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This paper describes the application of containment codes to predict the response of the fast reactor containment and the primary piping loops to HCDAs. Five sample problems are given to illustrate their applications. The first problem deals with the response of the primary containment to an HCDA. The second problem deals with the coolant flow in the reactor lower plenum. The third problem concerns sodium spillage and slug impact. The fourth problem deals with the response of a piping loop. The fifth problem analyzes the response of a reactor head closure. Application of codes in parametric studies and comparison of code predictions with experiments are also discussed.

1. Introduction

The response of the primary containment to hypothetical core-disruptive accidents (HCDA) in liquid-metal fast breeder reactors (LMFBR), depends on a number of considerations, including the characteristics of the HCDA energy source, propagation of the pressure waves, and interactions of fluid and internal structural components.

Because of the complexity of the problem, the early approach was to perform scaled model experiments, in which the HCDA was simulated by the use of high explosives and the sodium was replaced by water. However, to perform a meaningful test, not only must the reactor containment structure and internals be properly represented in the scaled model, but also the initiating conditions and the rate of energy release from the core must be properly simulated. Because of the different characteristics of the energy release in nuclear excursions and chemical explosives, it is very difficult to simulate an HCDA with chemical explosives.

A different approach to the problem is to use the basic conservation equations of mass, momentum, and energy and to determine mechanistically the response of the primary containment and the internal structures under the postulated nuclear excursion conditions. In recent years, the advent of the high-speed large digital computer has made it possible to use computer programs to solve the conservation equations and to determine the loadings, strains, and the response of the reactor primary containment numerically. This approach has been adopted by most fast reactor projects of the industrialized nations. ANL's REXCO-HEP, ICECO, MICE, and ICEPEL [1-4], AWRE's ASTARTE [5], BN's SURBOUM [6], IA's ARES [7] and EU's EURDYN [8] are a few of the computer programs developed for LMFBR containment analysis. Other computer programs which were developed for different purposes, such as PISCES, HEMP, ANSYS, TOODY, WHAM, STEALTH, CSQ, STRAW, and SADCAT [9-17] have also been applied to safety problems related to LMFBR containments.

This paper deals with the application of containment codes to LMFBRs in the United States. However, with the current activities in the LMFBR containment analysis, new containment programs are developing at a very fast rate, so are their applications to LMFBRs. Hence, the coverage is inherently limited by the author's personal knowledge of the various activities where the containment codes were actually applied, and the space limitations for covering such a diverse subject. Therefore, only the containment programs developed at Argonne National Laboratory (ANL) and the applications to LMFBRs which are known to the author will be reported herein. It is hoped that the material presented in this paper will provide an overall view of the safety related containment computer code systems under current usage or active development in the United States.

2. Computer Codes Developed for Containment analysis

The arithmetic operation in the computer is both discrete and finite. It is necessary in the computer analysis that a physical system be replaced by a discretized system. In discretizing a continuum system, the continuum is often replaced by a group of meshes. Two types of meshes are commonly used. If the mesh point is attached to a material particle and moves with the material, the mesh is called Lagrangian; if the mesh point is fixed in space and is invariant in time, the mesh is called Eulerian.

Although Lagrangian codes have been successfully used for analyzing the pressure wave propagations coolant slug impacts, and containment responses under HCDAs, their analyses are still limited to the early stage of the excursion, because the excessive zone distortion will deteriorate the accuracy of the numerical results. Although rezoning of the distorted Lagrangian mesh can extend the calculations to a longer time, there is an upper limit on how far one can push this rezoning technique in a Lagrangian calculation.

Since the Eulerian meshes are fixed in space, the Eulerian hydrodynamic equations are ideal for treating excessive fluid distortions. Thus, for problems involving large material distortions, Eulerian discretization becomes more attractive, especially in the calculation of sodium spillage and core gas-bubble migration. Unfortunately, the very feature of the Eulerian method which makes it useful in handling material distortions is also the one which makes it unsatisfactory for performing calculations having material interfaces and moving boundaries. In reactor containments, the reactor vessel, cover head, and internal structures are not rigidly fixed in space. As they displace under the applied pressure loads, their boundaries will intersect the Eulerian lines in various ways. Complex procedures are needed at the material interfaces and boundaries.

The approach taken by Argonne National Laboratory is to develop both the Lagrangian and Eulerian computer programs: using the Lagrangian code for the determination of the pressure wave propagation, the response of the reactor internals and the deformation of reactor vessel wall during the phase immediately following the accident, and the Eulerian code for studying the excursion phenomena such as slug impact, sodium spillage and core gas-bubble expansion and migration during the latter phase of the excursion. Currently the containment codes at ANL have advanced to the point of being able to predict reasonably well the response of the primary system. The phenomena of the HCDA can be followed up to the time of dynamic equilibrium in the system, when no further plastic work can be performed on the components. The analytical predictions of component response have been verified reasonably well by comparing with experimental test results. The amount of sodium spillage into the secondary containment can also be reasonably predicted. A brief description of ANL containment codes is given below.

REXCO-HEP

REXCO-HEP is a two-dimensional (r-z) Lagrangian code for calculating the primary system response in fast reactors. It treats not only the hydrodynamics, but also the elastic and plastic deformation of the solid reactor materials. Fluids are assumed to be compressible but nonviscous. The reacter vessel, core barrel, and core-support structure can be analyzed as thin shell structures.

The radial shields can be treated as segmented solids with no tensile strength in the circumferential direction. The reactor head can be treated as deformable circular plate; crushable materials can be placed underneath the reactor head to absorb the slug impact energy. The reactor vessel can be connected to the reactor head through connecting bolts or attached to the ground with the holddown bolts. At the fluid-solid interfaces, the fluids are allowed to slide along the solid surface but are forced to move together with the solid in the direction normal to the solid surface. A REZONE code has been developed for rezoning the distorted Lagrangian meshes occurring in the REXCO-HEP runs.

ICECO

ICECO code is a two-dimensional implicit Eulerian code for calculating fluid transients in fast reactors. Complete hydrodynamic equations are considered, including nonlinear convective and viscous dissipation terms. The numerical technique used is the Implicit Continuous Eulerian (ICE) technique developed by Harlow and Amsden [18]. The reactor vessel, core barrel, and core-support structure are treated as thin shell structures using the finite-element method. At the fluid-shell interfaces, the fluid velocities are calculated in the same manner as described in the REXCO-HEP code, so that the condition of nonpenetration of fluid particles in the normal direction of the shell surface is satisfied. It treats the sodium flow through the core-support structure openings as well as sodium flow through the openings on the head cover and at the reactor head and vessel wall juncture.

MICE

MICE code is an extension of the ICECO code. It is a multi-field 2-D implicit Eulerian code and is currently under development at ANL. A fluid is considered to be a field; the vapor phase of that fluid is considered to be another field; a different fluid is considered to be another field. Altogether five fields are allowed in the MICE code calculation. Each field is governed by a set of equations of mass, momentum and energy, and the equation of state of that material.

Momentum exchange between fields is accomplished by the use of the drag functions. Mass changes resulting from phase transitions are considered in the mass and momentum equations. Thus, interpenetrations of fields are allowed in the MICE calculation.

ICEPEL

ICEPEL code is also an extension of the ICECO code, but is developed specifically for the analysis of the heat transport piping systems. It has a generalized pipe component model which can be used to model valves, reducers, expansions, and heat exchangers. Together with the elbow and tee branching model, they can be used to analyze a complete LMFBR heat transport system both hydrodynamically and structurally under the effect of simultaneous pressure pulses in an HCDA.

STRAW, SADCAT

STRAW and SADCAT are two finite-element codes. Although they are basically structural codes, they have hydrodynamic capabilities. They can be used for the nonlinear transient analysis of plane two-dimensional, axisymmetric, and three-dimensional structures and continua in fast reactors. They also have static and modal capabilities. One of the main features that distinguish this series of codes from other finite element codes is in the type of coordinates system employed. A corotational coordinate system is used which rotates and translates with each element. This offers the capability to ultilize linear relationships for the stress-strain and force-displacement expressions which yields considerable computational efficiency.

A list of the containment codes is given in Table 1. It should be mentioned that these codes were chosen primarily on the basis of the fact that they have been used for the analysis of the LMFBRs. It is by no means a complete list of all the containment codes available in the United States for LMFBR containment analysis.

3. Application to LMFBRs

The major areas of application of containment codes to LMFBRs are: (1) to perform safety analysis in support of LMFBR licensing; (2) to perform parametric studies to determine the sensitivity of results to key unknowns; (3) to perform comparisons with experiments to validate the mathematical model used in the safety analysis.

3.1 *Application of Codes in Support of the Safety Analysis*

The containment codes developed at ANL have been applied to the safety analysis of the Fast Flux Test Facility(FFTF). Currently, they are being applied to the Demonstration Plant Safety Analysis. The propagation of pressure waves, the response of the reactor internals, the coolant slug impact, and the deformation of the reactor vessel wall during the phase immediately following the accident are analyzed by the Lagrangian code, REXCO-HEP. The flow of sodium coolant in the reactor lower plenum, the sodium spillage through openings, and the core gas-bubble expansion and migration are determined by the Eulerian containment codes, ICECO and MICE. The response of the primary piping system to HCDA loads is analyzed with the ICEPEL code; the response of structural components such as reactor head and core-support structure is obtained with a structural code SADCAT. Some of the applications are illustrated in the following sample problems.

3.1.1 *Response of the Primary Containment to an HCDA*

As mentioned earlier, the response of the primary containment to an HCDA is analyzed by a Lagrangian code. Because of the Lagrangian discretization, sufficient details can be included in the analysis. However, in order to perform a manageable HCDA analysis, the representation of the reactor must be simplified. Internals which must be included in the analysis are core blankets, shields, fisson-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. Figure 1 is the mathematical model of the FFTF reactor which illustrates how these internals can be included in the analysis.

The most important thing in the computer analysis is the modeling of the reactor internals. Some of the modeling techniques are described below.

The reactor core is represented by zones. Four to six zones for a reactor core are generally sufficient for an HCDA analysis. 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 expands.

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 the 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-

Table 1 Compilation of Containment Codes

Name Origin (Ref)	Problem Solves	Unique Capabilities	Discretization Fluid Structure	Integration Method
REXCO-HEP ANL [1]	2D + Shell	Sliding lines for fluid/solid meshes and fluid/shell interfaces Rezoning by REZONE code	FD (Lagrangian)	Explicit
ICECO ANL [2]	2D + Shell	Coupled Eulerian/Lagrangian Can handle extended fluid motion (e.g. Bubble motion and Na spill)	FD (Euler.) FE (Lagran.)	Implicit
MICE ANL [3]	2D + Shell	Multifields Allowing interpenetrations	FD (Euler.) FE (Lagran.)	Implicit
ICEPEL ANL [4]	2D + Shell	Extensive models for piping component - Can handle extended fluid motion	FD (Euler.) FE (Lagran.)	Implicit
PISCES-2DL PI [9]	2D	Sliding lines for fluid/solid meshes - Rezone and static option - Auto. coordinate generator	FD (Lagrangian)	Explicit
PISCES-2DELK PI [19]	2D + Shell	Coupled Eulerian/Lagrangian Implicit/Explicit Option	FD (Euler.) (Lagran.)	Explicit/ Implicit
HEMP LLL [10]	2D	Sliding lines capability for fluid/solid meshes Rezoning capability	FD (Lagrangian)	Explicit
TOODY SLA [12]	2D	Slide line capability for coupled fluid/solid meshes Rezoning capability	FD (Lagrangian)	Explicit
CSQ SLA [15]	2D	Extensive EOS options Conduction & radiation H. T. Up to 10 fluids	FD (Eulerian)	Explicit
STEALTH SAI [14]	2D + Shell	Continuous auto rezoner Finite-element shell option	FD FE (Lagrangian)	Explicit
STRAW ANL [16]	2D	Convected coordinates Can handle complex geometry	FE (Lagrangian)	Explicit/ Implicit
WHAM UICC [13]	2D + Shell	Convected coordinates Can handle complex geometry	FE (Lagrangian)	Explicit/ Implicit
SADCAT ANL [17]	3D	Convected Coordinates Can handle complex geometry	FE (Lagrangian)	Explicit/ Implicit

FD = finite-difference; FE = finite-element
ANL = Argonne National Laboratory
PI = Physics International Co.
LLL = Lawrence Livermore Laboratory
SLA = Sandia Laboratories
SAI = Science Applications, Inc.
UICC = Univ. of Illinois at Chicago Circle

gas plenum can be treated as a group of mixed materials containing gases, liquids, and solids.

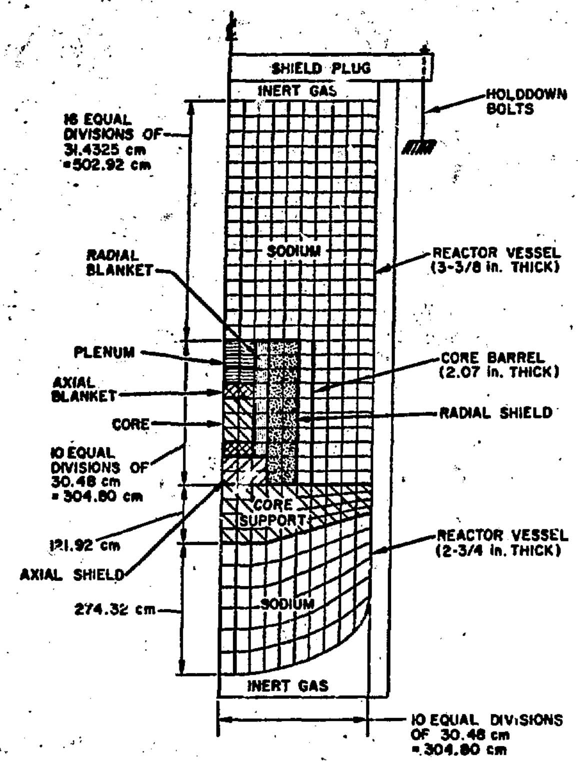

Fig. 1. Mathematical Model of the FFTF Reactor

Core barrel must be modeled as a thin shell. If the core barrel is to be treated as a 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 a 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 which can be done by shaping the Lagrangian zones to conform to the actual configuration. If the core-support structure is treated as an elastic-plastic material, the mass represented by those Lagrangian zones should be equal to that of the actual 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 lattice of shell elements is very similar to that of a core-support structure modeled as an elastic-plastic solid continuum material by shaping the Lagrangian zones to conform to the actual configuration. Connection between the core-support structure and the reactor vessel must be modeled properly. Results in [20] show that improper modeling of the core-support connection can lead to unrealistic results on the pressure loading in the reactor lower plenum.

Figure 2 is a series of computer-generated reactor configurations at various times during the excursion. These configurations show how the core-surrounding structures are deformed, and how the deformations of the core barrel, vessel wall, and other structures adjacent to the core become progressively larger with time. It also shows how the coolant slug is being pushed upward until it comes in contact with the reactor cover and how the upper vessel wall is deformed. The overall flow of the coolant can be followed visually from these configurations.

3.1.2 Effect of Core-support Structure Openings on the Pressure Loading in the Reactor Lower Plenum

As mentioned earlier, the Eulerian codes are ideal for treating excessive fluid distortions. Therefore, they can be used for analyzing sodium flow in the reactor lower plenum. However, because of the difficulties in the treatment of the material interfaces and moving boundaries, the Eulerian containment codes developed to date still can not treat a reactor configuration to have the same complexity as in the Lagrangian analysis. Therefore, the reactor configuration used in the Eulerian analysis is considerably simplified. Figure 3 is the reactor configuration used by Wang [21] in studying the effect of core-support structure openings on the pressure loading in the reactor lower plenum. It consists of a core gas-bubble, a radial shield, a core barrel, a core-support structure and a reactor vessel. The core-support structure is assumed to have perforated openings as shown in Fig. 4, where the perforation ratios for the first three zones from the center line are 0.15, 0.10, 0.15, respectively.

Because of the presence of the opening holes in the core-support structure, the coolant in the core region immediately below the core gas-bubble can be pushed down through the openings into the reactor lower plenum by the expansion of the core gas-bubble. This can be seen from the sequence of reactor configurations shown in Fig. 5. Figure 6 shows the pressure loadings in the lower plenum for the cases of with and without openings. As can be seen the omission of the coolant passage openings on the core-support

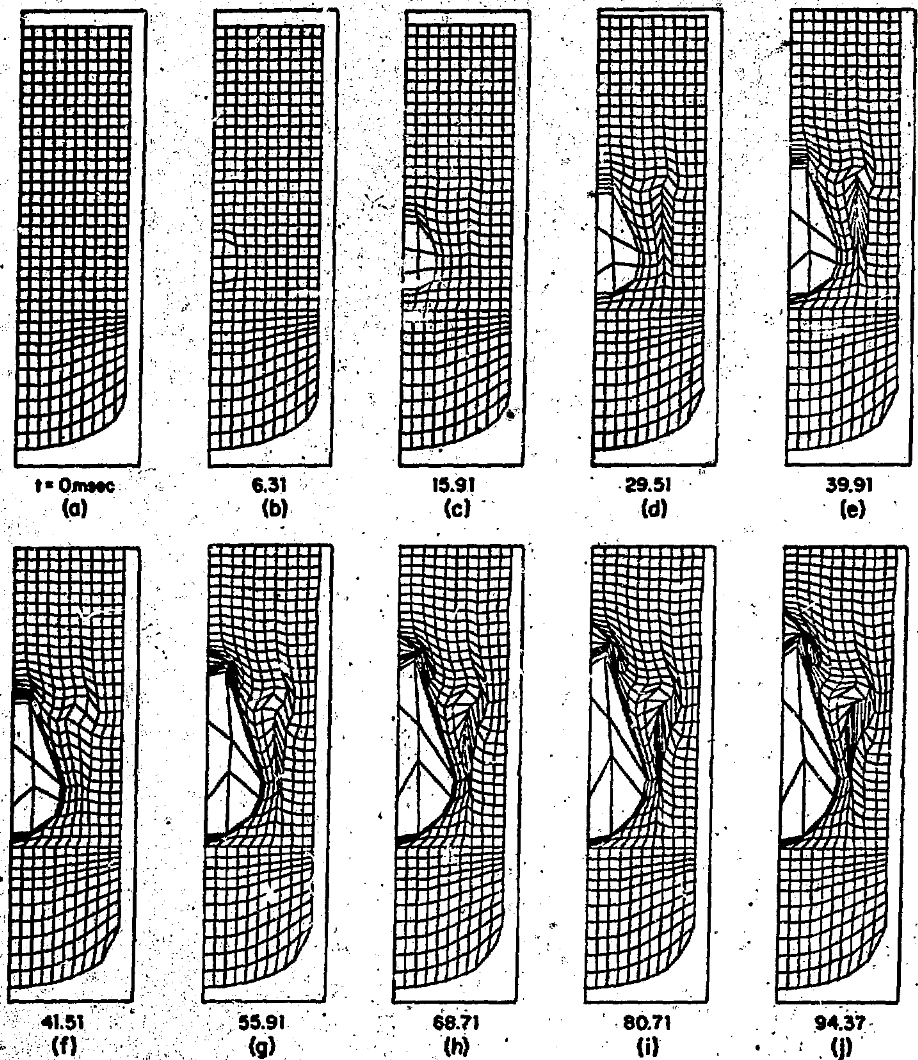

Fig. 2. Sequence of Reactor Configurations at Various Times during the Excursion

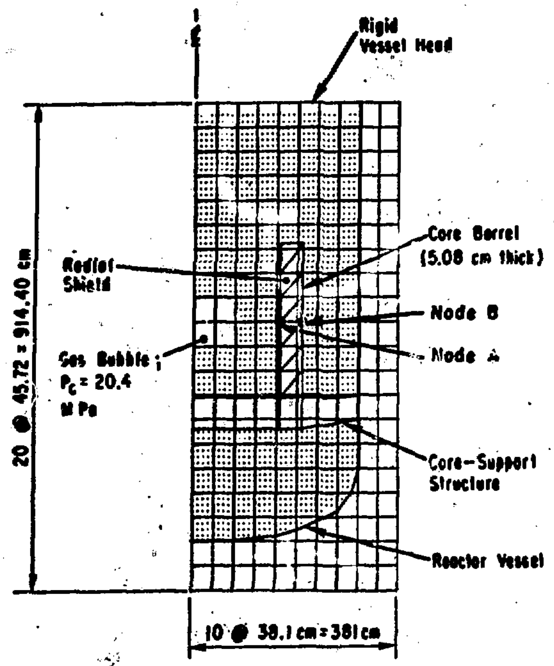

Fig. 3. Reactor Configuration in Eulerian Representation

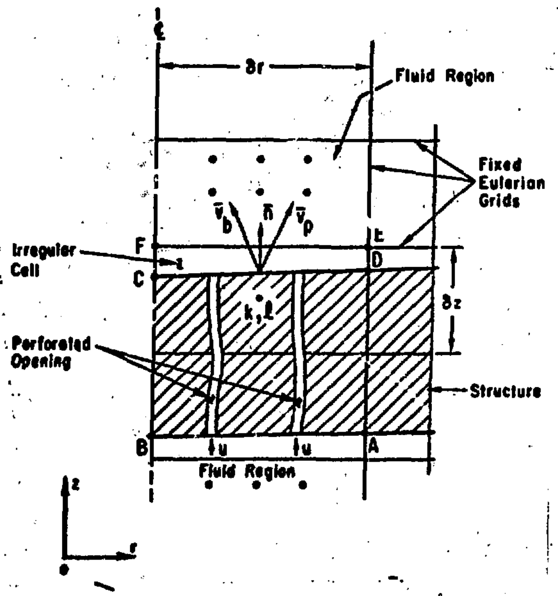

Fig. 4. Core-support Structure Perforated Openings

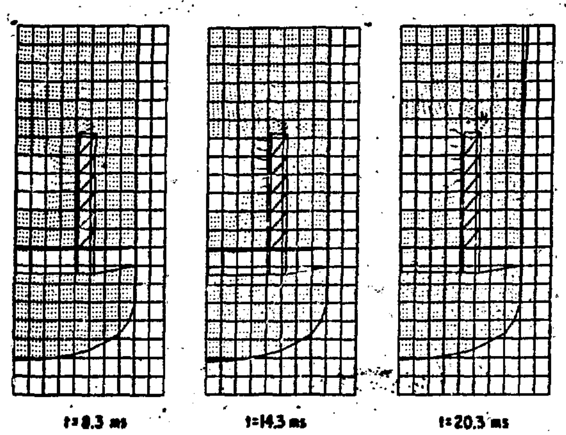

Fig. 5. Sequence of Reactor Configurations showing the Motion of Coolant through the Core-support Structure Perforated Openings

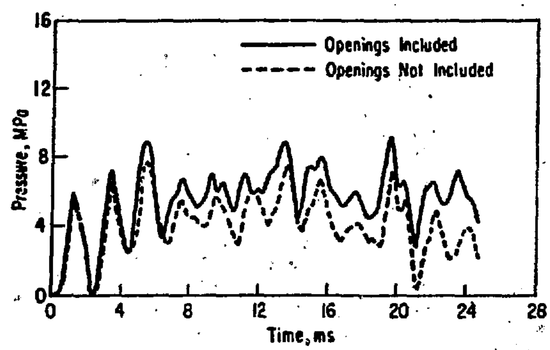

Fig. 6. Pressure Loadings in the Reactor Lower Plenum

structure can lead to an underestimation of the pressure loading in the reactor lower plenum. Of course, the degree of the underestimation is dependent on the perforation of the core-support structure.

3.1.3 Effect of Wall Opening and Sodium Spillage on Slug Impact

The effect of wall opening and sodium spillage on slug impact has been studied by Chu [22]. He applied the Eulerian code ICECO to a reactor configuration which consists of a core, a reactor vessel, a movable head and hold-down bolts, as shown in Fig. 7. For simplicity, the lower portion of the reactor was omitted in the analysis. The initial pressure in the core region was assumed to be 30 MPa. Three computations were performed: Case 1 assumed no gap opening as reactor head moved upward; Case 2 assumed that a gap opening was generated at the head vessel juncture as the reactor head moved upward; Case 3 assumed that there was a penetration hole opening in the reactor head cover in addition to the gap opening at the head vessel juncture.

Figure 8 shows the time history of the total slug force acting on the reactor head cover. As expected, the slug impact force with the penetration hole and side spillage (Case 3), as compared to that with side spillage only (Case 2) and that without spillage (Case 1), has been reduced. This reduction in impact pressures affects the collant motion, the upward motion of the reactor cover and the radial deformation of the vessel wall.

3.1.4 Response of a Piping Loop

The piping loop analyzed was a simplified primary coolant loop shown in Fig. 9, which was analyzed by Moneim [23] using the ICEPEL code. The loop consists of five pipes, two elbows, one reducer and one IHX. The flow of the primary coolant inside the IHX is simulated by the rigid zone boundaries shown by heavy lines in Fig. 9. The cross-hatched area represents the secondary fluid region.

The external walls of the pipes, reducer and the IHX are considered as deformable. The external walls of the elbows and the interior walls of the IHX are treated as rigid walls. Two step pressure pulses of 8 MPa and 4 MPa and 2 msec duration were applied, respectively, at the inlet of pipe 1 and the outlet of pipe 5 simultaneously. The system was assumed initially to be full of coolant at zero pressure.

Computed pressure traces at various locations in pipes and components are shown in Fig. 10 and 11, respectively. The wall deformation at the corresponding locations are shown in Fig. 12 and 13. In general, the pressure pulses at locations away from the pressure source are reduced by the wall deformation, if the magnitude of the pressure pulse exceeds the yield pressure of the pipe, and the plastic deformation are limited to a relative short length near the source of the pressure pulse. However, this is no longer true in the case of a piping loop having various components subjected to two pulses. As seen from Fig. 12, the radial wall deformation at node NP3 is larger than that at node NP2. This is caused by a sudden decrease of flow area in the reducer, which causes a pressure pulse to reflect back to the inlet of pipe 1. This pressure wave interacts with the incoming pressure wave, producing more plastic deformation to pipes near it as it travels back to the inlet of pipe 1.

Because two pulses are applied simultaneously at the two ends of the piping

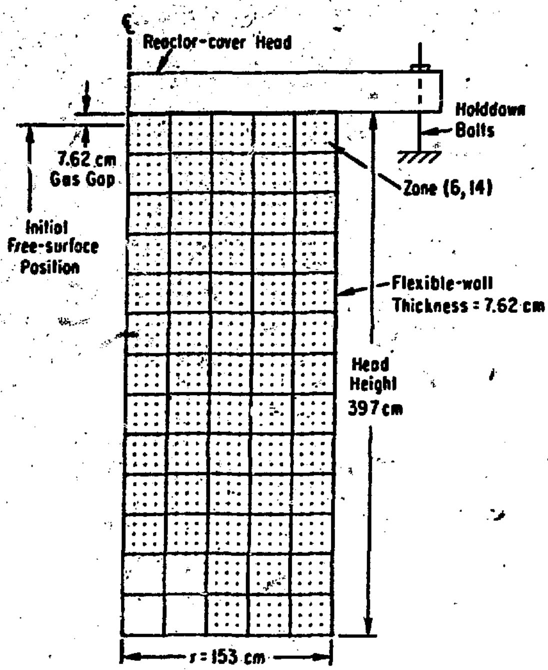

Fig. 7. A Simplified Reactor Configuration

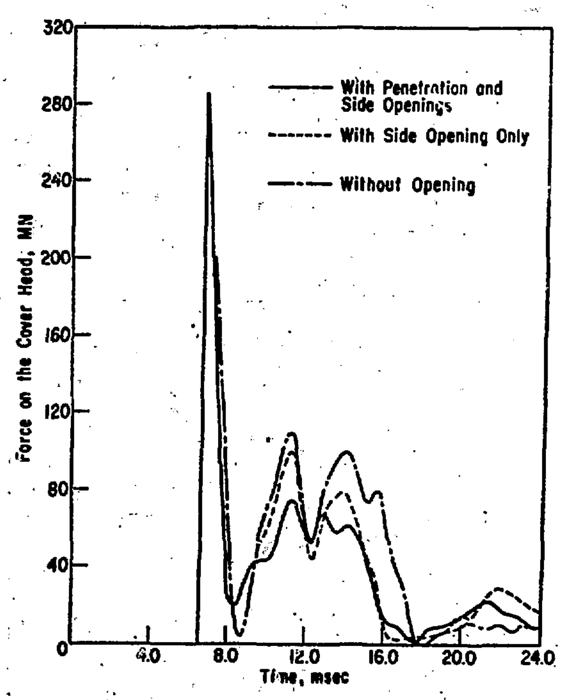

Fig. 8. Slug Force Histories on the Reactor Head Cover

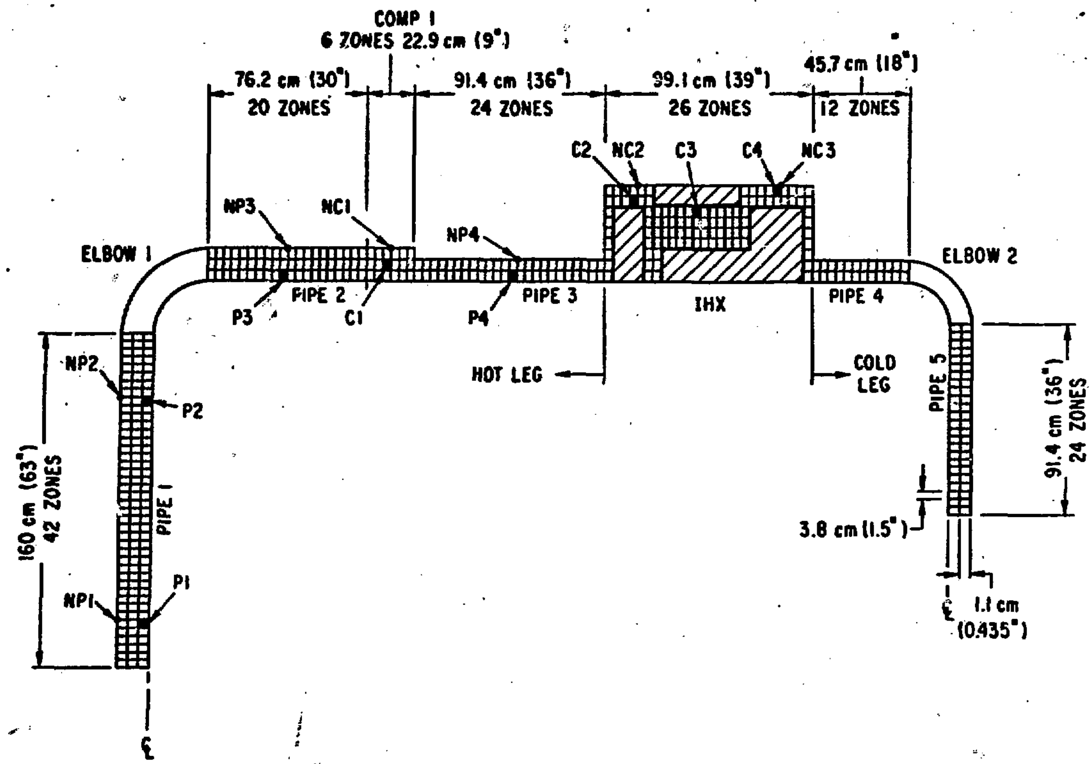

Fig. 9. A Simplified Primary Coolant Loop

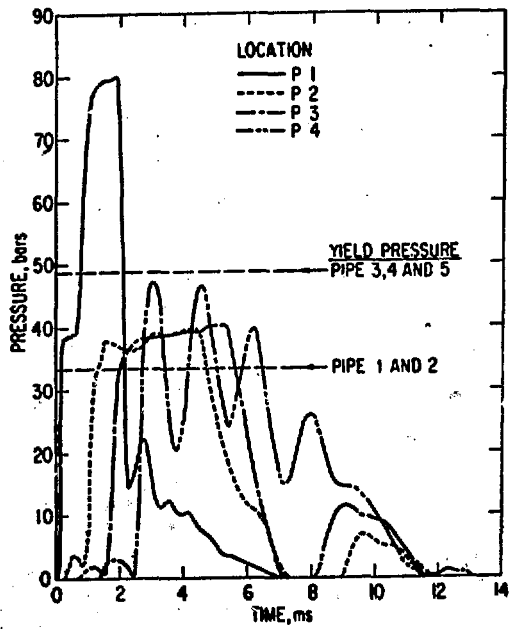

Fig. 10. Pressure Histories at Various Locations in the Pipes

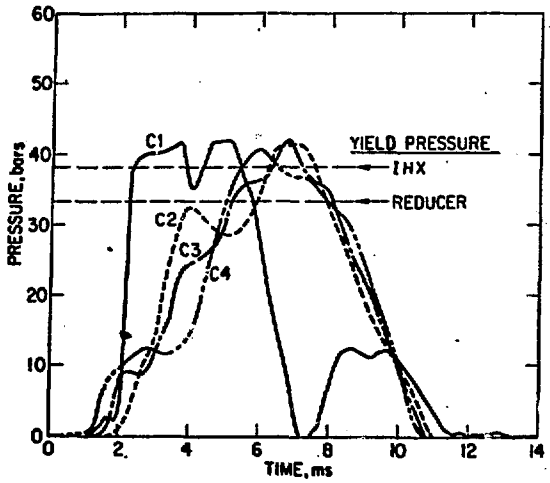

Fig. 11. Pressure Histories at Various Locations in the Components .

loop, they are bound to meet somewhere in the piping loop. As the two pulses meet, a new pressure pulse is generated which is usually large enough to cause further plastic deformation of the pipe wall. This can be seen from the pressure histories of nodes C2 and C3, shown in Fig. 11 where the second peak in those pulses is caused by the formation of a new pressure pulse when the two applied pulses meet inside the IHX.

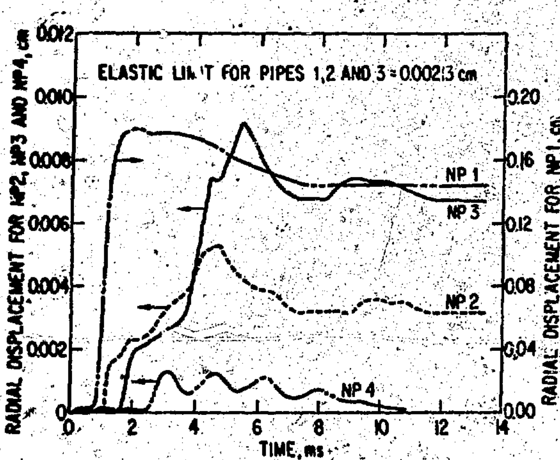

Fig. 12. Radial Displacements at Various Locations in the Pipes

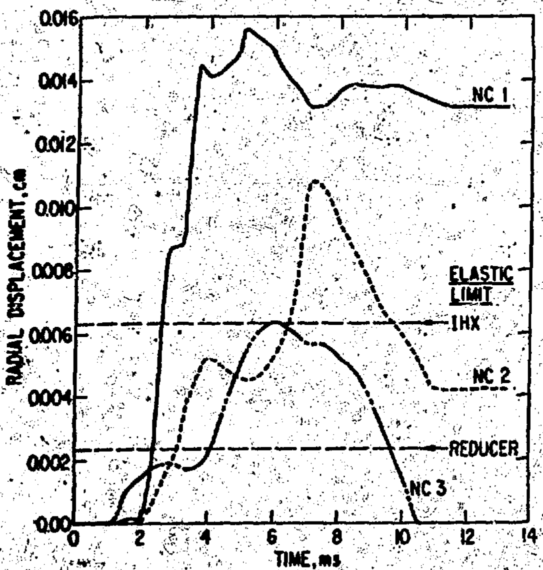

Fig. 13. Radial Displacements at Various Locations in the Components

3.1.5 Response of the Reactor Head Cover

The reactor head cover analyzed was the CRBR head cover. The calculation was performed by Kulak [24] utilizing the SADCAT code. The finite-element model of the triple rotating plug head is shown in Fig. 14. The plugs were divided into 148 triangular plate elements. The outer periphery of the large plug was treated as simply supported in the usual manner. The large-to-intermediate and intermediate-to-small plug boundaries were treated as special boundaries which permit the transfer of the transverse force from plug to plug, but not the transfer of moments or membrane forces. The coolant slug pressure force on the closure head was calculated with the REXCO code. The actual pressure is applied to the underside of the bottom shield plate.

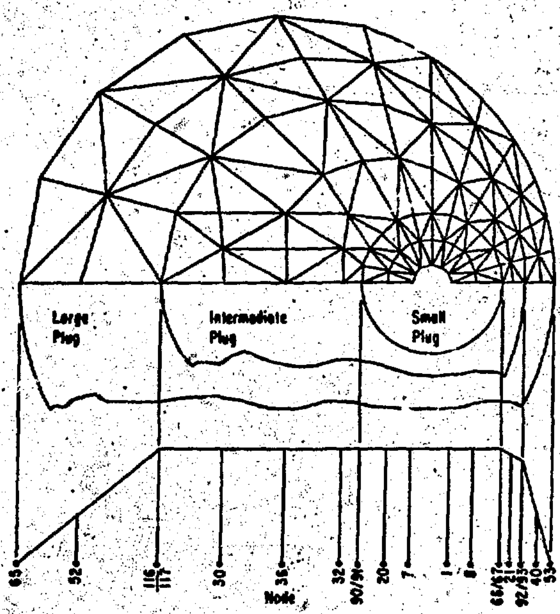

Fig. 14. Finite-element Model of the Closure Head and the Deformed Profile of the Triple Rotating Plug

Figure 14 also shows the deformed profile of the closure head along the diametral symmetry line. The cross-hatched areas in Fig. 15 indicate the finite elements that underwent plastic deformation. This illustrates that the deformations and the resulting stresses of a complex structure such as a reactor head closure can be analyzed with a structural code.

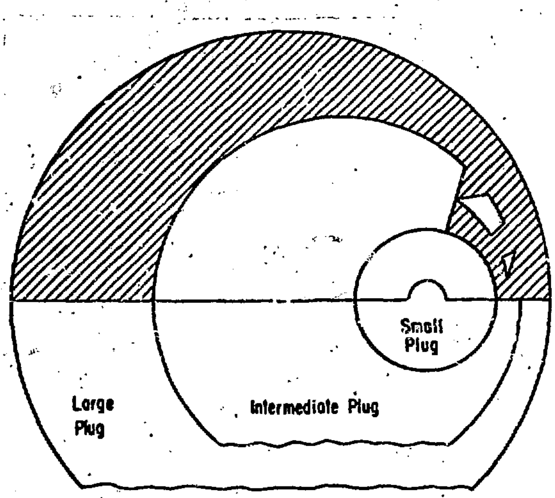

Fig. 15. Location of Plastic Regions

3.2 *Application of Codes in Parametric Studies*

C.mputer codes are ideal for performing parametric studies. They can be used to determin. the sensitivity of results to key unknowns. Such studies provide guidance for containment design as well as for experimental work. For example, in performing experiments, the sodium is often replaced by water. There, computer calculations can be performed to determine how the experimental results will be affected, if sodium is used in the actual test. The sensitivity of results to the variations of the coolant inertia and compressibility can also be determined rather easily. In the early design stage of the CRBR plant REXCO-HEP was used to perform scoping analysis to provide HCDA loads for the containment design [25]. Since REXCO-HEP scoping analysis was carried out in parallel with other calculations, the source term of the HCDA was not available at the start of the REXCO-HEP calculations. It was decided to use FFTT P-V relation of the 150 MW-sec accident as the source term of a reference case in the scoping analysis and perform a parametric study with different pressures, energy releases, and P-V relationships. The gas spacing above the sodium surface was also treated as a parameter in that study. Altogether five cases have been analyzed. The initial conditions of these five cases are given in Table 2.

The study produced enough information on the HCDA loads on various components and the response of the primary containment to HCDAs.

Currently a parametric study in energy partitioning following energetic HCDAs is being performed at ANL [26]. Numerical calculations are performed with REXCO-HEP to determine the redistribution of energy among reactor materials for increasingly energetic HCDAs. The level of energetics for each calculation will be characterized by the magnitude of reactivity ramp input into the core at prompt-critical. For a given ramp, VENUS-II [27] is used to determine the pressure vs. volume relationship for fuel vapor expansion from an oxide-fueled core which has been voided of coolant. This pressure vs. volume data is then used as input for the REXCO-HEP calculations. For each calculation, the partition of energy will be plotted as a function of time up to and including coolant slug impact on the reactor head cover.

PLBR is chosen as the reference reactor; the pressure vs. volume data for a $50/sec ramp case is taken as the base case. Calculations will then be performed for increasingly higher ramps until the extrapolated relationship between maximum coolant slug upward kinetic energy and ramp can be accurately predicted. Additional parametric calculations will be performed to determine the sensitivity of maximum coolant slug upward kinetic energy to variations of dimensions in the reference reactor model for a given ramp. The objective of these calculations is to identify reactor designs with optional inherent characteristics for limiting the potential for missile generation.

Table 2 Initial Conditions

Case No.	P-V Relation	Pressure	Work Energy MW-sec*	Gas Gap in.
1	FFTF P-V	FFTF Pressure	300	27
2	FFTF P-V	FFTF Pressure	300	51
3	FFTF P-V	75% of FFTF Pressure	225	27
4	FFTF P-V	150% of FFTF Pressure	450	27
5	$PV^{.8}$=Const.	Initial P=98.5 bar	225	27

*Based on expansion to one atmosphere pressure

3.3 *Comparison of the code Predictions with Experiments*

As stated earlier, the arithmetic operation in the computer is discrete and finite. It is necessary in the computer code to formulate the equations of interest in a discrete form, and to replace the physical system by a discretized system. Quite often, the real physical system is represented by a simplified mathematical model for obtaining a manageable solution in the computer analysis. How closely the computer solution approximates the true solution of the system is a concern of the safety analysis. If the computer codes are to be used with confidence, they must first be validated by comparison with analytical solutions and experimental data.

To establish the validity of the containment codes, extensive comparisons of the code predictions with analytical solutions and experiments have been undertaken at ANL [28,29]. It has been shown that the computer code can produce the same results as the analytical solutions of simple physical problems, such as wave propagations, fluid dynamics, and response of a shell under impulsive loading. It has been further demonstrated [30] that the computer code can predict accurately the response of the primary containment to high explosives, as well as to HCDAs, and can give guidance as to the effects of such accidents in a fast reactor system.

However, the answer obtained from the computer analysis depends upon the basic information and mathematical model used in the computer analysis. Even if the computer code has been proven to be perfect, the use of a improper mathematical model can lead to erroneous results. For example, the radial sheilds in a reactor are made of segmented plates. Although they have no tensile strength in the circumferential direction, they exhibit considerable tensile strength in the radial and axial directions. Therefore, they can not be represented by the hydrodynamic material. Modeling of the shield as hydrodynamic material can lead not only to underestimation of the slug impact loading on the reactor head and the upper vessel wall, but also to erroneous profile of the core barrel deformation.

To illustrate this, 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. 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. The profile of the core barrel deformation is shown in Fig. 16, where the mode of deformation in the two models is quite different.

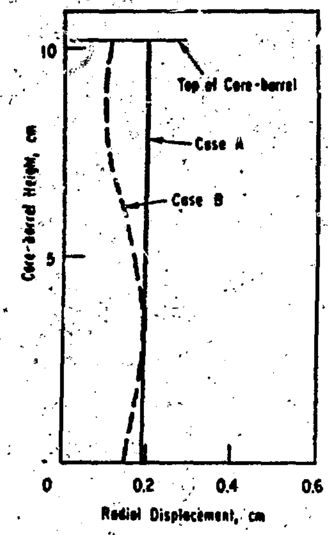

Fig. 16. Profile of the Core Barrel Deformation

Therefore, to verify the mathematical model and to understand the behavior of individual components and their effects on the primary system, experimental tests are needed. It is believed that through the analyses involved in making the comparisons, insights can be gained into the phenomena. These insights can lead to improvements in the mathematical model representations in the computer codes. It should be noted that mathematical models, no matter how sophisticated they are, are only as good as their capability of predicting results which are in agreement with the experimental data. Recently SRI has performed some tests such as rigid vessel test, simple vessel test with internal radial shield [31] and flexible pipe test [32], which were specifically designed for the purpose of validating the mathematical models used in ANL's REXCO-HEP, ICECO, and ICEPEL codes.

4. Concluding Remarks

The containment codes developed for the safety analysis have advanced to the point of being able to predict reasonably well the response of the primary system. The phenomena of the HCDA can be followed up to the time of dynamic equilibrium in the system. The amount of sodium spillage into the secondary containment can also be reasonably predicted.

In order to be able to analyze the entire sequence of events which occurred in an HCDA, one should use both the Lagrangian and Eulerian codes. Lagrangian codes should be used for the determination of the pressure wave propagation, the response of the reactor internals, and the deformation of the reactor vessel wall during the phase immediately following the accident. The excursion phenomena such as slug impact, sodium spillage, and core gas-bubble expansion and migration during the latter phase of the excursion must be analyzed with an Eulerian code. The response of a piping loop must also be analyzed with an Eulerian code.

Once a computer code has been validated by comparing with experimental tests, it can be used with confidence to perform analysis on reactor containments and piping loops. Future tests are needed only for the understanding of behavior of individual components and their effects on primary systems, so that appropriate mathematical models can be formulated for these components. It is believed that extensive explosive testing programs will not be necessary to assess the containment capability of reactor systems.

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References

[1] Y.W. Chang and J. Gvildys, REXCO-HEP: A two-dimensional computer code for calculating the primary system response in fast reactors, ANL-75-19, (June 1975).

[2] C.Y. Wang, ICECO-An implicit Eulerian method for calculating fluid transient in fast reactor containment, ANL-75-81 (December 1975).

[3] H.Y. Chu, Y.W.Chang, and S.H.Fistedis, MICE: A two-dimensional containment code for treating multifield fluid flow with interpenetrations, Trans. Am. Nucl. Soc.24 (November 1976)p.277.

[4] M.T.Abdel-Moneim, ICEPEL, a two-dimensional computer program for the transient analysis of a pipe-elbow loop, ANL-75-35 (July 1975).

[5] M.S.Cowler, ASTARTE-a 2-D Lagrangian code for unsteady compressible flow, theoretical description, AWRE-44-91-37 (March 1974)

[6] M. Stievenart et al.,Analysis of LMFBR explosion model experiments by means of the SURBOUM-II code, E 3/5, Trans. 3rd Int'l Conf. on Structural Mech. in Reactor Technology, London, UK. (September 1975).

[7] H. Lauber, Private communication

[8] J. Donea, S. Giuliani, and J.P. Halleux, Prediction of the nonlinear dynamic response of structural components using finite elements. T 2/4 Proc. Extreme Load Conditions and Limit Analysis Procedures for Structural Reactor Safeguards and Containment Structures, Berlin, Germany (Sept.1975).

[9] PISCES 2DL finite difference equations, 2DL-3, Physics International Co., San Leandro, CA (October 1971).

[10] M.L. Wilkins, Calculation of elastic-plastic flow, UCRL-7322, Rev. 1 (1969).

[11] G.J. DeSalvo and J.A. Swanson, ANSYS engineering analysis system user's manual, Swanson Analysis Systems, Elizabeth, PA. (1975).

[12] B.J. Thorne and W. Herrmann, TOODY, a computer program for calculating problems of motion in two dimensions, SC-RR-66-602 (July 1967)

[13] T.B. Belytschko, WHAM, waves in hysteretic arbitrary media and structures, Dept. of Materials Engineering, Univ. of Illinois at Chicago Circle (Feb. 1974).

[14] R. Hofmann, "STEALTH" a Lagrange explicit finite-difference code for solids, structural, and thermo-hydraulic analysis, EPRI NP-260 (August 1976).

[15] S.L. Thompson, CSQ-a two-dimensional hydrodynamics program with energy, flow and material strengths, SAND 72-D112, Sandia Laboratories (August 1975).

[16] J.M. Kennedy, Nonlinear dynamic response of reactor-core subassemblies, ANL-8065 (Jan. 1974).

[17] A.H. Marchertas and T.B. Belytschko, Nonlinear finite-element formulation for transient analysis of three-dimensional thin structures, ANL-8104 (June 1974).

[18] F.H. Harlow and A.A. Amsden, A numerical fluid dynamics calculation method for all flow speeds, J. Comp. Phys.8 (1971) p. 197.

[19] S.L. Hancock, Finite difference equations for PISCES-2DELK, a coupled Euler Lagrange continuum mechanics computer program, TCAM 76-2 Physics International Co. San Leandro, CA (April 1976).

[20] Y.W. Chang and J. Gvildys, On the modeling of reactor internals in the HCDA analysis, E 2/b, Trans. 4th Int'l. Conf. on Structural Mech. in Reactor Technology, San Francisco, CA (August 1977).
[21] C.Y. Wang, Analysis of sodium flow through the core-support-structure openings, Reactor Development Program Progress Report, ANL-RDP-57 (Jan. 1977) p. 6.14.
[22] H.Y. Chu, A quasi-Eulerian method for analyzing slug impact and coolant spillage in a fast reactor accident, ANL-RAS 76-14 (March 1976).
[23] M.T.A-Moneim and Y.W. Chang, Dynamic response of LMFBR primary coolant loop components to pressure pulses, ASME paper 76-PVP-52 (1976).
[24] R.F. Kulak, Head-cover analysis, Reactor Development Program Progress Report, ANL-RDP-57 (Jan. 1977) p. 6.35.
[25] Y.W. Chang, J. Gvildys, and J. Bratis, Primary system response of the demonstration plant to hypothetical core disruptive accidents (a parametric study using REXCO codes), ANL-RAS 73-41 (Dec. 1973).
[26] J. Bratis, Private communication
[27] J.F. Jackson & R.B. Nicholson, VENUS-II: An LMFBR disassembly program, ANL-7951 (September 1972)
[28] J.E. Ash and R.T. Julke, Comparison of a two-dimensional hydrodynamics code (REXCO) to excursion experiments for fast reactor containment, ANL-7911 (Jan. 1972)
[29] G. Nagumo and C. Fiala, Comparison of FFTF simple model tests with REXCO predictions, ANL-8071 (Feb. 1974).
[30] T.J. Marciniak et al., Analysis of FFTF primary containment complex model experiments, ANL-8062 (Jan. 1974).
[31] D.J. Cagliostro and C.M. Romander, Experiments on the response of rigid and flexible reactor vessel models to a simulated hypothetical core disruptive accident, SRI report, Stanford Research Institute, (Nov. 1976).
[32] C. M. Romander and D. J. Cagliostro, Experiments on the response of flexible piping systems to internal pressure pulses, SRI report, Stanford Research Institute (April 1977).