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Quarterly Technical Progress Report On Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research January-March 1981

April 1981

Prepared for the
U.S. Nuclear Regulatory Commission
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EG&G Idaho

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QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH JANUARY-MARCH 1981

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ABSTRACT

Water reactor research performed by EG&G Idaho, Inc., during January through March 1981 is reported. The Water Reactor Research Test Facilities Division performed two 10% cold leg break experiments, one with additional emergency core coolant injected into the upper head and one without. These experiments are described, with emphasis on the upper head injection. The Loss-of-Fluid Test (LOFT) Program completed a methodology to determine relationships between LOFT and commercial pressurized water reactor designs for transient initiating events. The relationship between LOFT and ZION for the case of a small break is discussed. The results of LOFT nuclear experiments L3-5/L3-5A and L3-6/L8-1, which address the Nuclear Regulatory Commission (NRC) pumps on-off issue, support the NRC position of requiring an early pump trip in small break situations. The Thermal Fuels Behavior Program completed Loss-of-Coolant Test 6, in which both the previously irradiated and unirradiated high pressure rods ballooned and ruptured during blowdown at temperatures in the high alpha crystalline phase of zircaloy. Instrumented Fuel Assembly (IFA) 511 experiments are being conducted in the Halden reactor in Norway to help resolve the uncertainty regarding the ability of electric heater rods to simulate the thermal response of nuclear fuel rods. Limited results indicate that there is a significant difference between the behavior of the two types of rods during reflood and quench. The Code Development Division completed development of the TRAC-BD1 code and released it to the National Energy Software Center. A brief description of the code is given and results of selected developmental assessment calculations are presented. The Code Assessment and Applications Division analyzed jet pump data that will be used to support the continued development and assessment of jet pump models in boiling water reactor thermal-hydraulic computer codes. A "blind" prediction for the LOBI test facility in Italy was completed, using the RELAP4/MOD6 thermal-hydraulic code. The 2D/3D Program has completed a series of tests in a vessel simulating the Japanese Slab Core Test Facility. The results, which characterize non-equilibrium, two-phase flow in simulated reactor vessels during reflood, are described.

SUMMARY

The Water Reactor Research Test Facilities Division emphasis has been directed at performing experiments and analyses to support the Nuclear Regulatory Commission in the assessment and improvement of computer models for small break loss-of-coolant accidents. Two 10% cold leg break experiments have been performed, one with additional emergency core coolant injected into the upper head and one without. These experiments are described, with emphasis on the effects of the upper head injection. Further experiments with different break sizes are planned in the future.

A methodology involving system computer codes has been developed to study the relationships between the Loss-of-Fluid Test (LOFT) Facility and the commercial PWR plant designs. The purpose is to determine the relevance and implications of LOFT experimental data to the commercial plants and the licensing process. The methodology, applicable for all transient initiating events, has thus far been used to identify the relationship between LOFT and the ZION pressurized water reactor (PWR) for a small pipe break event. The results show that in the first 1000 s of the transients in the two systems, the differences are due only to system setpoints, energy per unit volume, and core bypass. The analysis was completed on LOFT experiments L3-5/L3-5A and L3-6/L8-1, which address the pumps on-off issue in small break situations. The results of the experiments support the position of the NRC in requiring an early pump trip in PWR small break situations. The LOFT Augmented Operator Capability Program developed several new displays this quarter.

The Thermal Fuels Behavior Program completed (a) Loss-of-Coolant Test 6, which was performed to obtain information on the thermal and mechanical response of pressurized water reactor (PWR) design fuel rods subjected to simulated loss-of-coolant accident (LOCA) conditions during which the cladding peak temperature reached 1070 K; (b) the destructive examination of the Power-Cooling-Mismatch Test 7 nine-rod bundle; (c) the metallographic and radiochemical analysis of the Reactivity Initiated Accident (RIA) Test 1-4; (d) the analysis and reporting of the molten fuel-coolant interaction that occurred during the RIA-ST-4 test; and (e) a preliminary analysis of the results of the Instrumented Fuel Assembly (IFA) 511.2 and IFA-511.3 experiments performed in the Halden reactor in Norway. Both the previously irradiated and unirradiated high pressure Test LOC-6 rods ballooned and ruptured during blowdown at temperatures in the high alpha crystalline phase of zircaloy. The rupture times of the two rods differed significantly; the previously unirradiated rod deformed between 4 and 5.2 s after blowdown, indicating rapid ballooning, whereas the previously irradiated rod deformed over a much longer period of time from 8 to 18.2 s after blowdown, indicating significantly more deformation. The IFA-511 experiments are being conducted to determine the applicability of electric heater rod data to the understanding of core thermal-hydraulic and fuel rod response during the heatup and reflood phases of a LOCA. The ability of electric heater rods to simulate the thermal response of nuclear fuel rods has been questioned. In the IFA-511 tests, both nuclear and electric heater rods are being exposed to identical thermal-hydraulic heatup and reflood conditions to help resolve this uncertainty. On the basis of the results of relatively limited test data, there is a significant difference between the behavior of nuclear and electric heater rods during reflood and quench.

The Code Development Division completed development of the TRAC-BD1 code and released it to the National Energy Software Center. An important part of the final phase of the development was a series of developmental assessment calculations. A brief description of the TRAC-BD1 code is given and results of selected development assessment calculations are presented.

The Code Assessment and Applications Division analyzed jet pump data that will be used to support the continued development and assessment of jet pump models in boiling water reactor thermal-hydraulic computer codes. The assessment of the fuel rod analysis code, FRAPCON-2, was completed. This effort served to characterize the predictive capabilities of FRAPCON-2 and assist the user community in the use of the code and interpretation of results. A "blind" prediction for the LOBI test facility in Italy was completed using the RELAP4/MOD6 thermal-hydraulic code.

The 2D/3D Program is continuing to provide flow instrumentation for German and Japanese experiments, and design and analysis support to the NRC. A series of tests has been completed in a vessel simulating the Japanese Slab Core Test Facility. The results, which characterize non-equilibrium, two-phase flow in simulated reactor vessels during reflood, are described. Advanced instrumentation has continued its research efforts on specialized measurement devices.

FOREWORD

EG&G Idaho, Inc., performs water reactor safety research at the Idaho National Engineering Laboratory under the sponsorship of the U.S. Nuclear Regulatory Commission's (NRC) Division of Reactor Safety Research. The current water reactor research activities of EG&G Idaho, Inc., are accomplished in the Semiscale Program, the Loss-of-Fluid Test (LOFT) Experimental Program, Thermal Fuels Behavior, the Code Development Division, the Code Assessment and Applications Division, and the 2D/3D Program.

The Water Reactor Research Test Facilities (WRRTF) Division is responsible for a continuing series of small-scale, nonnuclear, thermal-hydraulic experiments having as their primary purpose the generation of experiment data that can be applied to the development and assessment of analytical models describing loss-of-coolant accident (LOCA) phenomena in water-cooled nuclear power plants. Emphasis has been placed on acquiring system effects data from integral tests that simulate the phenomena that could occur in a nuclear reactor during the depressurization (blowdown) and emergency core cooling process resulting from a large break (rupture) in the primary coolant system piping. Current emphasis is being placed on acquiring data on thermal-hydraulic phenomena likely to occur during reactor operational transients and during small pipe breaks. These data will be used to evaluate the adequacy of and make improvements to the analytical methods currently used to predict the transient response of large pressurized water reactors (PWRs). The WRRTF Semiscale test facility is now in a configuration that contains two active loops and a full-length electrically heated core scaled to a PWR.

The LOFT Experimental Program is a nuclear test program for providing test data to support (a) assessment and improvement of the analytical methods used for predicting the behavior of a PWR under LOCA (including small breaks) and operational transient conditions; (b) evaluation of the performance of PWR engineered safety features, particularly the emergency core cooling system; and (c) assessment of the quantitative margins of safety inherent in the performance of these safety features. The test program uses the LOFT Facility, an extensively instrumented 55-MW (thermal) pressurized water reactor facility designed for conduct of loss-of-coolant experiments (LOCEs) and anticipated transients. The test program includes ten series designations that begin with either a large, intermediate, or small break or an anticipated transient as the plant off-normal or accident initiating event. The many series of tests are intended (a) for evaluation of specific plant responses to initiating events from a variety of plant conditions, and (b) for assessment of emergency safety features, plant recovery procedures, and operator diagnostics.

Thermal Fuels Behavior is an integrated experimental and analytical program designed to provide information on the behavior of reactor fuels under normal, off-normal, and accident conditions. The experimental portion of the program is concentrated on testing single fuel rods and fuel rod clusters under power-cooling-mismatch, loss-of-coolant, reactivity initiated accident, and operational transient conditions. These tests provide in-pile experiment data for the evaluation and assessment of analytical models that are used to predict fuel behavior under reactor conditions spanning normal operation through severe hypothesized accidents. Data from this program provide a basis for improvement of the fuel models.

The Code Development Division is responsible for the development of codes and analysis methods; analytical research is conducted that is aimed at predicting the response of nuclear power reactors under normal, off-normal, and accident conditions. Computer codes are developed that primarily relate to an hypothesized LOCA in light water reactors. The codes are used to calculate the thermal-hydraulic behavior of reactor primary coolant systems, to calculate the environmental conditions in a reactor containment system during a LOCA, and to analyze fuel behavior during reactor steady state operation and during a variety of reactor operating transients.

The Code Assessment and Applications Division assesses the accuracy and range of applicability of computer codes developed for the analysis of reactor behavior. The assessment process involves the development of methods of analysis assessment, the analyses of many different experiments, and the comparison of calculated results with experiment data. Statistical evaluations of both the analytical and experimental results are part of the assessment process. Assessment results serve to inform the scientific community interested in reactor safety of the relative capabilities, validity, and range of applicability of NRC-developed codes.

The 2D/3D Program encompasses the 2D/3D instrument projects and analysis efforts and the water reactor research advanced instrumentation. The 2D/3D Program provides technical support to the NRC in its multinational (U.S., Germany, and Japan) experimental program to investigate the behavior of entrained liquid in a full-scale reactor upper plenum, and cross flow in the core during the reflood phase of a PWR LOCA. Advanced instrumentation develops new, specialized measurement devices and supports analytical development by enhancing state-of-the-art capabilities to measure physical phenomena.

More detailed descriptions of the water reactor research programs are presented in the quarterly report for January through March 1975, ANCR-1254. Later quarterly reports are listed and are available from the Technical Information Center, Department of Energy, Oak Ridge, Tennessee 37830, and the National Technical Information Service, Springfield, Virginia 22161.

ANCR-1262 (April-June 1975)
ANCR-1296 (July-September 1975)
ANCR-NUREG-1301 (October-December 1975)
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I. WATER REACTOR RESEARCH TEST FACILITIES DIVISION

P. North, Manager

1. PROGRAM STATUS

The program emphasis has been directed at performing experiments and analyses to support the Nuclear Regulatory Commission (NRC) in the assessment and improvement of computer models for small break loss-of-coolant accidents (LOCAs).

The NRC needs to determine whether current computer models can calculate the transient behavior of light water reactors equipped with emergency core coolant (ECC) injection into the upper head. A series of experiments has been designed to provide systems behavior data from the Semiscale Mod-2A system with a range of small break sizes, with and without upper head injection (UHI).

The first two experiments employed a 10% cold leg break. The experiment without UHI was designated S-UT-1, whereas that with UHI was designated S-UT-2. The major results from these two experiments are reported subsequently. Further experiments with smaller break sizes are planned, but heat loss from the Mod-2A system becomes more significant at the smaller break sizes. The remaining UHI experiments will, therefore, follow installation and testing of a heat loss makeup system.

2. RESULTS FROM MOD-2A TESTS S-UT-1 AND S-UT-2 (UPPER HEAD INJECTION TEST SERIES)

D. J. Shimeck

Tests S-UT-1 and S-UT-2 were the first tests conducted in the Semiscale Mod-2A system. The tests are part of the UT test series, which is being conducted to investigate the influence of upper head injection (UHI) of emergency core coolant (ECC) on system behavior during small break transients. Tests S-UT-1 and S-UT-2 were 10%, communicative, cold leg break experiments. Test S-UT-1 was performed without the upper head accumulator so as to establish baseline data on the performance of the Mod-2A system. Test S-UT-2 was then conducted with the same specified initial and boundary conditions but with the use of upper head injection. Important ECC system parameters are listed in Table 1.

Comparison of the two tests shows that the general system response was similar, with minor differences introduced due to the presence of the extra UHI water in S-UT-2. In both tests, the early transient response (0 to 500 s) was dominated by the two phenomena of gravity drain, and the formation of manometric liquid seals in the pump suction piping. As a result of pump suction seal action, a brief period of core dryout occurred during both tests beginning at about 50 s. In neither test did heater rod cladding temperatures exceed those at initial conditions. Once the vessel liquid levels recovered following the clearing of liquid from the pump suctions, the core remained adequately cooled for the remainder of the transients.

Table 1. ECC parameters

System and Parameter	S-UT-1	S-UT-2
Upper head accumulator		
Actuation pressure (kPa)	N/A	8500
Liquid volume injected (m^3)	N/A	0.0164
Intact loop accumulator		
Actuation pressure (kPa)	2770	2980
Liquid volume (m^3)	0.060	0.060
Intact loop HPIS[a]		
Actuation pressure (kPa)	13300	13500
Injection rate (L/s)	0.060	0.061
Intact loop LPIS		
Actuation pressure (kPa)	N/A[b]	1150
Injection rate (L/s)	N/A[b]	0.17

a. HPIS—high pressure injection system, LPIS—low pressure injection system.

b. Test S-UT-1 was terminated prior to LPIS injection.

The UHI ECC liquid did provide a better margin against core uncovery prior to intact loop accumulator injection at about 330 s. System responses for the latter portion of the transient were nearly identical for the two tests, with a slow refill of the vessel and system accompanied by accumulator-induced level oscillations. Test S-UT-2 was run to 2750 s to allow LPIS injection to begin. Although Test S-UT-1 was terminated at 1000 s, the results from S-UT-2 after 1000 s are directly applicable. The following paragraphs review more specific comparisons between the tests.

System pressures for the two tests are compared in Figure 1, where the occurrence of important phenomena are indicated. In Test S-UT-2, the upper head accumulator injected liquid during the period from 15 to 140 s. System depressurizations in both tests were nearly identical. A rapid drop in pressure occurred until nearly the entire system fluid mass became saturated (about 20 to 30 s). At that time, the resultant flashing caused a much slower depressurization rate. The pressure for Test S-UT-2 was slightly lower than that for S-UT-1 until approximately 150 s. It then exhibited a period of slower depressurization, corresponding to the end of upper head accumulator injection when the upper head fluid became saturated and began to flash. System pressure decreased more slowly until the upper head emptied at about 220 s.

The behavior and distribution of the upper head fluid were of particular interest in these tests. Figure 2 compares the calculated upper head liquid levels for the two tests. The accumulator flow kept the upper head nearly full in Test S-UT-2 during injection. The drain rate was then nearly identical between the two tests, with the upper head emptying within about 50 to 70 s. Temperature measurements throughout the upper head showed stratification of subcooled liquid throughout the period of accumulator injection. Comparison of flows through the various penetrations into the upper head for the two tests (these being the downcomer to upper head bypass line, guide tube, and two support columns) showed that UHI ECC fluid flowed both to the core and also to the cold leg. Figure 3 compares the volumetric flows through one of the support columns. Except for brief periods of reverse flow that accompanied

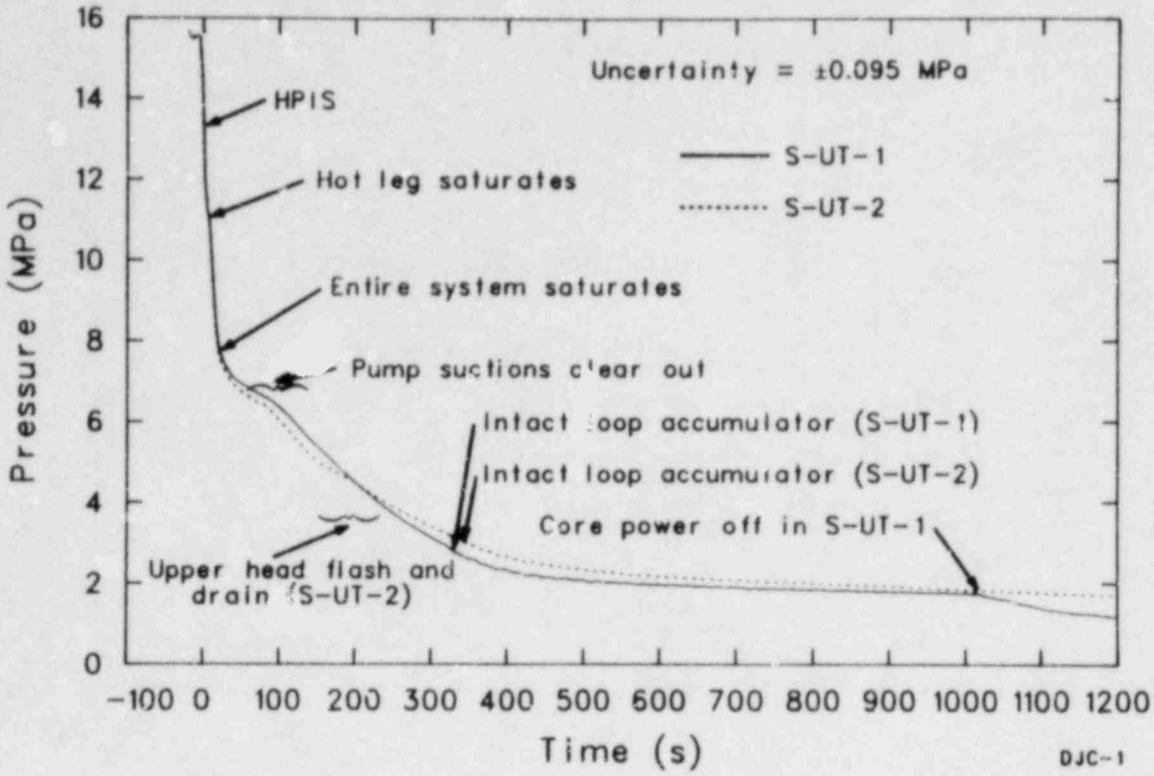

Figure 1. Comparison of system pressures.

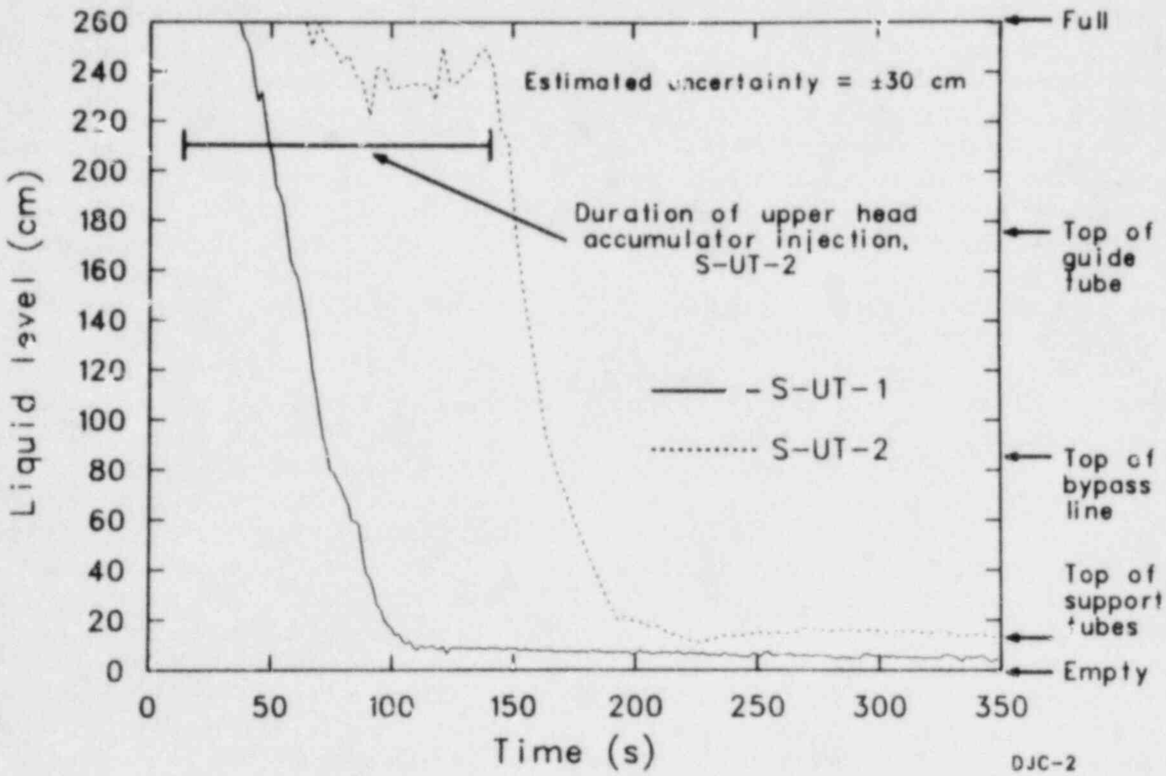

Figure 2. Comparison of collapsed liquid levels in the vessel upper head.

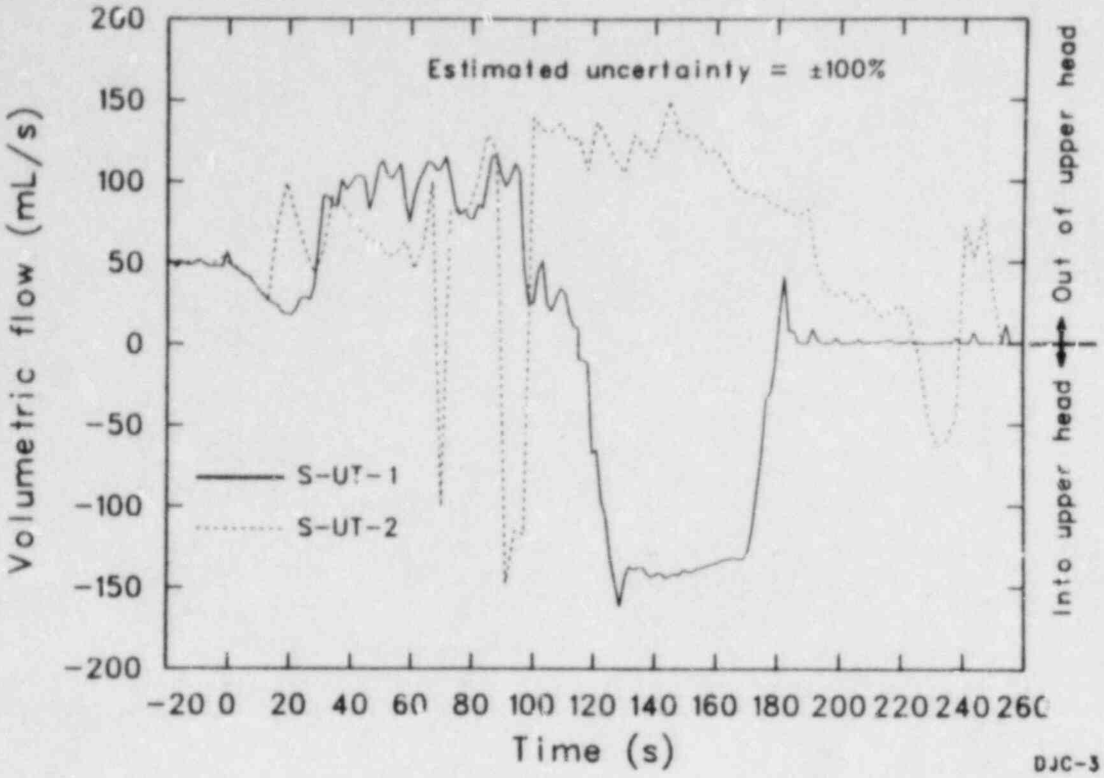

Figure 3. Comparison of volumetric flow rates through one of two support columns.

pump suction seal phenomena between 70 and 100 s, the flow was out of the head (to the core) in both tests until the head emptied. Similar behavior was observed in the bypass line to the downcomer. Although condensation-induced reverse flow of steam was observed in the guide tube, from the upper plenum to upper head, it did not represent a significant transfer of mass.

Measurements throughout the system indicated that UHI had little influence on early transient phenomena. The steam generator drain, and the formation, blowout, and sweepout of the pump suction liquid seals were essentially identical. This is to be expected since the upper head accumulator injection rate (an average flow of approximately 0.15 L/s) was small relative to the break flow rate (on the order of 0.7 L/s average) and the rate at which liquid was displaced from the pump suctions and core.

In neither test was there any significant heatup of the core heater rods. The rod temperatures at initial conditions were the highest recorded during the tests. A brief temperature excursion occurred due to the depression of the core liquid level as a result of pump suction seal formation. Figure 4 compares the calculated collapsed liquid levels in the core for the two tests. Figure 5 shows the temperature response of selected heater rod cladding thermocouples from Test S-UT-1. The injection of upper head accumulator fluid had little impact on the extent to which the core uncovered at 75 s. As the core level recovered at 90 s during Test S-UT-2, following loop seal blowout, measurements indicated that approximately an additional 2.5 L of liquid reentered the core and downcomer. This resulted in a rapid rewet of the entire core as opposed to the reduced filling in Test S-UT-1 that allowed the additional dryouts in the upper core seen in Figure 5. This amount of liquid could have entered the core region from the upper head via the support tubes as discussed previously. As evident from the core liquid levels, the boiloff rate prior to intact loop accumulator injection was different between the two tests, with the additional fluid in Test S-UT-2 keeping the core more covered than in Test S-UT-1. Once accumulator injection had begun (330 s in Test S-UT-1; 345 s in Test S-UT-2), system behavior

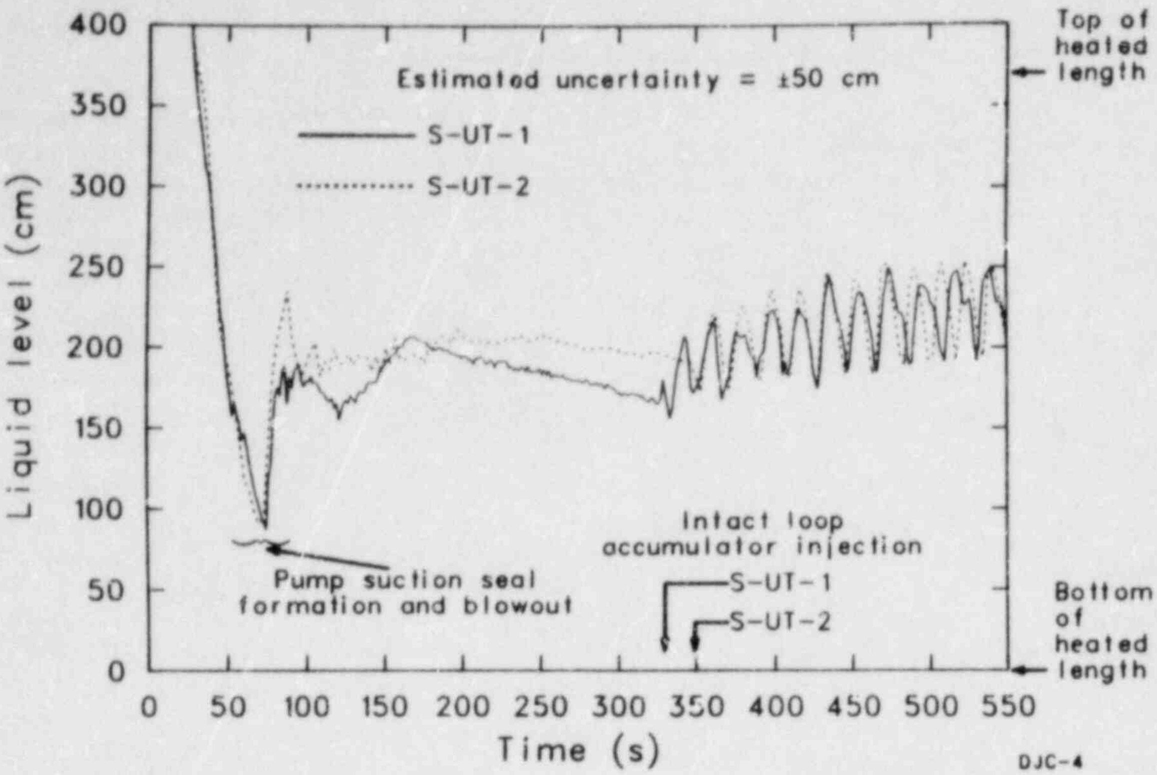

Figure 4. Collapsed liquid levels in the core.

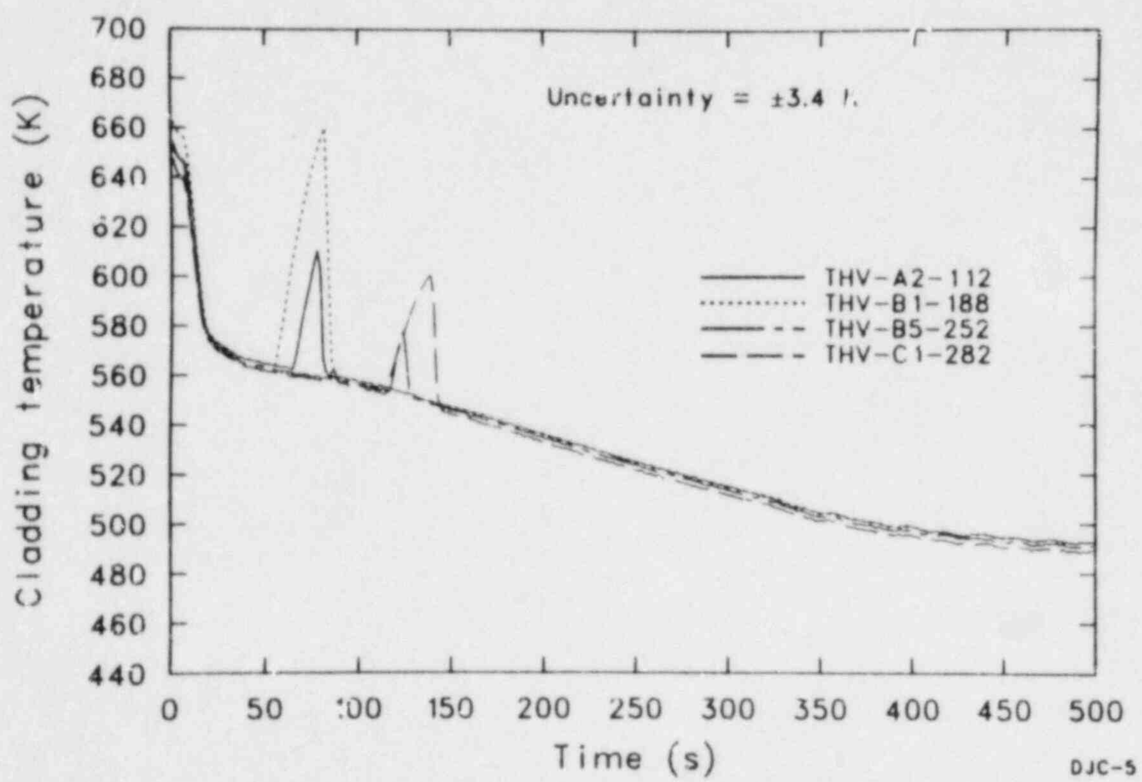

Figure 5. Response of selected heater rod thermocouples for Test S-UT-1.

was nearly identical. The liquid level oscillations induced by accumulator injection had no effect on core cooling and the vessel exhibited a slow refilling trend.

The occurrence of these liquid level oscillations was an interesting phenomenon common to both tests. Oscillations with a period of about 15 to 20 s were observed in many system parameters including the manometric core-downcomer liquid levels, system pressure, and flows. Figure 6 shows how the system pressure oscillations caused oscillations in accumulator injection flow. The mechanisms contributing to this phenomenon will be investigated further and addressed in the posttest analysis.

In summary, results from these tests have shown that ECC water injected from the upper head accumulator flows both to the break and to the core region. UHI had virtually no influence on the early (0 to 100 s) or late (after 350 s) portions of the transient relative to the behavior observed in Test S-UT-1. The UHI ECC liquid did provide a better margin against core uncovery prior to intact loop accumulator injection.

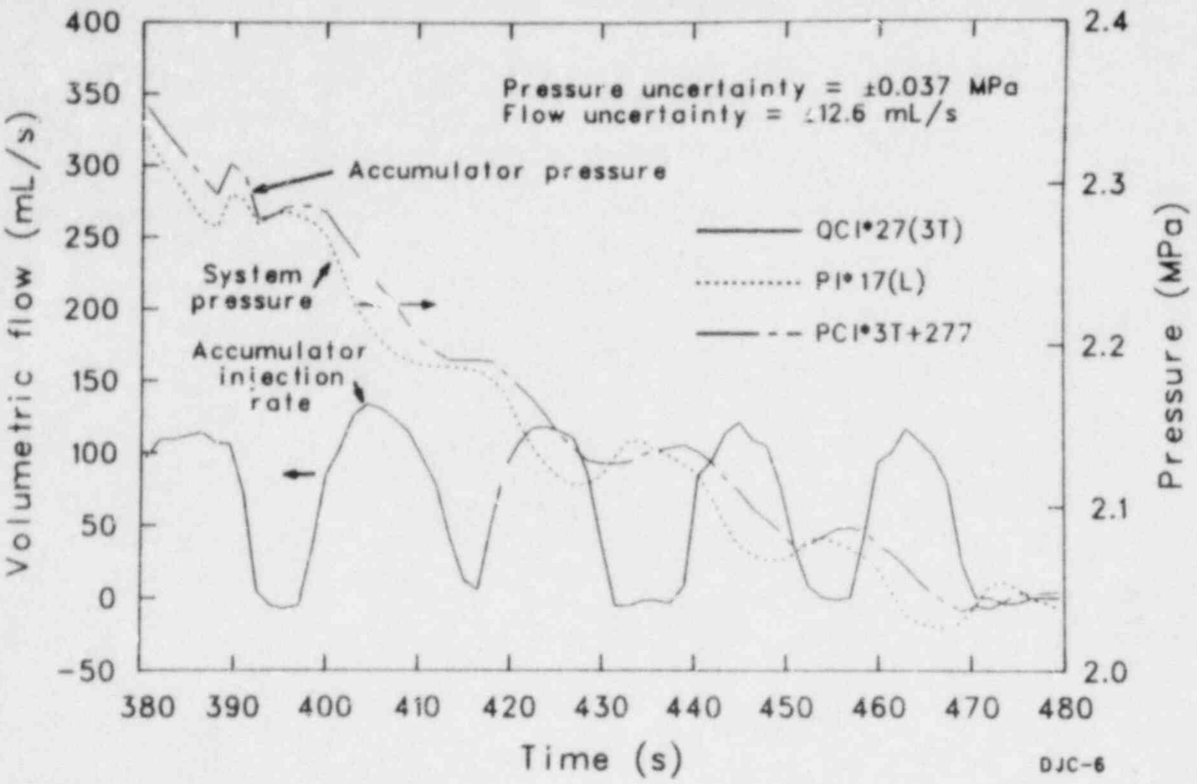

Figure 6. System pressure, accumulator tank pressure, and accumulator injection flow for Test S-UT-1.

II. LOFT PROGRAM OFFICE

C. W. Solbrig, Manager

The relevance and implication of experimental data from Loss-of-Fluid Test (LOFT) Facility accident simulations to commercial pressurized water reactor (PWR) plant designs and the licensing process is formally being studied. A methodology involving system computer codes has been developed, which is used to determine the relationships between LOFT and commercial plants. LOFT loss-of-coolant experiments L3-5/L3-5A and L3-6/L8-1 were conducted in response to the pumps on-off issue discussed in NUREG-0623.[1] The issue, of whether the primary coolant pumps be tripped following a small break loss-of-coolant accident (LOCA), as directed by the NRC, or be left running, arose from the Three-Mile-Island Accident. Resolving the issue was the main objective of the L3-5 and L3-6 phases of the experiments. The pumps on-off issue is described, along with the differences observed between the experiments. The results support the NRC position of tripping the primary coolant pumps in PWRs early during small break situations.

Development of safety related computer-generated graphics is being performed by the LOFT Augmented Operator Capability (AOC) Program. Task-analysis work has resulted in a response-tree display, and the modeling effort has led to a new steam generator model for use in a predictive steam generator level display. The displays will now be able to be tested using a newly installed communication link between the color graphic generation computer and a digital computer model of the LOFT plant.

1. RELATIONSHIP BETWEEN LOFT AND COMMERCIAL PWRs

V. T. Berta and T. L. DeYoung

Since the completion of the large break LOFT experiments L2-2 and L2-3 in 1979, the experimental information obtained from LOFT has been analyzed to determine the relevance and implication of the data to commercial PWRs and the licensing process. The results of analyses of the large break data showed that, for this size break, the transients in the LOFT reactor closely simulate the expected transients in commercial four-loop PWRs.[2,3] Since the completion of LOFT L2-3, other experiments have been performed with break size simulations in the small break category. The question of the relevance of LOFT was raised anew for the small break category since the transients involved are of an entirely different nature than those for large breaks. Consequently, a methodology has been developed for studying and evaluating the relationshps between LOFT and the various commercial PWR plant designs. The methodology encompasses the full range of accident initiating events and is reversible, with the starting point being either a transient in LOFT or a calculated transient in a commercial PWR.

The steps in the methodology for using a LOFT transient as the basis for a calculated transient from the same initiating event in a commercial plant are diagrammed in Figure 7. Three elements are employed:

1. The calculational mechanism
2. The operational package
3. The system model upon which the calculational mechanism operates within the constraints of the operational package.

The calculation mechanism is usually a computer code, such as RELAP4/MOD7, which embodies a collection of physics, correlations, and numerical solution techniques for reactor transient analysis. The operational package is that part of the code input deck covering input quantities such as operational setpoints, correlation selections, and multipliers and other constant values. The system model is that geometric representation of the plant that is constructed within the computer code model construction criteria.

These three elements are applied first to a LOFT transient. The calculation is compared to the measured transient and then revised until the

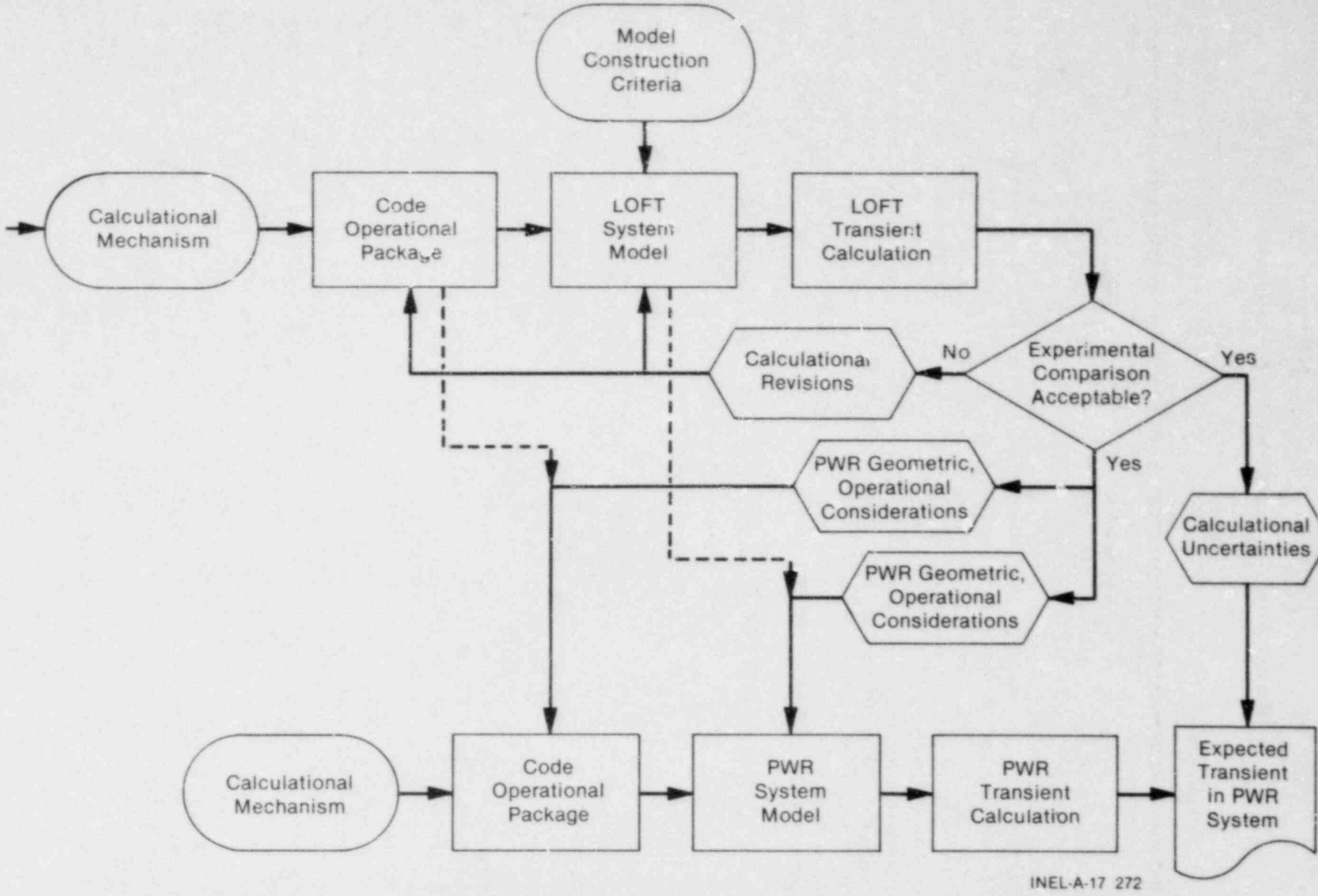

Figure 7. Methodology using LOFT experimental data to calculate commercial PWR transients.

agreement is considered to be the best possible. The differences between calculated and measured LOFT response are the minimum uncertainties that can be expected in a commercial plant calculation. The three elements are then applied to the commercial PWR. Changes to the operational package and system model of LOFT are made based only on operational and geometric differences between LOFT and the commercial PWR. The PWR calculation that results is, as a consequence, traceable to the LOFT transient for an evaluation of the soundness of the calculation and the minimum uncertainty associated with the calculation.

Confidence in the PWR calculation is increased by a further analysis, which involves analyzing the operational and geometric differences between LOFT and the commercial PWR with the intent of determining the origins of the principal differences in the transients in the two systems. The analysis is done by imposing on the commercial system the corresponding actual operational and geometric parameters of the LOFT system, either singly or in combination, to make the commercial PWR transient more like the transient in LOFT. In this way, the manner in which the LOFT transient transforms into the commercial PWR transient can be understood.

The methodology, as shown in Figure 7 and described previously, is reversible. The starting point can be the application of the three elements in the methodology to the commercial PWR. This is equivalent to the PWR calculations that have been and are being done by analysts in industry and government. The objective, then, becomes the transformation of the three elements for a calculation of a LOFT transient with the same initiating event. The calculation is compared to the LOFT transient data set and the differences are then evaluated to determine both the adequacy of the calculation and uncertainties. In the event the comparison is good, then the PWR calculation would also be judged good. In the event the comparison shows significant differences, then the PWR calculation is questionable in specific or all areas. Revising the LOFT calculation to better agree with the LOFT data set provides a feedback mechanism to produce a better PWR calculation. Finally, if a LOFT reference transient does not exist for a specific PWR initiating event analysis, then the importance of the issues involved can be evaluated to determine if justification exists for obtaining a new LOFT reference transient.

This methodology has been applied to the ZION commercial PWR in a calculation of the transient resulting from a 1-inch-diameter cold leg break. The LOFT reference experiment is L3-7 and the calculational mechanism used was the RELAP4/MOD7 code.[a] The results of the LOFT/ZION L3-7 calculations revealed firstly that the LOFT facility can be used to generate benchmark data sets for traceability of small break transient calculations in commercial PWRs of the ZION design. Secondly, relatively straightforward transformations were found to exist between LOFT and ZION, which enhance the confidence in the ZION-calculated small break transient. The differences between the LOFT L3-7 transient and the calculated ZION L3-7 transient in the first 1000 s were due to differences only in system setpoints, energy per unit volume, and core bypass from the downcomer to the upper plenum; a far lower number of sources than had been theorized. Similar transformations are believed to exist between LOFT and other PWR plant designs. Finally, the methodology permits evaluation of the uncertainties in the commercial PWR calculations. These findings indicate that a stronger calculational base may exist for evaluating plant licensing requirements and current plant operational limitations and also for establishing operator training and plant recovery procedures. The reversibility of the methodology provides a mechanism whereby justification can be defined for conducting new reference experiments in the LOFT facility when needed to aid in resolving issues as they arise.

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a. RELAP4/MOD7, Idaho National Engineering Laboratory Computer Code Configuration Control Number H013431B.

2. LOFT PUMPS ON-OFF EXPERIMENTS L3-5/L3-5A and L3-6/L8-1

V. T. Berta and G. E. McCreery

The question of whether pressurized water reactor coolant pumps should be tripped at the onset of a small break loss-of-coolant accident or be left running arose from the Three-Mile Island (TMI-2) accident. Evaluation of plant data after the accident indicated that

1. The core was adequately cooled when the primary coolant pumps (PCPs) were running

2. Natural circulation did not occur and core damage resulted when the PCPs were turned off.

As a result of this evaluation, operation of the PCPs during a small break LOCA was considered desirable by the NRC. Bulletins 79-05A, 79-06A, and 79-06B, issued immediately after the TMI-2 accident, specified that PCPs be kept running following high pressure injection actuation.

More extensive analyses by PWR vendors and by NRC staff were performed after these directives were issued and are summarized in NUREG-0623.[1] The analyses concluded that either delayed pump trip or continuous pump operation following a small break LOCA will lead to higher break mass flow rate and lower primary system mass inventory than in the case when the pumps are immediately tripped. As a consequence of lower primary system mass inventory, fuel cladding temperatures in excess of current licensing limits (1478 K) were predicted to occur.

As a result of these analyses, the NRC concluded that[1] "sufficient uncertainty exists in the thermal-hydraulic phenomenological modeling such that the quantitative results of these small break analyses with the pumps running cannot be accepted at this time (i.e., the specific bounds of the critical break size/critical trip time map). However, the staff does believe that the predicted overall qualitative behavior, supplemented with a basic understanding of the phenomena in question, is sufficient to conclude that small break LOCAs with the pumps operational or with delayed trip can result in more severe consequences than when the pumps are tripped early in the accident. Therefore, we have concluded that tripping of all of the reactor coolant pumps early in a small break accident is required at this time to preclude the occurrence of excessive fuel cladding temperatures."

The NRC issued two bulletins (79-05C and 79-06C) requiring coolant pump trip in the case of a small break LOCA. The bulletins also require that the PWR vendors propose and submit a design change that will assure automatic tripping of operating primary coolant pumps under all circumstances in which this action may be needed.

To gain more understanding of pump effects during small break LOCAs, the NRC sponsored two experiments, L3-5/5A[4] and L3-6/L8-1,[5] in the LOFT Facility. The two experiments were conducted from nearly the same initial conditions. Experiment L3-5/5A provided data on the case in which the pumps were tripped early in the transient; in L3-6/L8-1 the pumps were left running.

The pumps-on condition (L3-6) exhibited significant differences from the pumps-off condition (L3-5). L3-6 compared with L3-5 exhibited (a) higher break mass flow rate throughout the transient, (b) a more homogeneous mass distribution throughout the system, (c) substantially less liquid mass in the core (void fraction greater than 90% at the end of L3-6 compared with liquid full at the end of L3-5), (d) a slightly lower depressurization rate, and (e) a vessel liquid level that was less distinct and less easily measurable. A more benign transient occurred in the pumps-off experiment because of these differences and, therefore, the two LOFT experiments support tripping the primary coolant pumps early in a small break LOCA, as specified in NRC bulletins 79-05C and 79-06C.

During L3-6, pump-forced core flow was sufficient to provide high convection heat transfer from the fuel rods at void fractions greater than 90% at the end of the transient. The efficient heat transfer ceased when the pumps were tripped and coasted down, because the flow was no longer sufficiently high to entrain liquid drops. The LOFT pumps performed well and exhibited higher normalized head and torque in two-phase flow than the Semiscale pumps. The LOFT pumps are much more similar to the Combustion Engineering pumps in this regard.

The computer codes RELAP5 and TRAC-PD2 did not accurately predict depressurization rate and break mass flow rate. RELAP5 did not predict depressurization rate and break mass flow rate accurately because (a) the two-phase pump characteristics were not correct for the LOFT pump, (b) there were difficulties with the steam generator heat transfer calculation, and (c) phase separation in the tee leading to the break was not modeled. These areas require significant improvement for accurate calculation of pump effects on small break transients.

3. LOFT AUGMENTED OPERATOR CAPABILITY PROGRAM

M. A. Bray

The LOFT Augmented Operator Capability (AOC) program is developing diagnostic graphic techniques that can improve reactor operator performance. These display techniques are being implemented on a digital computer that drives color cathode ray tubes in the LOFT control room and technical support center.

During the quarter, the AOC program has made progress in several areas. Prior to the quarter, a Safety Parameter Display (SPD) was implemented which used a computer model of the LOFT system to normalize 11 key parameters, such as feed flow and steam flow, to measured reactor power. Deviations from normal are distinctively displayed to the operator. During March 1981, the SPD computer-modeled normalizations were refined by using measured LOFT data from several power levels. Trending capability was also added to the SPD during this quarter. Trend plots for each SPD parameter are available for instant callup. The trend plots show the previous 30 minutes of data and allow the operator to watch current data being added to the plots.

A Core Cooling Display (CCD) has been developed which complements the SPD. The SPD shows at-power plant relationships, but the CCD shows data that indicate the adequacy of core cooling after shutdown. The CCD shows derived parameters such as power history, derived decay heat, and heat removal capability (in kilowatts) of water supplied to the reactor.

Also during the quarter, the use of color by AOC displays has been standardized. An AOC color standard has been developed on the basis of military and nuclear industry practices and has been applied to all AOC displays. This effort has produced displays in which each color use has a particular, standard meaning. This reduces the amount of color variations and prevents color from becoming a distraction (noise) rather than a signal.

During the quarter, a cladding temperature display has been developed primarily for use in the LOFT technical support center during reactor experiments. This display indicates cladding temperatures axially in the core in relationship to a moving line that represents saturation temperature (calculated from system pressure). This display is based on a successful saturation temperature deviation display used in prior experiments and on posttest computer generated films that have been developed for explaining LOFT results. This display will be valuable to technical support center personnel for evaluating reactor safety during LOFT experiments in real-time.

An intertie has also been installed between a digital computer model of the LOFT reactor and the color graphic generation computer. This involves a communication interface between two computers that will allow AOC graphic displays to be exercised with the simulated computer model as well as with real LOFT data.

Finally, a computer model of the LOFT steam generator has been completed, which will be used to develop a predictive steam generator level display. This display will provide current state information and a prediction of future states based on current data and the steam generator computer model.

III. THERMAL FUELS BEHAVIOR

W. A. Spencer, Manager

Thermal Fuels Behavior (TFB) is an integrated experimental and analytical program designed to provide information on the behavior of reactor fuels under normal and accident conditions. The program is focused on the resolution of key safety issues regarding fuel behavior during power-cooling-mismatch accidents, loss-of-coolant accidents, reactivity initiated accidents, operational transients, and accidents such as Three Mile Island-2 that result in severe fuel damage. The program is structured to provide the data necessary to (a) confirm the adequacy of specific Nuclear Regulatory Commission licensing regulations designed to ensure plant safety, (b) resolve key safety issues and provide a data base from which new safety criteria and regulations can be established, (c) indicate where new or revised regulations may be appropriate, and (d) assess the computer models needed for licensing.

The experimental portion of Thermal Fuels Behavior is concentrated on the testing of single fuel rods and small clusters of fuel rods in the Power Burst Facility (PBF) to address safety issues related to fuel rod failure, maintenance of a coolable geometry, and the release of fission products during simulated accident conditions. The original Power Burst Facility test program, including the power-cooling-mismatch, reactivity initiated, and loss-of-coolant accidents, has been completed and the analysis and reporting of these test results are underway.

The followup experimental program in the Power Burst Facility is composed of operational transient, operational transient without scram, and severe fuel damage experiments. Results from the operational transient and operational transient without scram tests are expected to define damage mechanisms and failure thresholds and to help determine whether (a) a reactor should be derated following a severe operational transient, (b) regulations should be imposed to limit pellet-cladding interaction in high burnup rods, and (c) reactors should be modified to reduce the probability of an operational transient without scram. The severe fuel damage experiments are structured to provide key data regarding the primary fuel rod damage mechanisms that occur during an accident such as Three Mile Island-2, including fuel rod fragmentation and UO_2 dissolution, movement, and freezing. The quenching and long-term coolability of a previously molten and highly fragmented fuel rod rubble pile is of major concern.

Three nonprogrammatic test series are being performed to investigate the effects of cladding surface thermocouples on fuel rod behavior during blowdown and quench. The first series, TC-1, was completed in fiscal year 1980; the second, TC-3, was completed last quarter, and the final series, TC-4, will be performed next quarter.

Thermal Fuels Behavior also conducts in-pile testing of instrumented fuel assemblies in the Halden reactor in Norway. Both long-term irradiation to high burnups and transient testing are included. The long-term irradiations are designed to assess (a) the assumptions in the Regulatory Guides prescribing the inventory of radioactive fission gases in the fuel-cladding gap available for release during loss-of-coolant and fuel handling accidents, (b) the licensing guides for increase in fuel rod internal pressure and degradation of fuel-cladding gap conductance due to stable fission gas release at high burnups, and (c) the thermal and mechanical fuel behavior models used for licensing and safety analyses. Transient tests are also being conducted to determine how the thermal response of electrical heater rods compares with that of nuclear rods under loss-of-coolant reflood conditions. A discussion of the results of these tests is presented in Section 2.

1. PBF TESTING

P. E. MacDonald and R. K. McCardell

Loss-of-Coolant Test 6, which was performed to obtain information on the thermal and mechanical response of PWR design fuel rods subjected to simulated LOCA conditions during which the cladding peak temperature reached the high alpha (1070 K) phase, was completed this quarter. A Fuel Behavior Report for Reactivity Initiated Accident (RIA) Test 1-2 was issued,[6] and a draft of the Test PR-1 Test Results Report was completed. A report on the kinetics calculations for RIA experiments performed in the Capsule Driver Core (CDC) was issued.

The PBF LOCA Test Series consisted of four tests, each of which was designed to provide test fuel rod thermal and mechanical response information for the cladding temperature ranges corresponding to the alpha, alpha plus beta, and beta crystalline phases of zircaloy. Test LOC-6 was conducted to provide data on fuel rod response for cladding peak temperatures in the high alpha phase (1070 K). Test rod variables in the PBF LOCA tests include internal rod pressure, cladding state (previously irradiated or fresh), and test rod power, while a system depressurization typical of that expected during a PWR double-ended cold leg break is maintained.

Test LOC-6 was performed using four separately shrouded fuel rods (designated Rods 9, 10, 11, and 12) of PWR design, two of which had previously been irradiated in the Saxton reactor to burnups of 10 000 and 15 000 MWd/t (Rods 10 and 12), and two of which were unirradiated (Rods 9 and 11). One each of the unirradiated and irradiated rods was backfilled with helium to a pressure (2.4 MPa) representative of beginning-of-life PWR fuel rods (Rods 9 and 10) and the other two rods (Rods 11 and 12) were backfilled with helium to a pressure (4.8 MPa) typical of fuel rods at the end-of-operational life. Test LOC-6 consisted of preblowdown steady state operation to provide power calibration information and decay heat buildup, blowdown transient operation (coolant depressurization), and test termination by reflood and long-term quench cooling. Test conditions at the initiation of blowdown were: inlet coolant temperature of 590 K, system pressure of 15.4 MPa, and test rod peak power of 55.9 kW/m. The axial power profile along the fuel rods was shaped with the power flattened in the center third of the active fuel length to simulate conditions typical of the central region of the core in a PWR. Conduct of the LOCA transient began with the isolation of the in-pile tube from the PBF loop, and then the opening of quick-actuating cold leg blowdown valves to initiate the blowdown. The PBF reactor (driver core) power was controlled during blowdown by preprogramming the transient rods to follow a power function predetermined to provide the desired cladding peak temperatures (high alpha phase). Opening of the blowdown valves resulted in the rapid expulsion of coolant from the test train and complete depressurization within 25 s, simulating the expected depressurization of a PWR.

Preliminary results indicate that the goal of achieving cladding temperatures in the high alpha phase was attained. The cladding of both high pressure rods (one irradiated and one unirradiated) ballooned and ruptured. The cladding of the two low pressure rods did not rupture. The rupture times of the two high pressure Test LOC-6 rods differed significantly; the unirradiated rod deformed between 4 and 5.2 s after blowdown, indicating rapid ballooning, whereas the previously irradiated rod deformed over a much longer period of time from 8 to 18.2 s after blowdown, indicating significantly more deformation.

Test RIA 1-2 was performed to (a) determine the enthalpy required to fail previously irradiated, light-water-reactor-type fuel rods, (b) evaluate the failure mechanism, and (c) study the effect of beginning-of-life (BOL) and end-of-life (EOL) rod internal pressures on preirradiated fuel rod response during an RIA event. Four individually shrouded, zircaloy-clad, UO_2 fuel rods were tested. All four rods had been previously irradiated to 4800 MWd/t; two of the rods were prepressurized to beginning-of-life conditions for boiling water reactor (BWR) rods, and the other two were prepressurized to end-of-life conditions for BWR rods. Beginning at BWR hot-startup conditions, the rods were subjected to a power transient resulting in an axial peak, radial average fuel enthalpy of 775 J/g (total radial average energy deposition of 1005 J/g UO_2). The rods reached cladding peak temperatures ranging from 1520 to 1700 K during the transient, with the cladding temperatures being lower for the high pressure rods than for the low pressure rods. One of the low pressure rods failed as a result of multiple (22) longitudinal cracks in the cladding, but the other low pressure rod did not fail. The high pressure rods deformed more than the low pressure rods, with as much as 6.7% diametral strain, but did not fail. There was no obvious difference between the two low pressure rods that would explain the failure of one rod and not the other; however, the low pressure rod that did not fail and the two high pressure rods had been opened prior to the test for installation of a plenum pressure transducer and had been refilled with a 77.7% helium 22.3% argon mixture. Posttest evaluations indicate that the single rod failure occurred at an enthalpy insertion of 586 to 775 J/g UO_2. This failure threshold is consistent with previous RIA test results.

RIA-type transient tests were conducted in the Capsule Driver Core (CDC) at the Idaho National Engineering Laboratory during the late 1960s, and the data obtained from the CDC tests have been used by the United States Nuclear Regulatory Commission to evaluate and set reactor licensing criteria. Analyses of RIA tests performed more recently in the PBF have shown that energy deposition in the test fuel by delayed-neutron-induced fissions, after reactor shutdown and control rod insertion, is significant (~19% in the PBF) and is a factor in determining fuel failure thresholds during RIA tests. Reactor kinetics calculations were performed for the CDC in an effort to obtain information for reevaluation of the failure thresholds determined from the CDC tests. The results of the calculations indicate that for the CDC tests, delayed-neutron-induced fissions account for about 12% of the total energy deposition. These results will be used in additional analyses to identify specific corrections for the previously reported failure thresholds.

2. PROGRAM DEVELOPMENT AND EVALUATION

P. E. MacDonald and R. R. Hobbins

Power Burst Facility programmatic efforts and accomplishments during the quarter are reported in this section. The postirradiation examination of PBF tested fuel rods and topical analysis of fuel data are discussed in addition to the results from the Instrumented Fuel Assembly (IFA) 511 Test Program being conducted in the Halden reactor in Norway.

The destructive examination of the Test PCM-7 fuel rods was completed. The earlier nondestructive examination showed extensive oxidation on eight of the nine rods in the bundle as a result of extended time in film boiling during the test. Severe embrittlement within the film boiling zone of seven of the test rods produced cladding fragmentation, exposing the fuel column. The center rod had broken apart and fuel fragments had relocated between other rods in the bundle and to the lower spacer grid of the assembly. Despite this damage, the fuel columns remained intact throughout the nondestructive and destructive examinations. The results of the metallurgical examination of selected rod segments are being analyzed to determine the extent of severe oxidation of the cladding, limited fuel melting, and fuel oxidation from exposure to the coolant during film boiling. Several samples taken from separate rods and different elevations are being analyzed metallographically to determine axial and circumferential temperature profiles, determine a bundle temperature profile, and elucidate film boiling propagation from rod-to-rod not indicated by the cladding oxidation and embrittlement pattern of the nine-rod bundle.

The metallographic examination of the Test RIA 1-4 fuel rods was completed. The posttest analysis of the thermal-hydraulic system response during Test RIA 1-4 is in progress.

A topical report[7] on fuel analysis, "Molten Fuel-Coolant Interaction Occurring During A Severe Reactivity Initiated Accident Experiment," was completed and issued during the quarter. Fuel behavior results from the PBF RIA-ST-4 in-pile experiment were discussed and an analysis of molten fuel-coolant interaction (MFCI) was performed. The analysis showed that a high coolant pressure (35 MPa) recorded during the experiment was probably the result of an energetic MFCI initiated by a shock wave produced in the flow shroud at the time of rod failure. The MFCI was analyzed using a pressure detonation model. High coolant temperatures achieved during the experiment ($T < 940$ K) were due to the formation of superheated steam in the shroud, with the thermal-to-mechanical energy conversion ratio estimated to be ~0.3%.

The fine fragmentation of rod debris found in the RIA-ST-4 experiment was produced by a number of cooperating phenomena, from which three main mechanisms were proposed: (a) impact of molten debris on the shroud wall and in the coolant, (b) rupture of the frozen crust formed at the surface of debris particles by internal pressure arising from overheating entrapped liquid coolant droplets, and (c) coolant jet penetration of the particle crust. Mechanisms (b) and (c) were proposed on the basis of metallurgical and scanning electron microscopic examinations of debris. Phenomenological modeling of mechanisms (b) and (c) was performed, and the effects of key parameters were studied analytically.

A rough draft on the analysis of power-cooling-mismatch UO_2 fuel behavior was completed. The results of the study showed that UO_2 fuel damage accompanying PCM rod film boiling conditions varied from negligible to quite severe, depending on the linear operating power during the transient, the coolant conditions, and the duration of film boiling. The most severe transient operations produced central melting of the UO_2, which was, in general, contained at the center of the fuel pellets and produced no observable molten UO_2-cladding interaction. Fission-gas-induced swelling in previously irradiated rods produced cladding deformation, but did not seriously compromise fuel rod integrity and integral rod behavior as a result of transient operation.

2.1 IFA-511 Test Results
J. M. Broughton

The majority of experiments performed to understand light water reactor core thermal-hydraulic and fuel rod response during the heatup and reflood phases of a large break loss-of-coolant accident (LOCA) have been performed using electric heater rods instrumented with both external and internal cladding thermocouples. The applicability of these data have been subject to question because of uncertainty regarding the ability of electric heater rods to simulate the thermal response of nuclear fuel rods. In the IFA-511 Test Series[8] performed by the Halden Project, nuclear and electric heater rods have been exposed to identical thermal-hydraulic heatup and reflood conditions in an attempt to resolve this uncertainty. The test rods were instrumented with both external and internal cladding thermocouples to determine if external thermocouples provide an accurate measurement of cladding temperatures.

The results of the nuclear test, IFA-511.2,[9] and the test with Semiscale solid-type electric heater rods, IFA-511.3,[10] are discussed and compared below.

2.1.1 Test Design and Conduct. The tests were performed with a seven-rod bundle consisting of six peripheral rods symmetrically surrounding the center rod (see Figure 8). The heated length was 1.5 m. Five of the rods were instrumented with internal cladding thermocouples and one peripheral rod was instrumented with LOFT-type external thermocouples. The cladding thermocouples were positioned at various azimuthal orientations and at five different axial elevations, as shown in Figure 8.

The tests were performed by depressurizing the pressure flask containing the test bundle, and then allowing the rods to heat up to the desired temperature at a constant power of 1.0 to 3.0 kW/m. Reflood was first initiated by rapidly refilling the reflood pipes and pressure flask lower plenum. The bundle reflood then proceeded at a preselected flooding rate, which varied from approximately 2 to 10 cm/s. Twenty-one tests were performed in the IFA-511.2 Test Series and eleven tests were performed in the IFA-511.3 Test Series. The IFA-511.3 Test Series was prematurely terminated because of three failed heater rods. The test bundle will be rebuilt and the test series repeated.

2.1.2 Discussion of Base Case. The thermal response of the center nuclear rod during the highest temperature test, Run 5246, is shown in Figure 9. The measured cladding surface temperatures at four elevations between 0.15 and 0.90 m from the bottom of the fuel column, and the fuel centerline temperatures at 1.35 m are shown as a function of time after initiation of the transient. Loop isolation was initiated at 0 s and required approximately 13.5 s to complete. Loop blowdown was then initiated at 14.5 s and completed after approximately 30 s, and the system pressure then remained constant at about 0.15 MPa. Cladding temperatures were initially at about 500 K and did not change significantly until completion of loop isolation. The cladding temperatures increased approximately 25 K as the coolant flow stagnated after about 12 s, and then decreased following the coolant saturation temperature during blowdown. Dryout occurred within the rod bundle, starting at the top at about 40 s, and cladding temperatures then increased nearly linearly at approximately 12 K/s, which is about 95% of the calculated adiabatic heatup rate. The measured cladding peak temperature was 1103 K at 0.60 m at 134 s. At 128 s, the high rate reflood was initiated to fill the reflood piping system and test train lower plenum. When the reflood coolant entered the system, steam was generated via heat transfer from the piping, pressure flask, and test train structure and was rapidly expelled through the test bundle. This saturated two-phase mixture

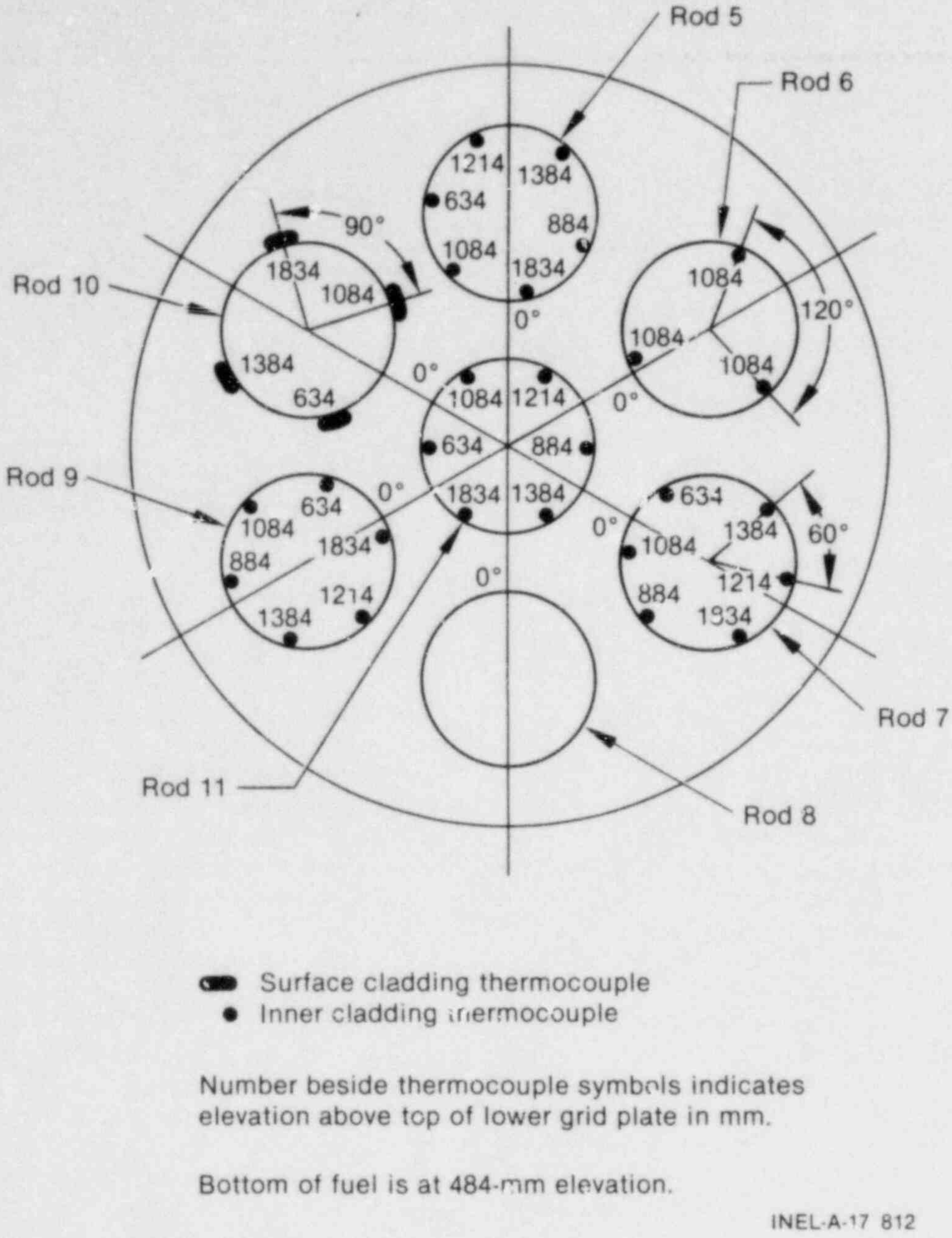

Figure 8. Schematic of the IFA-511.2 test assembly showing the axial and azimuthal location of the thermocouples.

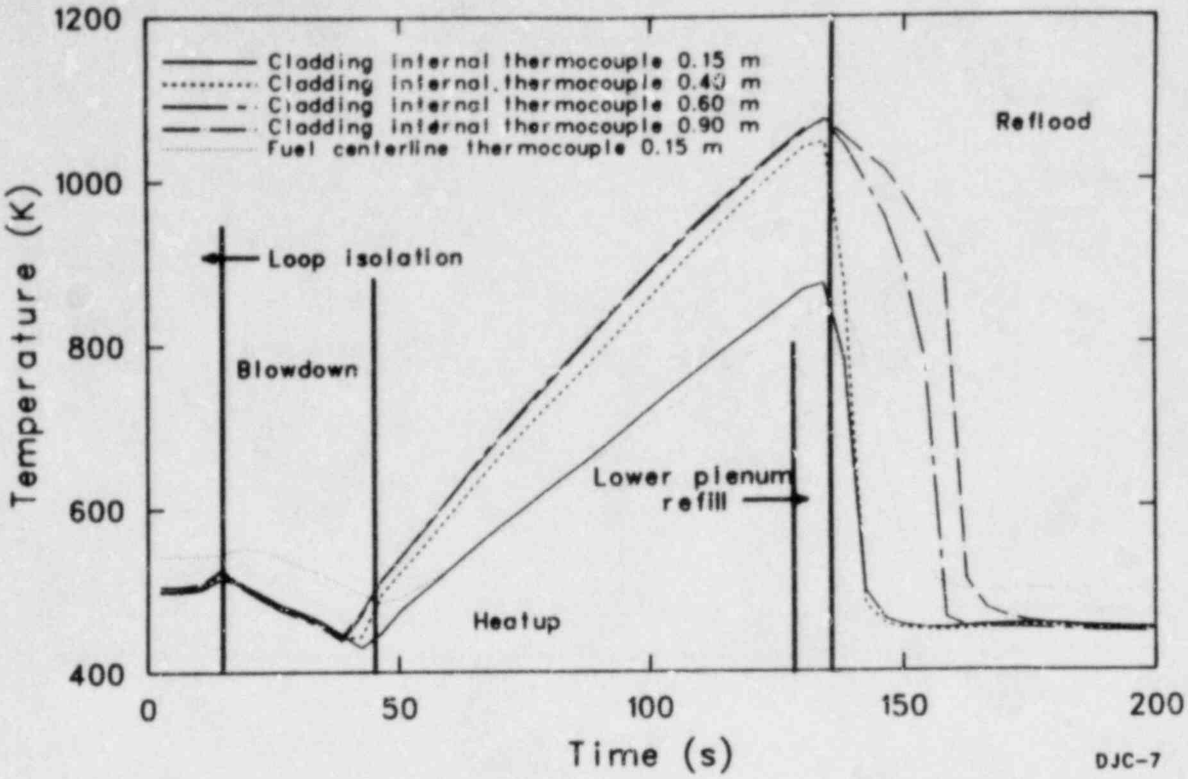

Figure 9. Thermal response of center rod (Rod 11) during the high temperature Run 5246.

(from liquid entrainment) significantly increased the rod surface heat transfer and for all tests, nuclear or electric, and rapidly terminated the cladding heatup.

The reflood rate after 134 s was almost 7 cm/s, which is relatively low, and a significant delay in time to quench was expected. However, cladding quench at the lower thermocouples, 0.15 and 0.40 m, occurred within 6 s, and the fuel centerline thermocouple indicated quenching of the fuel approximately 5 s later at 0.15 m. The cladding quench then rapidly progressed up the fuel rod, with the thermocouple at 0.9 m indicating quench at about 162 s. Also, there was, in general, no well-defined "quench temperature," which is generally indicated by a knee or point of inflection in the time-temperature plot in any of the IFA-511.2 tests. Instead, the cladding cooldown rate continually increased after initiation of reflood until the rod quenched.

Significant circumferential temperature differences on the peripheral tests rods were caused by radiation heat transfer to the cold wall of the pressure flask. This was determined by comparing measured cladding temperatures at the same elevation but on the rod inner surface facing the center rod and on the rod outer surface facing the pressure flask. Measured temperature differences ranged from 20 to 40 K, depending primarily on the magnitude of the absolute temperature, with the temperature difference increasing with higher cladding temperatures.

The repeatability of the system was of primary concern because of questions regarding the applicability and validity of direct comparisons between nuclear and electric rods. Of concern in any data comparison was attributing observed behavioral differences or similarities to the thermal and mechanical characteristics of the nuclear and electric rods and not to stochastic variation in the thermal-hydraulic boundary conditions. Therefore, three tests were performed repeating nearly exactly the initial conditions, reflood rate, and test sequencing for Run 5236, the base case. The response of the cladding internal and surface thermocouples at the 0.60-m elevation on the center rod (Rod 11) and a peripherial rod (Rod 10), respectively, are shown in Figure 10. During the blowdown and heatup, the three traces for

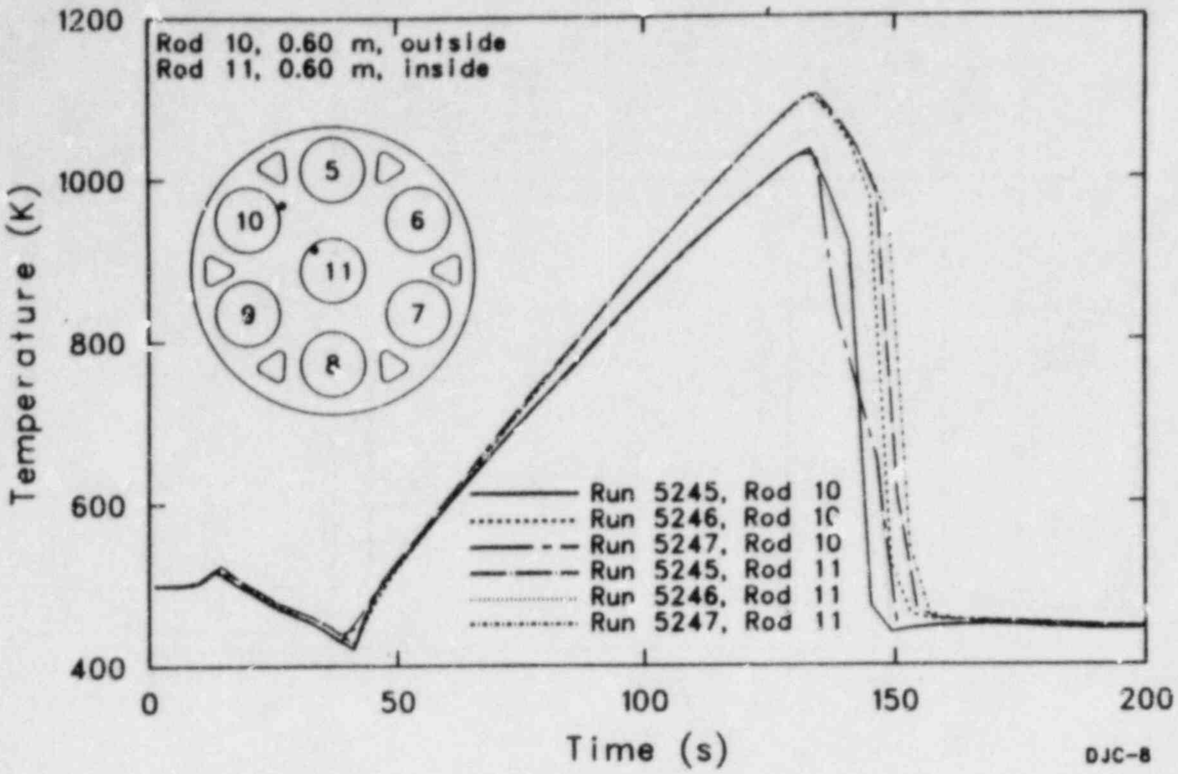

Figure 10. Comparison of measured inside and outside cladding surface temperatures during successive tests with identical test conditions.

each thermocouple nearly overlay. The heatup rate and measured cladding peak temperature of the internal thermocouple was significantly greater than the external thermocouple. This was caused by: (a) variation in the response time of internal and external thermocouples, and (b) the center rod was at a slightly greater power because of incorrect enrichment. During quench, there were slight variations in the thermal response of the fuel rods, but these are considered insignificant because the difference in quench time was only about 5 s.

2.1.3 Comparison of Cladding Internal and External Thermocouples. The comparative behavior of cladding internal and external thermocouples during the lower temperature test, Run 5236, and the base case, Run 5246, is shown in Figures 11 and 12, respectively. For both cases through blowdown and heatup until temperatures exceeded 700 K, the response of the external and internal thermocouples was nearly identical. However, after about 700 K, the cladding surface temperature as measured by the external thermocouple was less than the internal thermocouples and the difference increased thereafter. The measured cladding peak temperature was 24 to 40 K less, which was approximately 5% of the absolute temperature. The increased heat transfer within the bundle from flowing steam during the lower plenum refill terminated the temperature increase measured by the external thermocouples 5 to 10 s before a similar indication by the internal thermocouples. Throughout reflood, the indicated temperature of the external thermocouple was at least 50 K less than that indicated by the internal thermocouples, and the external thermocouples indicated quench 5 to 20 s earlier. The different thermal behavior indicated by the external thermocouples was primarily caused by heat transfer fin effects.

2.1.4 Effect of Rod Power, Cladding Peak Temperature, and Reflood Rate. Tests were performed in the IFA-511.2 series in which the rod peak power was increased from ~1 to ~3 kW/m and in which the time to reflood was increased such that the cladding peak temperature varied from about 580 to 1100 K. It was anticipated that the increased power and temperature would significantly delay the temperature turnaround and time to quench after initiation of reflood. This was not the case. Evidently, the thermal resistance across the fuel-cladding gap effectively decoupled the

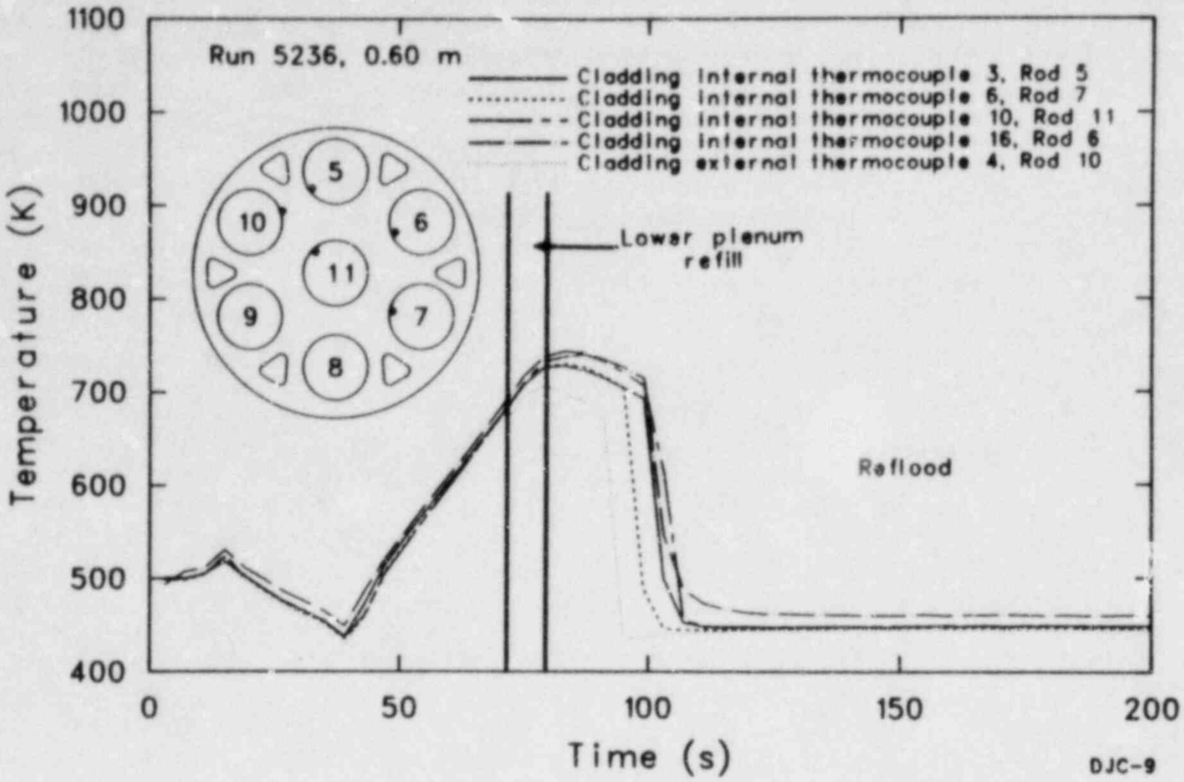

Figure 11. Comparison of cladding internal and external thermocouple measurements during the low temperature Run 5236.

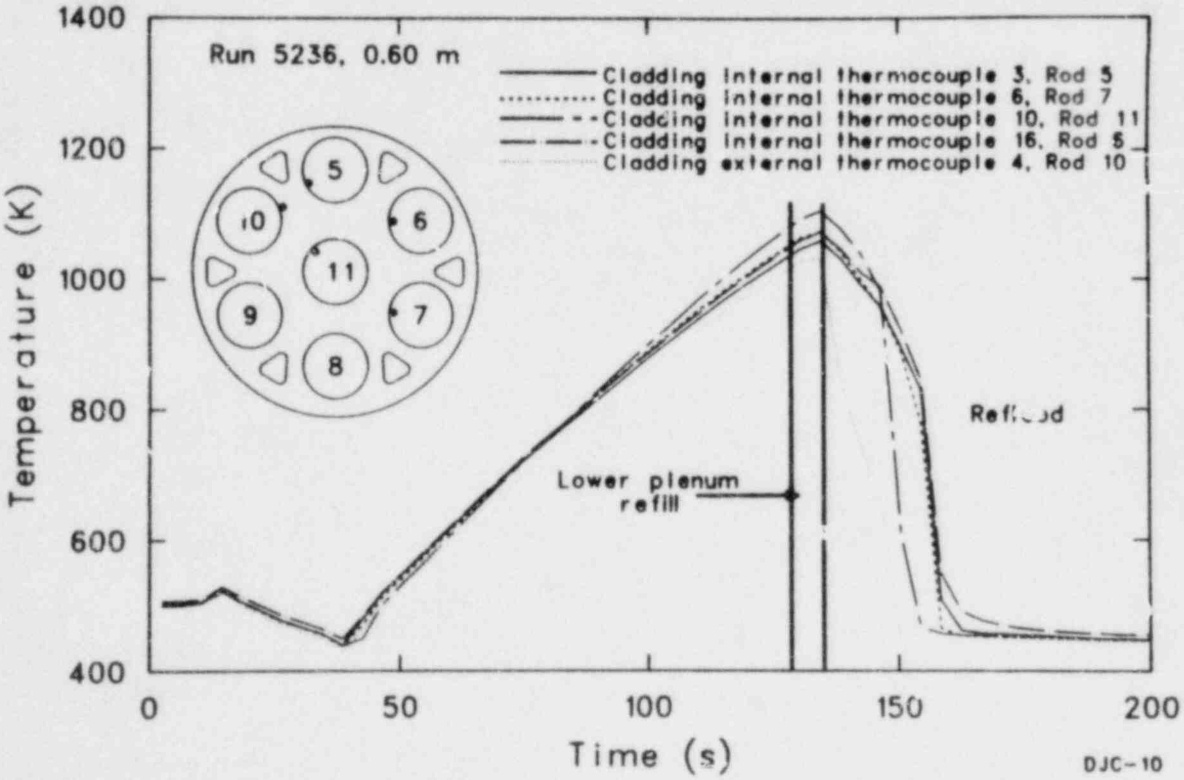

Figure 12. Comparison of internal and external cladding thermocouple measurements during the high temperature Run 5246.

cladding thermal response from the fuel and permitted the cladding to quench essentially independent of the fuel.

The reflood rate was varied from approximately 2 to 8 cm/s (12 to 55 g/s). The observed tendency was to cause higher cladding temperatures and delay quench with decreasing reflood rate. This general trend is consistent with previous results from out-of-pile electric heater rod heatup and reflood tests.[11-14]

2.1.5 Comparison of Nuclear and Electric Rod Behavior. The IFA-511.3 Test Series was intended as a duplication of the IFA-511.2 nuclear test series, except that the tests were performed with solid-type Semiscale electric heater rods. Unfortunately, three of the seven rods, Rods 5, 6, and 11, were failed at the start of the tests.

Plotted in Figure 13 is the measured cladding temperature (internal thermocouple) at the 0.60-m elevation for three tests, which were almost identical with the nuclear base case discussed previously. However, the traces are from Rod 7, a peripheral rod, and the thermocouple was located on the side of the fuel rod facing the failed center rod. The observed cladding thermal response during blowdown and heatup was basically the same as discussed previously. The measured cladding peak temperatures ranged from about 1060 to 1080 K but did not occur until approximately 10 s after initiation of bundle reflood. After the temperature turnaround at 145 s, the cladding temperatures gradually decreased until the rod quenched from about 790 K between 240 and 252 s for the three tests. The test repeatability was excellent as for the nuclear tests.

Also plotted in Figure 13 is a trace of the corresponding thermocouple from the identical nuclear test, Run 5246. During blowdown and heatup, the response of the nuclear and electric heater rods essentially overlay. However, the cladding temperature rise was terminated on the nuclear rod about 5 s after the initiation of system reflood at 128 s. This was about 12 s earlier than for the electric rod. The indicated quench of the nuclear rod was only 20 s after initiation of bundle reflood, compared with about 110 s for the electric heater rod. A similar comparison was made between the temperatures measured by the

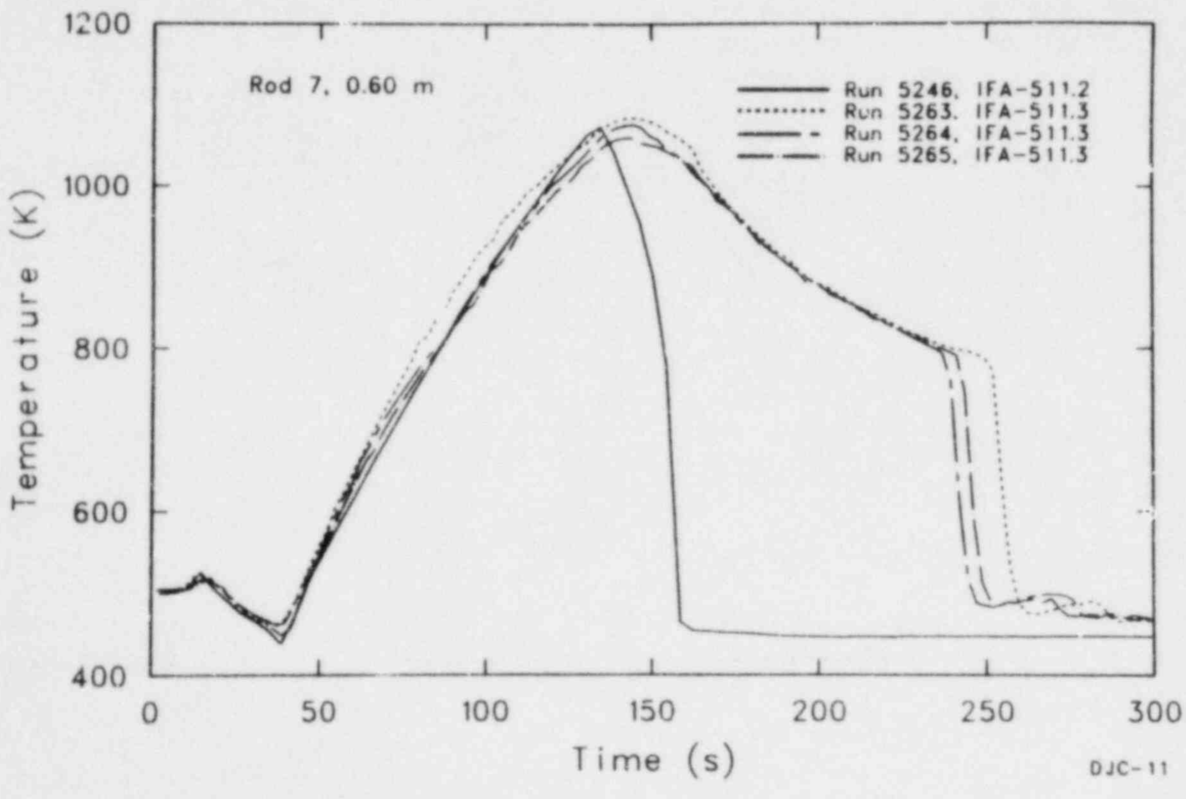

Figure 13. Comparison of electric and nuclear test rod response (internal thermocouples) during nearly identical test conditions.

external thermocouples, with similar differences in behavior observed. However, the difference in time to quench of the external thermocouples on the nuclear and electrical rods was about 50 s.

The three failed test rods may have introduced complicating factors into the bundle thermal-hydraulic behavior, which could have adversely affected the comparisons discussed in the preceding paragraph. The relatively cool, unheated flow channels could have caused local variations in the coolant conditions. The reduced heat input to the coolant during reflood resulted in a significant reduction in steam generation, approximately 43%, compared with the similar nuclear test. This should have resulted in less liquid entrainment and, thus, both a reduced heat transfer above the quench front and a faster rise rate of the liquid level within the bundle.

The comparative behavior of a nuclear test with approximately the same steaming rate as the electric rod bundle is presented in Figure 14. Again, the data are from the peripheral Rod 7 thermocouple at 0.60 m. The reflood rate within the electric heater rod bundle was approximately 8.5 cm/s, compared with 5.2 cm/s, for the nuclear test. The higher reflood rate into the electric heater rod bundle should have resulted in approximately a 66% faster rise rate of the liquid level within the bundle compared with the nuclear case. For the electric heater rod tests, the cladding peak temperature was about 825 K at 102 s, after which cladding temperatures decreased and the rod quenched from about 770 K between 170 and 180 s. The peak temperature for the nuclear rod was about 755 K at 90 s, and the rod quenched from 735 K at approximately 100 s. There was a significant increase in time to cladding peak temperature (~12 s), measured cladding peak temperature (~55 K), and time to quench (~75 s) between the electric heater rod and nuclear rod, although the liquid level rise rate should have been about 66% faster.

2.2 Conclusions

The following preliminary conclusions are based on the preceding discussions:

1. The reliability and repeatability of the system and test rod response were

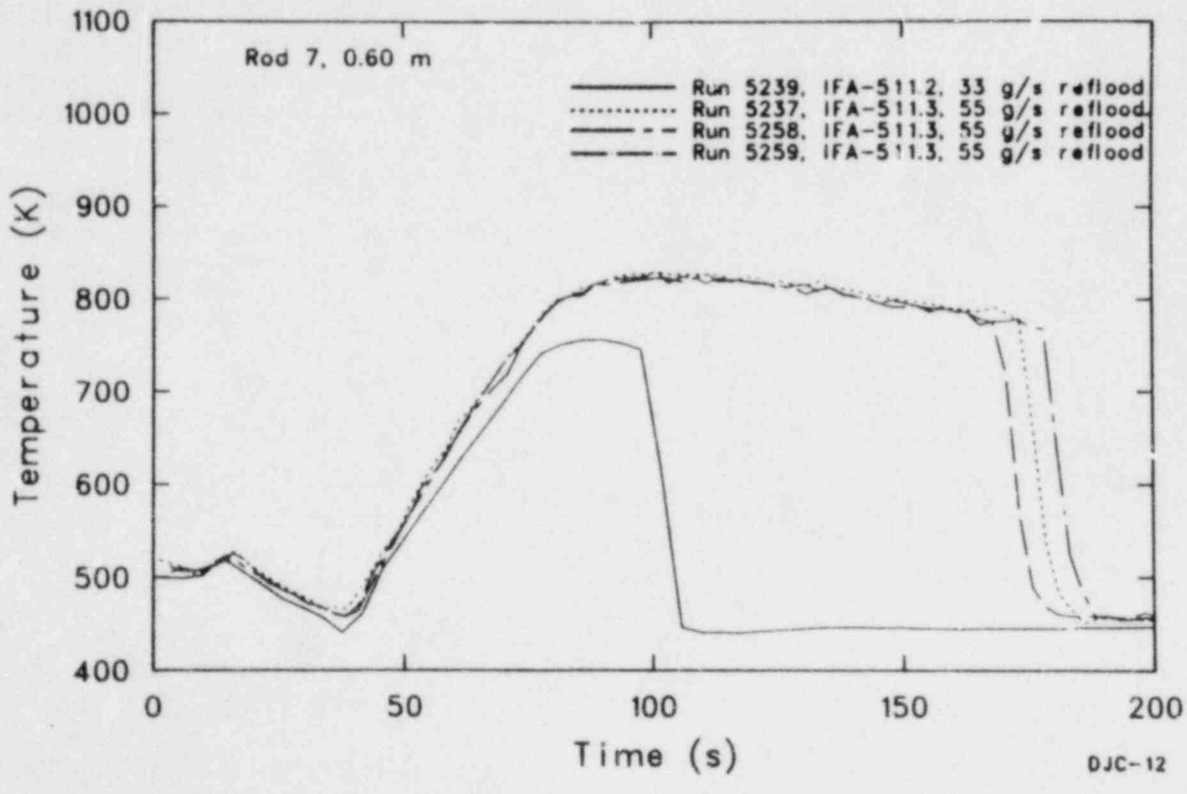

Figure 14. Comparison of electric and nuclear test rod response (internal thermocouples) during runs with nearly identical rod power and steaming rate during reflood.

excellent. The presence of cold walls created significant temperature differences around the circumference of the peripheral rods.

2. The response of the external cladding thermocouples was significantly different than the comparative internal cladding thermocouples during the reflood conditions tested.

3. Increasing the fuel rod power and cladding peak temperature did not significantly affect the thermal behavior of the nuclear rods during reflood, which is in agreement with the expectation from previous out-of-pile tests. Increasing the time to reflood resulted in rod behavior that was qualitatively similar to that observed in previous out-of-pile tests.

4. For the relatively limited test conditions against which the response of the nuclear and electric rods can be compared, the quench of the electric heater rods was delayed significantly and the cladding peak temperature was greater. Because of the failed heater rods in IFA-511.3, the test bundle will be rebuilt and the test series repeated.

3. TEST TRAIN DESIGN AND ASSEMBLY

J. P. Kester and K. G. Therp

The assembly of the test train for Test LOC-6, fabrication and assembly of the new coolant flow shroud assemblies for the TC-4 test train, and design and fabrication of the Operational Transient (OPTRAN) Test 1-1 fuel rod flow shrouds comprised the major efforts of the Test Train Assembly Facility during this quarter. The Test LOC-6 test train consisted of four individually shrouded fuel rods, two irradiated PWR design fuel rods, and two fresh PWR design fuel rods. The TC-4 test train is similar to the LOC-6 test train, except that all four test rods will be unirradiated, and two of the fuel rods will include a newly designed internal cladding thermocouple that is planned for possible future use in LOFT fuel bundles. Test LOC-6 was completed in January 1981, the Test TC-4 hardware will be completed in support of a May 1981 test date, and Test OPTRAN 1-1 is scheduled to be performed near the end of FY-81.

The Test LOC-6 fuel rod instrumentation consisted of four external cladding thermocouples per fuel rod, a plenum pressure transducer and plenum thermocouple in each fuel rod, and a fuel centerline thermocouple in each unirradiated fuel rod. The instrumentation on each fuel rod flow shroud consisted of two coolant differential thermocouples, two coolant inlet thermocouples, and two coolant outlet thermocouples. In addition, three inside shroud thermocouples and three outside shroud thermocouples are located on one of the preirradiated fuel rod flow shrouds and one of the unirradiated fuel rod shrouds.

The OPTRAN 1-1 test train configuration consists of four individually shrouded fuel rods installed in the previously provided Battelle Pacific Northwest Laboaratory four-quadrant test train. During the conduct of the test, two of the four rod/shroud assemblies will be replaced. The fuel rods for this test consist of six previously irradiated BWR/6-type fuel rods supplied by the General Electric Co. The Battelle four-quadrant test train is designed so that any of the four fuel rod/shroud assemblies, or the whole quadrant including the fuel rod, shroud, linear variable differential transformer, and flowmeter, can be replaced remotely and under water.

A supplemental final design review was held on the OPTRAN 1-1 test train design. The supplemental review was necessitated by revised test train requirements. These new requirements are the addition of two fuel rod flow shroud assemblies for the fuel rod changeout during the test, the deletion of fuel rod strain gages, and the addition of a fission product detection system (FPDS) sample injection and sampling line. The fabrication of the two fuel rod flow shrouds, fuel rod end caps, and components have been completed.

A final design review was held for the OPTRAN 1-2 test train. This test train is scheduled for completion at the end of FY-81, and incorporates some unique design features. The test train consists of two unirradiated and two irradiated BWR/6-type General Electric fuel rods. The fuel rods are enclosed in individual flow shrouds with a crossover tube from the outlet of each fresh fuel rod shroud to the inlet of the flow shroud of each of the irradiated fuel rods. Thus, the coolant flow through the irradiated fuel rod shroud will be preheated by a fresh fuel rod. Also, located at the upper end of each fresh fuel rod is a variable orifice for controlling the coolant flow rate to the associated irrradiated fuel rod. The variable orifice is remotely operated for control during the test. With these design features, conditions simulating a BWR main steam isolation valve closure event can be attained.

IV. CODE DEVELOPMENT DIVISION

F. Aguilar

The Code Development Division has a primary responsibility for the development of computer codes and analysis methods. The division provides the analytical research tools aimed at predicting the response of nuclear power reactors under normal, off-normal, and accident conditions. The codes developed in this division also provide a valuable analysis capability for experimental programs such as Semiscale, LOFT, and Thermal Fuels Behavior.

During the last quarter, the development of the TRAC-BD1 code was completed and the code released to the National Energy Software Center. An important part of the final phase of the development of TRAC-BD1 was a series of developmental assessment calculations. A brief description of the TRAC-BD1 code is given in Section 1 and the results of selected developmental assessment calculations are presented in Section 2.

1. TRAC-BWR DEVELOPMENT

W. L. Weaver, J. W. Spore, C. M. Mohr, G. L. Singer

TRAC-BD1[15] is a best estimate code for the analysis of loss-of-coolant accidents in boiling water reactors (BWRs). It is based on a developmental version of the TRAC code,[16] supplied by the Los Alamos National Laboratory, which contains a full nonequilibrium, two-fluid hydrodynamic model in both the one- and three-dimensional flow components. A significant number of changes, additions, and improvements have been made to the base code version to produce the TRAC-BD1 code. These can be divided into four main areas: (a) BWR component models, (b) hydrodynamic models, (c) heat transfer models, and (d) user-convenience features. Each of these areas is discussed briefly.

1.1 BWR Component Models

TRAC-BD1 provides distinct models for the hardware components that distinguish BWR systems: shrouded fuel bundles, jet pumps, and steam separator/dryers. The modeling of a BWR system is based on a new component called the CHAN, which simulates a fuel bundle and channel assembly. The CHAN is a one-dimensional flow component in which fuel rod and channel wall heat transfer models have been included. In modeling a BWR core region, CHAN components are connected across the core region of the VESSEL component, while the three-dimensional flow in the core bypass is calculated by the usual VESSEL hydrodynamics solution. This allows the hydrodynamic solution in the bypass to be separated from the solution within the CHAN components. There are leakage flow paths[17] and channel wall heat transfer models that do allow for communication between the flow inside the CHAN and the flow in the bypass. These models can be important for simulation of a BWR reflood transient, since penetration of the emergency core cooling system (ECCS) water from the upper plenum into the fuel bundle can be limited by the countercurrent flow limiting (CCFL) phenomenon that will occur at the fuel bundle upper tie plate.[18] The leakage flow paths from the bypass to the BWR fuel bundle provide another mechanism for allowing ECCS water to enter the fuel bundle. The channel wall heat transfer model can also provide a heat transfer path for removing energy from the fuel bundle even if CCFL is occurring for a significant length of time during a BWR LOCA. In the case of the BWR/6 ECC system, the low pressure core injection system is used to flood the core bypass, which will improve heat transfer through this path as well as provide ECCS water to the leakage flow paths in the bottom of the BWR fuel bundles.

A jet pump component (JETP) was also developed for TRAC-BD1. The momentum equations for the TRAC TEE component were modified so that they accurately represented the momentum exchange that occurs in a jet pump. The jet pump model assumes complete mixing in the throat section of the jet pump and represents irreversible pressure losses, with appropriate loss coefficients for abrupt or smooth area changes. The user only has to supply a minimum of geometric input for the JETP component because an input processor has been developed for this component.

In the TRAC-BD1 steam separator-dryer model, separation of steam and liquid is accomplished by appropriate choices for phasic loss coefficients in the separator/dryer region of the TRAC-BD1 VESSEL component. The user only has to identify the region in the VESSEL component that will contain the separator/dryer and the code will initialize the model automatically. The present model assumes 100% separation.

In addition to these BWR components, most of PWR TRAC components are available in TRAC-BD1. The component models available in TRAC-BD1 are VESSEL, PIPE, PUMP, VALVE, FILL, BREAK, TEE, CHAN, and JETP.

1.2 TRAC-BD1 Heat Transfer Models

Application of the TRAC computer program to BWRs requires additional heat transfer modeling capability beyond what was available in previous versions. Heat transfer models developed for TRAC-BD1 are:

1. Rod-to-rod, rod-to-coolant, and rod-to-channel-wall radiation heat transfer model
2. Channel wall heat transfer model
3. BWR departure from nucleate boiling model
4. Quench propagation model on the inside of the channel wall as well as on each of the rod groups, both bottom up as well as falling film
5. Improved heat slab modeling techniques
6. ANSI/ANS 5.1 decay heat model
7. Improved TRAC-PD2 wall heat transfer model.

Radiation heat transfer can be a significant mode of heat transfer in a BWR fuel bundle, especially if the bundle is being steam cooled due to complete shutoff of ECCS water penetration and if the channel wall is being cooled on the outside by a supply of ECCS water. This situation can occur if CCFL at the upper tie plate continues for a long time and if water from the low pressure coolant injection system has flooded the core bypass regions. The TRAC-BD1 radiation model is described in Reference 19.

Departure from nucleate boiling (DNB) in a BWR system cannot be described by a local condition correlation due to the nonuniform axial heat flux profile and the high steady state steam qualities that exist in a BWR fuel bundle. As a result, an integral correlation must be used. The integral correlation included in TRAC-BD1 is the CISE-GE boiling length correlation given in Reference 20.

The quench front propagation model employed in TRAC-BD1 is described in Reference 21 and is applied to each rod group within a CHAN component and to the inside of the channel wall. The quenching of the channel wall can be an important phenomenon to model in a BWR fuel bundle, since the quenched channel wall results in a lower sink temperature for radiation heat transfer from the rods and also results in a higher effective emissivity for channel wall surface (see Reference 17).

Improved heat slab modeling techniques were required to accurately simulate the control rod guide tubes, vessel wall, and other heat structures in the lower plenum of a BWR vessel. Pipe and jet pump wall heat transfer models were modified so that a user could simulate the heat transfer between the fluid inside of the guide tubes and the fluid in the lower plenum, as well as the heat transfer between the fluid inside the jet pumps and the fluid in the downcomer. Previous versions of TRAC restricted the user to lumped-parameter heat structure models in the vessel. This has been modified in TRAC-BD1 so that the user can specify as many nodes as desired to simulate the conduction heat transfer within a structure. This is a significant improvement for vessel wall heat transfer modeling.

Finally, the wall heat transfer correlation package was smoothed to eliminate discontinuities in the boiling curve that result in instabilities in the TRAC calculation.

1.3 TRAC-BD1 Hydrodynamics

TRAC-BD1 uses the same, two-fluid hydrodynamic equations[16,21] as previous versions of the TRAC code for both the one- and three-dimensional flow components. The semi-implicit numerical scheme used in previous versions of TRAC is used in TRAC-BD1.

In order to model choking, it was necessary to include a critical flow model in TRAC-BD1, since modeling choking with a fine nodalization of the break plane with only semi-implicit numerics is impractical. The critical flow model included in TRAC-BD1 is the RELAP5/MODO[22] nonhomogeneous equilibrium critical flow model. This model appears to be adequate for BWR applications, since in BWR LOCA analysis, nonequilibrium effects on critical flow are negligible.

A CCFL (see Reference 18) model has also been implemented into TRAC-BD1. On the basis of the data of Jones,[18] Tobin,[23] and Naitok,[24] a CCFL correlation using Kutateladze (scaling) was developed for the BWR upper tie plate. The general form of this correlation was recommended by Sun[25] for BWR 7 x 7 bundle upper tie plates and also by Sun[26] for BWR 8 x 8 upper tie plates and is given as

$$K_g^{1/2} + m\,K_f^{1/2} = K^{1/2}.$$

The constants chosen for [illegible]lation of BWR upper tie plates are m = 1[illegible]nd K = 3.2. For 8 x 8 bundles, Sun recommends a higher value for K. However, due to the dependence of K on the injection method of steam into the channel, it was decided to use the lower value, which appears to correlate both 7 x 7 and 8 x 8 data satisfactorily. Comparisons of the correlation with m = 1.0 and K = 3.2 with 7 x 7 bundle data can be found in Reference 27.

CCFL has also been observed at the side-entry orifices of a BWR fuel bundle (see Reference 27). Sun (see Reference 26) recommends a correlation similar to the above equation for the side-entry orifice, except that the m = 0.6 and the K are given as a function of the Bond number based on the wetted perimeter of the side-entry orifice. This correlation for the side-entry orifice is also available in TRAC-BD1.

Both the choking model and the CCFL model are implemented into the TRAC hydrodynamics solution as limit lines. For the choking model, the limit is a critical mixture velocity defined by the critical flow model. For the CCFL model, the limit is a critical liquid downflow rate defined by the CCFL correlation. In both cases, if the limit or critical velocity is exceeded by the normal TRAC hydrodynamics solution, then the linearized TRAC momentum equations are modified such that the hydrodynamics solution will follow the limit line defined by the appropriate correlation.

1.4 User Convenience Features of TRAC-BD1

A number of user convenience features or modeling capabilities have been added to TRAC-BD1. These new features included in the program are:

1. Increased input error checking
2. More readable output
3. Multiple pipe-to-vessel connection capability
4. Improved VALVE component, which allows for modeling banks of relief valves as well as motor-controlled valves
5. Downcomer level trip
6. Improved heat transfer modeling capability (discussed in Section 1.2)
7. Slab or cylindrical VESSEL noding option.

These features are self-explanatory, except the multiple pipe-to-vessel connection capability. This code feature allows more than one pipe to be connected to a single vessel hydrodynamic cell. Previous versions of TRAC allowed only one pipe connection per vessel cell. This multiple connection capability allows for coarser noding in the VESSEL component. The downcomer level trip is also an important BWR feature, since many of the BWR safety systems are initiated by a low downcomer level.

2. TRAC-BD1 DEVELOPMENT ASSESSMENT

R. W. Shumway and R. E. Phillips

An important aspect of the final phase of the development of the initial version of the TRAC-BD1 code was a series of developmental assessment calculations. The calculations were performed to provide an initial assessment of the various modeling capabilities of the TRAC-BD1 code. These calculations can be divided into three main areas: (a) separate effects hydrodynamic tests, (b) separate effects heat transfer tests, and (c) integral system tests.

The separate effects hydrodynamic tests were used to assess the hydrodynamic model in the TRAC-BD1 code. The tests simulated with the code were:

1. Full-Scale[28] and One-Sixth-Scale[29] Jet Pump Tests

2. Edwards Pipe Blowdown[30]

3. Semiscale Nozzle Flow Test S-02-1[31]

4. General Electric 8 x 8 Bundle CCFL Tests[32]

5. CISE Adiabatic Pipe Tests[33]

6. GE Small Vessel Level Swell Test 1004-3.[34]

The separate effects heat transfer tests that were used to assess the TRAC-BD1 heat transfer package were:

1. FLECHT Test 9077[35]

2. GOTA Spray Test 78[36]

3. GOTA Radiation Only Test[36]

4. THTF Test 3.06.6B[37]

5. TCHF Test 110[38]

6. TLTA Test 4904.[39]

The integral system tests that were used to assess the performance of the whole code were:

1. TLTA Test 6422[40]

2. A Small Break 10% Pump Suction Line Break BWR/6 LOCA

3. A Large Break BWR/6 200% Pump Suction Line Break LOCA.

The results of several of these simulations have been reported in previous quarterly progress reports and will not be repeated here. These previously reported tests are the One-Sixth Jet Pump tests,[41] the GOTA Radiation Only Test,[42] and the TCHF Test 110.[43]

Results of a single test from each of the three main areas will be presented here. The results of the remaining developmental assessment calculations can be found in the TRAC-BD1 manual (see Reference 15). The separate effects hydrodynamic test chosen was the Semiscale Nozzle Test S-02-1, which exercises the critical flow model in the TRAC-BD1 code. The measured upstream conditions (pressure and temperature) were used as input to the code. The computed mass flow rate through the nozzle as well as the mass flow rate measured by a drag disk are shown in Figure 15. The agreement as shown in Figure 15 is excellent. The separate effects heat transfer test chosen was TLTA Test 4904. This test exercises the transient critical heat flux correlation in TRAC-BD1, as well as the transition and film boiling heat transfer correlations. The TRAC-BD1 CHAN component was used to model the simulated fuel bundle in this test and the measured fuel bundle inlet flow rate was used to drive the TRAC simulation. The results of this computation are shown in Figure 16. Excellent agreement with the data is obtained when the recommended critical quality-boiling length critical heat flux correlation is used. Also shown are the results of the same simulation using the Biasi local conditions critical heat flux correlation used in the PWR versions of the TRAC code. Finally, several typical results from the simulation of TLTA Test 6422 are presented in Figures 17 and 18. TLTA Test 6422 is a simulation of a 200% pump suction line break in a BWR/6 plant. The average power fuel bundle was simulated and all ECCS systems were utilized. The computed and measured core inlet mass flow rates are shown in Figure 17. Particularly noteworthy is the correct prediction of the core flow rate surge at 12 s due to lower plenum flashing. The TRAC-BD1 computation lies within the error bounds of the data and reproduces all of the essential features of the measurement. Figure 18 shows the measured temperatures at the top of 2 of the 64 simulated fuel rods in the bundle. The measured cladding inner surface temperatures for a rod in the center of the bundle, as well as for a rod on the outside of the bundle, are shown.

These two measurements show the range of temperatures experienced by the various rods in a fuel bundle. The TRAC-BD1 simulation of this fuel bundle used a single fuel rod group to predict the behavior of the average fuel rod in the bundle. The computed cladding outer surface temperature is shown in Figure 18. The agreement between the data and the computed temperatures is good, when one compensates for the temperature drop from the cladding inner to outer surface and for the nonuniform radial void and velocity profiles in the actual bundle, which TRAC-BD1 is not able to

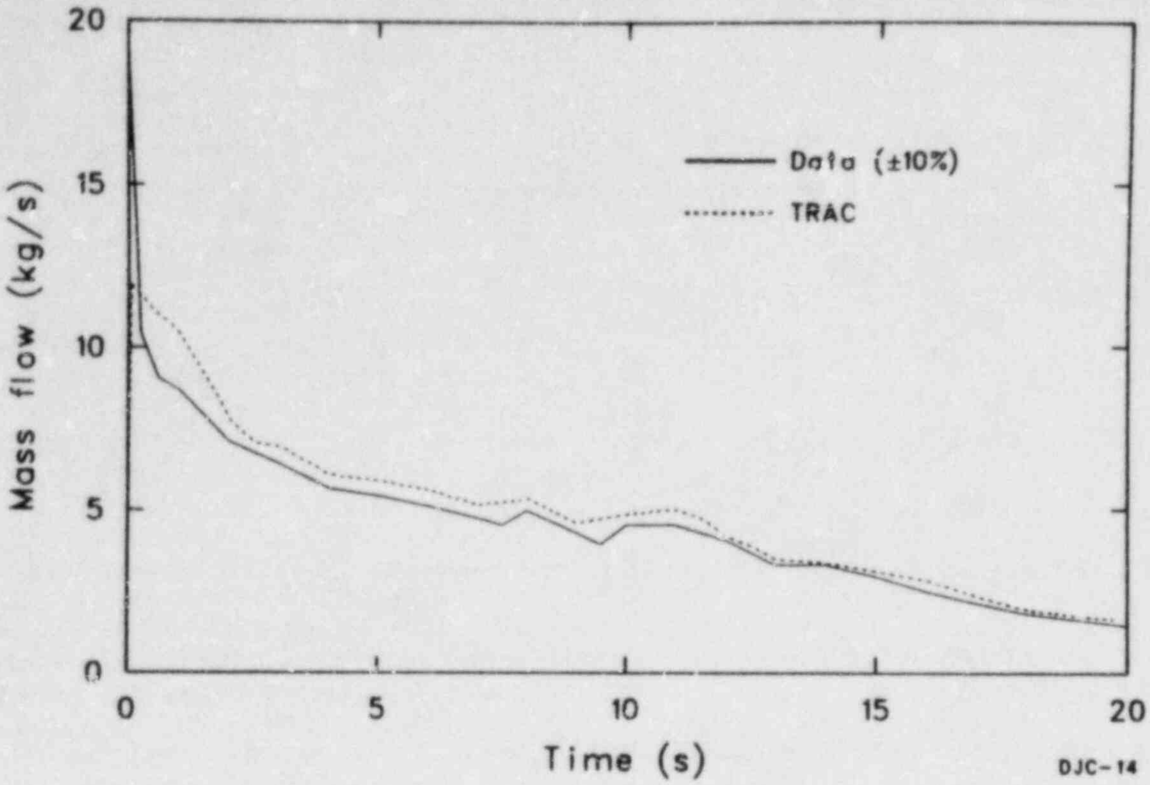

Figure 15. Hot leg break mass flow rate during Semiscale Test S-02-1.

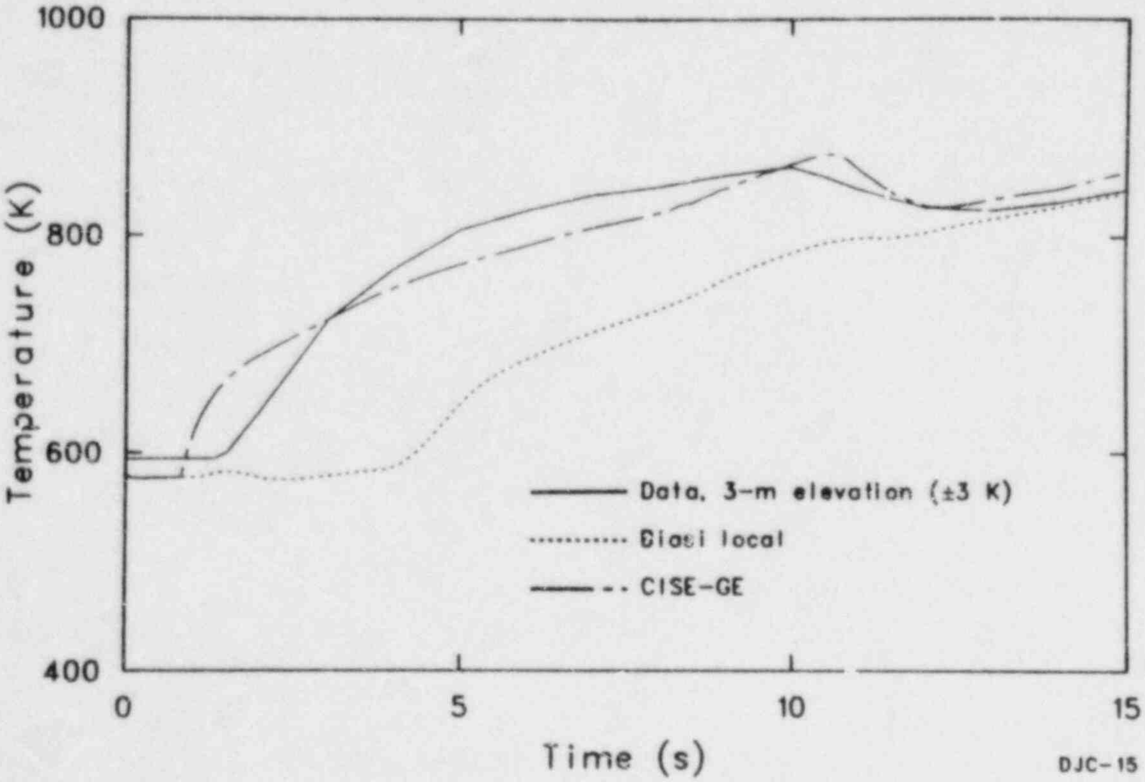

Figure 16. Rod temperature at the 3-m elevation during TLTA Test 4904-4?.

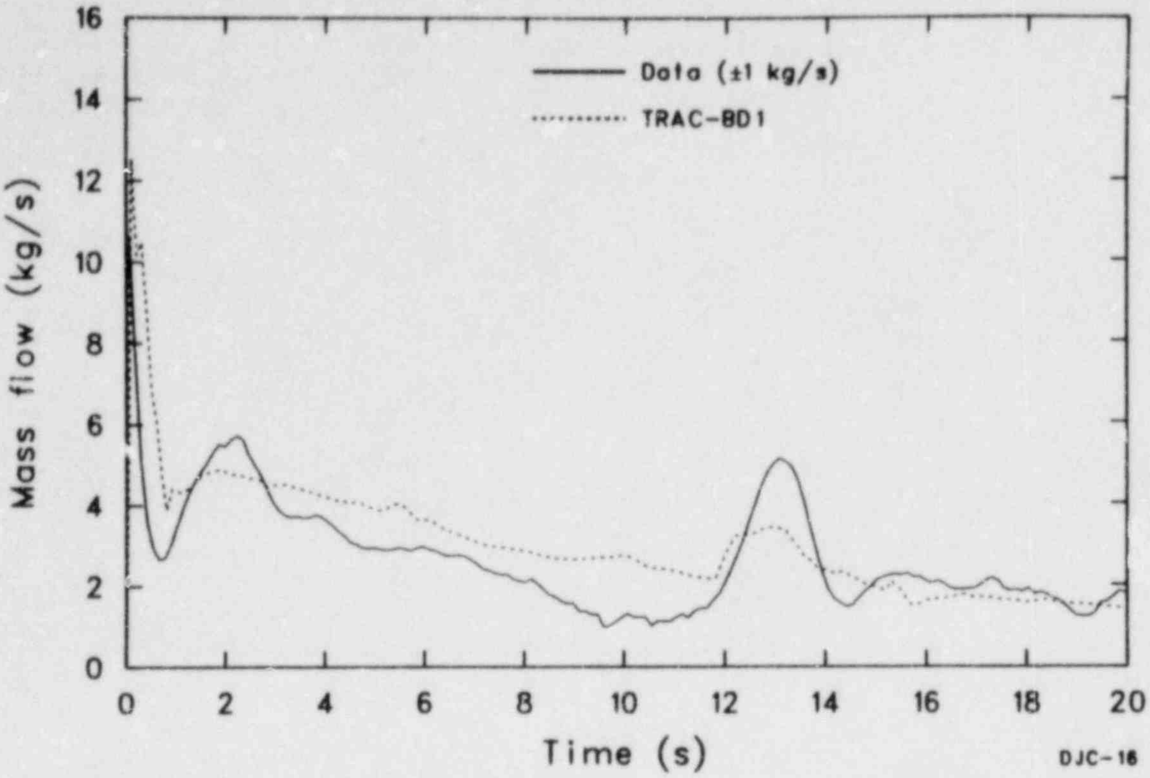

Figure 17. Core inlet flow during TLTA Test 6422-3.

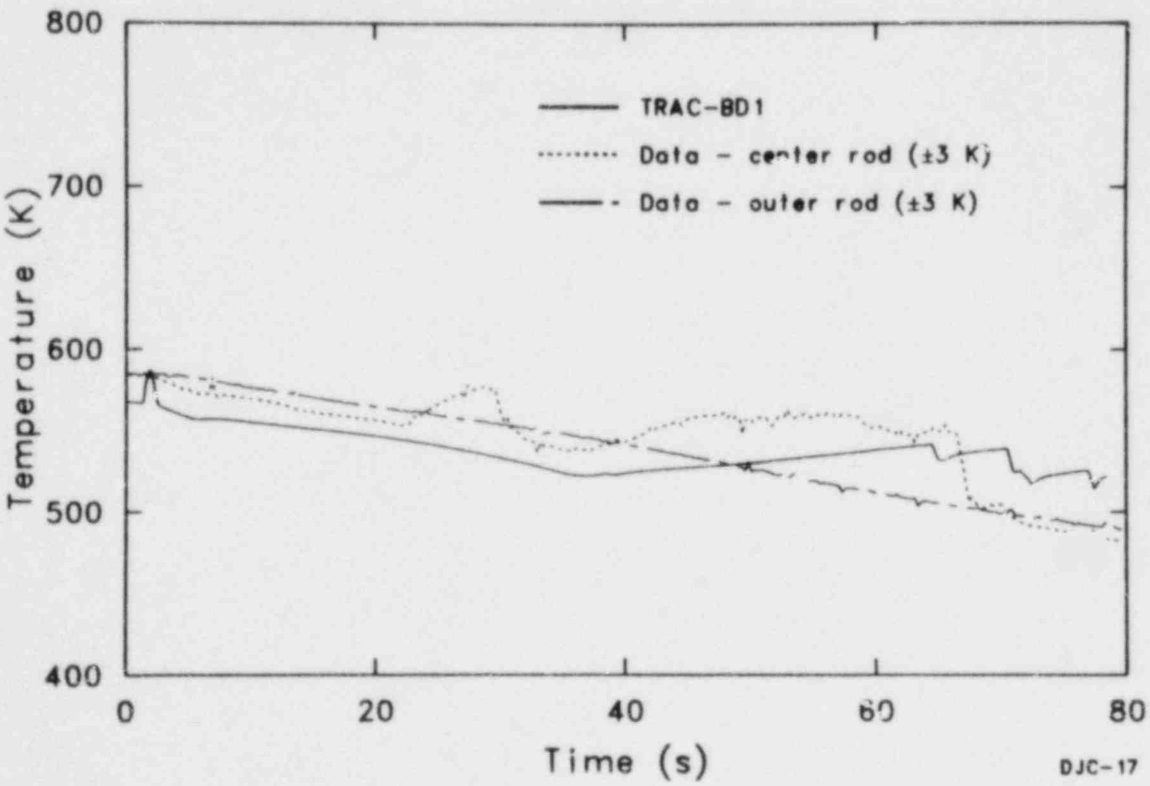

Figure 18. Fuel rod temperatures during TLTA Test 6422-3.

simulate. Taken as a whole, the agreement between the TRAC-BD1 simulation of the various tests and the measured data is excellent.

Finally, Table 2 presents a comparison of the computer central processing unit (CPU) time required for the execution of a 200% pump suction line break using TRAC-BD1 and RELAP4/MOD6. This table shows that TRAC-BD1 is faster than RELAP4/MOD6 for any combination of hydrodynamic cells and heat structures.

Table 2. Execution statistics, RELAP4—TRAC-BD1 comparison

Case: BWR/6 (218-624) plant 200% pump suction break

	RELAP4/MOD6	TRAC-BD1
Total cells	34	111
Vessel cells	8	32
Heat structures	45	129
Problem time	40 s	45 s
CPU time	2022 s	1675 s
CPU/problem time	50.5 s	37.2 s
CPU/(problem time)(cell)	1.49 s	0.34 s
CPU time/(problem time)(cell)(heat structure)	0.033 s	0.0026 s

V. CODE ASSESSMENT AND APPLICATIONS DIVISION

B. F. Saffell, Jr., Manager

The Code Assessment and Applications Division (CAAD) has a primary responsibility to the Nuclear Regulatory Commission (NRC) for the assessment of thermal-hydraulic and fuel behavior analytical codes. Data obtained from foreign and domestic experimental programs are used to assess the results of code calculations. The purpose of code assessment is to provide a quantitative assessment of the computer programs being developed for the NRC. The NRC/Reactor Safety Research (RSR) Data Bank is being developed to provide the data base and data processing capabilities needed for quantitative assessment activities. In addition to assessing codes, the CAAD is the technical advisor to the NRC on industry cooperative safety experimental programs. The purpose of this activity is to ensure that data from these experimental programs are adequate for assessment of thermal-hydraulic codes. The CAAD is also assisting in the NRC Standard Problem Program in which computer code simulations of nuclear safety related transient tests are performed by participants using calculation techniques (computer codes) of their choice. This program is a cooperative effort among the NRC, U.S. reactor vendors, and the international nuclear community. The latest program in which the CAAD is providing assistance to the NRC is the Severe Accident Sequence Analysis (SASA) task. The purpose of this program is to identify and analyze accident or upset sequences of events and to provide assistance during commercial reactor transients such as occurred at Three Mile Island.

In the following sections, results are presented for

1. An analysis of jet pump data to be used in technical support efforts

2. The completion of the assessment of FRAPCON-2, a fuel rod analysis computer code

3. A "blind" prediction for the LOBI test facility in Italy using RELAP4/MOD6, a thermal-hydraulic computer code.

1. ONE-SIXTH-SCALE JET PUMP EXPERIMENT DATA ANALYSIS

G. E. Wilson

The Idaho National Engineering Laboratory one-sixth-scale jet pump experiment[44] was conducted to provide a data base over a wide range of on- and off-design performance conditions. This data base was considered necessary for the continued development and assessment of jet pump models in boiling water reactor (BWR) thermal-hydraulic codes. To satisfy the objective of the test program, over 200 steady state, subcooled data points were generated.

Subsequent to the completion of the experiment, a data analysis[45] was performed for the steady state data points. These studies addressed (a) evaluation and correction of a measurement bias in the differential pressure (DP) cells as a function of system pressure, (b) evaluation of the sensitivity of the calculated M-N data[a] to potential DP cell and flow measurement errors, (c) characterization of the M-N relationships and their uncertainties and (d) recommendations for the use of the data base in code development and assessment activities.

On the basis of the study results, the data base is not considered to be unduly sensitive to flow measurement errors. The portion of the data base encompassing positive drive flow with large reverse suction flow (i.e., $M > -0.8$) shows considerable scatter, which appears to be related to potential DP measurement errors. This portion of the data base is not considered suitable for code development or assessment. The remaining data base is not considered to be unduly sensitive to DP measurement errors.

Figure 19 exemplifies the M-N characterization of a selected portion of the data base. The solid center line in the figure is the mean N calculated with standard linear regression techniques, assuming the data points are well represented by a power

a. M and N are dimensionless numbers used to characterize jet pump performance. M is associated with suction and drive mass flows. N characterizes the energy change in the suction and drive flow streams.

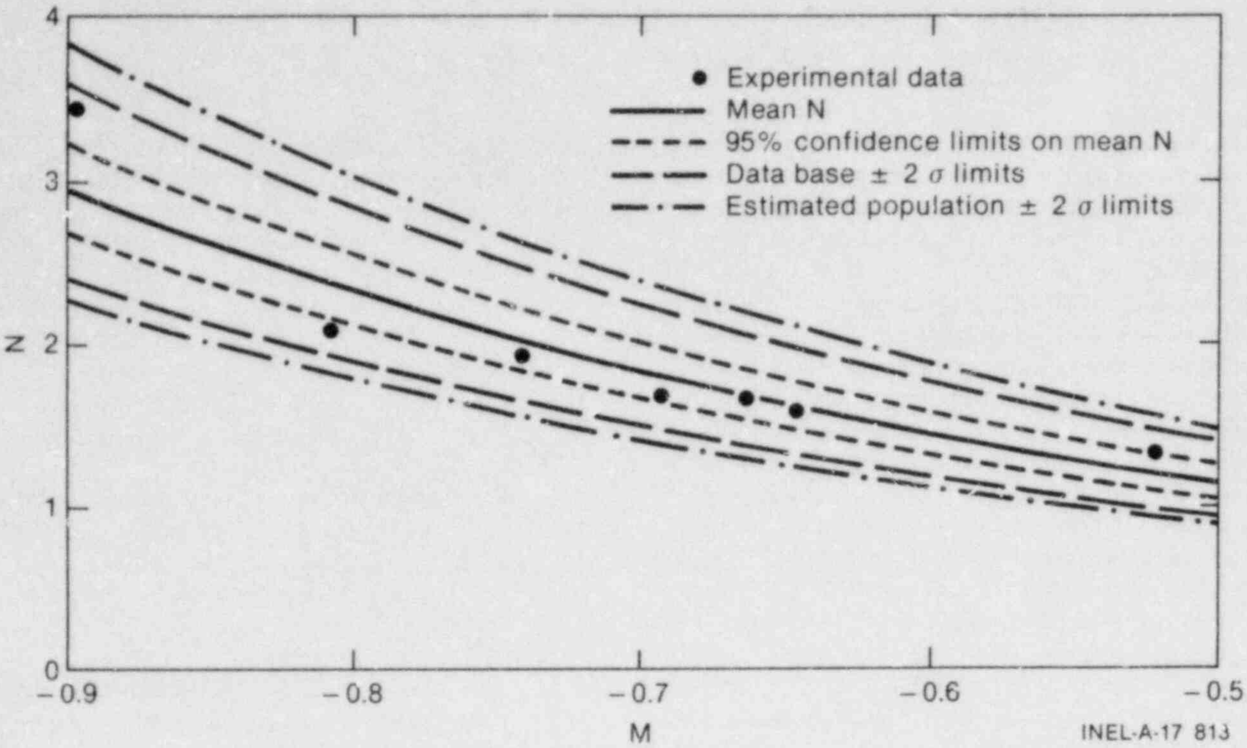

Figure 19. Jet pump data base characterization for positive drive flow with -09 ≤ M ≤ -0.5.

curve form. As such, this curve represents a best estimate of the pump performance exclusive of random errors, as projected from the existing experimental data. The dashed lines next to the mean N curve indicate the uncertainty in the mean N curve at a 95% confidence level. The next set of dashed lines (large dash) identify the region in which approximately 95% of the experimental data points lie. The outermost dashed lines show the region in which one would project 95% of all data points (existing and new) to occupy if the new data points came from the same population. Characterizations of all of the acceptable data base are given in Reference 45.

Several ways in which the characterized data base can be used to evaluate jet pump models are:

1. Use the appropriate system parameters from the experimental data as boundary conditions for the jet pump model to develop calculated M-N data points. These calculated data points are then compared with the data base 95% range on a point by point basis.

2. Use the calculated M-N data points as described above to develop a calculated mean N curve. This curve could then be compared to the 95% confidence band on the experimental mean N.

3. Select arbitrary boundary conditions to drive the code, comparing the resultant calculated M-N data points with the largest (population) 95% range.

In Approaches 1 and 3, if 95% of the model-calculated data lie within the specified bands, one can say that with 95% confidence, the code produces results that are as reliable as the experimental data base. A similar judgment can be made, providing the calculated mean N of Approach 2 lies within the 95% confidence limits for the experimental mean N. Ideally, boundary conditions for each and every experimental data point would be used in Approaches 1 and 2; however, a reduced number of calculations may be satisfactory, providing a sound, statistically based selection technique is used. Similar considerations relative to the number of data points are appropriate to Approach 3. The experimental system parameters necessary to execute all of the suggested approaches are contained in the extensive data tables provided in Reference 45.

2. INDEPENDENT ASSESSMENT OF THE STEADY STATE FUEL ROD ANALYSIS CODE, FRAPCON-2

E. T. Laats, R. Chambers, N. L. Hampton

The steady state fuel rod analysis program, FRAPCON-2,[46,a] was independently assessed[47] for the United States Nuclear Regulatory Commission (USNRC). The general goals of this assessment were to characterize code predictive capabilities, to identify the most appropriate fuel rod mechanical deformation and fission gas release models residing in FRAPCON-2, to make recommendations for future model development, and to aid the general code user when running the code and interpreting its results.

During these studies, FRAPCON-2 calculations were compared with experimental data from some 750 test rods. First, an analysis was conducted of the three rod deformation models and five fission gas release models residing in FRAPCON-2. The results indicated that FRACAS-II was, overall, the most appropriate rod deformation model, and FASTGRASS Mod-1 was the most appropriate fission gas release model. Emphasis was then placed on assessing FRAPCON-2 capabilities in the areas of thermal, deformation, and internal gas response when the selected deformation and gas release models were used concurrently.

The following conclusions were drawn:

1. The overall predictive capabilities of FRAPCON-2 were superior to those of its predecessor, FRAPCON-1, especially in the area of thermal performance.

2. Fuel centerline and off-centerline temperature calculations were in general agreement with experimental data. Best agreement was noted for rods with radial geometries, fill gas, and operating power levels that were typical of commercial fuel rods during beginning-of-life (BOL) operation. Less agreement was noted for these commercial-type rods that operated at high burnup levels. Also, less accurate calculations were obtained for rods of atypical geometries and nonhelium fill gases, regardless of burnup.

3. On the basis of the fuel temperature conclusions, the stored energy calculations were thought to be most accurate for rods of commercial design, operating at BOL. A tendency to undercalculate the stored energy will most likely occur when a rod is fabricated with nonhelium fill gases, or when the burnup of a fuel rod is high, or both.

4. The onset of pellet-to-cladding gap closure was calculated to occur within the range of the measured values. Best agreement between FRAPCON-2 and the data was obtained when the observed power level at closure was between 15 and 30 kW/m. Much calculational difficulty and numerical inaccuracy was obtained for rods that had small as-built gap sizes (e.g., diametral gap less than 1% of the pellet diameter).

5. Following the onset of gap closure, the rate of increase of the cladding strain with increasing power was overestimated until hard gap closure was attained.

6. Fission gas release was most accurately calculated when the measured release was less than 25% of the gas generated. Also, the calculated amount of gas release was less than the measured for rods that attained high burnup levels.

7. Internal pressure calculations were most accurate for rods with a large plenum volume, and least accurate for small plenum, pressurized rods.

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a. FRAPCON, Mod-2, MATPRO Version 11 (Rev. 1) Idaho National Engineering Laboratory Code Configuration Control Number H019882B.

3. A RELAP4/MOD6 PREDICTION OF THE LOBI PREX TEST

C. B. Davis

A "blind" prediction of the Loop Blowdown Investigations (LOBI) Pre-Prediction Exercise (PREX) Test was completed using RELAP4/MOD6.[a] The LOBI facility[18] is a 1/700-scale model of a large pressurized water reactor and is located at the Joint Research Centre in Ispra, Italy. The PREX Test[49] represented a 200% double-ended cold leg break. The PREX Test initial conditions were approximately 15 MPa primary system pressure, 565 K cold leg temperature, 603 K hot leg temperature, and 5.38 MW core power.

The PREX Test was the first integral systems blowdown experiment conducted in the LOBI facility. Thus, predictions of the test provided a unique opportunity to assess the capability of thermal-hydraulic computer codes to predict blowdown and refill phenomena in a facility that had not previously produced data for code development or assessment. Representatives from 16 organizations, including the INEL, and five nations participated in the PREX to evaluate their respective computer codes. Initial and boundary conditions measured during the test were supplied to all PREX participants. However, the predictions were made without knowledge of the experimental results and, thus, were "blind" predictions.

a. RELAP4/MOD6, Idaho National Engineering Laboratory Configuration Control Number H010171B.

The INEL RELAP4/MOD6 prediction adequately represented the thermal-hydraulic behavior measured during the PREX Test. The maximum deviation between predicted and measured pressure in the primary coolant system during the blowdown was less than 0.5 MPa. The fluid densities were generally calculated accurately throughout the primary coolant system. The deviation between predicted and measured fluid density in the lower plenum and the vessel-side of the broken cold leg was generally less than the measurement uncertainty. This agreement in calculated and measured densities indicated that the code adequately represented the important phenomena of phase separation in the lower plenum and downcomer. The thermal response of the core heater rods was also calculated accurately, as illustrated in Figure 20 which shows predicted and measured cladding temperatures at the peak power zone. The predicted cladding temperature had the correct trends, including the occurrence of critical heat flux 1 s after the initiation of blowdown and rewet 3 s later. The predicted cladding peak temperature was within 5 K of the measured cladding peak temperature and was 20 K higher than the corresponding average measured cladding temperature.

The accuracy of the "blind" INEL prediction of the PREX Test provided a positive indication of the adequacy of the user guidelines developed during the assessment of RELAP4/MOD6.

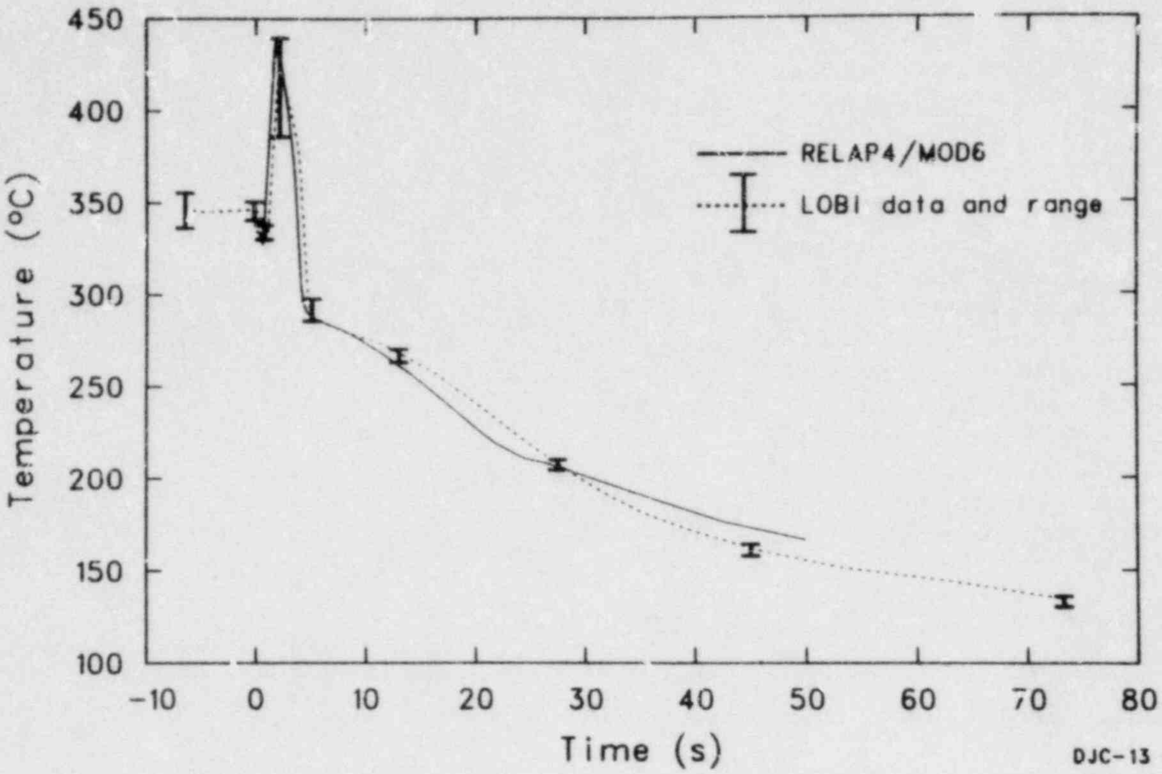

Figure 20. Heate^r rod temperatures at the peak power zone.

VI. 2D/3D PROGRAM OFFICE

R. E. Rice, Manager

The 2D/3D Program includes the 2D/3D Instrument Projects and Advanced Instrumentation Projects. The objectives of the 2D/3D Program are the experimental investigation of the refill and reflood phases of a postulated loss-of-coolant accident, and development and assessment of computer codes suitable for describing such behavior. EG&G Idaho is providing flow instrumentation for German and Japanese experiments, as well as design, test, and analysis support to the NRC. A series of tests has been completed in a vessel simulating the Japanese Slab Core Test Facility. The results, which characterize nonequilibrium, two-phase flow in simulated reactor vessels during reflood, are described in the following section. Advanced Instrumentation efforts support all EG&G Idaho experimental programs through the development of specialized measurement devices.

1. JET DISINTEGRATION TESTS

J. C. Lin and P. D. Wheatley

Five tests to qualitatively study jet disintegration and condensation have been completed this quarter. The jet disintegration tests were conducted using the Simulated Slab Core (SSC), which was installed in the two-phase loop at the LOFT Test Support Facility. The SSC and hot leg spool piece modeled the upper plenum and hot leg portion of the Slab Core Test Facility operated by the Japan Atomic Energy Research Institute (JAERI).

The objectives of the jet disintegration tests were to (a) determine the coherence of an emergency core coolant (ECC) jet in an upper plenum, (b) evaluate water buildup in the vessel and entrainment into the hot leg, (c) evaluate the severity of condensation shocks as cold water mixes with hot steam, and (d) extend the current data bases for further countercurrent flow limiting (CCFL) model development.

Figure 21 shows the SSC and a view of the simulated upper core support plate (UCSP). Steam, at about 410 K, was injected into the lower portions of the SSC while liquid, approximately 130 K subcooled, was injected above the UCSP, just below the hot leg spool piece. The liquid flow rate was approximately 24 kg/s in all of the tests. The steam flow rates for each test were:

Test	Steam Flow (kg/s)
3D-LD-1	0
3D-LD-2A	3.03
3D-LD-3	4.46
3D-LD-3.5	5.91
3D-LD-4A	7.89

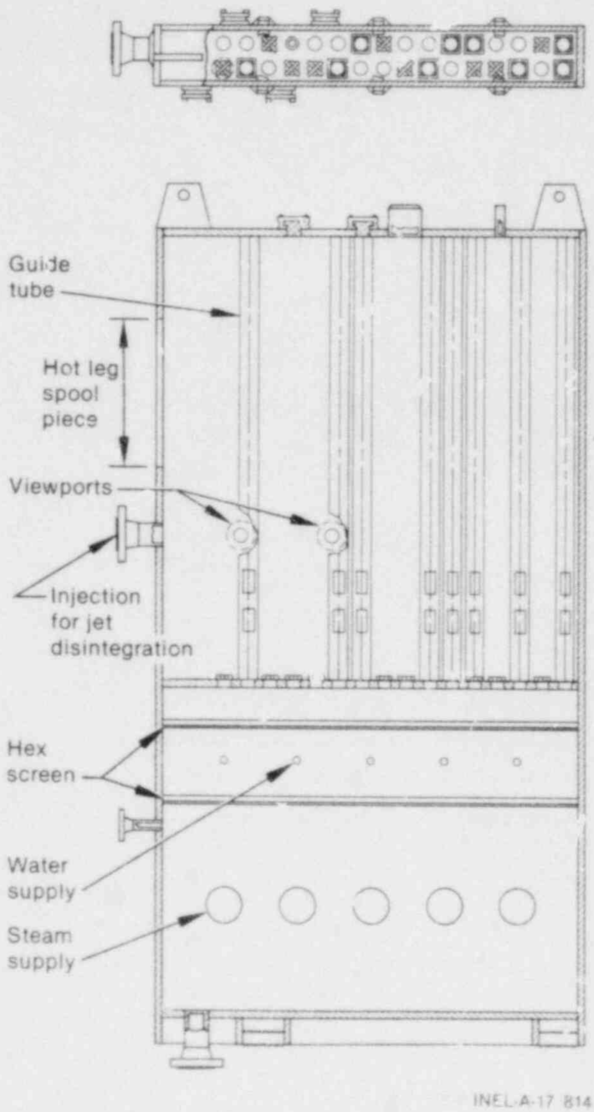

Figure 21 Simulated slab core.

Thermocouples were located at selected openings in the UCSP and on a screen below it. Thermocouple responses were evaluated to determine if steam or subcooled liquid was present. Figure 22 shows the response of thermocouples SC-TE-1 and SC-TE-5 for a typical test. SC-TE-1 was located on top of the UCSP and showed more liquid subcooling than SC-TE-5, located below the UCSP. Both thermocouples showed intermittent subcooling, or wetting, indicating CCFL breakdown, which was observed in the tests with intermediate steam flow. As the injected steam flow rate was increased, the duration of CCFL breakdown was decreased. The highest steam flow test indicated very little CCFL breakdown, and nearly all of the ECC fluid was carried to the hot leg.

Figure 23 shows the mass flows measured in the hot leg spool piece for each of the five tests. A large flow into the SCC was measured in the hot leg spool piece, for the low steam flow test (Run 3D-LD-2A), due to steam-water mixing and steam condensation above the UCSP. During the intermediate flow tests (Runs 3D-LD-3 and 3D-LD-3.5), oscillatory flow was measured in the hot leg spool piece. This was caused by unstable countercurrent flow limiting at the UCSP in the slab core. For the high flow test (Run 3D-LD-4A), a large flow out of the SCC was measured in the hot leg spool piece. The test data indicated that significant ECC fluid can be removed through the hot leg for both intermediate and high steam flows.

Posttest calculations were conducted to evaluate the capability of the TRAC-BD1[50] code to calculate the jet disintegration tests.[a] The calculated fluid temperatures showed trends similar to the experimental data. Figure 24 shows the thermal response of thermocouple SC-TE-5, located below the UCSP, compared with the TRAC-BD1 calculated fluid temperature for Test 3D-LD-3. The calculation and data were in good agreement during the first 15 s. The data returned to near saturation temperature at 15 s, whereas the calculation remained subcooled until about 25 s. Timing of the calculated liquid breakthrough did not match the data in all locations, but the magnitude of liquid subcooling was calculated reasonably well. TRAC-BD1 calculated trends

a. The TRAC-BD1 computer code and the calculational model are stored under Idaho National Engineering Laboratory Configuration Control Numbers F00118 and F00117, respectively.

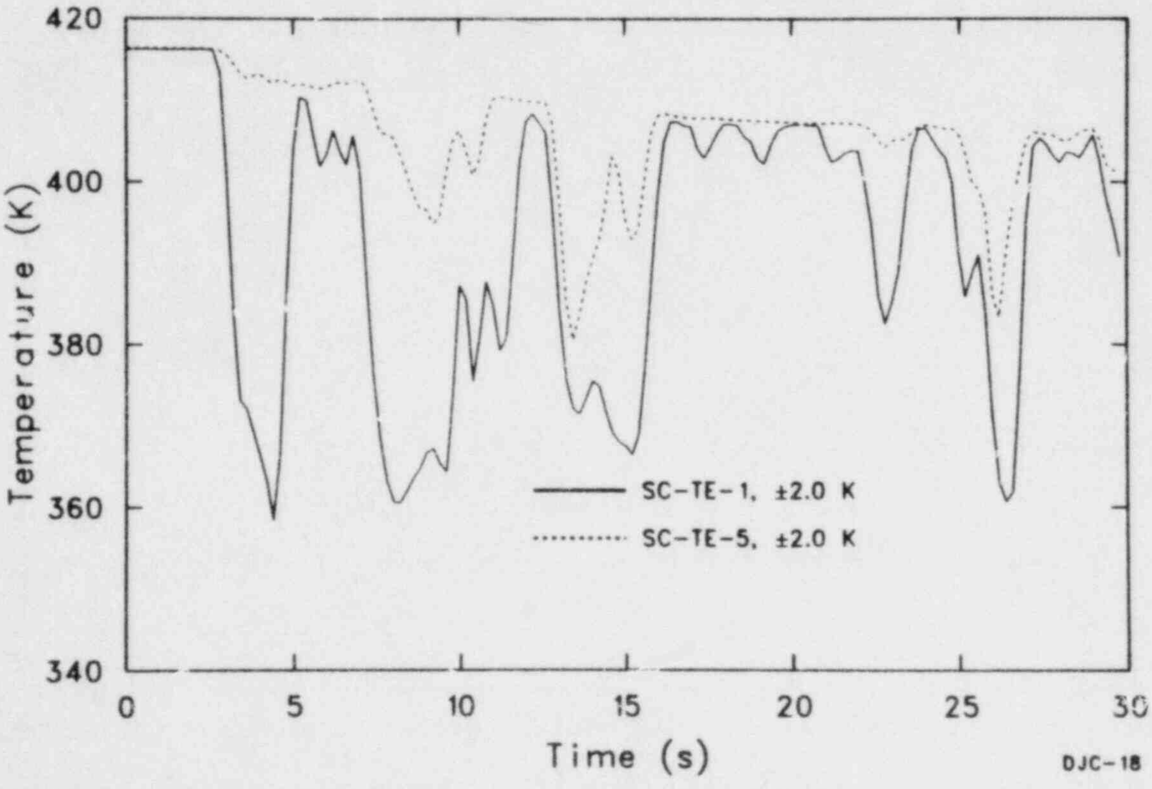

Figure 22. Comparison of fluid temperatures above and below the upper core support plate.

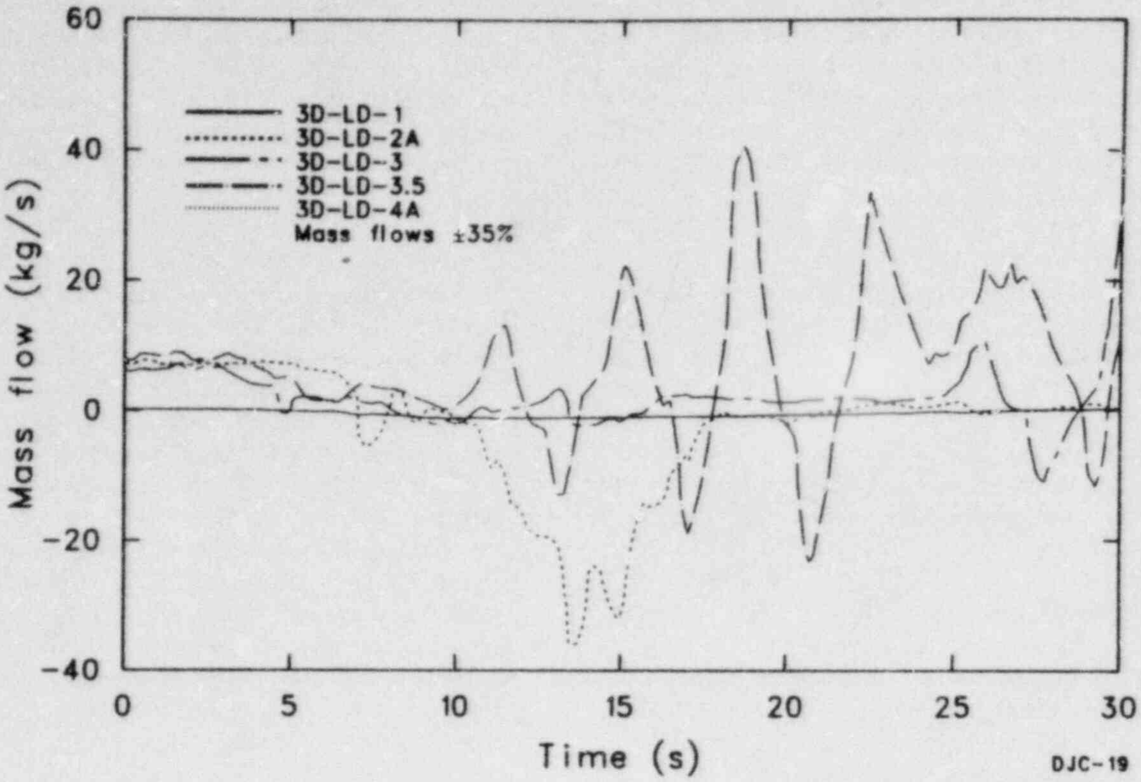

Figure 23. Mass flow in the hot leg spool piece.

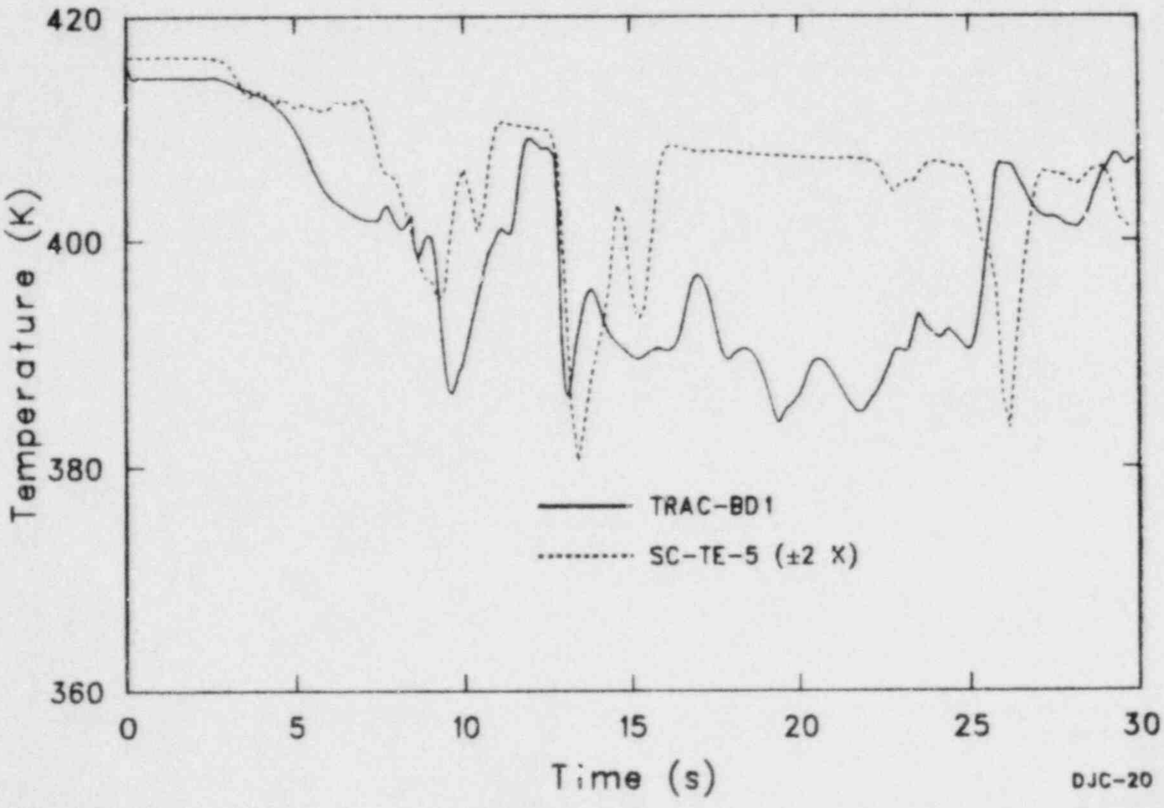

Figure 24. Temperature SC-TE-5 comparison with TRAC - calculated fluid temperature (3D-LD-3).

similar to the experimental data for Tests 3D-LD-3 and 3D-LD-4, which had steam injection into the SSC.

Several countercurrent flow limiting correlations were reviewed, and the modified Wallis correlation[51] was selected, due to geometric considerations, for comparison with the data. The modified Wallis correlation selected is

$$k_{g,e}^{1/2} + 0.689 k_1 = 2.02$$

where

$$k_{g,e} = k_g \quad \lambda \frac{C_p \cdot \Delta T}{h_{fg}} \left(\frac{\rho_L}{\rho_g}\right)^{1/2} k_{1,in}$$

was suggested by Tien[52] to include the condensation effect.

The data obtained from jet disintegration tests and CCFL correlations are shown in Figure 25. For the low steam flow test (3 kg/s), the data were below the CCFL correlation. This means that the water penetration was not yet limited by CCFL. For intermediate steam flow tests (4.4 and 6.0 kg/s), the data agreed with the CCFL correlation. This implies that the water penetration was limited by CCFL. For the high steam flow test (8 kg/s), the data were above the CCFL correlation, implying there was no water penetration.

In addition to the data, the CCFL correlation used in the TRAC code is shown in Figure 25. The CCFL correlation used in the TRAC code is lower than the Wallis CCFL correlation because the TRAC correlation was developed from BWR upper tie plate CCFL data. However, the well-mixed model used in TRAC results in a lower effective Kutateladze number, $K_{g,e}$, than that obtained from the test.

In summary, the selected jet disintegration test matrix spanned a wide range of phenomena at the upper core support plate. The low steam flow test allowed liquid penetration and backflow through the hot leg spool piece. The intermediate steam flow tests showed countercurrent flow limiting at the UCSP, with intermittent breakdown. The high steam flow test indicated CCFL, with large amounts of emergency core coolant liquid being

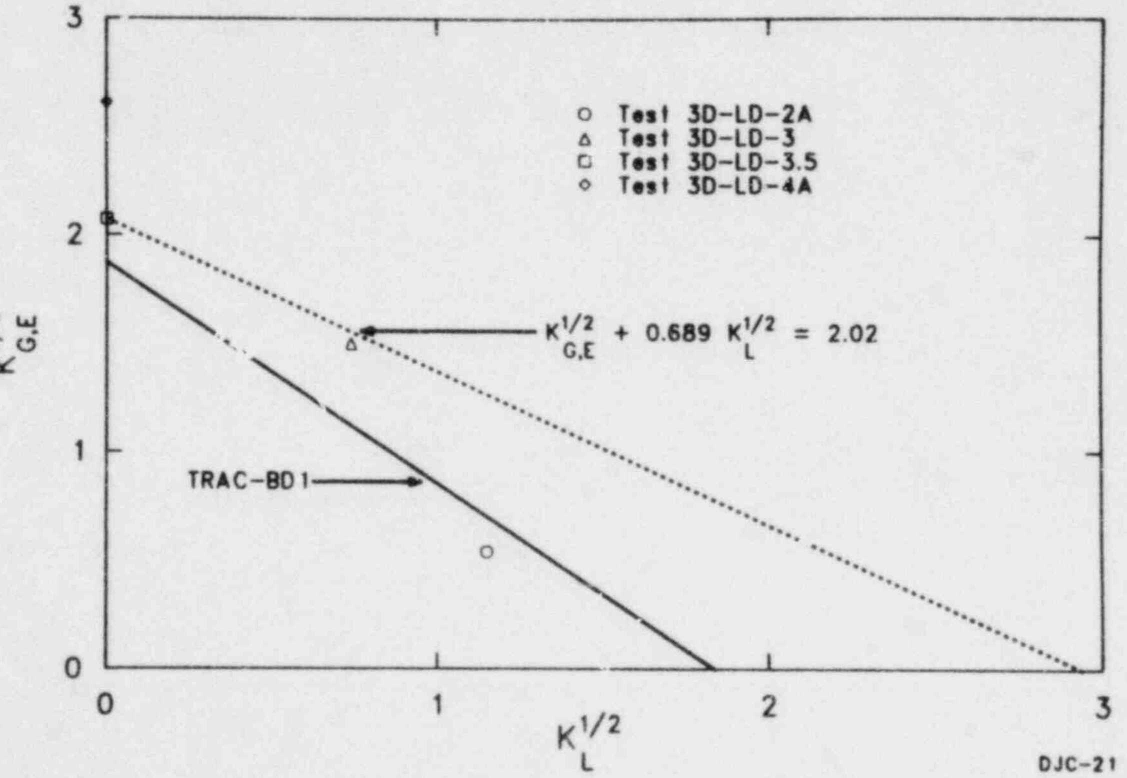

Figure 25. Comparison of CCFL correlations with jet disintegration data.

swept out of the hot leg. Liquid carryover to the hot leg during the intermediate and high steam flow tests would reduce the available mass in the upper plenum and could contribute to steam binding in the steam generators. The jet disintegration test data showed good agreement with the selected CCFL correlation, thus adding to the data base. The TRAC-BD1 computer code also proved capable of calculating the thermal-hydraulic responses of the simulated slab core for tests with steam injection rates similar to those expected in a BWR upper plenum.

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