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Nuclear Reactor Safety

October 1—December 31, 1980

Los Alamos
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

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October 1—December 31, 1980

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NUCLEAR REACTOR SAFETY

Compiled by

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ABSTRACT

Development of the fast-running Transient Reactor Analysis Code (TRAC) version (PF1) continued during the quarter with numerical improvements and addition of a stratified-flow model. Independent assessment of the detailed version (PD2) continued with several Loss-Of-Fluid Test (LOFT) small-break tests, a PKL reflood test, and five Marviken critical-flow tests. Analysis efforts in the 2D/3D project concentrated on detailed investigations of Cylindrical-Core Test Facility (CCTF) Core I tests and calculated flow oscillations in the primary loops of the German pressurized water reactor (PWR). Investigations were completed of PWR transients involving emergency feedwater unavailability. Other Light-Water Reactor (LWR) safety progress included the use of the three-dimensional version of the SALE code to study hot-leg injection into the upper plenum and the effect of guide tube cross section on momentum flux.

Efforts in Liquid-Metal-Cooled Fast-Breeder Reactor safety included studying transition-phase phenomena in an SNR-300-type reactor geometry using SIMMER and performing Upper Structure Dynamics experiments to examine rupture disk performance.

In High-Temperature Gas-Cooled Reactor (HTGR safety, improvements were made to the Composite High-Temperature Gas-Cooled reactor Analysis Program (CHAP) code, and system transients in the Fort St. Vrain reactor were calculated. Other work in this area included thermal stress analyses of core support block response during firewater cooldown following a loss-of-forced-circulation accident.

Tests were run on steel cylinders to determine the effects of the Area Replacement Method on buckling strength as part of the Structural Margins-to-Failure program. In addition, a literature review was completed of models and experiments to determine damping and stiffness of reinforced concrete structures.

I. INTRODUCTION

This quarterly report summarizes technical progress from a continuing nuclear reactor safety research program conducted at the Los Alamos National Laboratory. The reporting period is from October 1 to December 31, 1980.

This research effort concentrates on providing an accurate and detailed understanding of the response of nuclear reactor systems to a broad range of postulated accident conditions. All of the funding for the work reported here is provided by the US Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research.

The report is organized according to project. Major sections deal with Light-Water Reactors (LWRs), Liquid Metal-Cooled Fast-Breeder Reactors (LMFBRs), High-Temperature Gas-Cooled Reactors (HTGRs), and Reactor Structural Margins Research.

II. LIGHT-WATER REACTOR SAFETY RESEARCH

Light-water reactor safety research reported in this section includes Transient Reactor Analysis Code (TRAC) development, assessment, and applications; severe accident sequence analysis; thermal-hydraulic research for reactor safety; and video probe system development.

A. TRAC Independent Assessment

(T. D. Knight, Q-9)

During the quarter we continued to analyze Loss-of-Fluid Test (LOFT) Facility, Primarkreislaufe (PKL) Facility, Marviken, and Semiscale tests. For LOFT, pretest predictions were completed for the small-break pumps-off/on Tests L3-5 and L3-6, and posttest analyses were run for small-break Tests L3-1 and L3-7. A posttest analysis for PKL Test K9 (International Standard Problem 10) was completed. The analyses that were initiated in the previous quarter of five Marviken critical flow tests were completed. We have commenced analyses of five of the Semiscale Mod-3 pumps-on/pumps-off tests.

1. LOFT Small-Break Calculations

(T. D. Knight, Q-9)

This quarter we concentrated on producing pretest predictions for Tests L3-5 (early pump trip) and L3-6 (delayed pump trip). Also, posttest analyses were made for Tests L3-1 and L3-7. The posttest analyses and the pretest predictions were made with the same code (TRAC-PD2) and the same basic input deck.

For small-break tests in LOFT, the performance of the secondary system is more important than it is in the large-break tests. Therefore, the secondary system, consisting of the steam-generator secondary and steam line to the isolation valve, was represented with more fluid cells. The new nodalization scheme provides recirculation in the steam-generator secondary and a more representative secondary liquid inventory. A new valve type was added to the code to describe the opening and closing characteristics of the steam flow control valve. A coarse nodalization scheme represented the break orifices, and the additive friction factor was used to adjust the flow based on a critical flow model.

The overall LOFT system model was modified to represent known leakage paths. In particular, the reflood-assist bypass lines were connected, and the steady-state flow was approximately 6% of the total intact-loop flow. Also, the heat losses from the primary system to the environment were included in the input model.

The calculated results for Tests L3-1 and L3-7 have been compared to data.[1,2] The system depressurizations (Figs. 1 and 2) are in good agreement with the data, and the pressurizer empties at the correct time. For Test L3-1 the accumulator injection begins and ends at the correct times. The comparisons for the steam-generator secondary pressures are sensitive to, among other things, the amount of leakage permitted through the steam flow control valve. The calculated break flow is, on average, representative of the experimental data, and the system-pressure comparisons indicate that the proper magnitude of the flow is calculated. The liquid level in the vessel always remains above the core, and the core remains well cooled.

Preliminary data comparisons for Tests L3-5 and L3-6 indicate that the code-predicted system depressurization was too rapid. The differences in depressurization rates were probably a result of overpredicting the flow at the break in both tests; however, the break flow for Test L3-1, a similarly sized break, was good. The timing of events in the calculations was compressed because of the more rapid pressure decay. TRAC correctly indicated a larger depletion of liquid inventory in the primary system for the pumps-on test than for the pumps-off test.

2. Comparisons with PKL Reflood Test K9

(B. E. Boyack, Q-9)

An experiment conducted in the PKL facility at Erlangen, West Germany, is one element in the TRAC-PD2 assessment program. The PKL facility was designed to simulate the reactor core and primary cooling system of a 1300-MW pressurized water reactor (PWR). The reactor coolant system is represented by three loops, one of which simulates the broken reactor coolant loop during a postulated loss-of-coolant accident (LOCA). One of the remaining loops has a double capacity and simulates two intact reactor coolant loops. Elevations of the electrical core and steam generators simulate PWR elevations. Hydraulic cross sections are scaled by the ratio of rod simulators in PKL to fuel rods in the 1300-MW reactor. The core contains 340 electrical rods, of which 337 are heated.

PKL Test K9 simulated the gravity refill and reflood phases in a PWR following a double-ended guillotine break in one cold leg with subsequent cold-leg injection of emergency cooling water. Test K9 has been designated as International Standard Problem Number 10.

Calculated and measured[3] temperature histories of a rod located near the center of the core are shown in Fig. 3. Although other test rods in the same general core region exhibited slightly different characteristics during the test, the results are representative. The calculated results are from TRAC-PD2 modified to correct an error in the gravity head term in the axial momentum equation for the three-dimensional vessel component. TRAC calculated the reflood phenomena in the lower-to-middle core regions quite well. Calculated peak cladding temperatures exceed observed values by 30 to 50 K. The calculated time for the peak cladding temperature in the lower core occurred 20 s later than observed in the test. Calculated times for the cladding temperature peaks in the middle and upper core are within the rod-to-rod variations observed in the test. A significant deviation is the failure to predict the early top-down quench observed in the test. TRAC did not predict quenching at the top of the core during the calculated 375-s transient.

The pressure history in the upper plenum of the test vessel is shown in Fig. 4. Although the correct trends are exhibited during the first 100 s of the transient, the pressure is underpredicted when condensation occurs during the early phase of coolant injection. The pressure is then overpredicted following refill as the lower core begins to quench.

In summary, the comparison of the TRAC-PD2 calculation and the experimental results shows that TRAC correctly predicts the qualitative features of the test and also closely predicts many of the quantitative features. TRAC does not predict the top-down quench in the upper core. Also, the predicted rates of condensation and evaporation appear too large for the gravity refill and reflood phenomena examined in PKL Test K9.

3. Comparisons with Marviken Full-Scale Critical Flow Tests

(M. S. Sahota, Q-9)

TRAC-PD2 calculations were made to simulate the blowdown behavior for five different Marviken full-scale critical flow tests. The test matrix included the use of nozzles varying in nominal diameter from 0.2 m to 0.5 m, and in length-to-diameter ratio from 0.3 to 3.7. Tests were conducted at initial pressures in the range of 4 to 5 MPa with conditions varying during the tests from 50 K subcooled to low-quality saturated fluid.

Test 4 was selected as a base case for the study, and Tests 13, 20, 22, and 24 were used for further comparisons. The tests selected for the study were based primarily on their differences in (1) nominal initial subcooling in the vessel, (2) nozzle diameter, and (3) nozzle length-to-diameter ratio. Table I summarizes the initial and boundary conditions for the tests selected for data comparisons.

Figures 5 and 6 show mass flow and pressure comparisons for Test 4, which represents the typical qualitative behavior for all the tests. Also shown in these figures are the TRAC-P1A calculations, which are virtually identical to TRAC-PD2 results. The two data curves for the mass flow present rates derived from velocity (pitot static) measurements and vessel differential pressure measurements. The pitot static data curve is valid throughout the transient, whereas the vessel differential pressure curve is valid only after approximately 5 s. The pitot tube data have a smaller uncertainty band after 5 s than those for the vessel differential pressure; uncertainty bands are therefore superimposed on pitot static data only.

The calculated mass flow agrees very closely with the initial peak, underpredicts the subcooled blowdown by about 15%, and agrees very well with the saturated part of the blowdown. The mass flow underprediction during subcooled blowdown is thought to result from the code not predicting correctly nucleation delay in the vapor generation. Rapid flashing of the liquid causes choking to occur at a lower mass flow. The agreement between the prediction and the data for the system pressure is excellent except during the first 3 s when the dip in the experimental data is not calculated, again because of delayed nucleation. After the dip, the code underpredicts the pressure by an average of about 50 kPa, which is within the data uncertainty.

To summarize for the other tests, the calculated pressures are generally within about 100 kPa of the data except during the first 5 s when the code does not calculate the pressure dip. The maximum deviations in the fluid temperature calculation are about ±5 K. The mass flows are generally calculated within the data uncertainty for the saturated blowdown and underpredicted by about 15% during the subcooled blowdown. In general, the code results are better for the larger length-to-diameter ratio nozzles, as flashing nonequilibrium effects are less important.

B. TRAC Code Development and Assessment

(D. R. Liles, Q-9, and J. H. Mahaffy, Q-8)

Emphasis was placed this quarter on developing the fast running version, PF1. Much of the work is a continuation of the testing and debugging of the effort that was initiated the previous quarter. Several numerical improvements were made to PF1 and a horizontal stratified-flow model was added. The code support effort was directed toward the release of PD2 and its conversion to the VAX computer.

1. PF1 Numerical Improvements

(J. H. Mahaffy, Q-8)

A number of changes were made to PF1 to improve its performance. The logic for changing donor cells if the velocity changes sign during a Newton iteration was upgraded. The tee momentum source terms were changed to make them analogous with those in PD2. The previous finite difference form of the $V\nabla V$ term could introduce substantial erroneous momentum sources in transient calculations when the change in void fraction was large from one cell to the next. Terms to correct this effect were added to the code.

2. Horizontal Flow Model for PF1

(D. R. Liles, Q-9)

In the horizontal portions of reactor piping, fluid stratification can occur during small-break LOCAs. A package generally based on A. Dukler's analysis[4] was implemented in PF1 to provide the capability to handle this stratification more accurately. The procedure consists of calculating a transition velocity based on a Froude number. If the vapor velocity is less than 80% of this transition reference, then the interfacial shear and wall shear are calculated using the Blasius correlation. In the range 80 to 100%, an interpolated shear is used; if the vapor velocity exceeds the reference velocity, then the standard dispersed or churn correlations are employed. Initial testing with a model of the German hot-leg injection nozzle has shown that the predictions definitely are affected by the inclusion of this model.

3. Code Support

(R. P. Harper, M. R. Turner, and P. J. Giguere, Q-9)

Distribution packages for the official release version of TRAC-PD2 were prepared and sent to 21 users in the US and overseas. In addition to the source listing for TRAC, the magnetic tape provided in the package contained graphics postprocessors, utility programs, sample problems, and the TRAC-PD2 assessment problems. Documentation contained in the package included a draft of the TRAC-PD2 User's Manual, a graphics postprocessor User's Manual and Programming Guide, and comments regarding the release version of the TRAC-PD2 computer program.

The graphics postprocessor, TRAP, was improved to allow production of plots that are more suitable for publication. When run in the FORMAL mode, TRAP will use a label look-up file to obtain descriptions for variables that are being plotted. In addition, several alphabets, subscripting, superscripting, and Greek and mathematical symbols are now available. Finally, a new feature was added to TRAP making it possible to override any default labels including the title and subtitle.

Progress continued this month on converting TRAC-PD2 to run on the VAX computer. A single-precision version was completed and successfully tested. Work is now under way to create a double-precision version. Future work in this area includes implementation of graphics and dump/restart capabilities and improving efficiency.

A study was begun this quarter on what can be done to improve the input for TRAC-PF1 in terms of user convenience. Possible improvements include free-format input, input preprocessors, dump editors, and the capability to override any variables already initialized after restarting from a dump file. Work was also begun on improving the dump/restart capability in TRAC-PF1.

C. TRAC Applications to 2D/3D

(K. A. Williams, Q-8)

The work described in this section includes the application of TRAC to full-scale LWR transients and to the large-scale German and Japanese 2D/3D experiments. During this quarter we began a detailed investigation of the refill phase of the Cylindrical Core Test Facility (CCTF) Core-I tests. This effort identified a significant discrepancy between the emergency core coolant (ECC) flow rate reported in the Quick-Look Report and that given by reading the data tape. We are repeating CCTF Runs 12 and 20 using the actual experimental data for the ECC flow rates. We have also begun a thorough investigation into the sensitivity of code models on calculated ECC flow and pressure oscillations in a German PWR. This study considers not only the sensitivity to constitutive models, such as condensation rates, but also the fluid model formulation; that is, drift-flux vs two-fluid. We are also reviewing pertinent experimental data through a subcontract with Creare, Inc. The results of this study will have direct bearing on the Upper Plenum Test Facility (UPTF) design and operation.

1. CCTF Refill Analysis

(T. H. Brown, Q-8)

Analysis of CCTF test data has revealed discrepancies regarding the amount of ECC water injected from the accumulator into the primary system and the amount of water mass addition to the vessel. These discrepancies became apparent from a TRAC calculation for Run 12 that showed a much greater downcomer filling rate than observed in the test. This is depicted by Figs. 7 and 8.

A comparison of the water mass addition to the vessel with the injected ECC water mass is shown in Figs. 9 and 10 for Runs 12 and 20, respectively. The curve labeled ECC (JAERI) is calculated from the average flow rates and durations [5,6] provided by the Japan Atomic Energy Research Institute (JAERI), and the curve labeled ECC (Flow Meters) is calculated from data taken from flow meters in the accumulator line and cold-leg ECC injection lines. The former has been used for ECC flow in TRAC calculations up to this time. Water mass additions in the various regions in the vessel are calculated from vertical differential pressure data. The curve labeled Vessel is the sum of all the water masses in the various regions of the vessel. Because there is no downcomer bypass during this time period (indicated by no change in the water level data in the containment tank), there should be a reasonable match between the ECC mass injected and the mass addition to the vessel. However, the plots indicate that at the termination of accumulator flow, the ECC mass injected is several hundred kilograms greater than the mass addition to the vessel. If the ECC mass injected is in error by this amount, then in the TRAC calculation, the water level in the downcomer after termination of accumulator injection may be too high by a few meters.

The source of this discrepancy will be investigated further. Possible sources include error in the flow meter measurements, air or vapor in the ECC lines, and reverse flow of ECC water into the loop seals. For Run 12 there is a significant discrepancy between the ECC injection estimated by JAERI and the ECC injection as determined from the ECC flow meter data. This discrepancy probably results from JAERI's method of estimating the average ECC flow rates and durations from the actual flow meter data. For all subsequent TRAC posttest analyses we will use the flow meter data.

2. Analysis of Flow and Pressure Oscillations in the German PWR (GPWR)

(M. W. Cappiello and F. E. Motley, Q-8)

TRAC-PD2 calculations predicted large flow and pressure oscillations in the primary loops of the German PWR during a cold-leg LOCA. The oscillations are caused in the following manner. A water plug forms in the hot leg at the point of ECC injection, condensation effects cause the plug to move toward the steam generator, the plug is partially vaporized, and is expelled back toward the vessel. The pressure spikes caused by the plug movement range as high as 8 bars over the nominal.

In the actual GPWR hot legs, ECC water is forced toward the vessel by an internal pipe. The pipe exit is about one meter from the vessel; the internal pipe flow area is about 10% of the hot leg. Because ECC water is introduced into the hot legs in such a manner, actual plug formation was questioned. A study was conducted to investigate the TRAC-PD2 results in more detail.

Results show that the one-dimensional drift-flux formulation used in TRAC-PD2 nonvessel components accurately inputs the momentum of the injected ECC water. However, interfacial drag and heat transfer areas are calculated based on a flow regime of annular or mist flow because there is a high void fraction in the pipe. Thus, the momentum is dispersed by the large drag and any small resistance to flow will cause plug formation. To model correctly the phenomena involved, a more detailed formulation is required. The results of this sensitivity study with the drift-flux formulation were documented in a technical memorandum (LA-2D/3D-TN-81-1). Currently a TRAC two-fluid model with a stratified-flow constitutive package is being used as the next step in our investigation.

3. Sensitivity of SCTF Calculational Model Execution to Certain TRAC Input Parameters

(S. T. Smith, Q-8)

Statistical correlation methods were used to determine the sensitivity of the TRAC model for the Slab Core Test Facility (SCTF) to certain TRAC input parameters to optimize the calculations projected for FY 1981. By means of nonparametric statistical methods, it was determined that reasonable convergence for minimum running time can be attained with a specific set of input timing and convergence criteria, thus effecting a savings in the anticipated cost of these calculations.

4. Comparison of TRAC Calculational Results for SCTF and the Reference GPWR During Reflood

(S. T. Smith, Q-8)

A calculation through reflood and full rod quenching was performed with the TRAC-PD2 model for the SCTF. The model was intended to simulate a typical German pressurized water reactor with combined hot-leg and cold-leg ECC injection. The results of this calculation were compared with a similar calculation for the German PWR. There was good overall agreement, with similar phenomena observed in both calculations for such parameters as vessel liquid inventories, mass flow rates, upper plenum subcooling, and fluid temperatures and pressures.

D. Severe Accident Sequence Analysis (SASA)

(R. D. Burns, Q-7)

Quarterly progress included the completion of investigations of PWR transients involving emergency feedwater unavailability at Zion Station (a large four-loop Westinghouse PWR). The investigation provided a technical basis for procedures to deal safely with feedwater unavailability. This helps fulfill the goal of the Laboratory SASA effort to investigate strategies for dealing with accidents involving unanticipated multiple equipment failures. The basic understanding of system behavior obtained through the Zion feedwater transient investigations also provides considerable insight into general accident behavior at Zion and other reactors.

Simulations of specific transients were performed with a modified version of TRAC-PD2. Key results of the investigations involved evaluations of strategies for dealing with feedwater unavailability. Guidelines for the application of "feed-and-bleed" were developed.

E. TRAC Calculational Assistance and User Liaison

(C. E. Watson, J. L. Elliott, and D. E. Lamkin, Q-7)

A formal report was prepared on the complete loss-of-feedwater (main coolant pumps on and no operator intervention) accident for a Westinghouse four-loop plant. A further calculation (with operator intervention) also was completed. In this calculation, the following actions were taken.

1. The steam generator atmospheric relief valves were fully opened at about 1 min and remained open thereafter.
2. The high-pressure part of the ECC system (two high-charging pumps plus two high-pressure injection pumps) was activated and the pressurizer relief valve opened and remained open.

This combination of operator actions would lead to recovery of the system from this accident. Other procedures are being evaluated that also would allow the system to recover under more extreme conditions.

Two problems were investigated under user liaison during this period. The first involved a simulation of a CREARE test with fully implicit (as opposed to semi-implicit) break pipe to decrease the run time. However, the result was a slower calculation with many more iterations. We believe the problem can be traced to a very slow convergence rate caused by unexplained variations in the Jacobian of the system. The problem is being investigated further. The second problem we investigated involved a mass conservation error in the ACCUM module. In this problem, approximately 17% more liquid is injected than is present. Modification of the relative velocity correlation in this module corrects the error. Several minor user questions were also answered. One means of communicating these solutions is by the TRAC Newsletter; a draft of the next TRAC Newsletter was prepared.

F. Thermal-Hydraulic Research for Reactor Safety Analysis

(B. J. Daly, T-3)

Research efforts during the past quarter were concentrated in the following areas.

1. Development of a critical flow model for TRAC, providing a limit to the flow velocity and an estimate of mass loss as a result of a small break in a pipe.

2. Analysis of German Standard Problem 4 using the SOLA-LOOP network code to provide a blind calculation of the effect of valve closing on flow in a discharge pipe.

3. A study of condensation processes when a subcooled liquid jet is injected into a transverse flow of saturated steam, and comparison with experiment.

4. A study of the effect of guide tube cross section on momentum flux in the UPTF during hot-leg injection.

5. Adaptation of the three-dimensional version of the SALE code to study hot-leg injection of ECC into the UPTF facility.

The discussion below focuses on the latter two areas of interest.

In the previous study[7] of hot-leg injection of emergency core coolant into an upper plenum, it was found that numerical predictions of liquid mass flow through the bottom of the upper plenum exceeded the deduced experimental measurements.[8] A possible cause of the greater liquid mass flow through the bottom boundary of this part of the upper plenum in the numerical calculations was an excess transfer of horizontal to vertical momentum as a result of using a stair-step representation of the circular guide tubes. This explanation was supported by the observation that the calculational cells that showed the greatest vertical liquid mass flow were located on the front face of the columns.

To test the theory that the use of a stair-step representation of a round cylindrical column results in excess transfer of horizontal to vertical momentum, we performed a study of water flow past a single cylindrical column of stair-step and circular cross section in an air environment. We also examined flow past a square column as part of this study.

The results of this study showed qualitative differences in liquid flow distributions and vertical liquid mass fluxes when using circular and noncircular cross-section columns. In the case of the circular column, much of the water remained attached to the column as a film, whereas in the noncircular cases considerable water was deflected laterally from the front face of the column. This resulted in different distributions in liquid mass flux through the bottom boundary for the different column cross sections. For the stair-step cross-section column, most of this liquid flux was concentrated in the stair-step region in one corner of the column. This was consistent with the observations of the earlier study. In the square-column calculation the vertical liquid flux was more uniform along the front face of the obstacle. In the calculation of flow past the circular cylinder, the vertical liquid mass flux was much more uniformly distributed, with the maximum occurring in the wake of the obstacle. The total mass flux through the bottom of the upper plenum for the circular cross-section column was about 60% of that in the noncircular cases.

As a result of this study, it was determined that improved accuracy could be obtained using a numerical procedure that permitted a more nearly circular column cross section. Therefore, we have been adapting the three-dimensional version of the SALE code[9] to these upper plenum studies. Figure 11 shows a velocity plot of ECC injection from a hot leg into the upper plenum obtained with the SALE code. In addition to providing a better representation of the column cross section, this code also permits calculation over a broader region of the upper plenum.

G. Video Probe Systems Development

(V. S. Starkovich, Q-8)

Two video probe systems are currently under development as part of the multinational 2D/3D experimental program. The video probe system is a self-contained instrument designed to provide visual observation of flow regimes and two-phase flows during simulated LOCA tests by means of a miniature television camera packaged to permit operation in a high-temperature, high-pressure (625 K, 40 bar), steam-water environment. The two systems currently under development consist of two probes to be installed on SCTF and one probe plus a lighting fixture to be installed on the PKL facility.

The pressure boundaries of the two video probes to be sent to Japan are undergoing modifications necessary to meet ASME code requirements. The material acquisitions, ASME code analysis, and mechanical drawings have been completed for the alteration procedure, and the design changes have been approved by JAERI. The alterations are expected to be completed by mid-February, with pressure boundary certification tests to follow. Installation is scheduled for early April.

Modifications to the electronic support system for SCTF are also under way. These modifications will permit JAERI personnel to switch the support electronics system from the two SCTF probes to the three CCTF probes by the use of front-panel jumper cables and switches only.

One of the SCTF video probes was installed into a test vessel to obtain realistic video tapes of flow phenomena within a simulated SCTF end box. Additional experiments were also performed to examine potential lighting problems that may arise in future probe designs. A Laboratory report summarizing the status of the SCTF probe systems is in preparation.

A single video probe system along with a separate dc lighting fixture will be provided to PKL. Both the probe and the lighting fixture will be installed in the upper core support plate region of the facility. The probe head will consist of five windows; four for lighting (both dc and strobe) and one for viewing. The instruments are designed according to ASME code requirements for operation in a 40-bar, 575-K steam-water environment.

III. LIQUID-METAL-COOLED FAST-BREEDER REACTOR SAFETY RESEARCH

Efforts in LMFBR safety research were concentrated on a study of transition-phase phenomenology in an SNR-300-type reactor geometry and on performing a large number of Upper Structure Dynamics (USD) experiments to examine rupture disk performance.

A. Transition-Phase Phenomenology in SNR-300

(W. R. Bohl, Q-7)

In connection with travel to the Kernforschungszentrum, Karlsruhe, West Germany (KfK), a brief study was performed on transition-phase phenomenology in an SNR-300-type reactor geometry. With the assistance of the KfK staff, the SIMMER-II code was upgraded to be consistent with the September 1980 version at the Los Alamos National Laboratory and a sample comparison produced results identical to those obtained at Los Alamos. The SIMMER-II code was then modified further to better simulate the expected response of the SNR-300 reactor (with a Mark-la core) to an unprotected loss-of-flow (LOF) accident.

A SAS3D calculation was used to provide initial conditions to SIMMER-II. Transformation of the SAS3D configuration to SIMMER was accomplished with the assistance of a KfK-modified version of an Argonne National Laboratory (ANL) code, which converts the Lagrangian mesh to an Eulerian mesh. Two mechanistic transition-phase cases then were calculated successfully with SIMMER-II.

These cases suggested that in a LOF accident with this SNR-300 design, significant fuel removal from the core might occur before formation of a whole-core pool. In this situation, recriticality in a whole-core pool mode would be difficult. Such fuel removal could result either from early core disassembly or by a more benign fuel removal process. These cases also suggested that plausible calculational sequences eventually could be constructed without the accident terminating in an energetic recriticality. In addition to the mechanistic cases, there was some examination of an open pool with sufficient fuel for recriticality. Such a pool was found to be quite sensitive to the initial conditions assumed.

B. Upper Structure Dynamics Experiment

(V. S. Starkovich, Q-8, and D. Wilhelm, KfK)

Approximately 40 USD tests were performed using helium and propanol to examine the performance of different rupture disk designs under different operating conditions. The tests also provided information for SIMMER-II comparisons. A rupture disk test series was necessary to determine the cause of the rupture disk problems that we were experiencing. Before these experiments the disks would fail in one of two ways--either the disk would break at a pressure considerably less than its design pressure, or the disk would fail to petal fully. The problem was particularly acute for low-pressure propanol tests.

The results of the test series indicated that a two-petal, prescored, rupture disk would open properly for both helium and propanol under the desired combinations of pressures, temperatures, and void fractions provided that a mechanical breaking mechanism was used to initiate the rupture. Consequently, a number of breaking mechanisms were examined to determine their suitability for use in the USD experiment.

The use of an electrical solenoid was unsuccessful. The selected solenoid was unable to impart enough impulse to the disk petals to make them open properly. The most reliable method was found to be an explosively driven rod striking the disk from below with sufficiently rapid momentum transfer to ensure full petalling. An explosively driven actuator has been fabricated for use on the USD experiment and is now in the initial testing stage. If successful, the new mechanism will ensure full petalling of the rupture disk under all operating conditions and will assist in controlling experiment initiation for better synchronization with high-speed cameras.

In addition to providing information needed for the design of the new rupture-disk-breaking mechanism, the successful helium simulant tests provided information that was useful for studying the fluid dynamic models within SIMMER-II. Figure 12 presents a comparison of typical measurement and SIMMER-II prediction of the transient pressure in the USD experiment core following breakage of the rupture disk. The measured pressures from successive experiments generally agree to within ±3%. Because the experiment used helium only, it was considered to be nearly adiabatic gas expansion without condensation. The good agreement between experiment and analysis suggests that the fluid dynamic models within SIMMER-II are adequate for such applications. Comparisons of SIMMER-II predictions and measurements for several propanol experiments are in progress. A paper summarizing these comparisons has been submitted to the June 1981 meeting of the American Nuclear Society, to be held in Miami Beach, Florida.

IV. HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY RESEARCH

Progress in high-temperature gas-cooled reactor (HTGR) safety is reported in this section including (1) CHAP code development and its application to system transients in the Fort St. Vrain reactor and (2) thermal stress analyses of core support block response during firewater cooldown following a loss-of-forced-circulation accident.

A. Simulation and Accident Analysis

(K. R. Stroh and J. F. Kotas, Q-9)

Our primary concern in this task is the development, verification, and application of Gas-Cooled Reactor consolidated plant simulation computer programs. The Composite HTGR Analysis Program (CHAP) consists of a model-independent systems analysis mainframe named LASAN and model-dependent linked modules, each representing a component, subsystem, or phenomenon of the overall HTGR plant model. The Los Alamos Systems Analysis (LASAN) code has steady-state, transient, and frequency-response solution capabilities. The standardized modular structure of LASAN/CHAP facilitates modification of component models, modification of solution algorithms, and the addition of new features. An initial version (CHAP-1) modeled the 3000-MW(t) HTGR. A version modeling the Fort St. Vrain (FSV) HTGR is being developed (CHAP-2).

During the reporting period, work on CHAP/LASAN proceeded on several fronts. Comparison of CHAP-2 simulations with FSV data and subsequent model refinement continued. Work on code documentation is accompanying the model refinement. Two improvements were made in LASAN that aid the CHAP development effort. The LASAN "dynamic" steady-state option is now operational in CHAP-2. This option attempts to follow a dynamic path to steady state with ever-increasing virtual time steps (up to 10^6 s/step). This option has been compared with the previously used transient-to-steady-state method for the HTGR model, and yields comparable results with greatly reduced computing times.

The model for the general proportional plus integral plus differential (PID) controller in LASAN has been improved. An error that had been unduly limiting the time step was identified and corrected. Modeling of the voltage saturation limits of the various operational amplifiers has been added. In addition, the PID subroutine was modified to allow the code user to take manual control of a controlled variable. Manual control can be accomplished through the symbolic input package, or by variable replacement during interactive execution. The controller model exhibits continuous action over the automatic-manual-automatic sequence. This feature enhances our ability to initialize a transient properly, and to simulate transients for an off-normal plant or those with operator intervention.

The adjustment of parameters to achieve various FSV steady-state conditions is continuing. Only those parameters that are not well known experimentally are being adjusted. Deviations in steady-state values are now less than 5% for all CHAP-2 primary system variables. This effort will continue, but we are now concentrating on transient simulations and refinement of the dynamic aspects of the model. Simulations are in progress for both slow and rapid depressurization accidents and for a control rod withdrawal-at-power accident. These simulations have led to refinement of the modeling for the Plant Protective System and the Operational Protective System.

LASAN is now operational on an AMDAHL computer at the Gesellschaft fur Reaktorsicherheit (GRS) in the Federal Republic of Germany (FRG). A GRS representative will visit Los Alamos during February to evaluate the latest version of CHAP, and to assess its applicability to the German High-Temperature Reactor. As of this writing, LASAN is operational at five other installations. We expect to use the AMDAHL version of LASAN as a basis for conversion of CHAP-2 to the VAX 11-780 during the next quarter.

B. Core Support Block Thermal Stress Analysis

(T. A. Butler, Q-13)

Before a meeting held with the NRC on November 7, 1980, we performed several transient analyses for a 72% power case to determine the response of the core support block (CSB) during firewater cooldown following a loss-of-forced-circulation accident. As expected, the predicted thermally induced stresses were much lower for the transient case than for the previous steady-state analyses. This is principally because of the large mass, and therefore long time constant, of the CSB. A second factor that lowered the stress considerably was the use of new thermal conductivity data supplied by General Atomics Company.

For the baseline case, using data that were generated by the Oak Ridge National Laboratory (ORNL) with the ORECA computer code and assuming cyclic symmetry, we predicted maximum principal stresses at the keyway location of 260 psi. Additional transient runs were made to determine how this value varied with input parameters such as radial heat flow and convection coefficients.

The result was that the maximum possible stress is approximately two times the nominal value of 260 psi.

We reported these results at the November 7, 1980, meeting with the NRC. Subsequent discussions at this meeting revealed that ORNL had not been using the latest thermal conductivity data for their ORECA calculations. The new thermal conductivity is a factor of three higher than that used in all previous calculations. ORNL was asked to repeat their calculations for the 72% power case and perform similar calculations for a 105% power case with the new thermal conductivity data. We agreed to perform calculations for the 105% power case using the new ORECA predictions and, based on the results, possibly repeat the calculations for the 72% power case.

Results of the ORECA calculations were, as expected, that the thermal gradients across the core support structure were significantly lower than previously predicted. At the same time, the radial heat flow between adjacent CSBs is greater with a different time profile (Fig. 13). Because our previous parameter studies showed that the thermal stresses at the keyway are relatively sensitive to the radial heat flow, the differences in the time profiles shown in Fig. 13 are of some concern.

We have not yet performed complete transient calculations with the new ORECA data. Extrapolations based on our previous parameter studies do indicate that if all six sides of a CSB conduct heat at the rates shown in Fig. 13, the stresses might exceed minimum PGX graphite strength at the keyway. However, the ORECA results also indicate that the heat flow from the CSBs is highly asymmetric. We can approximate the effect of the asymmetries by applying data generated during some of our initial two-dimensional calculations. Those calculations indicate that typical asymmetries will lower the maximum stresses by approximately 40%, which would make the maximum principal stresses at the keyway acceptable. We are presently performing new two-dimensional calculations that use asymmetries predicted in the most recent ORECA calculations. These, along with complete three-dimensional transient calculations, will be performed for both 72% and 105% power cases.

V. REACTOR STRUCTURAL MARGINS RESEARCH

Progress in the Structural-Margins-to-Failure program is reported in this section. The primary objective of this program is the determination of the failure load of certain nuclear plant structural components. In the first part of the program, Category-I structures and structural systems (exclusive of the containment building) are examined for their response to seismic, missile, and blast loads. In the second part, the static and dynamic buckling of steel containments is examined.

A. Structural Margins to Failure--Category-I Structures Program

(E. Endebrock, R. Dove, C. A. Anderson, E. Ferdinand, Q-13, and R. Bartholomew, Q-15)

After discussions with the peer review panel members (held in Knoxville, Tennessee, on September 16, 1980), the tentative experimental plan submitted in September 1980 has been revised. The revised experimental plan is being incorporated into a project status report, which includes all phases of this project. This report will be forwarded to the Structural Engineering Research Branch of the NRC and to selected peer review panel members during January 1981.

The review of the literature on the use of models and the experimental determination of damping and stiffness of reinforced concrete structures has been completed. The results of this review had a major impact on the experimental program proposal for this project.

As part of our planning of an experimental program, correspondence was initiated with three Japanese construction companies that are active in nuclear power plant construction. As of January 1, 1981, we had received replies from the Technical Research Institute of Ohbayashi-Gumi, Ltd. and the Kajima Institute of Structural Technology. Reports on their experiments on reinforced concrete structures, which they have supplied, will be most helpful; however, neither of these construction companies has done much experimental work on massive shear walls. Both of these construction companies expressed interest in the results of our work.

An analytical solution for the frequency response of a damped two-degree-of-freedom structural system was developed to aid in the design of test structures. An example of the results from this solution is shown in Fig. 14. Using this solution we will perform parametric studies to gain insight into the behavior of two-degree-of-freedom damped structural systems.

An investigation was conducted into the analytical difficulties encountered when a generalized damping mechanism is postulated for Category-I structures. A rough draft report[10] was written that includes a survey of current analytical generalizations of damping, together with an assessment of the utility of such generalizations to the program at hand. The inherent limitations of nonlinear, nonproportional, energy dissipative functionals of both the position and velocity coordinates when subjected to both deterministic and stochastic excitation forces are also assessed. It appears (from the results of the survey) that significant generalizations of energy dissipative (damping) mechanisms have not been made and one must rely on numerical methods involving the solution of coupled nonlinear differential equations. The method of linear superposition does not hold for nonlinear systems, therefore, the computer time becomes enormous when the structural configuration consists of many elements and each problem has to be solved individually. It may be possible that specific computer codes can be developed for certain classes of problems involving specific nonlinear damping repeated at a discrete number of places within a composite structure.

B. Structural Margins to Failure--Containment Buckling

(J. G. Bennett, T. A. Butler, R. C. Dove, and J. A. Romero, Q-13)

Testing of steel cylinders to determine the effect of the ASME Area Replacement Method (ARM) on buckling strength has been completed and a report is in preparation. The initial tests verified the conclusion[11] that the buckling strength of a cylindrical shell under an axial load will be increased if the ARM rules are followed. Later tests indicated that this conclusion must be qualified. The data for the completed test series are shown in Fig. 15.

When a penetration is introduced into a cylindrical shell, the effect of the penetration on the buckling strength depends not only on the size of the penetration, but also upon the buckling strength of the unpenetrated shell. If the buckling strength of the unpenetrated cylinder is so low that the penetration does not lower it further, then reinforcing the penetration can have little or no effect on the buckling strength. If, however, the shell is of such a quality that by introducing the penetration, the buckling strength is lowered, then reinforcing the penetration may benefit the buckling strength and may even increase it over that for the unpenetrated shell as reported in Ref. 11.

Steel containments are in the class of shells commonly defined as "fabricated" shells. A fabricated shell is defined here as a cylinder constructed by normal rolling and welding shop practices to normally specified engineering tolerances. Because no imperfection limits are specified, a fabricated shell can be expected to have considerably larger magnitude imperfections when compared with laboratory specimens that are carefully made by machining, milling, or electroforming techniques. Consequently, steel containments will exhibit buckling strengths in accordance with the above discussion.

Based on our experimental results, we have reached the following tentative conclusions for shells having a radius to thickness (R/t) ratio of 460 and a penetration (of radius r) characterized by the ratio of $r/\sqrt{Rt} \leq 3.6$.

1. For a fabricated cylinder that has a low ratio of actual buckling load to classical buckling load (P/P_C, the "knockdown" factor), the axial buckling load will be reduced only slightly or not at all by the introduction of the penetration. Reinforcing the penetration by the ARM will have little or no effect on the buckling strength of these shells. In this case, the margin to failure can only be ensured by a conservative knockdown factor such as suggested by the ASME code case N-284 (Ref. 12).

2. Reinforcement of a penetration in accordance with the ASME ARM should ensure that, if the buckling strength of the shell is not governed by the initial imperfections, the effect of the penetration will be reduced by the reinforcement and the margin to failure will be increased above the value ensured by the use of a conservative knockdown factor.

A meeting was held on November 20, 1980, at Chicago Bridge and Iron (CB I) for the purpose of establishing the guidelines for computer code bench mark buckling experiments that will be carried out this fiscal year. CB I, NRC, Los Alamos, and the Lockheed Missiles and Space Company (LMSC) were represented. Several points were generally agreed upon. The models should be representative of containments, thus in the R/t range of about 500. Ring stiffeners with spacing representative of typical containments should be used and the material should be mild steel with a yield strength of about 207 MPa (30000 psi). The rings should be large enough to force inter-ring buckling and thus the imperfection sensitivity of the structure should be reduced. Loading configurations were discussed, as well as methods for testing in such a manner as to obtain more than a single test per model. Models will be designedand tested that have penetrations that interrupt no rings, a single ring, and multiple rings.

LMSC has also sent a letter to Los Alamos outlining a proposed bench mark ring-stiffened cylinder model to be tested under axial load. This model does not have a penetration. This design is under study to determine if it is representative of a containment-like shell structure.

TABLE I

MARVIKEN TEST DESCRIPTIONS

Test No.	Nozzle Straight Section Length (m)	Nozzle Diameter (m)	Length-to Diameter Ratio	Initial Pressure (MPa)	Initial Subcooling Near Vessel Bottom (K)	Initial Subcooling At Nozzle Inlet (K)	Initial Water Level[a] (m)	Initial Elevation of Transition Zone[a,b] (m)	Test Period (s)
4 (Baseline)	1.500	0.509	2.95	4.94	37	63	17.59	8.0-10.5	49
13	0.590	0.200	2.95	5.09	31	95	17.52	7.0-10.0	148
20	0.731	0.500	1.46	4.987	7	77	16.65	0.5-2.5	58
22	0.731	0.500	1.46	4.93	52	95	19.64	15.5-17.0	48
24	0.166	0.500	0.33	4.96	33	83	19.88	15.5-17.0	54

[a]Water level referenced to the bottom apex of test vessel.

[b]The transition-zone level is defined as the elevation below which water is subcooled.

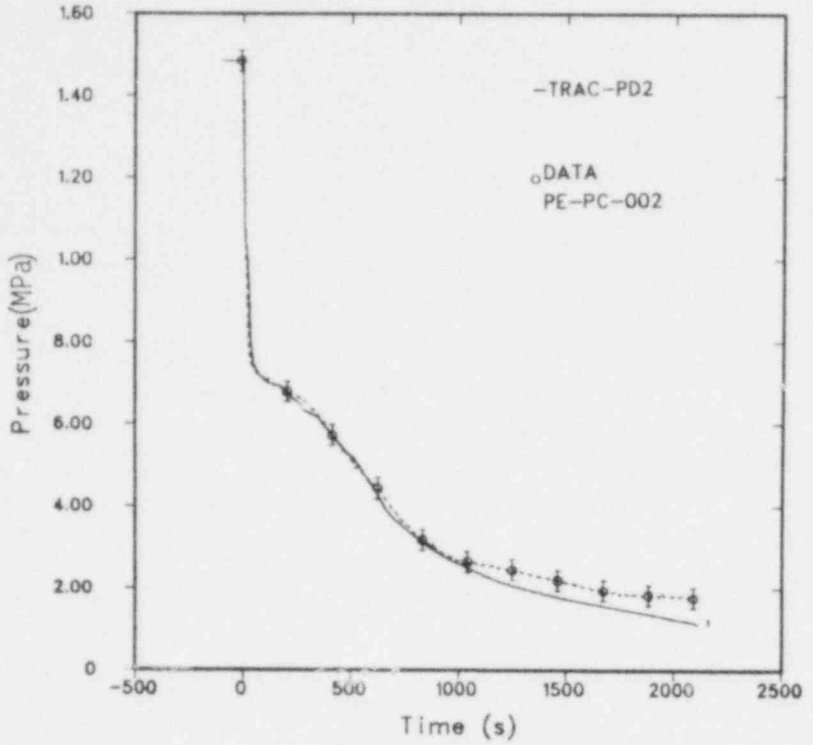

Fig. 1 Comparison of the TRAC-PD2 calculation to the intact-loop hot-leg pressure for LOFT Test L3-1.

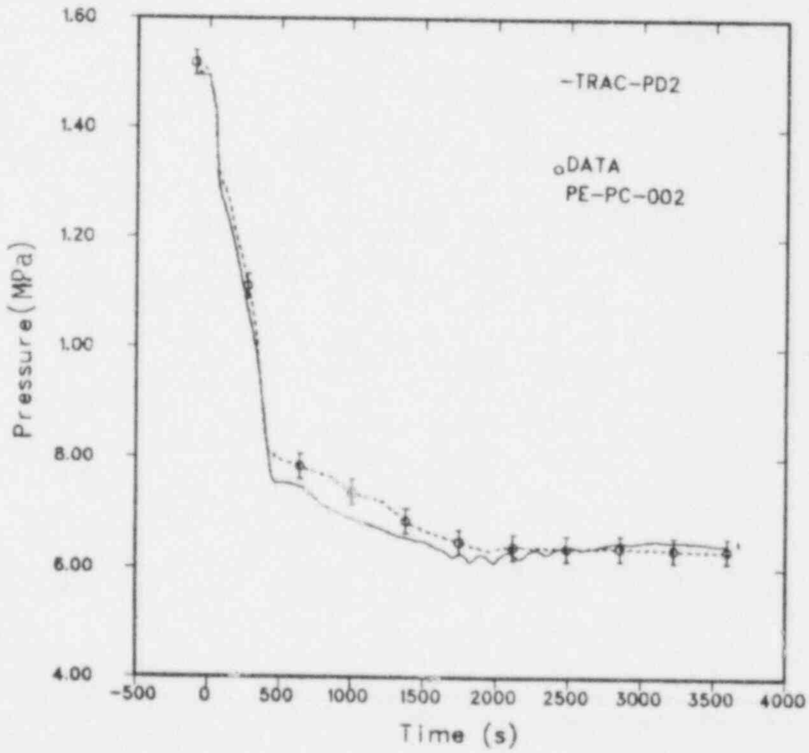

Fig. 2. Comparison of the TRAC-PD2 calculation to the intact-loop hot-leg pressure for LOFT Test L3-7.

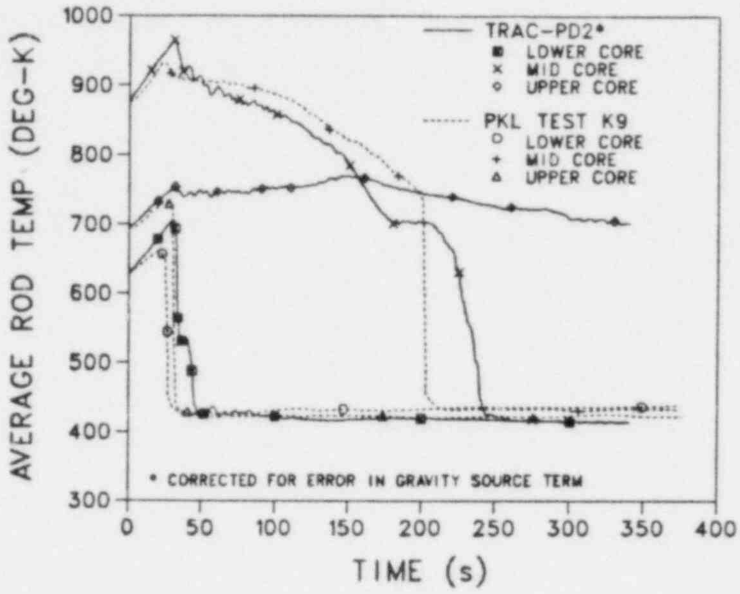

Fig. 3. Cladding temperatures for PKL Test K9.

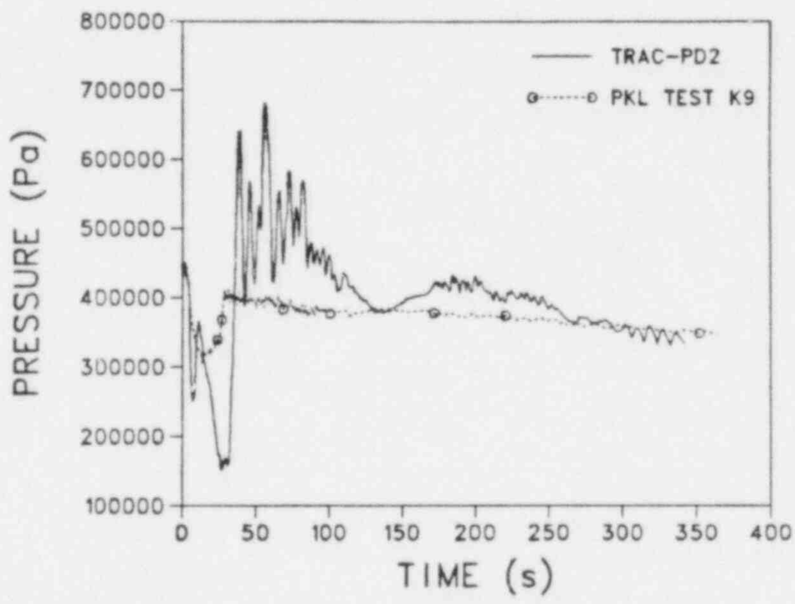

Fig. 4. Upper plenum pressure for PKL Test K9.

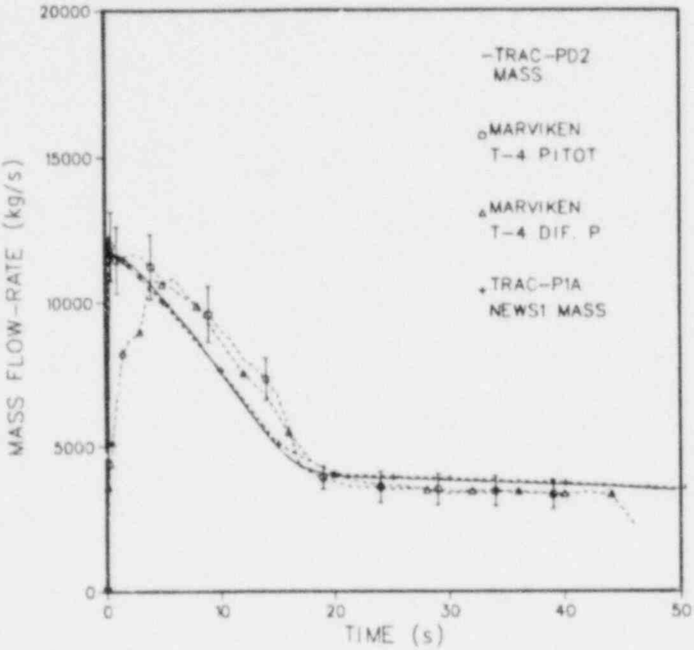

Fig. 5. Comparison of experimental nozzle mass flow for Marviken Test 4 with TRAC PD2 and TRAC-P1A calculations.

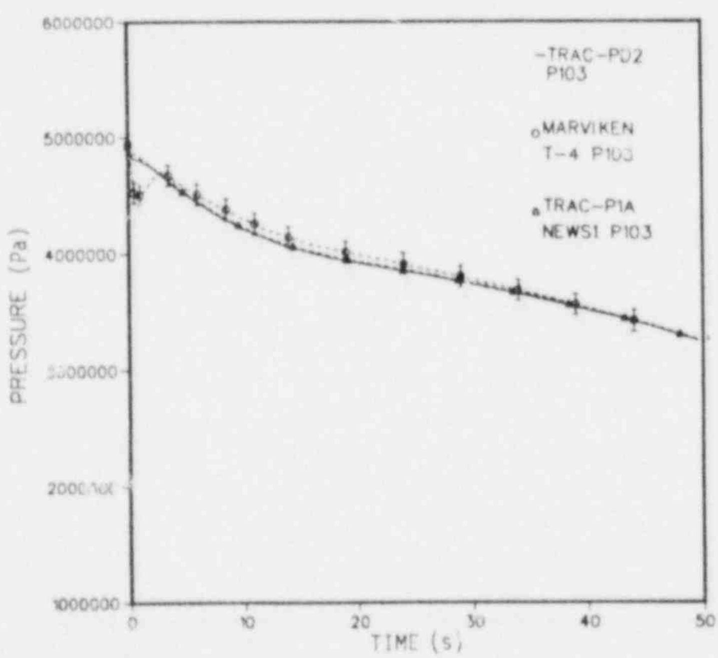

Fig. 6. Pressure comparisons of TRAC-PD2 and TRAC-P1A calculations with the experimental data near the vessel top for Marviken Test 4.

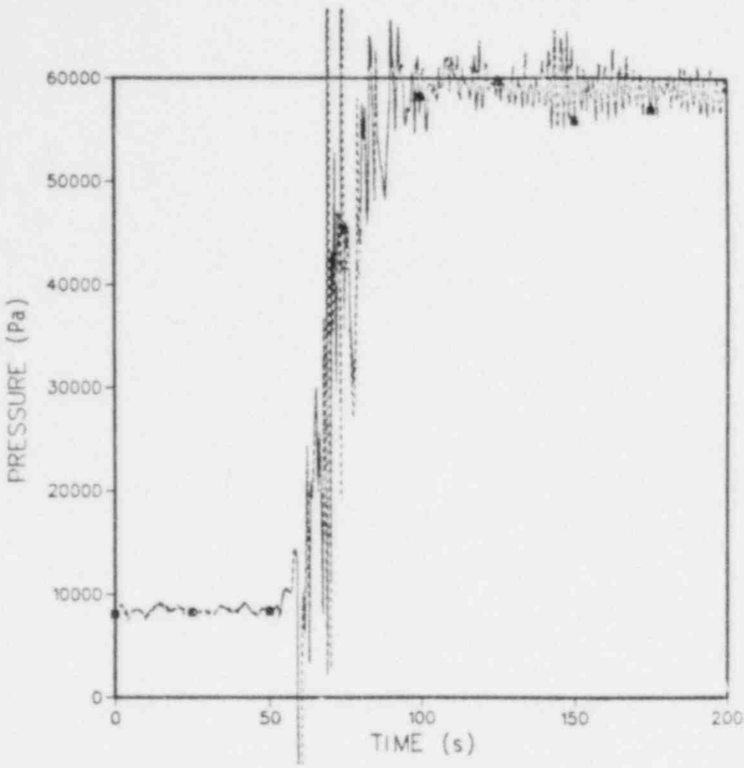

Fig. 7. Pressure difference in downcomer between bottom of vessel and level of loops calculated by TRAC at four azimuthal locations for CCTF Test C1-03 (Run 12). Accumulator injection into lower plenum is estimated at 52 s, switchover to cold legs is estimated at 70 s, and termination of the accumulator injection is estimated at 75 s.

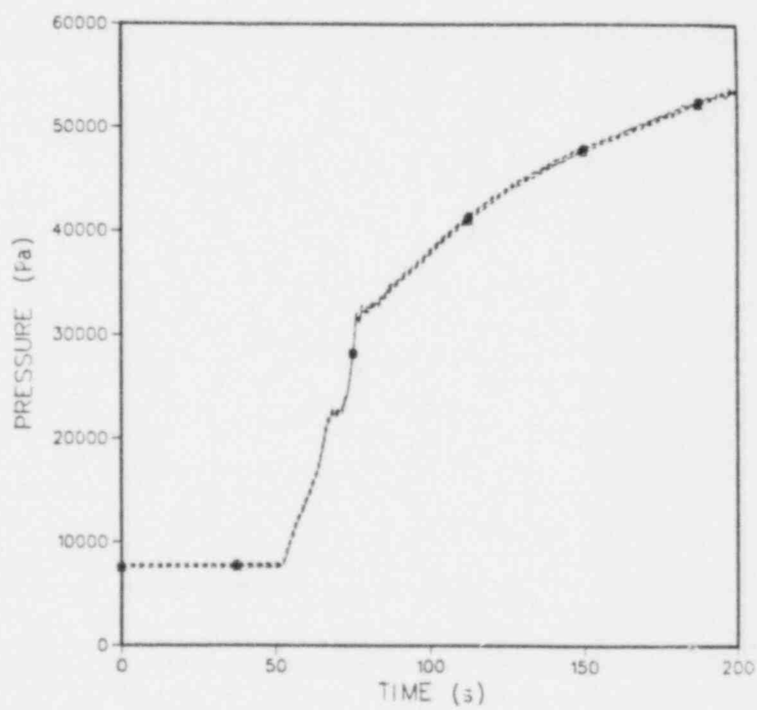

Fig. 8. Pressure difference in downcomer between bottom of vessel and level of loops calculated from test data at four azimuthal locations for CCTF Test C1-03 (Run 12). Accumulator injection into lower plenum is estimated at 51 s, switchover to cold legs is estimated at 70 s, and accumulator injection termination is estimated at 75 s.

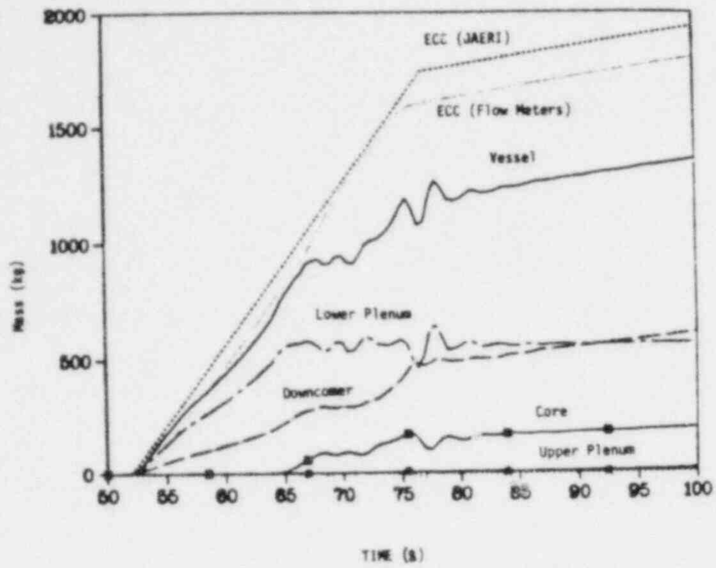

Fig. 9. Water masses added to primary system calculated from test data for CCTF Test C1-03 (Run 12). Accumulator injection into lower plenum is estimated at 53 s, switchover to cold legs is estimated at 62 s, and accumulator injection termination is estimated at 72 s.

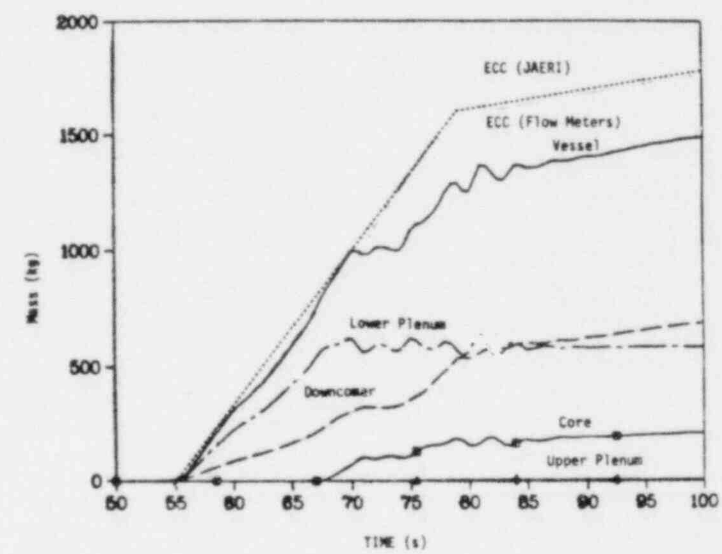

Fig. 10. Water masses added to primary system calculated from test data for CCTF Test C1-11 (Run 20). Accumulator injection into lower plenum is estimated at 55 s, switchover to cold legs is estimated at 73 s, and accumulator injection termination is estimated at 79 s.

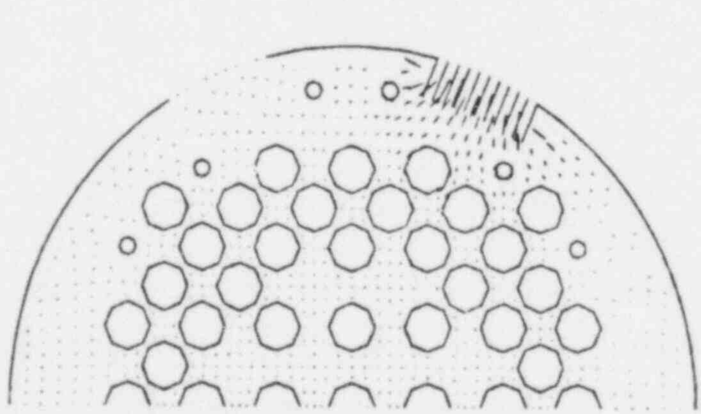

Fig. 11. A velocity plot showing liquid injection from a hot leg into the upper plenum for UPTF facility.

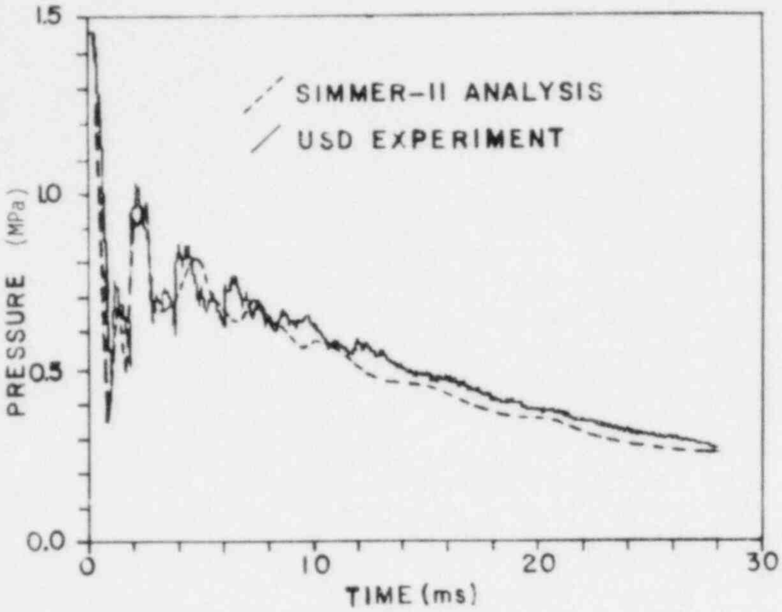

Fig. 12. Comparison of measurement and SIMMER-II prediction of core pressure during a USD experiment using helium gas. The rupture disk breaks at time zero and the piston impact occurs at 28 ms, corresponding to sodium slug impact.

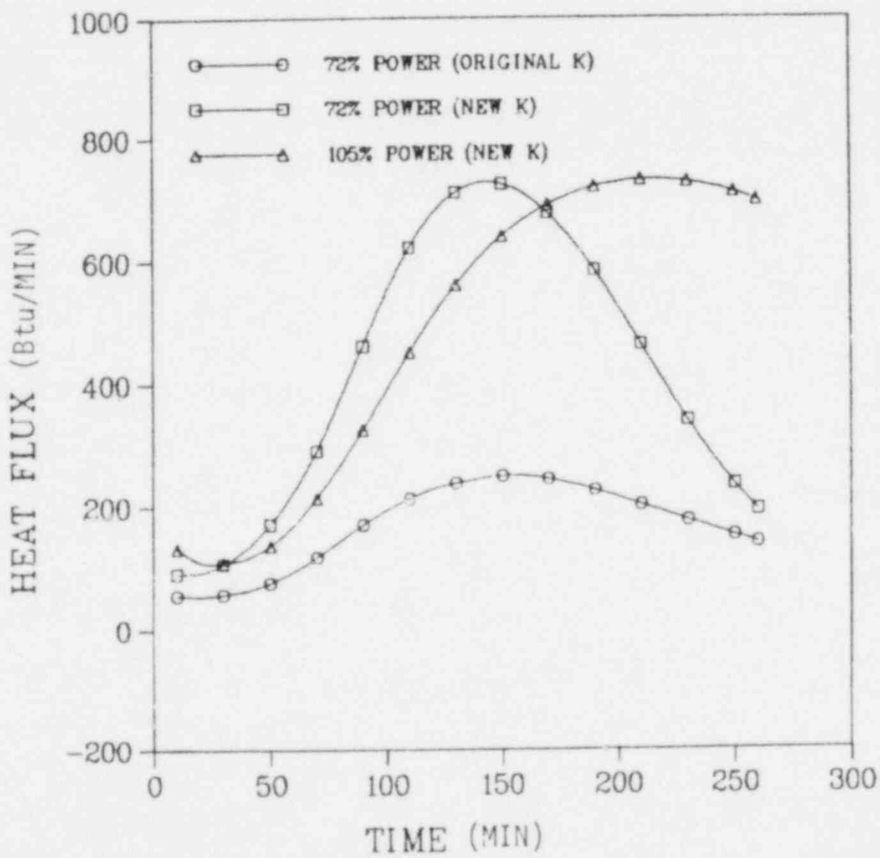

Fig. 13. Radial heat flow between adjacent core support blocks.

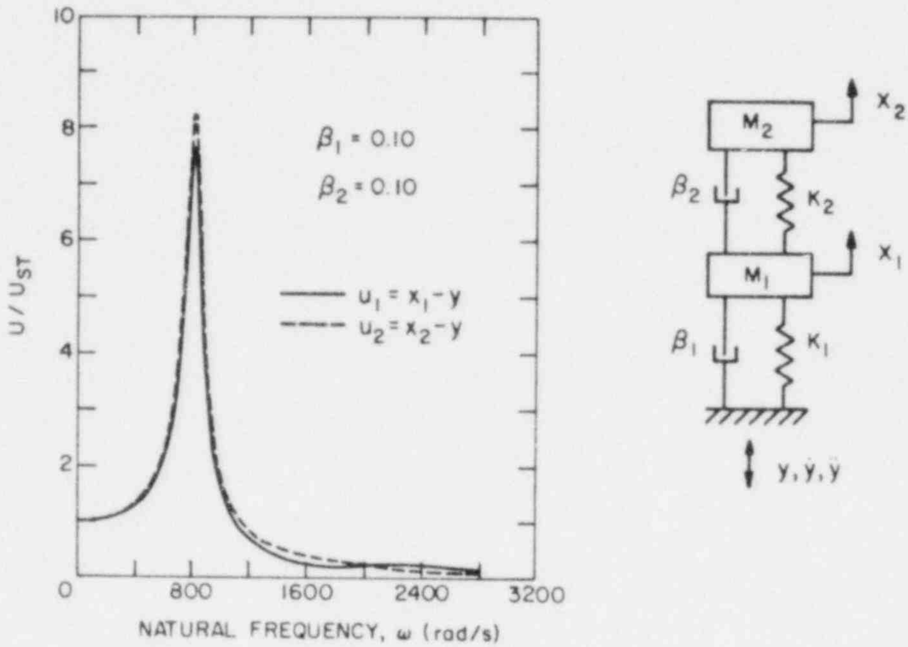

Fig. 14. Frequency response of a damped two-degree-of-freedom structural system.

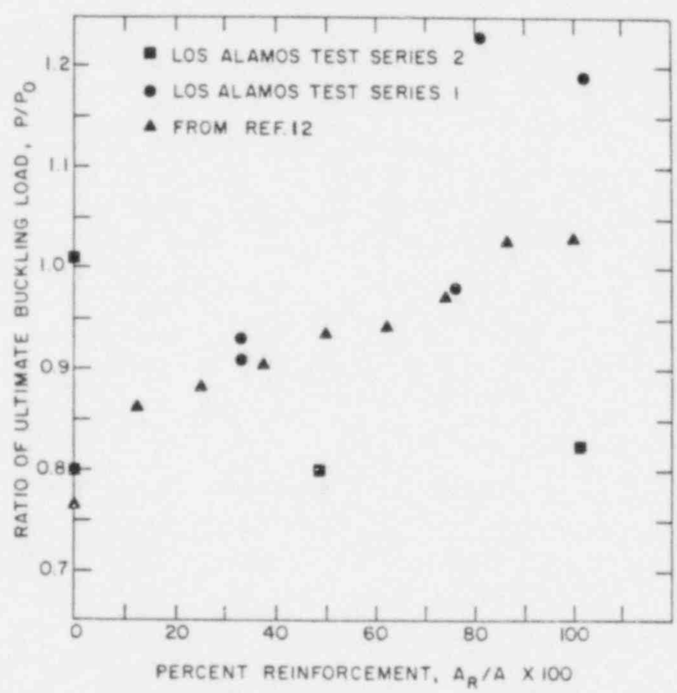

Fig. 15. Ratio of ultimate buckling load vs per cent reinforcement.

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