall. In Case A, the radial shield was assumed to be solid elastic-plastic material with no tensile strength in the circumferential direction, and the coolant was permitted to slide along the inner surface of the shield. In Case B the radial shield was modeled as compressible hydrodynamic material where the sliding of coolant was also permitted.

Although the total core work energy released is about the same in these cases, the upward axial kinetic energy of the coolant is somewhat different. It can be seen from Fig. 3 that the maximum kinetic energy is higher in Case A than in Case B. As a result, the slug impact pressure on the reactor head and upper vessel wall is higher in Case A than in Case B. This can be seen from the profile of the vessel wall deformation shown in Fig. 4. The profile of the core barrel deformation is shown in Fig. 5 where the mode of deformation in the two models is quite different. Thus, the modeling of the shield materials as hydrodynamic materials can lead to unconservative results on slug impact loads, as well as on deformations of the vessel wall and core barrel.

2.4 Fission-gas Plenum

The fission-gas plenum can be treated as a group of mixed materials containing gases, liquids, and solids. Therefore, it can also be represented by zones. Unfortunately, none of the existing equation-of-state data are available for such mixtures. Therefore, it is necessary to develop a method of constructing an equation of state from available data for individual materials.

The method used is similar to that employed by Goranson et al [2] in determining the dynamic compressibility of metals. The basic assumptions employed are: (1) all components of the mixture within a mesh zone are under the same pressure and at same temperature and (2) all components remain intact under pressure. Detailed formulations are given in Reference [3]. Since the compressibility of the fission gases is relatively high, the work of the core expansion energy at the beginning of the excursion is used up in compressing the fission gases, in packing the other fission-gas plenum materials and reducing them to a smaller volume. Thus, the core energy released to the surrounding coolant is reduced until the fission gases have been reduced to a very small volume and other materials have been closely packed and come in contact with each other. Figure 6 shows that the fission-gas plenum zones have been compressed from the original configuration to a compact configuration at the early stage of the excursion, while the other zones remain relatively unchanged.

2.5 Core Barrel

The core barrel is the first structural member placed next to the reactor core. Therefore, the modeling of the core barrel is very important in the HCDA analysis. There are two ways in which the core barrel can be modeled: one by continuum solid materials and the other by thin shell structures. Here the shell structure is defined as a slender solid member in which the deformation is governed by the deformation of the midsurface or midline. In HCDA analysis, the choice of a continuum or thin shell approach depends both on the geometry of the core barrel and on the loading and response that is of interest. If the wave propagation through the thickness is of importance, such as in the case of the radial shields, a continuum approach is appropriate. If the thickness of the core barrel is relatively thin compared to other dimensions, and the wave propagation through the thickness of the core barrel is of no interest, then it is advantageous to model the core barrel as a thin shell structure. Thus the motion of the core barrel can be defined by that of the midplane, hereby reducing the number of degrees of freedom and improving the stability of numerical computations. If the core barrel is to be treated as continuum solid material, the thickness of the core barrel will have to be divided up into at least three to five zones, so that bending strength of the shell can be properly included in the analysis. This not only brings the requirement for a large number of zones, but also limits the time steps to very small values if the explicit integration scheme is used. Therefore, the core barrel should be modeled as thin shells if it is a slender member. Again, sliding lines must be provided at both sides of the core barrel.

Since the problems in the HCDA analysis are nonlinear in both geometry and material properties, the shell equations used should be those for large deflections and for elastic-plastic and strain hardening materials. The shell formulations can be either in finite-difference or finite-element forms. Detailed formulations are given in References [4,5]. Although there are substantial differences in the mathematical treatments, the results obtained with the finite-difference and finite-element methods are comparable.

2.6 Core-support Structure

The core-support structure can be treated either as an elastic-plastic solid continuum material with overall bending stiffness equivalent to the actual structure or as a lattice of shell elements similar to the actual structure. If the core-support structure is treated as an elastic-plastic material, it is modeled by shaping those Lagrangian zones to conform to the actual configuration as closely as the Lagrangian zones would allow. The mass represented by these Lagrangian zones should be equal to that of the actual core-support structure, which can be obtained quite easily. However, the equivalent bending stiffness is very hard to obtain. Fortunately, the core-support structure in the LMFBR design is very rigid. Numerical experimentation shows that the deflection of a core-support structure made of a lattice of shell elements like the actual structure is very similar to that of a core-support structure modeled as elastic-plastic solid continuum material by shaping the Lagrangian zones to conform to the actual configuration.

It should be mentioned that the core-support structure is not an axisymmetric structure even though it has certain degrees of symmetry. Therefore it can not be accurately modeled as an axisymmetric structure in the 2-D analysis. It is believed that some conservatism is needed in the calculation of the slug impact force on the head, downward thrust on the vessel support straps, pressure loading on the wall, and deformation of the core barrel and vessel wall. Thus, modeling of the core-support structure as an elastic-plastic solid material instead of a lattice of shell elements is not an unrealistic model, for it provides some needed conservatism. It should be pointed out that the mass of the core support structure is kept the same as that of the actual structure. The inertia effect which is very important in the transient analysis is, indeed, properly accounted for in the model. The only disadvantage of this model is that the pressure pulses at the lower reactor plenum below the core-support structure obtained from this model will be underestimated because of the overstrong core support which prevents the coolant slug from moving downward. More will be said later on this coolant downward motion.

Perhaps the most important element in the modeling of the core-support structure is the modeling of the connection between the core-support structure and the reactor vessel wall. In LMFBR designs, the core-support structure is usually connected to the reactor vessel wall through a skirt connection. This skirt connection between the core-support and the reactor vessel must be properly modeled. Omission of the skirt connection can lead to unrealistic results on the pressure loadings in the lower reactor plenum. The following example illustrates that the improper modeling of the core-support connection can affect the results on pressure loadings in the lower reactor plenum, as well as displacements of the core-support structure.

Two core-support structure models are shown in Fig. 7, where one models the core-support structure to be supported by a skirt, Case A, while the other models the core-support structure to be directly attached to the vessel wall, Case B. In both models the cross-hatched area indicates the core-support structure which is modeled as an elastic-plastic solid continuum. Figure 8 shows the pressure loadings at the inlet nozzle in the lower plenum for both Cases A and B. As the skirt model allows the core-support structure to move downward easier than the other model, the pressure at the inlet nozzle in Case A becomes as high as 2.2 MPa before slug impact, whereas for Case B it does not exceed 1.2 MPa. After slug impact, the pressure history of Case A exhibits several high spikes of the order of 5.0 MPa, which are absent in Case B. However, the pressure loadings at the outlet nozzle in the middle region of the reactor vessel (shown in Fig. 9) are very much the same, indicating that the modeling of the skirt connection does not affect the outlet nozzle pressure significantly. Similarly, the slug impact force histories for the two cases show no noticeable differences. This can also be inferred by comparing the average velocity of the sodium in the upper 12 zones for the two cases.

To show the effect of the skirt modeling on the motion of the core-support structure and vessel wall, axial displacements at points A, B, and C are shown in Figs. 10a, b and c respectively, as a function of time. As can be seen from those curves, the displacements in the case of with-skirt connection are sufficiently larger than those without-skirt. Similarly, there are substantial differences in the vessel wall radial displacements due to the downward motion of the core-support structure. For instance, the final radial displacement at point A at the end of the calculation is -0.92 cm for Case A and -0.04 cm for Case B. These larger displacements result in higher stresses in the vessel wall and especially at the connection of the skirt and vessel wall juncture.

As the core-support structure is modeled as an elastic-plastic solid continuum material, it does not permit sodium flow through the coolant-passage openings. Thus the pressure in the lower reactor plenum obtained in the Lagrangian analysis is further reduced due to the assumption which requires the mass in each zone to remain constant, while in the actual case, the sodium above the core-support structure is pushed to move downward through the coolant passage holes into the lower reactor plenum by the expanding core gas.

To study the effect of sodium flow through the core-support structure openings on the lower reactor plenum pressure loading, one can use either an Eulerian code or a modified Lagrangian code which permits the flow of sodium through the openings in the core-support structure, as described in Reference [6]. To remove the effect of the skirt downward motion, the reactor configuration used is the one shown in Fig. 7b where the core-support structure is directly connected to the vessel wall. The coolant passage opening area is given as some percent of the total zone area ranging from 0.25% at areas underneath the reactor core zones to 0.10% at areas underneath the radial shield zones. The pressure loading at the location of the inlet nozzle is given in Fig. 11, together with the case which permits no sodium flow through the coolant passage openings. In this analysis, the sodium mass that flows through the core-support structure openings into the lower reactor plenum is assumed to be distributed: (1) to the zones immediately below the openings, or (2) to all zones below the core-support structure. As can be seen from Fig. 11, the effect of coolant passage openings on the pressure loading in the lower reactor plenum is substantial. Thus, they can not be ignored in the mathematical model.

3. Comparison with Experimental Data

To demonstrate the validity of the mathematical model and modeling technique, a comparison of the code calculations with the experimental data is given below. The experimental data chosen for this purpose is the SRI complex model tests. Detailed comparisons of these tests have been given in Reference [7]. Here, only the important ones, which will demonstrate the validity of the mathematical model used for predicting the response of the primary system to an HCDA, are compared.

As in the experiments, the calculations did not indicate vessel failure at any point. At the core midplane, the maximum strain measured was 1.06% for the 1/30th-scale model, and 1.20%

for the 1/10th-scale model; whereas, the calculated values were 1.0% for both models. At the vessel top, the maximum strain measured was 2.23% for the 1/30th-scale model and 1.88% for the 1/10th-scale model; whereas, the respective calculated values were 3.4% for the 1/30th-scale model and 3.2% for the 1/10th-scale model. Discrepancies for the upper vessel may well be explained by the fact that the analytical model does not allow a fluid loss, so that all energy must be deposited in the system; whereas, the experimental models lost water from the vessel. Maximum core-barrel strain measured was 5.39%; whereas, that calculated was 6.5%.

Slug impact time provided another interesting comparison. Experimental measurements indicated slug impact at 4.3 msec for the 1/10th-scale model, whereas the calculations indicate that the impact at the centerline is at 4.044 msec with complete impact several tenths of a milisecond later, the peak force being at 4.3 msec. Thus, the impact times are quite similar.

4. Conclusions

The modeling of the reactor internals has been discussed in detail. The general approach to the mathematical model is to produce a simplified reactor model so that it can be analyzed by the computer code, and to include all important elements in the mathematical model so that it will accurately describe the response of the reactor primary system under the HCDA loads.

Good agreement between the SRI experiments and computer calculations lends credibility to the mathematical model and the modeling technique described above.

It should be noted however, that the answer obtained from the computer analysis depends upon the basic information used in the computer analysis, even if the mathematical model and modeling technique are perfect. Thus, to improve computer results, a better understanding of the energy-release mechanisms and more reliable equation-of-state data for the core and other reactor materials over the appropriate pressure and temperature ranges are also needed.

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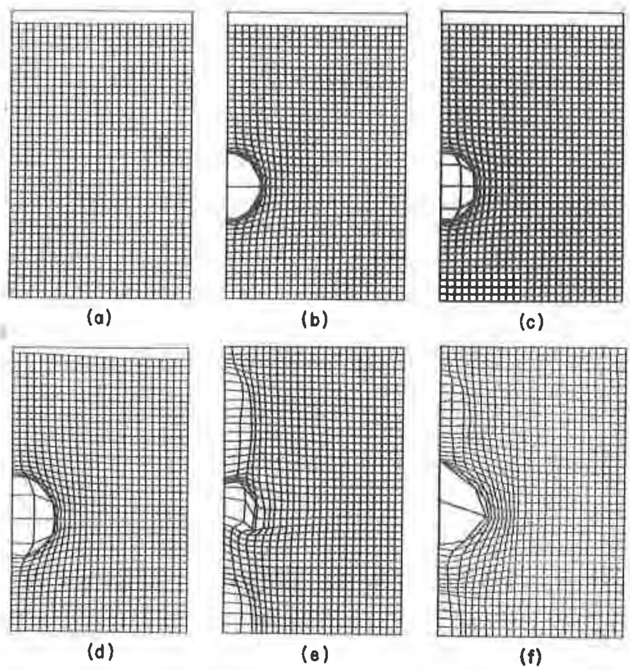


Fig. 1. Core Gas-bubble Configurations

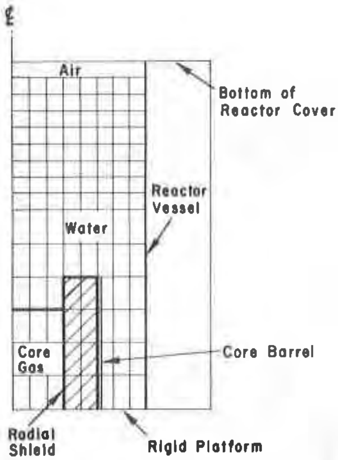


Fig. 2. Simple Reactor Configuration

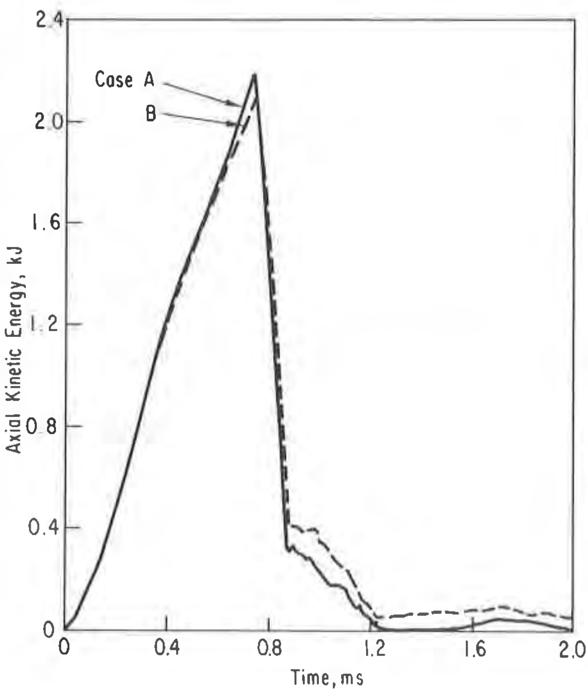


Fig. 3. Upward axial kinetic energy

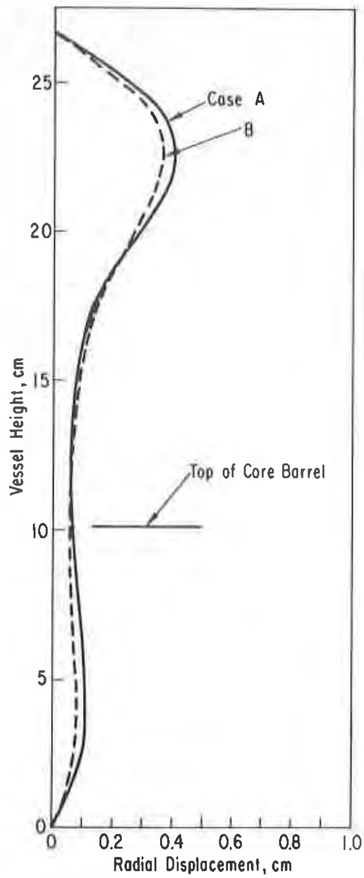


Fig. 4. Profile of Vessel Wall Deformation

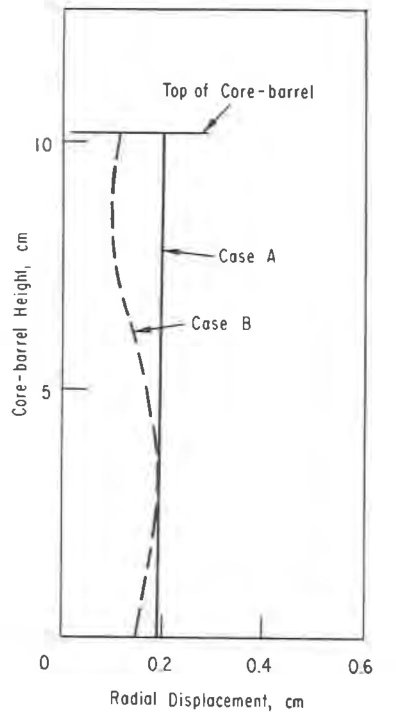


Fig. 5. Profile of Core Barrel Deformation

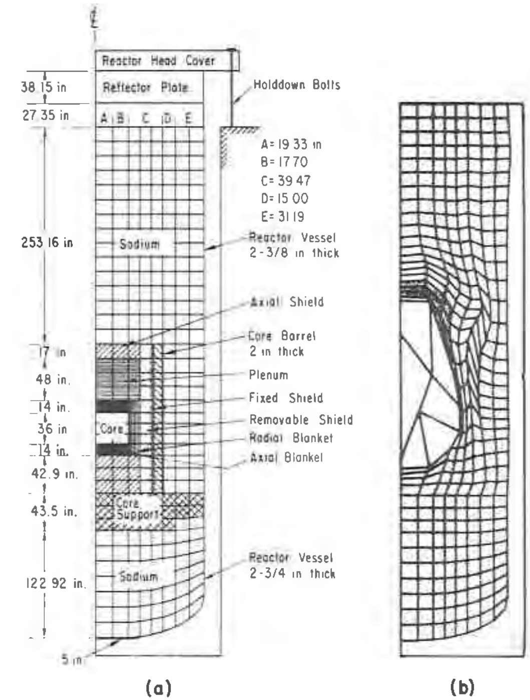


Fig. 6. Configuration of Fission-gas Plenum zones:
(a) Original; (b) Deformed

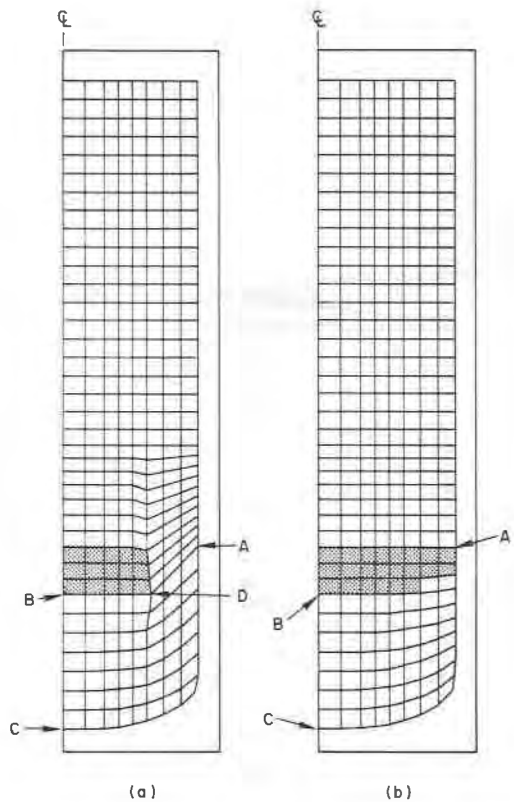


Fig. 7. Mathematical model of the Core-support Structure; (a) supported by a skirt; (b) directly attached to the vessel wall

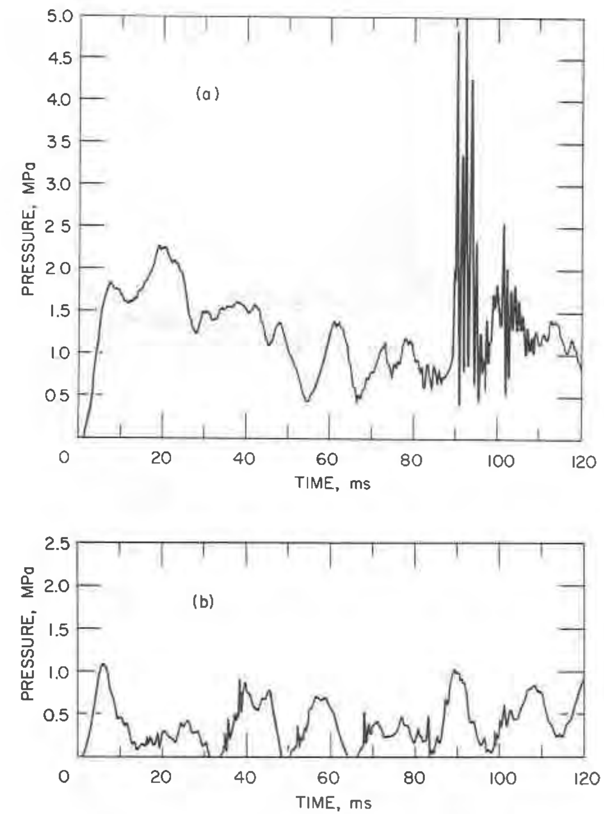


Fig. 8. Pressure Loadings at the inlet nozzle in the lower plenum: (a) Case A (b) Case B

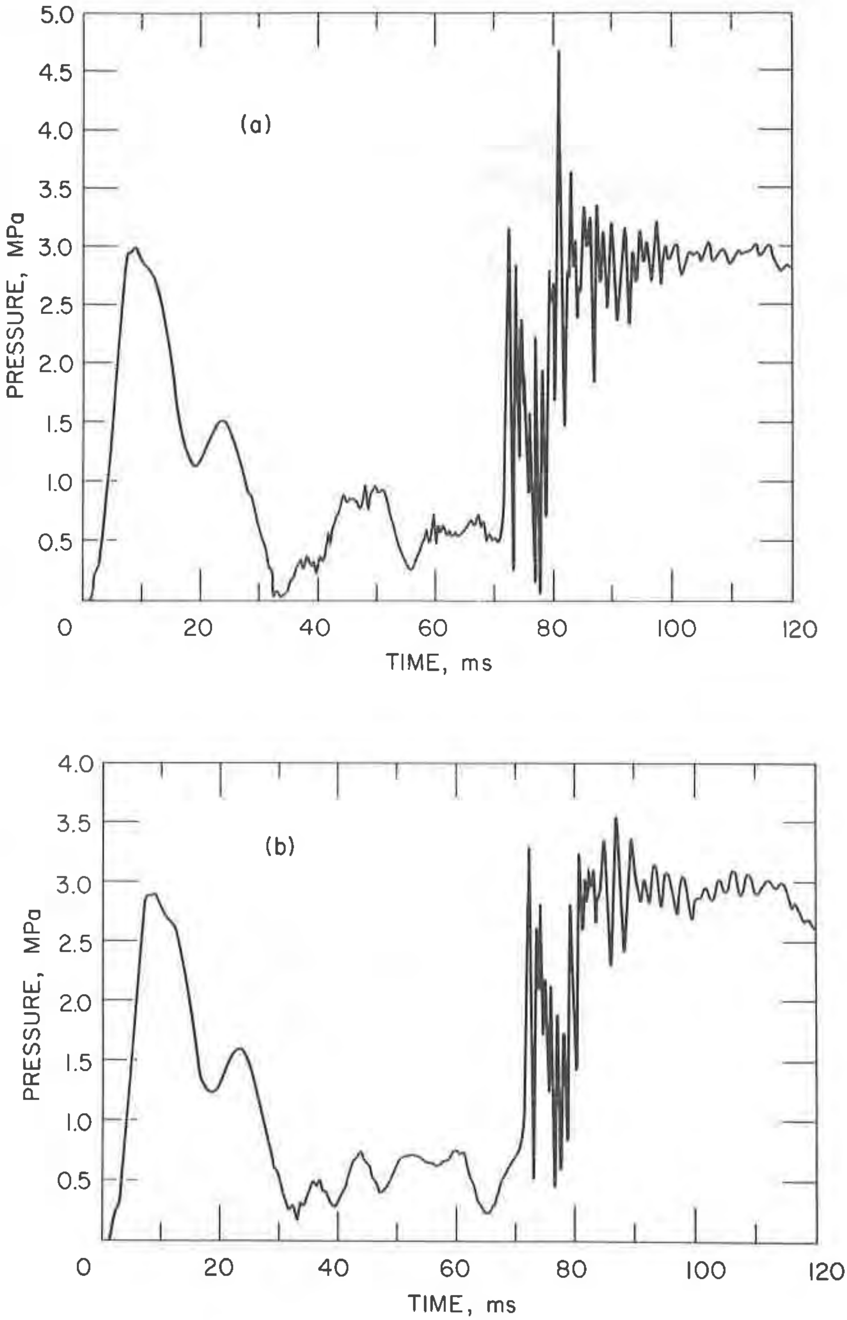


Fig. 9. Pressure Loadings at the Outlet Nozzle in the middle region of the reactor vessel: (a) Case A; (b) Case B

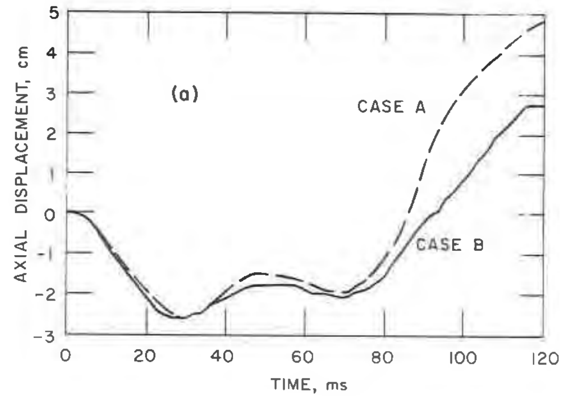
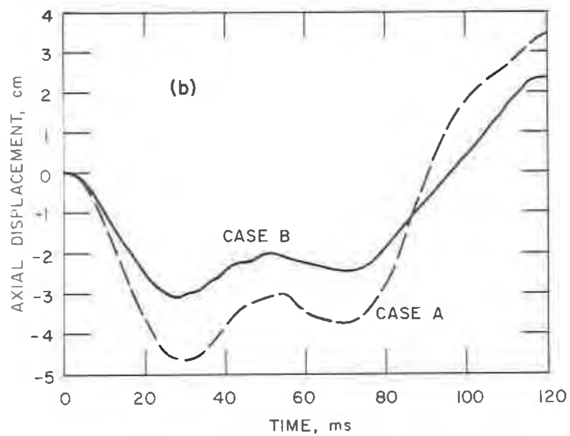
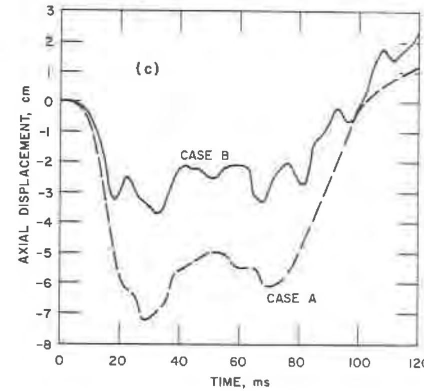


Fig. 10. Axial Displacements of: (a) Point A;



(b) Point B;



(c) Point C

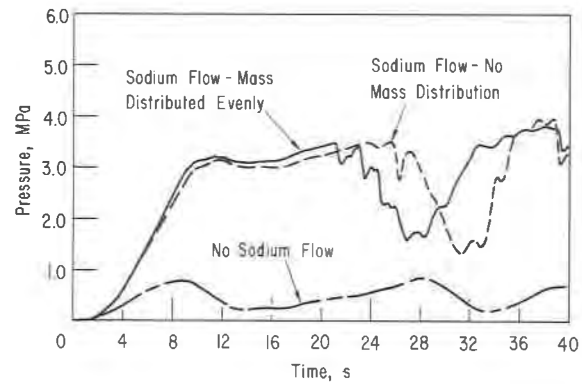


Fig. 11. Pressure Loadings in the lower reactor plenum for the cases of with and without sodium flow through the coolant passage openings on the core-support structure