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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 JULY-SEPTEMBER 1980

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Water Reactor Research

Published November 1980

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ABSTRACT

Water reactor research performed by EG&G Idaho, Inc., during July through September 1980 is reported. The Semiscale Program focused activities on facility conversion to system configuration Mod-2A and evaluated the performance of RELAP5/MOD1 in predicting thermal-hydraulic phenomena for future Semiscale experiments. The Loss-of-Fluid Test (LOFT) Experimental Program completed (a) an analysis of Loss-of-Coolant Experiment (LOCE) L3-7 conducted in the LOFT facility, (b) calculations of reference gamma flux profile data in the LOFT reactor vessel for future instrument development, and (c) preparation of a LOFT system baseline input deck for the RELAP5 code. The Thermal Fuels Behavior Program completed (a) the nine-rod bundle Test PCM-7; (b) the experiment predictions and operating specification for the Test OPTRAN 1-1; (c) analysis of the results of the RIA-ST-4 experiment, the fission product monitoring during Test PCM-1 and the RIA-ST-4 experiment, and the measurements of the release of radioactive gases from UO_2 during irradiation in the Halden Reactor; and (d) nondestructive examination of the Test RIA 1-4 nine-rod cluster assembly. The Code Development and Analysis Program completed a rigorous developmental assessment of FRAPCON-2 and continued development of the TRAC-BD1 code. The Code Assessment and Applications Program updated the U.S. Nuclear Regulatory Commission/Reactor Safety Research Data Bank and performed studies to assess the TRAC-P1A computer program. The 2D/3D Program delivered instrumentation to the Japanese Slab Core Test Facility and developed a new gamma densitometer, a new tomographic densitometer, and a thermocouple liquid level probe.

SUMMARY

Activities in the the Semiscale Program for this quarter focused on the conversion of the facility to a new system configuration, Mod-2A. The modified system will include a new, full-length intact loop steam generator, primary coolant system pipe heat tracing, a replacement of the electrical core simulator, and improved instrumentation. It is anticipated that the Mod-2A system will become operational in November 1980. A RELAP5/MOD1 analysis was performed for a previously conducted test (S-SB-P1) to evaluate the ability of the computer code to calculate important phenomena occurring in Semiscale during a small, cold leg break test with early pump trip. The study indicates that RELAP5/MOD1 accurately calculated the important thermal-hydraulic phenomena and, therefore, should prove to be a useful analytical tool for the prediction of future Semiscale experiments.

The LOFT Experimental Program completed (a) an analysis of Loss-of-Coolant Experiment (LOCE) L3-7 conducted in the LOFT facility, (b) calculations of reference gamma flux profile data in the LOFT reactor vessel for future instrumented development, and (c) preparation of a LOFT system baseline input deck for the RELAP5 code. LOFT Experiment L3-7 was conducted to investigate (a) primary system response to a small break when the break flow is of the same order as high pressure injection system flow, (b) steam generator heat removal modes, and (c) plant recovery procedures. Significant findings from the experiment were that natural circulation began early in the transient and persisted throughout the transient, and that plant recovery can be completed with primary system feed-and-bleed and the purification system. The gamma flux calculations originated from observation in previous experiments that radiation detectors in the LOFT core are sensitive to water density variations and from an industry need to measure liquid level in the reactor vessel. The gamma flux data provide a reference for conceptual design of radiation detectors for liquid level measurement. The data show that such measurements are possible. The RELAP5 LOFT baseline input deck was developed to provide a reference for LOFT calculations wherein all such RELAP5 calculations would have a known starting point, with proven accuracy in model and LOFT system correspondence. Large break and small break transient calculations with this input deck as a starting point are underway. Initial results have verified this concept for RELAP5 calculations.

The Thermal Fuels Behavior Program completed (a) the nine-rod bundle Test PCM-7, which was performed to evaluate the potential for film boiling and fuel rod failure propagation under severe power-cooling-mismatch conditions; (b) the experiment predictions and experiment operating specification for Test OPTRAN 1-1, which will simulate BWR-6 reload fuel behavior during anticipated transients representative of a turbine trip without steam bypass in a BWR-4 reactor; (c) analysis of the results of the RIA-ST-4 experiment with respect to molten fuel-coolant interaction to explain the large pressure pulse (35 MPa) that was recorded about 2 ms after rod failure; (d) a comprehensive analysis of the results of fission product monitoring following fuel rod failures during Test PCM-1 and the RIA-ST-4 experiment; (e) nondestructive examination of the Test RIA 1-4 nine-rod cluster assembly; and (f) analysis of measurements of the release of radioactive xenon, krypton, and iodine from UO_2 during irradiation in the Halden reactor. In addition, models of the Halden Instrumented Fuel Assembly-511 (IFA-511) test rig were developed for both the BWR-TRAC and RELAP5 computer programs.

The Code Development and Analysis Program completed a very rigorous developmental assessment of FRAPCON-2, a code that calculates steady state fuel behavior after long-term burnup operation. The assessment results confirm the quality of FRAPCON-2, and show the superior capability of FRAPCON-2 over its predecessor to model both fuel rod thermal and mechanical response. Development of the advanced boiling water reactor thermal-hydraulics code, TRAC-BD1, continues to make good progress, as the originally defined scope of model development was completed ahead of schedule.

The Code Assessment and Applications Program updated the data base in the U.S. Nuclear Regulatory Commission/Reactor Safety Research Data Bank. The effects of steam generator tube ruptures on system thermal-hydraulic behavior during hot leg and cold leg breaks were examined

in studies using the TRAC-P1A computer program. As part of the overall assessment of TRAC-P1A, code calculations of the Semiscale Test S-04-6 were compared with measurements.

The 2D/3D Program is composed of the 2D/3D Instrument Projects and advanced instrumentation development. The 2D/3D Instrument Projects delivered instrumented spool pieces, drag transducers, fluid distribution grids, and turbine meter probes for use in the Japanese Slab Core Test Facility. Development has been completed on a new densitometer system. Major achievements in the advanced instrumentation area include completion of a 14-inch tomographic densitometer and development of low flow velocimeters. Also, a heated thermocouple liquid level probe has been developed.

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, the Thermal Fuels Behavior Program, the Code Development and Analysis Program, the Code Assessment and Applications Program, and the 2D/3D Program.

The Semiscale Program consists of 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 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.

The Thermal Fuels Behavior Program 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 and Analysis Program 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 Program 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 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 below. Copies of the quarterly reports 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)
ANCR-NUREG-1315 (January-March 1976)
TREE-NUREG-1004 (April-June 1976)
TREE-NUREG-1017 (July-September 1976)
TREE-NUREG-1070 (October-December 1976)
TREE-NUREG-1128 (January-March 1977)
TREE-NUREG-1147 (April-June 1977)
TREE-NUREG-1188 (July-September 1977)
TREE-NUREG-1205 (October-December 1977)
TREE-NUREG-1218 (January-March 1978)
TREE-1219 (April-June 1978)
TREE-1294 (July-September 1978)
TREE-1298 (October-December 1978)
TREE-1299 (January-March 1979)
TREE-1300 (April-June 1979)
EGG-2003 (July-September 1979)
EGG-2012 (October-December 1979)
EGG-2031 (January-March 1980)
EGG-2048 (April-June 1980)

CONTENTS

FIGURES

TABLE

QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH JULY-SEPTEMBER 1980

I. SEMISCALE PROGRAM

P. North, Manager

The Semiscale Program performs and analyzes results of small-scale, nonnuclear, thermal-hydraulic experiments for the purpose of generating experimental data that can be used to develop and assess analytical models describing operational transient and loss-of-coolant accident (LOCA) phenomena in water-cooled nuclear power plants. Program emphasis is on acquiring system effects data from integral tests that characterize the thermal-hydraulic phenomena likely to occur in the primary coolant system of a pressurized water reactor (PWR).

1. PROGRAM STATUS

Program emphasis was directed at performing analyses to support the Nuclear Regulatory Commission (NRC) in assessment and improvement of computer models for small break LOCAs. Tests S-SB-P1, S-SB-P2, and S-SB-P7 were conducted earlier this year to evaluate the effect of pump operation on overall Semiscale Mod-3 system thermal-hydraulic response during a small break LOCA. Results from analyses of these tests were reported in an earlier quarterly report.[1] A RELAP5[a] posttest calculation was performed for one of these tests (Test S-SB-P1) and a comparison of the calculated results with the test data is reported in Section 2.

During this quarter, conversion to a new Semiscale system configuration (the Mod-2A system) was in progress. The system modifications, which were designed to allow closer simulation of a pressurized water reactor (PWR), include a new, full-length steam generator, primary coolant system pipe heat tracing, replacement of the electrical core simulator, and improved instrumentation. It is anticipated that the Mod-2A system will be operational in early November 1980.

2. COMPARISON OF RELAP5/MOD1 CALCULATION WITH SEMISCALE TEST S-SB-P1

B. W. Murri

RELAP5 posttest calculations for Tests S-SB-P1 and S-SB-P7 are being performed to assess the code's ability to calculate the influence of primary coolant pump operation on overall Semiscale system behavior. Tests S-SB-P1 and S-SB-P7 simulated 2.5% communicative cold leg breaks in a PWR system, with initial conditions that approximate those typical of a full-sized commercial PWR operating at full-load conditions. The pumps were tripped at a system pressure of 12.4 MPa during Test S-SB-P1 and at 3.28 MPa during Test S-SB-P7. A calculation for Test S-SB-P1 has been completed. Results from this calculation are compared with test data to

a. Experimental version, RELAP5/MOD1, Cycle 144.

illustrate the ability of RELAP5 to calculate important phenomena occurring in Semiscale during an early pump trip test. A full assessment of the calculated pump operation effects will be made after the calculation for Test S-SB-P7 has been completed.

The RELAP5 model for the Semiscale Mod-3 small break calculations includes 132 hydrodynamic volumes and 96 heat slabs, with a total of 817 mesh points. It includes a single-channel core with an external downcomer and an upperhead bypass line. The two separate Mod-3 loops are modeled; one representing the three intact loops of a PWR, and the other representing the single broken loop for cold leg loss-of-coolant accident simulations. Each loop has its own steam generator, with a steam separator model at the secondary side between the upper downcomer and the steam dome. The emergency core coolant (ECC) injection systems include a high pressure injection system (HPIS) and a polytropic accumulator model, which allows air to be injected into the system as it empties (for Test S-SB-P1, the accumulators were valved off). Heat losses to the environment are modeled in the hot and cold legs, pump suctions, downcomer, lower plenum, core barrel, and upper plenum. Thermal conductivities of the piping insulation and heat transfer coefficients from the pipe walls to the atmosphere were adjusted to obtain an estimated value of the Semiscale Mod-3 system heat loss ([illegible]0 kW at steady state, full power).[2]

A comparison of the upper plenum pressure response for Test S-SB-P1 with the posttest calculation is shown in Figure 1. The calculation accurately matches the test data through the subcooled decompression phase. As the hot leg fluid reaches saturation (at about 35 s), the calculation diverges from the test data. This is due to a slightly higher hot leg fluid temperature (and associated saturation pressure) in the calculation from the beginning of the transient. The pressure profiles converge at about 700 s, because RELAP5 calculates a higher two-phase flow rate out of the break between 300 and 750 s (Figure 2). After 1000 s, the differences in the calculation and test data can be primarily attributed to inaccuracies in the modeling of system heat losses.

The calculated break flow for Test S-SB-P1 is shown in Figure 2. A reliable break flow measurement was not available for this test; however, the

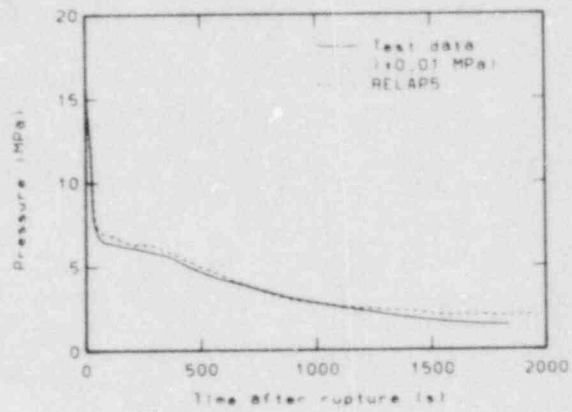

Figure 1. Comparison of calculated and measured primary system pressure for Test S-SB-P1.

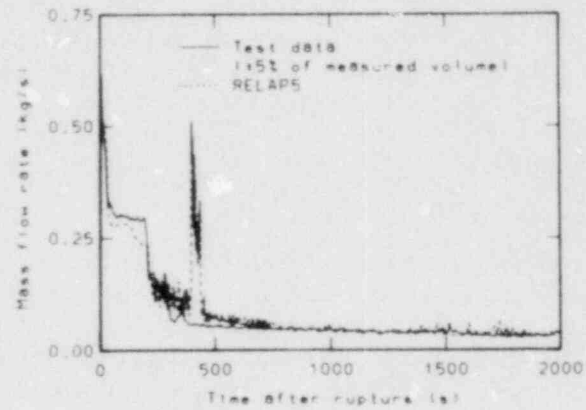

Figure 2. Comparison of calculated break flow rate for Test S-SB-P1 with measured break flow for Test S-SB-2.

intitial conditions and operating procedures for Test S-SB-P1 were, as nearly as possible, identical to those for Test S-SB-2[3] until the accumulators were actuated at 660 s in the latter test. For the time period of concern, the break flow rate measured in Test S-SB-2 should be nearly identical to that which would have been measured in Test S-SB-P1, and is, therefore, overlayed with the calculated break flow rate in Figure 2. RELAP5 underpredicts the subcooled break flow, accurately calculates the time of transition to two-phase (approximately 200 s), and then slightly overpredicts the two-phase break flow, with the exception of a large surge at about 400 s. Following the clearing of liquid from the intact loop pump trap (360 s in the test), a large slug of fluid drained from the broken loop steam generator and passed through the hot leg into the vessel, which

RELAP5 accurately predicted (Figure 3). A RELAP5 code error caused the break flow to become unchoked during this period of time and allowed a surge of fluid from the vessel to accelerate out the break. This problem has been corrected and the code should provide more accurate results for future calculations.

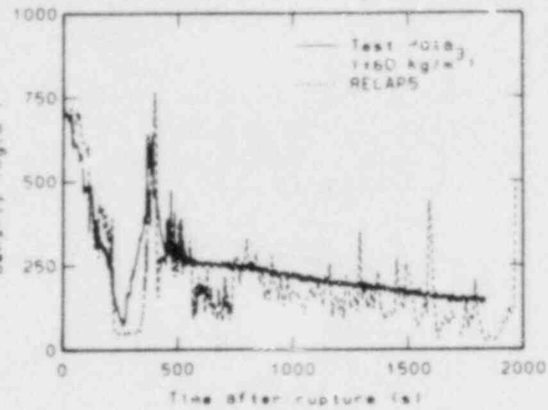

Figure 3. Comparison of calculated and measured broken loop hot leg density for Test S-SB-P1.

Figure 4 presents a comparison of the core collapsed liquid level for Test S-SB-P1 with the posttest calculation. RELAP5 conservatively calculates an earlier and more extensive uncovering of the core than was measured, but in general, follows major trends of the data. The collapsed liquid level dropped below the top of the core at about 140 s in the test and continued to drop until about 1050 s, at which time the HPIS flow rate exceeded the break flow rate. In the calculation, the liquid level drops below the top of the core at about 50 s, partially recovers following the pump seal clearing, and continues to drop until recovery begins at about 950 s. The calculated liquid level begins to recover at 360 s as fluid drains back from the broken loop steam generator. The core cladding peak temperatures resulting from the measured and calculated periods of core uncovering (Figure 5) occurred at a height of two-thirds the distance to the top of the core. Both curves follow the saturation temperature until about 750 s, when dryout occurs at an elevation of

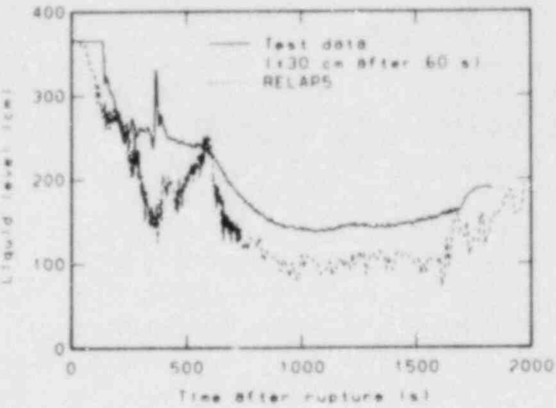

Figure 4. Comparison of calculated and measured core collapsed liquid level for Test S-SB-P1.

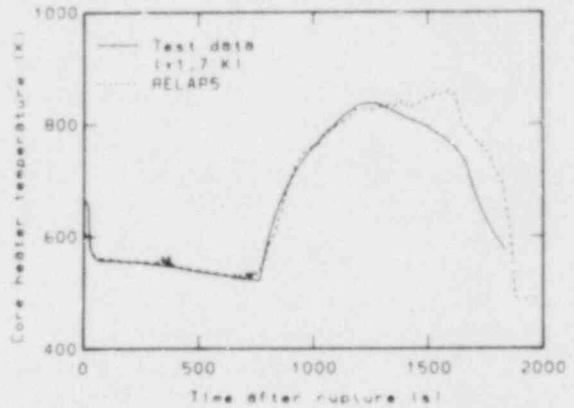

Figure 5. Comparison of calculated and measured core cladding peak temperature for Test S-SB-P1.

250 cm. The temperature excursion, which reached a maximum of 840 K, was calculated quite accurately, with RELAP5 overpredicting the cladding peak temperature by only 50 K.

From the present analysis, it was concluded that RELAP5 is capable of calculating important phenomena occurring in Semiscale during an early pump trip, small, cold leg break test. This provides insight into the capability of the code to predict similar phenomena in a full-sized commercial PWR.

II. LOFT PROGRAM DIVISION

C. W. Solbrig, Manager

The LOFT Experimental Program successfully completed Experiment L3-7, the fourth experiment in the small break Test Series L3, in the LOFT Facility on June 20, 1980. Experiment L3-7 simulated a single-ended offset shear break of a small (1-in. diameter) pipe connected to the cold leg of a four-loop, large pressurized water reactor. The data obtained from Experiment L3-7 will contribute to knowledge of small break loss-of-coolant accident phenomena, including primary loop natural circulation, steam generator heat transfer, and plant recovery following break isolation.

Calculations of the axial gamma flux profile in the LOFT reactor vessel were completed to provide reference data for the conceptual design of a liquid level measurement using a gamma-flux-sensitive device as the sensing element. The axial profile calculations extend from the bottom of the core to 2.0 m above the top of the core and include consideration of empty versus full vessel and time after reactor scram. The calculations indicate that such a sensing element is feasible.

A LOFT system baseline input deck for thermal-hydraulic calculations with the RELAP5 computer code has been developed that is applicable to large and small break and anticipated transient analysis. The input deck is designed to accept modifications required for each specific transient type. Calculations of large and small break LOFT transients, currently in progress, have demonstrated the validity of the baseline input deck concept.

1. LOFT NUCLEAR LOSS-OF-COOLANT EXPERIMENT L3-7

G. E. McCreery

Loss-of-Coolant Experiment (LOCE) L3-7 simulated a single-ended offset shear break of a small (1-in.-diameter) pipe connected to the cold leg of a four-loop, large pressurized water reactor.

The primary objectives of LOCE L3-7 were to establish a break flow approximately equal to high pressure injection system (HPIS) flow when the primary pressure was in the range of 6.9 MPa, to establish conditions for steam generator reflux cooling, to isolate the break and stabilize the plant at cold shutdown conditions, and to analyze the data obtained to investigate associated phenomena. The initial test conditions and sequence of events were consistent with the objectives.

Prior to the break, the nuclear core was operating at a steady state maximum linear heat generation rate of 52.8 + 3.7 kW/m. Other significant initial conditions for LOCE L3-7 were: system pressure, 14.95 ± 0.34 MPa; core outlet temperature, 576.1 ± 0.5 K; and intact loop flow rate, 478.8 ± 8.8 kg/s. At 36 s after the break occurred, the reactor scrammed on a low system pressure signal. Within 10 s after scram verification, the pumps were manually tripped and coasted down. Pump coastdown was followed by the inception of natural loop circulation. Between 1800 and 5974 s, the HPIS was turned off to hasten the loss of fluid inventory and to establish the conditions considered favorable for reflux flow in the primary loop. Starting at 3600 s, operator-controlled steam bleeding (by opening the main steam bypass valve early and the main steam valve later in the transient) and steam feeding (using both the auxiliary and main feedwater systems) were used to decrease primary system pressure.

Later in the experiment, 7302 s, the quick-opening blowdown valve was closed, which isolated the break. System mass depletion stopped, and all decay heat energy not lost to the environment was removed by the steam generator. Primary system pressure gradually increased, causing the fluid in the system to become subcooled. Subsequently, the purification system was used to bring the reactor to a cold shutdown condition, and the experiment terminated.

Conditions were established in LOCE L3-7 for natural circulation in the primary loop. Measurable natural circulation continued from 61 s into the transient until after the purification system started at 18 180 s. Single-phase natural circulation was fully established within about 35 s, starting at 61 s, and continued until about 375 s, at which time the upper plenum temperature reached saturation and the transition to two-phase natural circulation began. Two-phase natural circulation was fully established by about 1500 s, when the core temperature differential approached zero (Figure 6).

Single-phase natural circulation was reestablished after the break was isolated at 7200 s, as indicated by core and steam generator differential temperature (Figure 6). By that time, the fluid in the system had become subcooled. Fluid velocities were lower than during two-phase natural circulation. Core velocity, calculated from decay power and temperature differential, was approximately 0.05 m/s from 7800 to 15 000 s.

The primary flow regimes during two-phase natural circulation in the steam generator were calculated, through use of flow regime maps, to be churn flow changing to slug flow. During two-phase natural circulation, the dominant heat transfer modes were calculated to be convection and condensation of steam within the fluid. Convection contributed the higher heat transfer rate because the tube walls remained covered with liquid. During single-phase natural circulation, convection heat transfer dominated. The combination of natural loop circulation and steam generator heat transfer was sufficient to remove decay heat throughout the experiment. Measured steam generator inlet-to-outlet and primary-to-secondary temperature differences confirm that the primary-to-secondary heat transfer rates are high. However, instrumentation is not available in the steam generator to measure densities or localized temperature distributions that would help confirm calculated flow regimes and localized heat transfer phenomena.

An objective of LOCE L3-7 was to establish conditions for reflux flow. The conditions thought to be necessary for reflux flow were established; however, the flow mode did not occur, although condensation heat transfer did occur in the steam generator, and the intact loop partially voided during the period of condensation. Reflux is characterized by countercurrent flow of condensed liquid in the steam generator to the hot leg. Both the lower intact loop hot leg turbine meter and the pulsed-neutron-activation velocity measurement indicated positive flow rather than countercurrent flow. Thus, if a condensate film did fall down the steam generator tubes to the inlet plenum, it mixed with the two-phase fluid, which had a dominant, positive velocity through the hot leg and steam

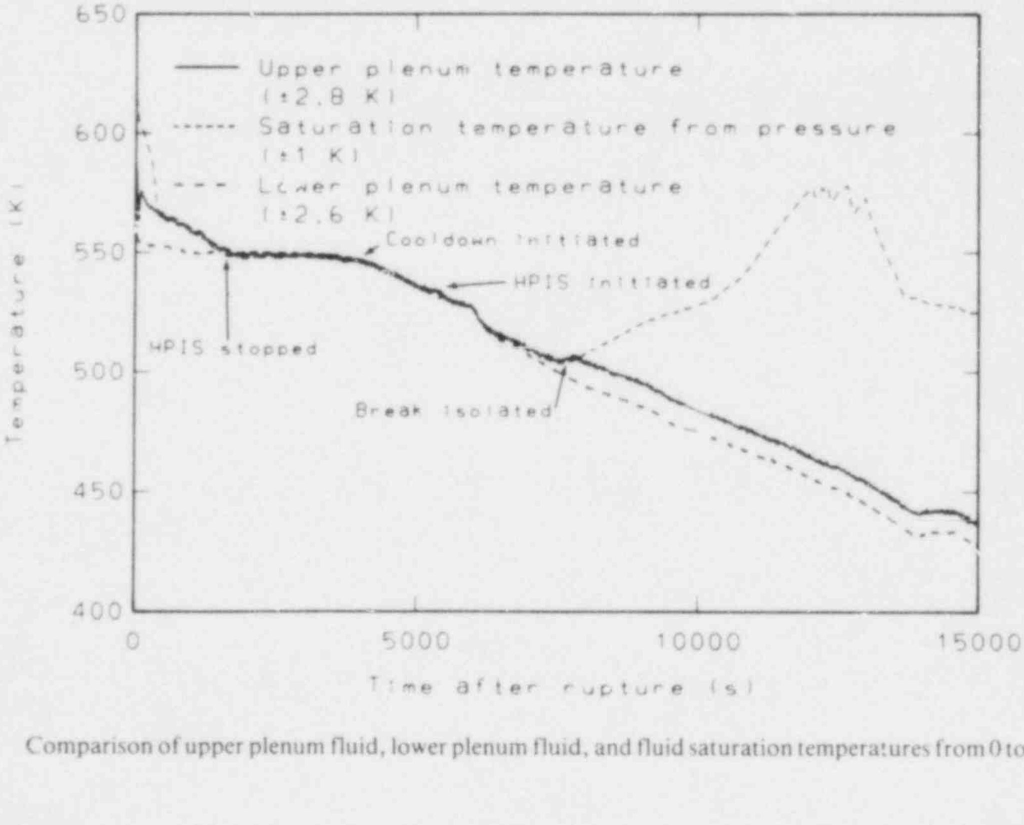

Figure 6. Comparison of upper plenum fluid, lower plenum fluid, and fluid saturation temperatures from 0 to 15 000 s.

generator. A possible reason that the reflux flow mode did not occur is that there was insufficient voiding in the intact loop hot leg.

When the break was isolated at 7305 s, the system gradually started to repressurize (Figure 7). The gradual increase in system pressure was induced by the increase in system fluid inventory driven by the HPIS. Superheated steam in both the pressurizer and vessel upper head was compressed and gradually condensed. This nonequilibrium process caused the pressurizer to start refilling at 8200 s; before the vessel head was refilled. At 12 000 s, operator-initiated primary system bleeding through the power-operated relief valve was required to control primary system pressure.

Preexperiment computer calculations of the LOCE L3-7 transient were made by EG&G Idaho, Inc., using the RELAP4[a] and RELAP5[b] codes, and by Los Alamos Scientific Laboratory using the TRAC-PIA code. The RELAP4 and TRAC calculations terminated at 1800 and 3600 s, respectively. The RELAP5 calculation terminated at 11 000 s, when the system was predicted to have refilled. All three codes adequately predicted early (0 to 1200 s) system depressurization, but overpredicted the time of pressurizer emptying (Figure 7). The TRAC code predicted the system would slightly repressurize between 1800 and 3600 s, but the data showed no increase during the period. The system pressure trends were predicted by RELAP5, except for predicting higher than measured system pressure and not predicting the gradual system pressure rise after the break was isolated. The prediction of higher system pressure between 1800 and 3600 s was probably due to weepage in the secondary system main steam stop valve that occurred during the experiment, but was not modeled in the pretest calculation.

Postexperimental analysis of LOCE L3-7 is continuing. Conclusions based on the preliminary analyses and experiment assessment include:

1. The reactor vessel water level did not drop below the outlet nozzles; thus, the core remained covered during the entire transient. No fuel rod damage resulted.

2. Natural circulation was initiated within 61 s after the break occurred, and continued until after the purification system was started at 18 180 s.

3. The steam generator was an effective heat sink throughout the experiment. Both convection and condensation heat transfer occurred in the steam generator.

4. The reflux flow mode was not measured, although condensation occurred in the steam generator and the intact loop hot leg partially voided.

5. Effective system decay heat removal through the steam generator continued after the quick-opening blowdown valve was closed, although the primary system pressure increased.

6. Computer calculations predicted the dominant phenomena, in the proper time sequence, except for (a) system depressurization during the period from 1200 to 3600 s due to expected steam control valve weepage and (b) gradual system repressurization after the break was isolated, due to superheated steam and possibly some noncondensible gases in the system.

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a. The experimental RELAP4 code used was RELAP4/MODG, Version 92, (experimental version of RELAP4/MOD7), Idaho National Engineering Laboratory Configuration Control Number H00718B. The new object deck, which includes changes to correct known coding errors and to incorporate the LOFT steam valve control logic into the code, was RLP4G92LFT04, Idaho National Engineering Laboratory Configuration Control Number H011681B.

b. The version of the code used was RELAP5/MOD"0". The source deck and update input data deck are stored under Idaho National Engineering Laboratory Configuration Control Numbers H005785B and H005985B, respectively.

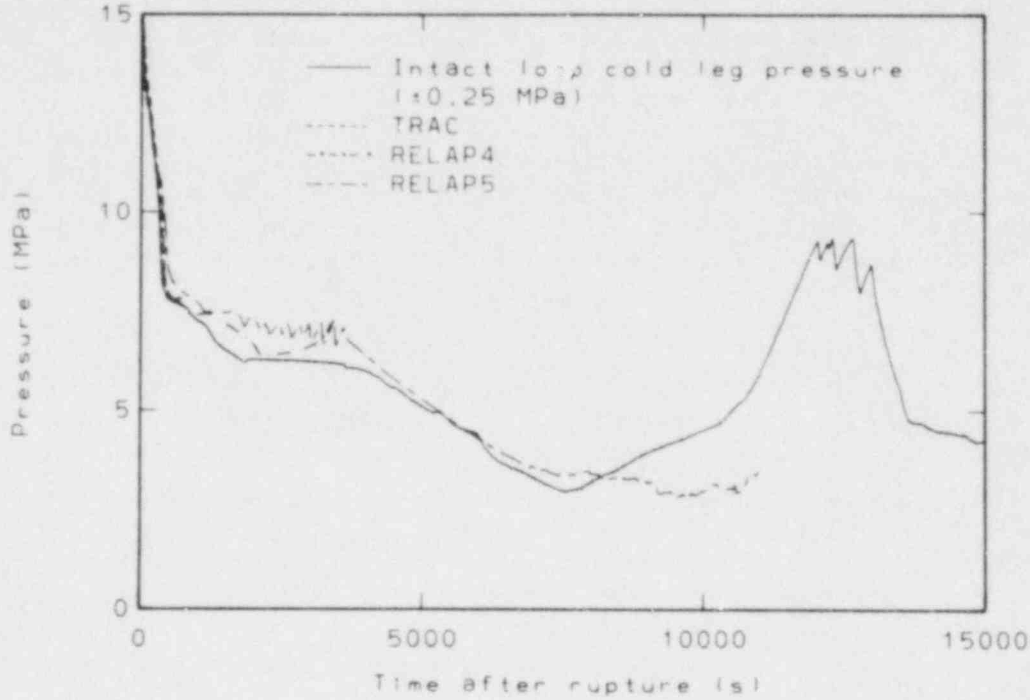

Figure 7. Comparison of measured system pressure with predictions from 0 to 15 000 s.

2. REACTOR VESSEL LIQUID LEVEL MEASUREMENT

V. T. Berta

The liquid level in the reactor vessel of a pressurized water reactor (PWR) is not measured in current plant process instrument systems. In certain types of off-normal conditions, such as a small break in the pressurizer, lack of this measurement constitutes a severe deficiency in monitoring plant status and in optimizing plant recovery. The LOFT facility is an ideal experimental facility in which to develop such measurements as the reactor vessel liquid level, which can be incorporated in PWR process instrument systems and operator information displays.

Liquid level information has been obtained in the LOFT experiments thus far conducted using the conductivity probe[4] as the sensing element. This type of sensor, however, is not sufficiently durable to last for a core lifetime. In large break experiments L2-2 and L2-3, another instrument, the self-powered cobalt-emitter neutron detector, showed a strong correlation to changes in water density.[5] Instruments of this type have a proven ability to withstand the reactor vessel environment for the lifetime of a core load. The neutron detectors, sensitive to both neutron and gamma flux, produce an output that is dependent in the same manner on the variation in water density.

As a first step in the conceptual design of a liquid level measurement using the self-powered radiation detector as the sensing element, the reactor vessel radiation levels for several hours after a reactor scram must be known. Only the gamma radiation will persist in sufficient magnitude for this length of time. Thus, effort was concentrated on gamma flux calculations in the LOFT reactor vessel out to 10 000 s after reactor scram.

Initial calculations of the gamma flux have been completed. The calculations estimate the gamma dose rate profile along the axial centerline of the LOFT reactor with an uncertainty of 10%. The profile calculations extend from the bottom of the core to 2.0 m above the top of the core. The results of the calculations show that in the core region, the fission product decay gammas dominate the dose rate profile; however, above the core, the fission product dose rate is rapidly attenuated and the activation product gammas dominate. Also of interest is the difference in dose rate between a water-filled vessel and an empty vessel. The difference increases from 20% at the core midplane to 66% 2 m above the core. The results indicate that a self-powered detector using a platinum emitter may be possible as a sensing element for a liquid level indication. Other types of gamma detectors may also be applicable.

These gamma flux results are currently being used for the conceptual design of a gamma detector for liquid level sensing in the LOFT reactor vessel. The results of the initial calculations are sufficiently positive to continue development of such an instrument.

3. DEVELOPMENT OF A RELAP5 LOFT BASELINE INPUT DECK

E. J. Kee and P. J. Schally

Thermal-hydraulic calculations are used extensively in the LOFT Program in the areas of experiment planning, safety analysis and prediction, and data analysis. A large part of these calculations are performed with large systems computer codes such as RELAP5. In order to provide a common starting point for RELAP5 analyses, development of a LOFT baseline input deck was undertaken for the RELAP5 code, which could be used as the base for analysis of large break, small break, and anticipated transients. The large phenomenological differences among the large and small break and anticipated transients necessarily require analysis models specifically defined for each transient type. The base input deck is designed to be applicable to all of the transients, with only minor modifications being required for each specific transient type. The development of the input deck has been completed and is briefly described in this section.

Figure 8 shows the nodalization scheme that is used in the input data. The input data set is based on the LOFT system information available as of July 1980.[6] The entire LOFT system consists of 113 volumes, 122 junctions, and 43 heat slabs (33 in the reactor vessel, 6 between primary and secondary sides of the steam generator, and 7 in the secondary side of the steam generator).

The nodalization in Figure 8 reflects the small break requirement of calculating liquid levels accurately. The nodalization is much finer than that required for large break transients; however, in order to avoid large computing times, excessively detailed nodalization was avoided. Also, the volume size was chosen such that all volumes have approximately the same flow length. When branches existed, multiple connections were used to facilitate the inclusion of parallel flow paths as a modification for specific transient analyses. Momentum flux cancellation in tees was achieved by having the junction connection to both the inlet and outlet of the main flow path. With this method, all junctions to the branch begin and end at the same elevation.

The input data set has been entered on the INEL computer system. Changes to the data set reflecting new LOFT information are made with the computer system update procedures only, which guarantees maintenance of a frozen version of the input data set. Calculations of large and small break transients using this data set have demonstrated the validity of the data set as a baseline reference for LOFT thermal-hydraulic calculations.

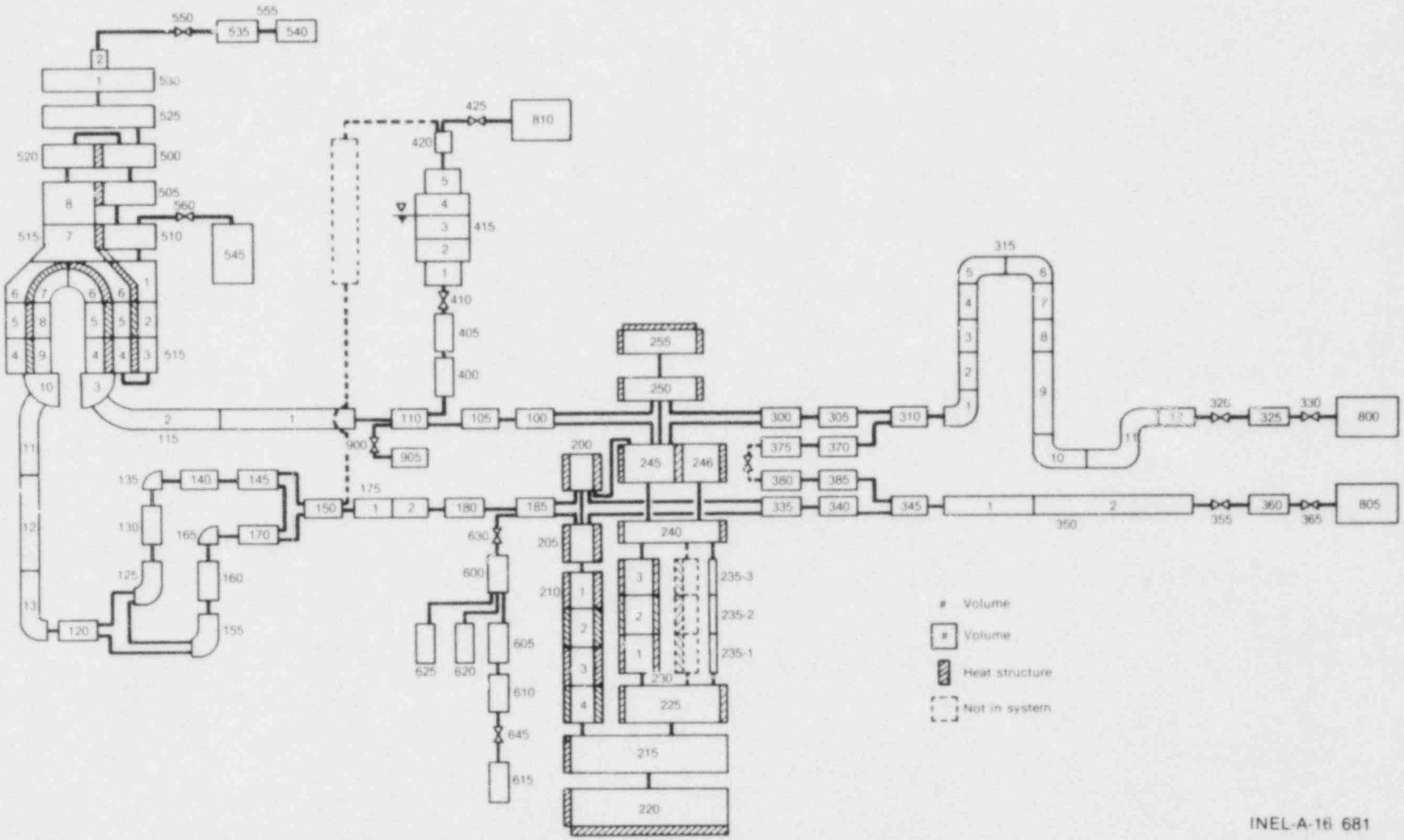

Figure 8. Nodalization scheme of the LOFT system.

III. THERMAL FUELS BEHAVIOR PROGRAM

H. J. Zeile, Manager

The Thermal Fuels Behavior Program (TFBP) 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 the Thermal Fuels Behavior Program 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 safety issues addressed by the power-cooling-mismatch accident tests are: (a) what is the margin between the occurrence of departure from nucleate boiling and fuel rod failure; (b) can a coolable geometry be maintained following power-cooling-mismatch accidents; (c) do film boiling and fuel failure propagate; and (d) will energetic molten fuel-coolant interactions occur during a severe power-cooling-mismatch accident?

The reactivity initiated accident tests, which were completed last quarter, have addressed the following key safety issues: (a) will there be a loss-of-coolable core geometry when light water reactor fuel is subjected to a radial average peak fuel enthalpy near the Nuclear Regulatory Commission licensing limit of 1172 J/g; (b) will energetic molten fuel-coolant interactions (vapor explosions) occur during a severe reactivity initiated accident and result in the production of a significant pressure pulse; and (c) what is the mechanism and threshold enthalpy for failure of light water reactor fuel during a reactivity initiated accident?

The information obtained from the loss-of-coolant accident tests will be pertinent in answering two key safety issues: (a) will ballooning during a severe, double-ended cold leg break loss-of-coolant accident lead to coplanar blockage and subsequent loss-of-coolable geometry; and (b) is out-of-pile ballooning data representative of nuclear fuel rod behavior? Two tests remain in this test series. When these two tests are completed, the original Power Burst Facility test program will be accomplished.

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 help to 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.

Two nonprogrammatic test series are being performed to investigate the effects of cladding surface thermocouples on fuel rod behavior during blowdown and quench. The first test, TC-1, has been completed.

The Thermal Fuels Behavior Program 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.

1. PBF TESTING

P. E. MacDonald and R. K. McCardell

Test PCM-7, a nine-rod bundle test, was conducted and preliminary results are briefly described in the following section. Analyses of the results from Tests LOC-3 and LOC-5 and Test PR-1 were begun. The Test OPTRAN 1-1 experiment operating specifications and experiment predictions were completed.

The OPTRAN 1-1 test will simulate BWR-6 reload fuel behavior during an anticipated transient representative of a turbine trip without steam bypass in a BWR/4 reactor. The test rods will not experience boiling transition, because the test is directed toward evaluating the probability of stress corrosion cracking assisted pellet-cladding interaction. Five transients with progressively higher energies are planned for OPTRAN 1-1.

Test PR-1 utilized four, separately shrouded, boiling water reactor (BWR) type fuel rods tested simultaneously in the PBF in-pile tube. The test consisted of three parts designed to (a) provide fuel rod thermal response information under steady state and power oscillation conditions; (b) investigate the conditions at the onset of boiling transition and quench, and evaluate the potential for two-phase hydrodynamic instabilities during power-cooling-mismatch experiments; and (c) obtain information regarding fuel temperature distributions and rod failure limits during severe reactivity initiated accident (RIA) type power excursions. A single rod failure was indicated during the three RIA power bursts. One rod survived the entire testing sequence.

Tests LOC-3 and LOC-5 were conducted as part of the Loss-of-Coolant Accident (LOCA) Test Series in the Power Burst Facility. In these tests, a system depressurization typical of that expected during a double-ended break in a pressurized water reactor was maintained. The rod power histories were varied such that the desired cladding temperatures of 1190 K for Test LOC-3 and 1350 K for Test LOC-5 were achieved. These two temperatures represent the points of minimum ductility in the alpha plus beta phase transition and the maximum ductility of beta-phase zircaloy cladding, respectively. Each LOCA test was conducted with four PWR-type fuel rods. The rod internal pressure and cladding state were varied between different test rods. The rod internal pressures were typical of PWR rod internal pressures at beginning-of-life and end-of-life. The rod cladding state was either fresh or previously irradiated. The test design will allow for an evaluation of the influence of internal rod pressure and prior irradiation on cladding deformation. The LOC-3 and LOC-5 posttest examination revealed that all seven test rods that experienced the thermal-hydraulic conditions typical of a double-ended cold-leg break deformed and failed. The diametral burst strains ranged from 26 to 53%, and the majority of bursts were in the high power region.

1.1 Test PCM-7 Results

D. T. Sparks

An imbalance in the heat generation rate of a light water reactor core and the heat removal rate of the coolant has long been recognized as a potential fuel damage mechanism if the rods depart from nucleate boiling. Extensive cladding failure could result in unacceptable levels of fission products released to the primary coolant system. The pressurized water loop located in the Power Burst Facility enables power-cooling-mismatch tests to be performed under simulated light water reactor conditions. Test PCM-7, the final test in the Power-Cooling-Mismatch Test Series, was performed this quarter. The test was conducted with nine pressurized-water-reactor-type test rods arranged in a cluster geometry. The objectives of the experiment were to evaluate the potential for film boiling and fuel rod failure propagation in a small cluster under severe power-cooling-mismatch conditions. Brief descriptions of the test design and performance, and an overview of the preliminary test results are presented in this section.

1.1.1 Test Design. A nine-rod bundle of unirradiated, pressurized water reactor design (15 x 15) fuel rods was used for Test PCM-7. Fuel enrichments of 20, 35, and 93 wt% ^{235}U were used for the corner, side, and center rods, respectively, to provide a relatively flat rod-to-rod power generation profile. The rods were arranged in a 3 x 3 array and positioned within a common zircaloy flow shroud by a series of five Inconel grid spacers on a pitch of 14.3 mm. The test rods were instrumented with cladding surface thermocouples and/or linear variable differential transformers (LVDTs) to detect high temperature film boiling operation. Three of the fuel rods also contained fuel centerline thermocouples and rod internal pressure transducers. The fuel bundle, flow shroud, and instrumentation were positioned within the PBF in-pile tube by a support structure referred to as the test assembly. Instrumentation on the test assembly provided coolant environmental conditions (temperature, pressure, and volumetric flow rate) and local neutron flux.

1.1.2 Test Conduct. Performance of Test PCM-7 consisted of a self-powered neutron detector (SPND) calibration, a power calibration and conditioning period, two scoping flow reductions, and the post-departure-from-nucleate-boiling test phase. The scoping flow reductions were conducted at constant test rod power (53 kW/m peak) and the flow was reduced to determine the onset of boiling transition. An inlet temperature of 602 K and coolant pressure of 15.4 MPa were maintained during the scoping flow reductions.

The coolant flow rate, temperature, and pressure were held constant, and the test rod peak power was increased from 30 to 55 kW/m to initiate the transient. The shroud coolant flow rate during the power increase was 1.1 L/s, and the coolant inlet temperature and pressure were the same as during the scoping flow reductions. The fuel rod cluster was allowed to remain in high temperature film boiling operation for about 25 min, after which the coolant flow was increased to rewet the rods.

1.1.3 Film Boiling Results. An illustration of the film boiling scenario within the cluster is shown in Figure 9. The figure represents an interpretation of test rod instrumentation and shows the onset of boiling transition, quench, and approximate film boiling axial extent on each rod. Also shown are the relative power and shroud coolant flow rate during the post-boiling-transition testing phase. The order of film boiling occurrence in the cluster is evident from Figure 9. The first rod to attain high temperature film boiling was the central rod of the cluster (Rod 207-5) during the power increase. Subsequent indications of film boiling were observed on two side rods (207-2 and 207-6), a corner rod (207-9), a side rod (207-4), a corner rod (207-1), and finally, at a later time, a corner rod (207-3). This order implies that the onset of boiling transition was unrelated to rod position in the cluster, and was not a systematic propagation through the assembly.

Although film boiling propagation in the cluster was not detected, the possibility of rod-to-rod interactions via hydraulic coupling cannot a priori be ruled out. As seen in Figure 9, the quench and rewet of Rod 207-6 just prior to the onset of boiling transition on Rod 207-3 at power and coolant conditions more conducive of sustaining film boiling than quench, may be an indicator of the inherent coupling between the bundle rods. Other factors such as rod bowing, surface condition, or gap conductance may also be postulated to result in the observed response, and will require additional investigation.

Approximately 1104 s after the first indication of film boiling, rod failure was detected from an increase in loop gamma activity. The long time at high temperatures prior to rod failure suggests that the failure was due to either high temperature nil-ductility or thermal shock induced embrittlement during rewet. Additional breakup of previously failed rods or of other highly embrittled rods was likely during the rewet sequence (flow increase).

1.1.4 Comparison of Bundle Test With Single-Rod Tests. Of fundamental interest in the conduct of bundle tests is the comparison between the conditions at the onset of boiling transition and quench of cluster rods and singly shrouded (thermally isolated) test rods. Figure 10 illustrates the comparison of conditions at the onset of boiling transition for two power-cooling-mismatch (PCM) cluster tests (PCM-5 and PCM-7) and several

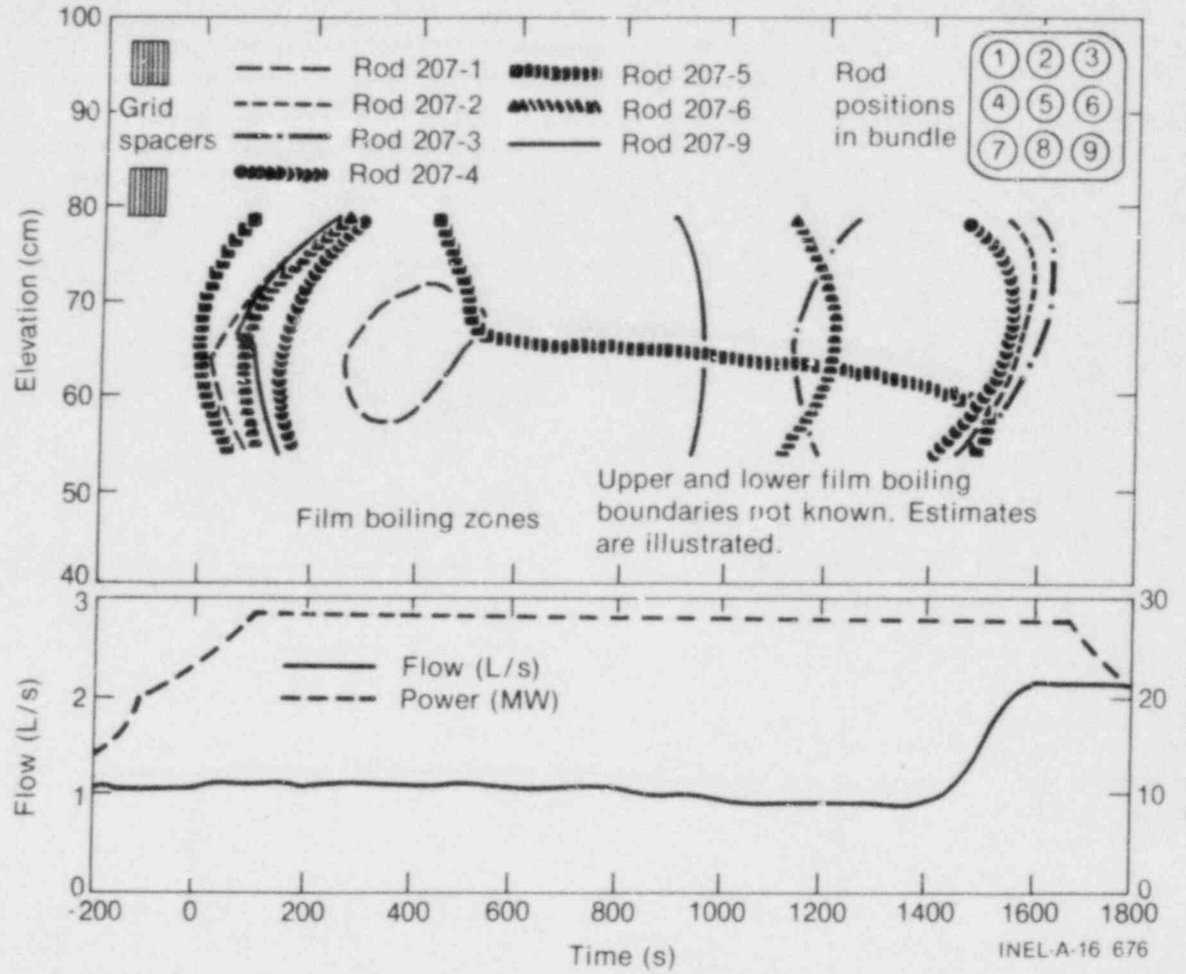

Figure 9. Test PCM-7 fuel rod film boiling histories showing estimated duration and axial extent fro on-line measurement data.

singly shrouded PCM test rods. As indicated, the central rod conditions at the onset of boiling transition for the bundle tests are fairly consistent with the trend of single-rod test results. The peripheral rod data (side and corner bundle rods) are slightly outside the overall trends. In general, a higher test rod power at a given mass flux is required for the peripheral rods to initiate boiling transition. Such behavior may indicate a slightly more effective cooling of the side and corner rods, perhaps due to coolant intermixing and cross flow, or rod bowing and the near presence of the cooler shroud wall.

The conditions at the onset of boiling transition for the central bundle fuel rods correspond well with those expected for an individually shrouded test rod. The boiling transition data base obtained for individually shrouded rods is, therefore, apparently applicable for determining the conditions at the onset of boiling transition for a fuel rod located within the inner regions of a fuel bundle.

Figure 11 illustrates the PCM Test Series data for both the onset of boiling transition and quench. The abscissa of the figure represents a best-fit regression of the PBF boiling transition and quench data.[7] The estimated 97% confidence level of the correlation is shown as the dashed lines of Figure 11, and is approximately 20%. As illustrated, the Test PCM-7 conditions at the onset of boiling transition and quench agree well with the derived PBF empirical correlation. Of notable importance in Figure 11 is the indistinguishable difference between the conditions at the onset of film boiling (open symbols) and the conditions at the onset of film boiling destabilization or quench (solid symbols). This observation suggests that the onset of boiling transition and the return to nucleate boiling processes proceed on similar paths, with little or no hysteresis. This implies that the classical transition boiling heat transfer regime is difficult to attain under the Test PCM-7 conditions and is traversed very rapidly in the quenching process.

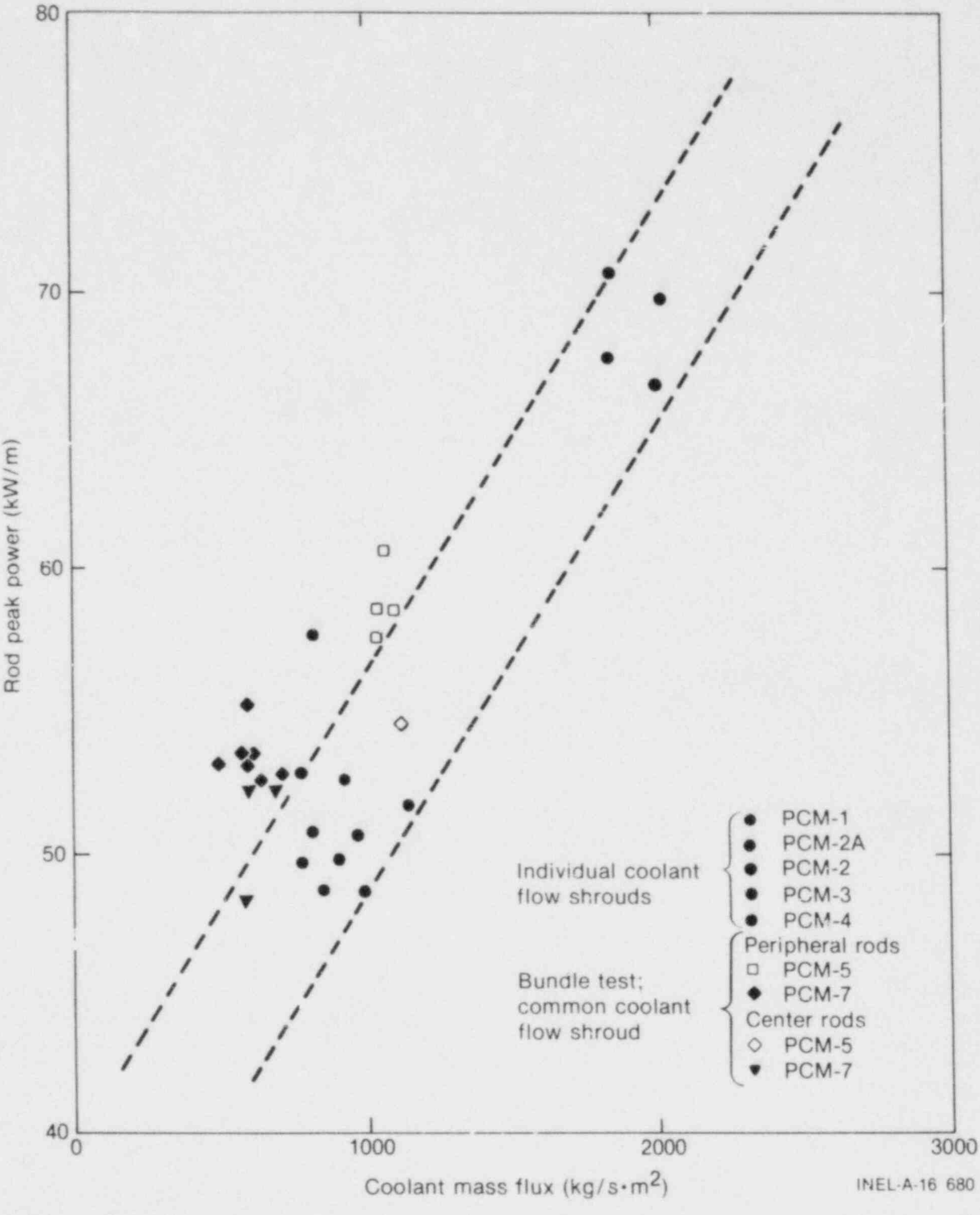

Figure 10. Comparison of the conditions at first indication of film boiling for PCM test series.

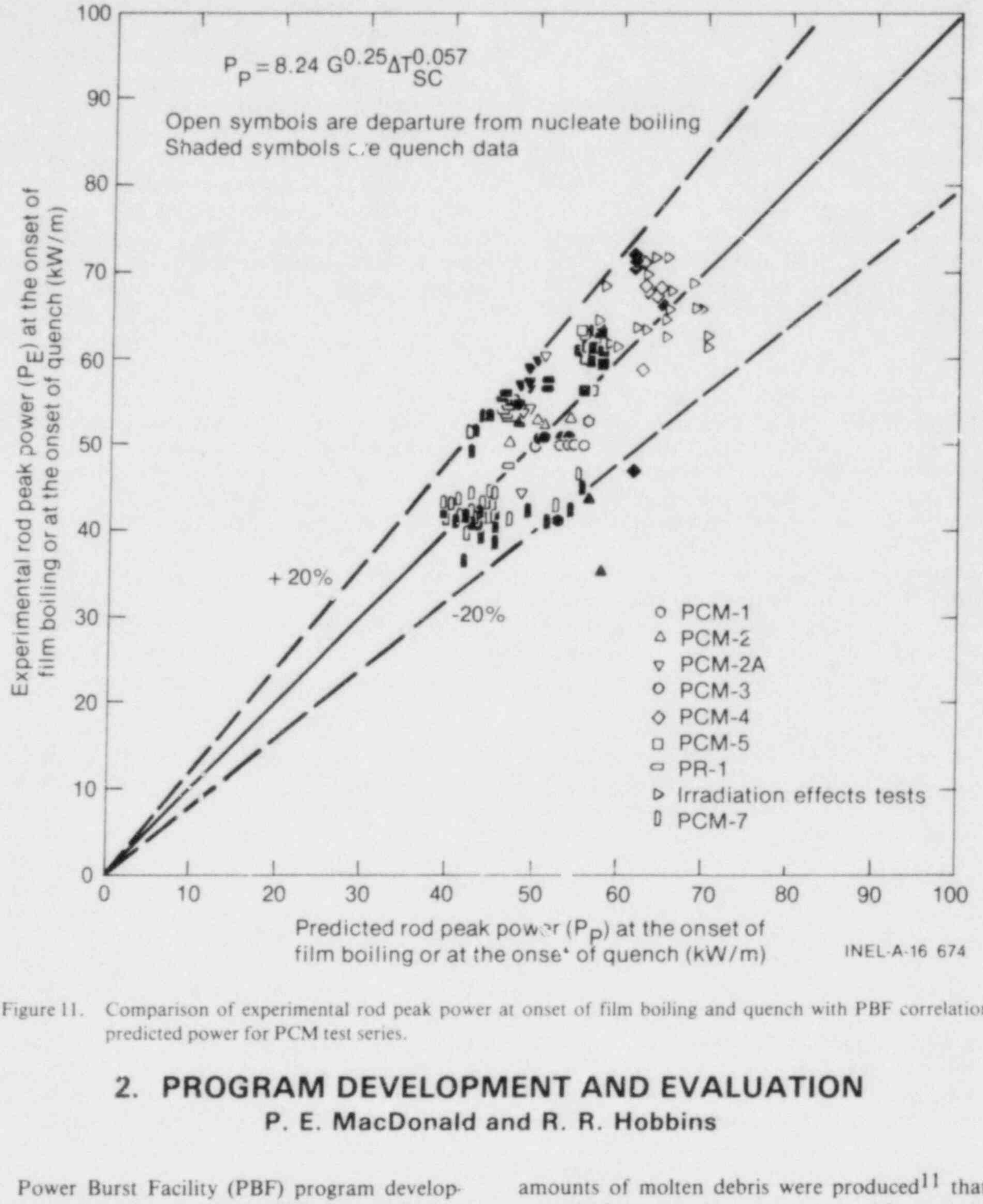

Figure 11. Comparison of experimental rod peak power at onset of film boiling and quench with PBF correlation predicted power for PCM test series.

2. PROGRAM DEVELOPMENT AND EVALUATION

P. E. MacDonald and R. R. Hobbins

Power Burst Facility (PBF) program development and data analysis, the PBF fission product measurement system, the postirradiation examination of PBF tested fuel rods, and Halden fuel behavior research are reported in this section.

The Reactivity Initiated Accident-Scoping Test-4 (RIA-ST-4) experiment[8,9] has been analyzed relative to molten fuel-coolant interaction.[10] In this experiment, extensive amounts of molten debris were produced[11] that severely fragmented upon contact with water in the test shroud. A coolant peak pressure of about 35 MPa was recorded ~2 ms after test rod failure. It is concluded that such pressurization was due to an efficient molten fuel-coolant interaction, which may be explained by the pressure detonation model[12] for vapor explosion. The shock wave due to the release of gas from the RIA-ST-4 rod failure triggered fine fragmentation and initiated coherent thermal interaction between the molten debris particles and the coolant.

The fragmentation characteristics of the debris particles in the RIA-ST-4 experiment were analyzed.[13] Two fragmentation mechanisms have been identified: (a) rupture of the frozen crust on the surface of the debris particles due to internal pressurization caused by overheating droplets of liquid coolant entrained in the debris; and (b) the perforation of the crust surface by coolant jets developed during the collapse of the vapor film on the surface of the particles.

The effects of core coolant conditions on potential shock pressurization of the core due to a molten fuel-coolant interaction occurring during a postulated core meltdown accident were also studied.[14] In a molten debris-coolant interaction (liquid-liquid system), rapid energy transfer to the coolant may cause a shock pressurization of the core, with a small amount of overheating (that is, mild fragmentation and intermixing of the debris particles with the coolant). Increasing the initial coolant pressure in this case results in a reduction of the peak pressure induced due to a certain amount of coolant overheating. Conversely, if the core coolant is initially a two-phase mixture, a larger amount of energy would be necessary (that is, fine-scale fragmentation and efficient intermixing of debris particles with coolant) to cause a shock pressurization of the core. In such a case, given an initial steam quality in the coolant, increasing the initial coolant pressure increases the coolant peak pressure induced.

A comprehensive report on the results of fission product monitoring during Test PCM-1 and the Reactivity Initiated Accident-Scoping Tests was issued during this quarter.[15] The report is the first detailed description of short-lived fission product behavior during PBF tests and completes an important milestone in the development of gamma spectra data processing. The new processing routines will be used on a regular basis for additional test data reduction and will permit detailed and more rapid reporting of the fission product behavior data collected by the PBF fission product detection system.

The relative isotopic release fraction histories of up to 20 short-lived fission products per test are included in the report. The measured release of the important iodine isotopes was found to be significantly different for tests with and without fuel melting. Iodine release fractions found in the coolant were only about 2 to 4% in two tests that included fuel melting, but on the order of 30% in two tests that included fuel powdering and no fuel melting. These results suggest that the fuel in the Three Mile Island-2 accident (in which iodine release fractions of about 40 to 60% were measured in the coolant) most likely underwent extensive powdering rather than appreciable melting. Transport of iodine in the test loop coolant was found to be dependent on loop thermal-hydraulic conditions, with a substantial reduction (factor of 10) in suspended or dissolved iodine occurring when loop temperature and pressure were reduced.

A comparison of the relative release fractions from the PBF tests to release fractions suggested in NRC regulatory guides, the Reactor Safety Study, and release fractions reported for the Three Mile Island accident indicated that fission product release resulting from fuel powdering can be as great as the release expected from fuel melting.

The analysis, interpretations, and discussions of results from the Reactivity Initiated Accident (RIA) Test Series, Test RIA 1-1, conducted in the Power Burst Facility reactor, have been completed.[16] Four light water reactor (LWR) type test fuel rods, two previously irradiated and two previously unirradiated, were subjected to a single power transient during Test RIA 1-1, resulting in an estimated axial peak, radial average fuel enthalpy of 1193 J/g (1402 and 1319 J/g peak fuel enthalpy near the pellet surface of the previously irradiated and unirradiated test rods, respectively). The total radial average energy deposition for the test was 1528 J/g UO_2. All four test rods failed as a result of the RIA power burst. Test RIA 1-1 produced rapid fuel-cladding heatup and melting, film boiling and oxidation embrittlement of the cladding, and quenching fragmentation of the test rods. Fuel rod breakup, together with fuel melting and severe fuel swelling, produced coolant flow shroud blockages. The flow blockage for irradiated rods was complete, whereas for unirradiated rods it was partial. No fuel was lost from the flow shrouds of the irradiated rods, but up to 27% of the fuel was lost to the coolant loop from the unirradiated rods. High strain rate deformation, induced by pellet-cladding mechanical interaction from rapid thermal expansion of the fuel, produced cladding failure early in the test when cladding temperatures were low.

The nondestructive examination of the Test RIA 1-4 nine-rod bundle was completed and metallography started during this reporting period. The results of the nondestructive examination showed that although all nine rods failed, coolable geometry was maintained. No evidence of gross fuel melting was observed, although one instance of fuel swelling and relocation into the coolant channel was found. Preliminary metallography indicates that cladding failures occurred due to pellet-cladding mechanical interaction brittle fracture. Examination of the cladding showed areas of thickening and thinning, along with areas of partial cladding melting.

An analysis of measurements of the release of radioactive xenon, krypton, and iodine from UO_2 during irradiation in the Halden reactor was completed. A comparison was made of the measured release fractions with the release fractions calculated by the proposed American Nuclear Society (ANS) 5.4 model. The comparison shows that the ANS 5.4 model calculates release fractions that are generally within a factor of 2 to 4 of the measured release fractions. These results support the current proposal of the Nuclear Regulatory Commision to use the ANS 5.4 model to revise the assumptions in Regulatory Guides 1.3, 1.4, 1.77 and 1.25 for fission product release during loss-of-coolant, rod ejection, and fuel handling type accidents.

A model of the Halden Instrumented Fuel Assembly-511 (IFA-511) test rig has been developed for both the BWR-TRAC and RELAP5 computer programs. Initial calculations indicate that slight modifications will be necessary for calculated results to match the IFA-511-2 experimental results.

Calculations, using the MITAS II computer system, were completed to determine the circumferential temperature distribution for the center and an outer rod in the IFA-511 five- and seven-rod geometries. The results of these calculations will be used in the planning of cladding ballooning experiments in the next phase of the IFA-511 program. For both rod bundles, the center rod has a negligible circumferential temperature variation. The temperature difference for an outer rod ranges from 57 to 64 K for a five-rod bundle and from 75 to 82 K for a seven-rod bundle, with a variation within a bundle type due to fuel rod average linear heat rates of from 40 to 20 W/cm.

IV. CODE DEVELOPMENT AND ANALYSIS PROGRAM

F. Aguilar

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

An important achievement of the past quarter is the rigorous developmental assessment of FRAPCON-2, a code that calculates steady state fuel behavior during long-term burnup operation. The assessment results, which are reported in Section 1, confirm the quality of FRAPCON-2 and show the superior capability of FRAPCON-2 over its predecessor at modeling both fuel rod thermal and mechanical response. Development of the advanced boiling water reactor thermal-hydraulics code, TRAC-BD1, continues to make good progress, as the originally defined scope of model development was completed ahead of schedule. The status of TRAC-BD1 development is reported in Section 2.

1. FRAPCON-2 DEVELOPMENTAL ASSESSMENT

G. A. Berna

FRAPCON-2 is a computer code that calculates the steady state behavior of light water reactor fuel rods during long-term burnup operation. Development of FRAPCON-2 is a joint effort of EG&G Idaho, Inc., and Pacific Northwest Laboratory that began with the development of the FRAPCON-1 code. FRAPCON-2, like its predecessor, models basic phenomena including heat conduction through the fuel and cladding, elastic-plastic cladding deformation, fuel-cladding mechanical interaction, fission gas release, fuel rod internal gas pressure, heat transfer between fuel and cladding, cladding oxidation, and heat transfer from cladding to coolant. The code contains all needed rod surface heat transfer coefficient correlations, water properties, and material properties. The code calculates the uncertainties in fuel rod variables due to known uncertainties in fuel rod fabrication variables, material properties, and rod power and cooling. In addition, the code is designed to generate initial conditions for the transient fuel rod analysis codes FRAP-T5[17] and FRAP-T6, (see Reference 2, pp. 19-20.)

The final step in the development process for FRAPCON-2 was to assess the correctness of the model additions and determine any unanticipated effects of the new models on overall code results. This process is known as developmental assessment, and was performed by comparing FRAPCON-2 calculations with a limited set of experimental data and with FRAPCON-1. An independent assessment of FRAPCON-2 is currently in process in the Code Assessment and Applications Program. The independent assessment activity will evaluate the accuracy of the constituent models of FRAPCON-2 by comparing the code calculations with an extensive set of experimental data.

To assess the adequacy of the model additions and their effects on overall code results, comparisons were made between FRAPCON-2 calculations and experimental data for fuel rod temperature, internal gas pressure, and deformation. Table 1 identifies the test rods used in the FRAPCON-2 developmental assessment, and includes fuel rod data for varying fuel density, fuel enrichment, fuel-cladding gap, fill gas pressure, and operating history. Thus, a wide range of fabrication variables is provided for comparison with the FRAPCON-2 code. Table 1 also indicates the different model options used for the comparison of FRAPCON-2 with experimental data. As a result, a matrix of 54 FRAPCON-2 cases was evaluated for the developmental assessment.

Typical results of the fuel centerline calculations are presented in Figure 12, which shows calculated fuel centerline temperature versus rod power level (FRACAS-I rod deformation model) compared with temperature data from Instrumented Fuel Assembly 508 (IFA-508) Rods 11 and 13, respectively. Rods 11 and 13 correspond to small and large gap rods, respectively. The comparisons indicate that FRAPCON-2 gives closer agreement

Table 1. FRAPCON-2 developmental assessment cases

Experiment	Mechanics Option Utilized		
	FRACAS-I[a]	FRACAS-II[a]	PELET[a]
GC 2-2, 522 Rod 3	1	1	1
GC 2-2, 522 Rod 4	1	1	1
GC 2-3, 523 Rod 1	1	1	1
GC 2-1, Rod 503	1	1	1
PCM-4, Rod 15	1	1	1
IFA-430, Rod 3	1	1	1
IFA-431, Rod 3	1	1	1
IFA-432, Rod 1	1	1	1,3,4
IFA-508, Rod 11	1	1	1
IFA-508, Rod 13	1	1	1
H. B. Robinson, Rod K	1,2,3,4,5,6	1	1,3,4
Studsvik	1,2,3,4,5	1,3	
IE-1, Rod 7	—	1	—
IE—2, Rod 11	—	1	—
IE—2, Rod 12	—	1	
IE—5, Rod 20	—	1	—

a. Gas release model utilized:

1 - MacDonald-Weisman
2 - Booth Diffusion
3 - Beyer-Hann
4 - ANS 5.4
5 - FAST/GRASS
6 - GRASS.

with experimental temperature data than FRAPCON-1. This was the general trend observed from the developmental assessment.

The major thrust in developing the FRAPCON-2 code was in improving the fuel rod deformation models. FRAPCON-1 makes use of a rigid pellet deformation model with no fuel pellet mechanical relocation. Comparison of FRAPCON-1 results with experimental deformation data has shown this assumption to be inadequate in modeling the mechanical interaction between the fuel pellet and cladding. FRAPCON-2 makes use of deformable pellet models with pellet mechanical relocation. This provides a basic capability for analyzing pellet-cladding mechanical interaction. The general trend observed from the developmental assessment is that FRAPCON-2 calculations agree more closely with the experimental deformation data than does FRAPCON-1. This was evidenced by an improved calculation of pellet-cladding lockup and maximum cladding elongation resulting from lockup.

To benchmark differences between FRAPCON-1 and FRAPCON-2 calculational results for commercial power reactor fuel rods, two standard power reactor fuel rods were analyzed. The fuel rods chosen correspond to a typical boiling water reactor (BWR) design (7 x 7 rod array) and a typical pressurized water reactor (PWR) design (15 x 15 rod array). The BWR and PWR fuel rods were each analyzed to benchmark FRAPCON-2 for extended fuel burnup conditions. In addition, a PWR fuel rod from the Zion nuclear power plant was analyzed using operating power histories that would be expected in normal plant operation. For the benchmark comparisons,

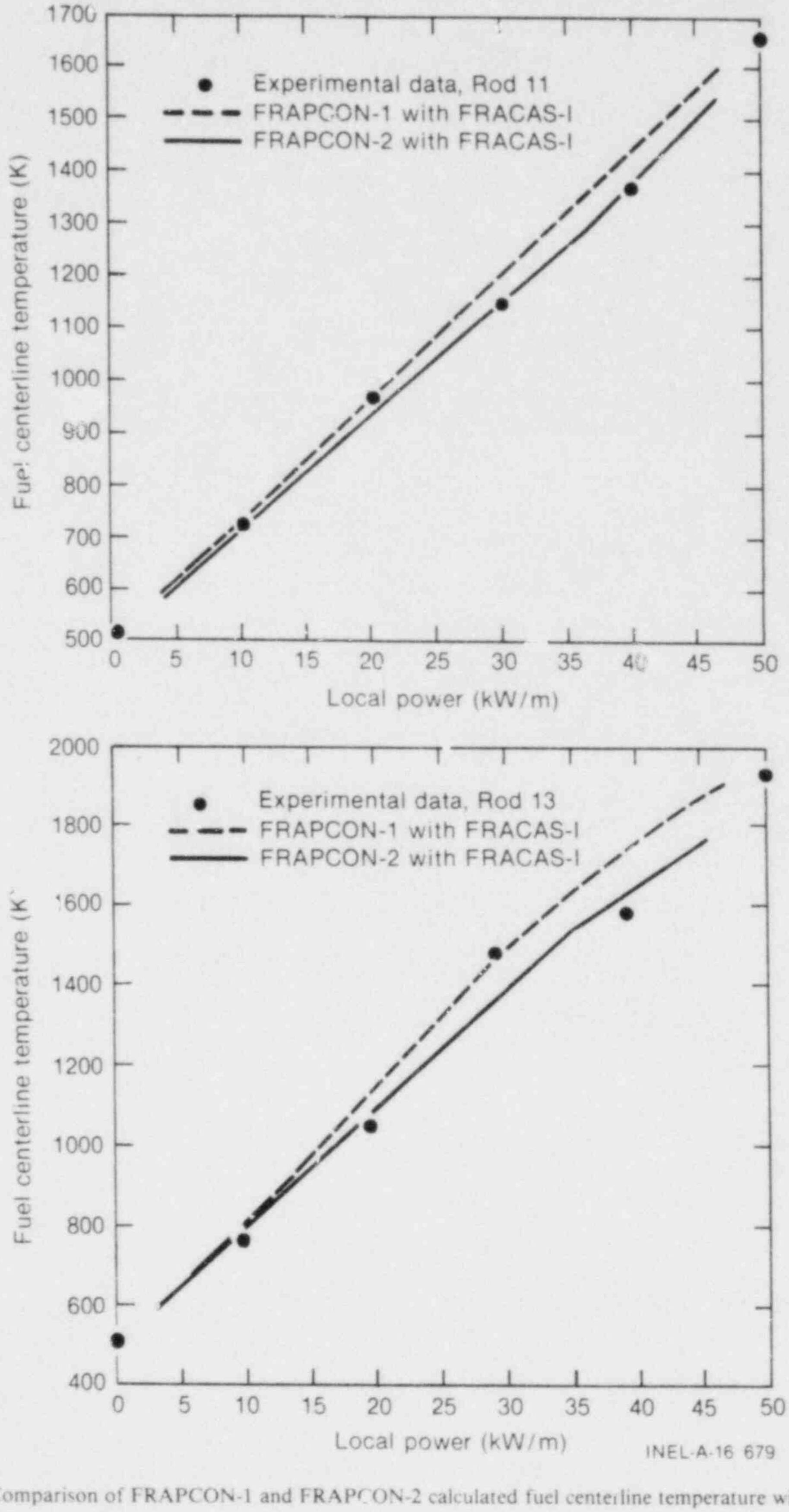

Figure 12. Comparison of FRAPCON-1 and FRAPCON-2 calculated fuel centerline temperature with experimental data.

each of the three FRAPCON-2 rod deformation models was used, but only the MacDonald-Weisman gas release model was exercised. This choice of modeling options resulted in a matrix of nine FRAPCON-2 cases.

In conclusion, the developmental assessment of FRAPCON-2 has shown the code to be superior to FRAPCON-1 as a fuel rod behavior analysis tool in both thermal and deformation modeling capability.

2. DEVELOPMENT OF THE TRAC CODE FOR BWR ANALYSIS

J. W. Spore, W. L. Weaver
C. M. Mohr, K. Trambauer

Significant progress has been made on the development of TRAC-BD1,[1] a computer program for the detailed thermal-hydraulic analysis of loss-of-coolant accident transients in BWR systems. During the quarter, a new model to limit countercurrent flow and the capability to connect more than one TRAC component to a single vessel cell were completed, tested, and documented. With these new capabilities, the original scope of model development is complete. In addition to BWR component models (shrouded fuel bundles, jet pumps, and separators and dryers), TRAC-BD1 has two-fluid hydrodynamics, a choked flow model, an upgraded "heat slab" model, an improved heat transfer capability for PIPE components, and a decay heat model that incorporates the recent ANS/ANSI Standard.

The scope of TRAC-BD1 model development was expanded during the quarter to include a boiling transition model that incorporates flow history, modification of the momentum equations to account for area changes in both one- and three-dimensional components, and modification of the TRAC-BD1 jet pump model to achieve compatibility with improved momentum equations. The boiling transition model will be supplied by the General Electric Company. Substantial progress was made at the Idaho National Engineering Laboratory (INEL) toward modifying both one- and three-dimensional momentum equations.

The backward difference form of the momentum flux terms in the TRAC motion equations does not permit coarse noding in the vicinity of smooth and abrupt area changes. If coarse noding is used, reversible pressure changes are not recovered. It is proposed to correct this deficiency by approximating momentum flux terms with central differences so that the reversible component of the pressure differential across an area change is calculated correctly. The irreversible component is calculated with an additive loss coefficient.

The central differencing technique proposed for TRAC-BD1 has been found to be numerically stable. It has also led to improved agreement of the TRAC-BD1 jet pump model with data. A jet pump model with backward differenced momentum flux terms requires as many as 18 cells to obtain the correct pressure rise through the diffuser. A jet pump model with only one cell in the diffuser is compared with 1/6-scale INEL jet pump data in Figure 13. The comparison indicates that central differenced momentum flux terms yield good representation of the INEL jet pump behavior.

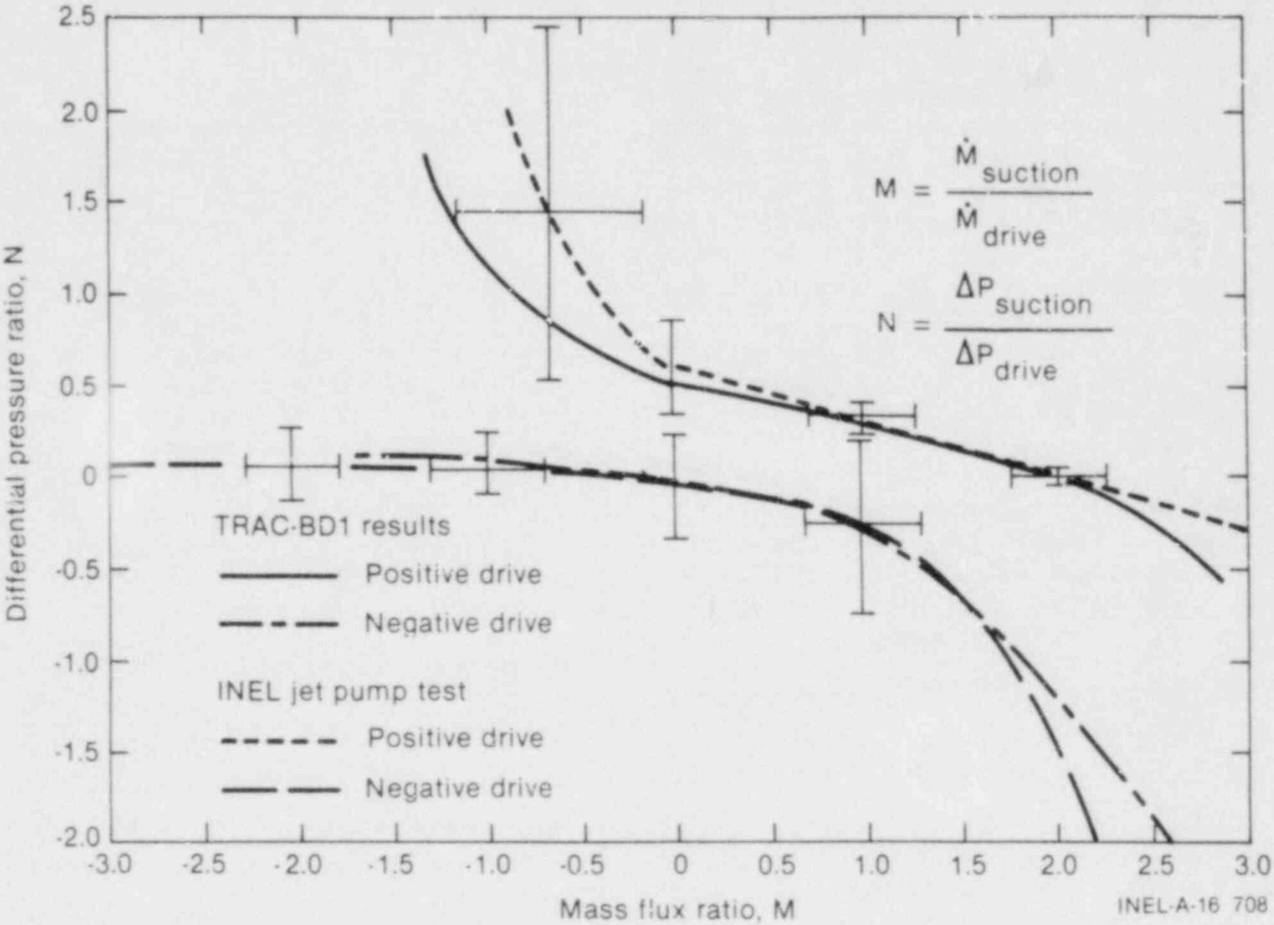

Figure 13. Comparison of INEL 1/6-scale jet pump data with model having single cell in diffuser.

V. CODE ASSESSMENT AND APPLICATIONS PROGRAM

J. A. Dearien

The Code Assessment and Applications Program (CAAP) has a primary responsibility to the NRC for the assessment of thermal-hydraulic and fuel behavior analytical codes. Data obtained from experimental programs such as LOFT, Semiscale, and the Thermal Fuels Behavior Program 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. In support of code assessment activities, the NRC/Reactor Safety Research (RSR) Data Bank is being developed to facilitate the processing of experimental data and comparisons of the experimental data with calculations performed using the analytical codes. In addition to assessing codes, the CAAP 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 CAAP 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. Another program in which the CAAP 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 that which occurred at Three Mile Island.

The following sections summarize the current status of the NRC/RSR Data Bank and present results from two code assessment activities.

1. NRC/RSR DATA BANK

N. R. Scofield

The idea of a central, computerized data bank to support water reactor safety research first originated in 1976. This NRC/RSR Data Bank provides a centralized data base from which code assessment, applications, and development groups can draw qualified experimental data quickly and easily both at the Idaho National Engineering Laboratory (INEL) and at other offsite NRC support facilities.

The Data Bank is an integral part of the INEL Scientific Data Management System (ISDMS), a generalized scientific data processing system supported by the major data producing and handling programs, Loss-of-Fluid Test (LOFT) Program, Semiscale Program, and Thermal Fuels Behavior Program (TFBP) organizations. Because these organizations use the same computer programs as ISDMS, plotted data comparisons are easily made. In addition, RELAP and TRAC calculations, as well as results from other thermal-hydraulic codes, are readily incorporated into the ISDMS network.

During the past year, data from 75 tests were added to the Data Bank, thus enlarging the data base as well as establishing three new data sources. The solid-line ellipses on the map shown in Figure 14 reflect current data sources available through the Data Bank. Those ellipses with dashed lines will be added in the coming year, or when the data become available.

2. TRAC-P1A CALCULATIONS FOR HOT AND COLD LEG BREAKS IN A PWR, INCLUDING THE EFFECT OF STEAM GENERATOR TUBE RUPTURES

J. R. Larson, M. A. Bolander, and P. D. Wheatley

Calculations have been performed with a four-loop model of a typical pressurized water reactor with the TRAC-P1A computer program.[18] The calculations covered a range of postulated loss-of-coolant accident break conditions. The objectives of the study were to determine the general behavior of the code, obtain baseline information for use in further studies, and determine areas in

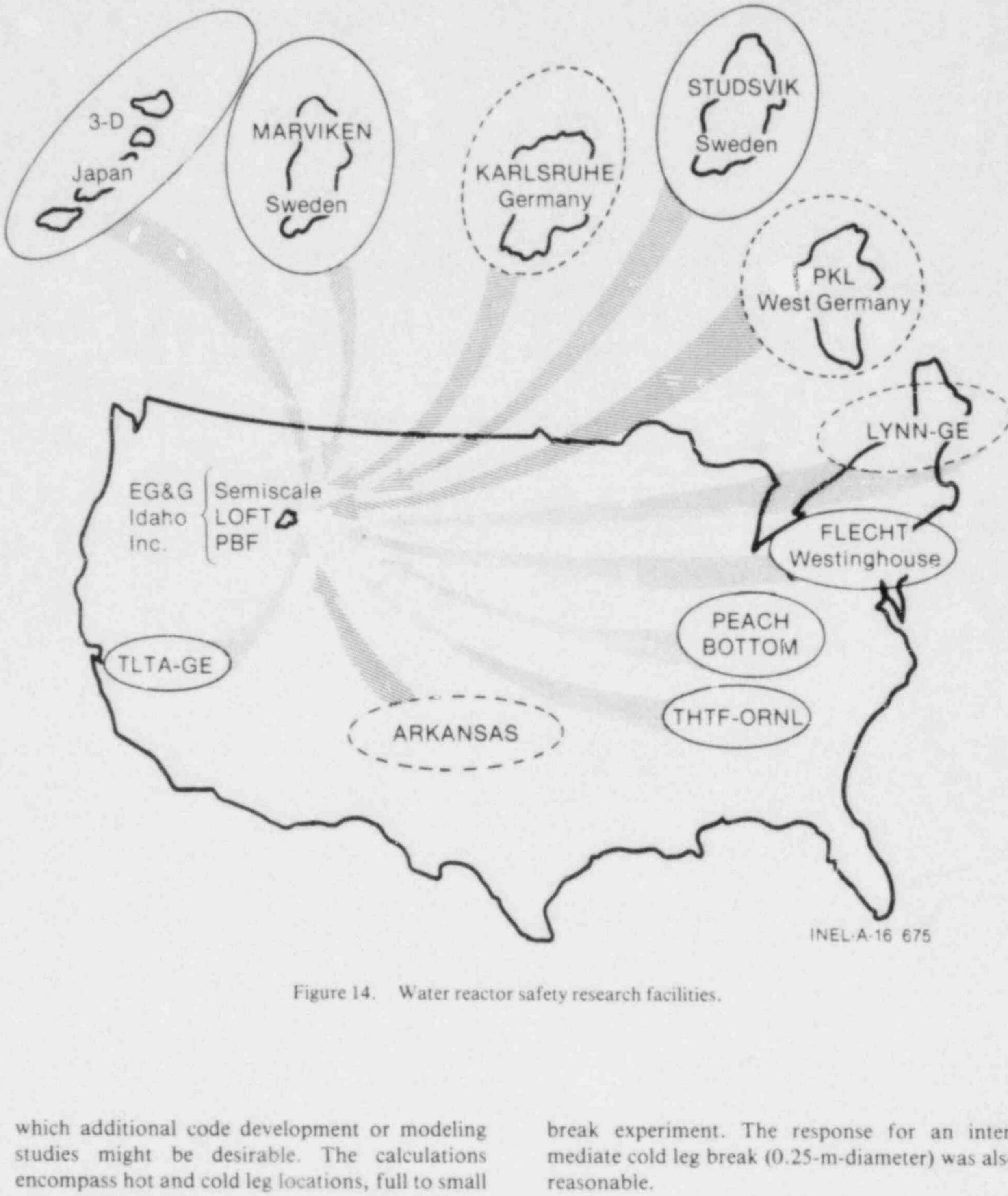

Figure 14. Water reactor safety research facilities.

which additional code development or modeling studies might be desirable. The calculations encompass hot and cold leg locations, full to small sizes, and include steam generator tube rupture as an additional parameter. The calculations cover the blowdown, refill, and reflooding phases of the accident.

For the 200% cold leg break, the calculated responses were typical of those of other calculations and the behavior of selected parameters compared well with results of a large cold leg break experiment. The response for an intermediate cold leg break (0.25-m-diameter) was also reasonable.

For the small cold leg break (0.10-m diameter), the calculated primary system depressurization appeared adequate. However, the calculated response of the steam generator secondary side was not satisfactory. The steam generator secondary side pressure remained at the relief valve setpoint. The cause of this secondary pressure response is not completely understood. A preliminary check of a mass balance on the secondary side indicated a numerical loss of mass about equal to the auxiliary feedwater flow rate. This numerical loss of mass contributed to the calculated steam generator secondary pressure response.

Two calculations were performed for the 200% cold leg break, including ruptured steam generator tubes. The calculations simulated both a small and a large number of ruptured tubes; each case sized to obtain a desired leakage mass flow rate from the steam generator secondary side to the primary side. The case simulating small leakage mass flow rate was selected because this flow rate was equivalent to the rate that caused the largest increase in cladding temperature during the Semiscale Mod-1 tests.[19] The large mass flow rate case was selected because it was the largest rate tested in the Mod-1 system.

The two calculations exhibited significant differences in system thermal-hydraulic behavior when compared to the calculation without ruptured tubes. The differences were caused by liquid mass from the ruptured steam generator entering the upper plenum and traveling down through the core. Vapor generated in the core continued flowing through the lower plenum and up the downcomer, thereby retarding accumulator and emergency core coolant liquid from reaching the lower plenum. Differences in the lower plenum liquid volume fraction are shown in Figure 15. For the calculation without ruptured tubes, the lower plenum was filled and liquid forced into the core by the accumulator gas discharge, which began the bottom flooding process. The calculation for the small rupture indicated a gradual refill of the lower plenum with liquid furnished by the emergency core coolant systems after the accumulator gas discharge. With lower plenum refill occurring, the bottom flooding process began and the increase in cladding temperature was arrested. The calculation for the large rupture indicated that the rupture mass flow prevented liquid from the accumulators and emergency core coolant systems from reaching the lower plenum until the ruptured steam generator became depleted of coolant mass.

Although the calculation for the large rupture was terminated prior to bottom flooding, it appeared that refill would have been completed prior to achieving cladding temperatures in excess of those calculated for the small rupture.

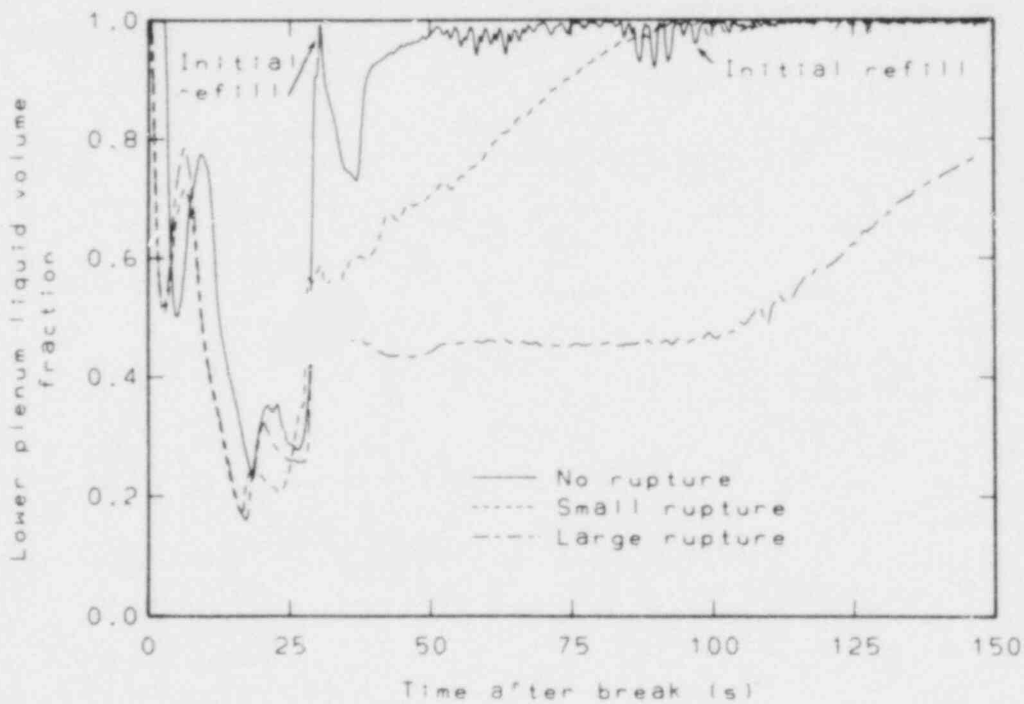

Figure 15. Lower plenum liquid volume fraction as a function of time after break with a small and large number of ruptured steam generator tubes compared with the calculation with no rupture.

Calculations were also performed for a 200% hot leg break with and without the rupture of a small number of steam generator tubes. The overall thermal-hydraulic response of the two hot leg break calculations was similar during the blowdown and refill stages of the transient; however, during the reflood stage of the transient, the time of core reflood and rod cladding quench was delayed in the tube rupture calculation. Figure 16 illustrates the upper core rod cladding temperature response of both calculations. The rods experienced a slight heatup early in time due to a partial voiding of the core. At 20 s in the tube rupture calculation, liquid from the ruptured steam generator tubes was being injected into the primary hot leg and into the vessel upper plenum. The fluid traversed around the upper plenum and exited through the break. The increase in liquid at the break increased the two-phase density at the break, decreasing the break volume flow rate. The decrease in break volume flow rate increased the system pressure, thus retarding the core mass flow rate and delaying the beginning of upper core reflood and rod cladding quench, as seen in Figure 16.

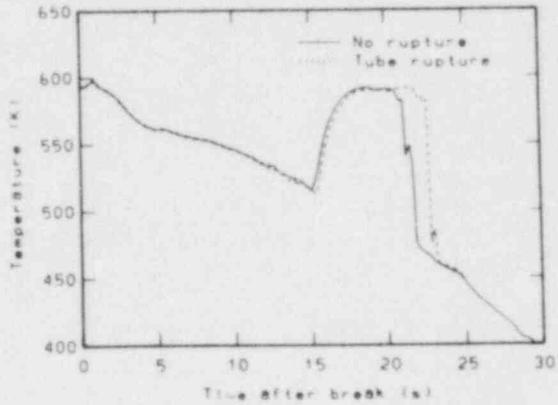

Figure 16. Cladding temperature as a function of time after a 200% hot leg break with a small number of ruptured steam generator tubes compared with the calculation with no rupture.

The TRAC-P1A computer code calculated the thermal-hydraulic behavior of the hot leg break transients and the effects of the ruptured steam generator tubes reasonably well. Core mass flow remained positive throughout the calculations, and the flow patterns within the vessel appeared to be reasonable. Effects from the tube rupture delayed core reflood and rod cladding temperature quench times as expected.

3. AN ANALYSIS OF SEMISCALE MOD-1 TEST S-04-6 USING THE TRAC-P1A COMPUTER PROGRAM

A. C. Peterson

Results from a TRAC-P1A simulation of Test S-04-6, a 200% cold leg break experiment, were compared with experimental measurements. These comparisons were performed as part of the overall assessment of the capabilities of TRAC-P1A being performed for the NRC.

An indication of the capabilities of TRAC-P1A to calculate local phenomena is indicated in Figure 17, which shows the calculated and measured mass flow rate in the broken loop cold leg. TRAC-P1A calculated a lower mass flow rate than was measured during the initial 3 s of the transient. From 3 to about 8 s after rupture, a higher mass flow rate was calculated than was measured. The difference between the measured and calculated mass flow rates between about 20 and 35 s after rupture indicates that more emergency core coolant was bypassed out the cold leg break during the experiment than was calculated. From about 40 to 60 s after rupture, more

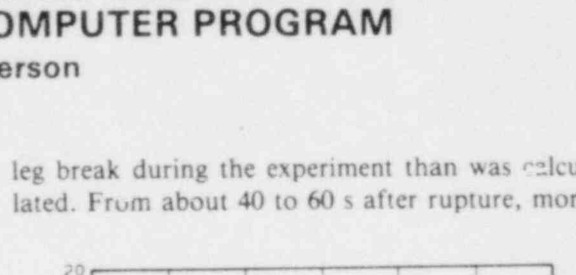

Figure 17. Test S-04-6 alculated and measured mass flows in the broken loop cold leg.

emergency core coolant bypass was calculated than was measured. In general, as illustrated in Figure 17, the mass flow rates were not accurately calculated by TRAC-P1A.

Comparisons of the rod cladding temperature response indicated that adequate calculations were obtained at core locations where an early critical heat flux (CHF) occurred. At core locations where a delayed CHF was measured, the calculated rod cladding temperatures were higher than the measured temperatures. These results illustrate that the hydraulic phenomena that result in a delayed CHF were not calculated by TRAC-P1A.

VI. 2D/3D PROGRAM

R. E. Rice, Manager

The 2D/3D Program includes the 2D/3D Instrument Projects and advanced instrumentation development. The 2D/3D Instrument Projects contribute technology and instrumentation to a multinational (U.S., Japan, and the Federal Republic of Germany) experimental program that investigates two- and three-dimensional phenomena in simulated pressurized water reactor loss-of-coolant reflood tests. Advanced instrumentation efforts support all EG&G Idaho experimental programs through the development of specialized measurement devices. Indirect support is also provided to analytical efforts by allowing data to be gained in previously unmeasurable areas.

1. 2D/3D INSTRUMENT PROJECTS

J. B. Colson

The objectives of the projects 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, and design and analysis support to the NRC. The following instruments have been delivered to Japan for installation in the Slab Core Test Facility (SCTF): two instrumented spool pieces for measuring two-phase mass flow rates in the piping; three drag disk transducers for measuring fluid velocity in the downcomer; two fluid distribution grids for measuring the distribution of liquid in the upper plenum and the downcomer; sixteen turbine meter probes for measuring fluid velocity at various locations in the pressure vessel; and the preliminary copy of the SCTF computer program subroutines for use in processing data from EG&G Idaho-supplied instruments on the Japanese computer. A training session in the operation of SCTF instruments was conducted at the INEL for Japanese personnel.

2. SCTF GAMMA DENSITOMETER

J. B. Colson and R. R. Rohrdanz

A gamma densitometer has been designed, fabricated, and tested for use in the Slab Core Test Facility in Tokai, Japan. The densitometer will be used to measure the coolant fluid density during the refill and reflood phases of a simulated loss-of-coolant accident. The design is based on the attenuation of a gamma beam as a function of density.

The densitometer was designed to maximize measurement accuracy during high void fraction, low density, two-phase flow in the range of 0.7 to 70 kg/m^3, and to measure up to 1000 kg/m^3 at reduced performance. Environmental criteria included a severe environment of superheated steam up to 1073 K and thermal shock. Other design criteria included 350 ms response time, SI (metric) hardware design, less than 2 mr/h surface radiation from the source, 300 series stainless steel construction, and restricted envelope limitations.

Two models were designed for application at four different regions of the test facility, for a total of twenty-three density measurements. One model, shown in Figure 18, was designed to measure the fluid density of the coolant in the core. As such, the gamma beam was collimated to be 2 mm wide by 39 mm high and focused between heater rods in adjacent simulated fuel bundles. The beam path through the fluid is about 250 mm long. The other model is designed to measure the density in the core end box, the upper plenum, and the hot leg piping where the area is more open. In these locations, a circular gamma beam collimated to 25 mm in diameter is used.

To meet the design criteria, a low-energy source was selected to have the best sensitivity and accuracy over the desired operating range of densities. The ^{241}Am gamma ray at 60 keV was selected. This source has the added advantages of

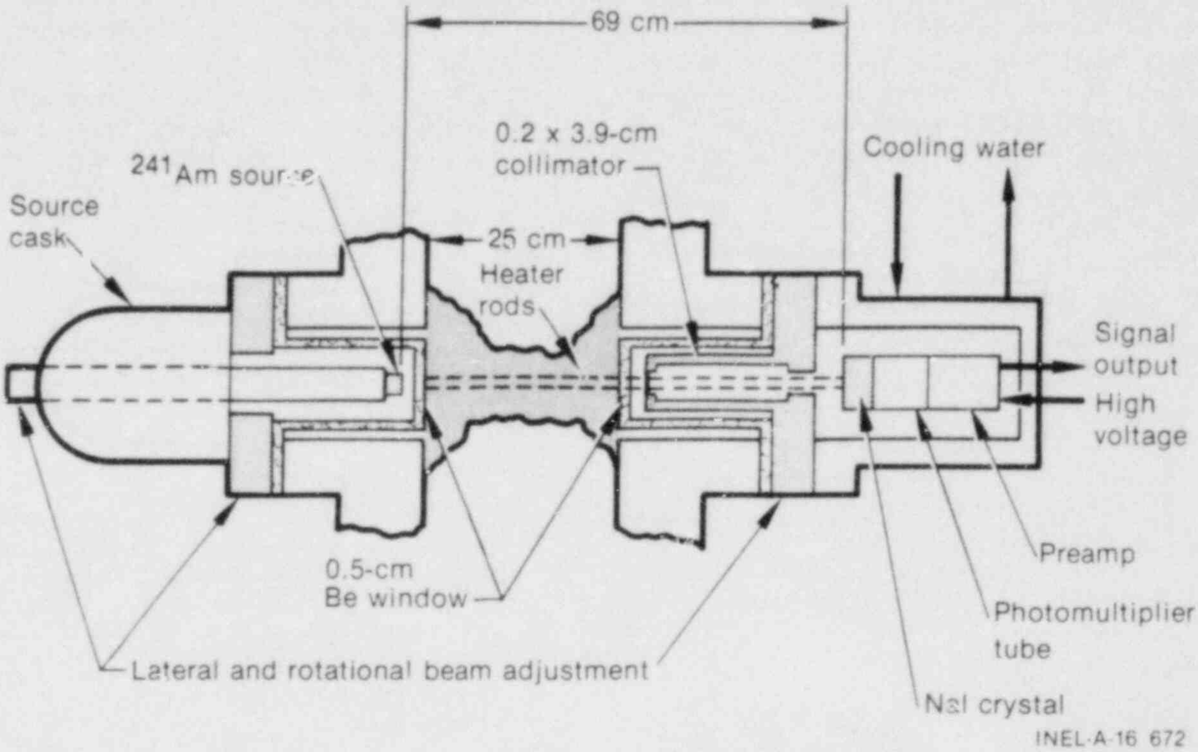

Figure 18. Schematic of rectangular beam densitometer.

long half-life (hence, no significant decay) and is easily shielded to a low radiation level. Since ^{241}Am is essentially a surface source due to its self-shielding, the maximum usable source strength is about one curie. This provides a maximum count rate to the detector of about 500 000 counts/s. This high count rate is necessary to minimize statistical error in the measurement.

To detect and process this count rate, a high speed system was developed using a feedback-stabilized photomultiplier tube in conjunction with a sodium iodide detector. The feedback circuit controls the photomultiplier high voltage to stabilize the pulse height output. High speed electronics are used to minimize the effects of dead time in the counting amplifier. The amplifier output is passed through a single-channel analyzer into a scaler, providing a digital output summing the counts in a 300-ms interval. The outputs of all 23 channels are combined into a single serial RS-232C format to be processed by the facility data acquisition system.

This design has eliminated the need for liquid nitrogen cooling of previous low-energy designs using Ge/Li detectors. Also, no dead time correction is needed and the use of direct digital processing has eliminated any error due to analog-to-digital and digital-to-analog conversion.

The statistical error due to the random decay of the source and the calibration errors contribute the major portion of the measurement error. The statistical error is calculated to be less than ± 1 kg/m^3 $\pm 0.3\%$ of reading, for a 95% confidence level over the range of 0.7 to 70 kg/m^3. The calibration errors were shown to be less than ± 1.7 kg/m^3 $\pm 0.7\%$ of reading.

3. ADVANCED INSTRUMENTATION

W. H. Roach, Manager

A tomographic densitometer for two-phase flow in large-diameter pipes has been completed and is ready for installation in the LOFT Two-Phase Flow Test Facility. Low flow velocimeters, using heated and cooled thermocouples, have been developed and tested. A holographic camera system has been completed and will be used for studies of bubble growth and heat and mass transfer. Velocity measurements at low void fractions and calibrations have been completed with the laser doppler low flow velocimeter. The heated thermocouple liquid level probe, completed and tested, is more fully explained below.

3.1 Heated Thermocouple Liquid Level System
Jay V. Anderson

A heated thermocouple liquid level system has been developed and tested by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory to demonstrate rapid liquid level detection for nuclear reactor application. The heated thermocouple approach was selected on the basis of long life and radiation resistance requirements and previous experience with heated and unheated thermocouples.[2, 20-22] System response, using economical components, has been demonstrated to be 0.1 s under ambient and pressurized water reactor (PWR) pressure and temperature conditions.

The heated thermocouple liquid level system consists of a transducer, a controller, and a discriminator or microprocessor, depending on response time requirements. The prototype transducer shown in Figure 19 consists of a Type K thermocouple, a nichrome heater element, nickel heater current conductors, magnesium oxide insulators, and a stainless steel sheath. Length of the heated section was kept short to reduce heater power, minimize response time, and produce a quasi-point measurement. The controller, shown in Figure 20, consists of an electronic ice point, amplifiers, operating temperature voltage equivalent, and current-limited heater power supply.

The theory of controller operation is unique in that it controls the heater power to maintain a preselected transducer temperature, which is chosen to be hotter than the maximum plant temperature, when the transducer is in the gas phase (dry). When the transducer is dry, the thermocouple voltage amplified by the instrumentation amplifier is essentially equal to the operating temperature voltage equivalent and the controller operates linearly to maintain the preselected temperature. Amplifier gain is selected to force the heater power supply into current limiting when the

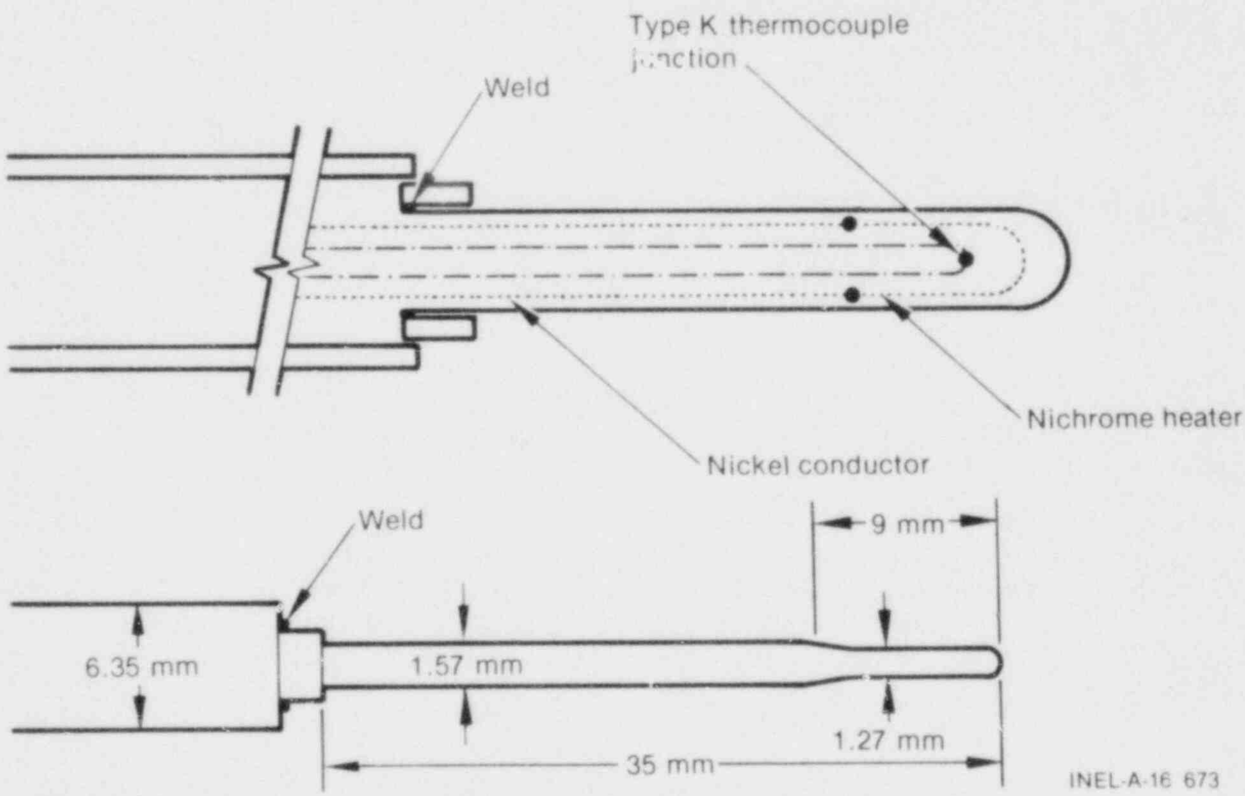

Figure 19. Illustration of heated thermocouple liquid level transducer.

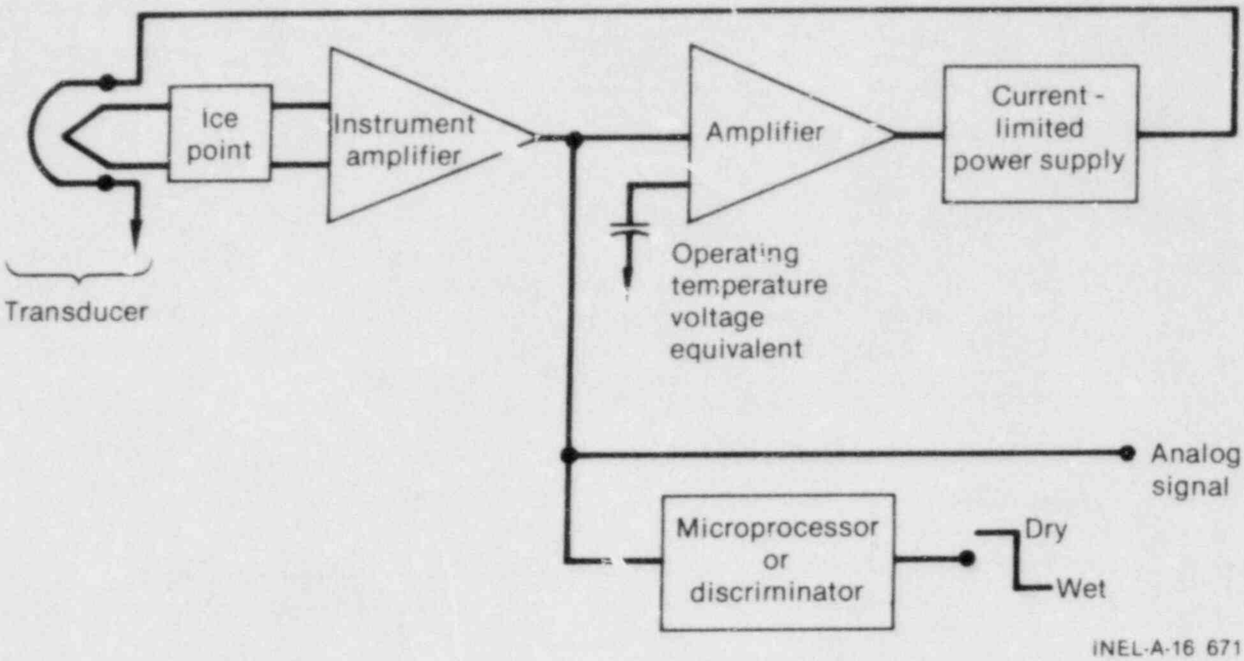

Figure 20. Block diagram of the heated thermocouple liquid level system.

transducer is immersed in water or struck by a droplet. This current-limited heater power ensures a rapid wet-to-dry transition; the dry-to-wet transition is already rapid due to transducer/water heat transfer characteristics. Thus, the controller serves to ensure rapid response and protect the transducer from excessive temperatures resulting from constant power operation. Wet/dry discrimination is performed by a comparator (discrimination) on the basis of temperature, or through the use of a microprocessor on the basis of temperature gradient.

Discrimination response under ambient (3.25 s) and PWR temperature and pressure (0.25 s) conditions, based on a preselected temperature of 645 K and a discriminator level halfway between this temperature and the 616 K water temperature, is shown in Figure 21. Also shown is the response to droplets, and flowing and static ambient temperature water. Pressurized water reactor conditions were simulated through the use of a tip-over autoclave.

To reduce the response time, a microprocessor was used to form the derivative (temperature gradient) of the analog signal, to reject the low temperature gradient of the flowing-to-static transition, and to accept the high temperature gradient of the wet-to-dry transition. This digital differentiation is accomplished by comparing the temperature before and after a 0.1-s delay instigated upon detection of a transition; thus, the system is said to have a 0.1-s response. This same 0.1-s delay is applied to the dry-to-wet transition. Microprocessor response to ambient temperature water and droplets, which is also attainable under PWR conditions, is shown in Figure 22.

Viability of the use of a heated thermocouple, controller, and microprocessor has been demonstrated as an economical and simple liquid level measuring system with a 0.1-s response time. Previous experience with thermocouples suggests that a radiation-resistant, long-lived, durable, and reliable heated thermocouple can be qualified for application in nuclear reactors.

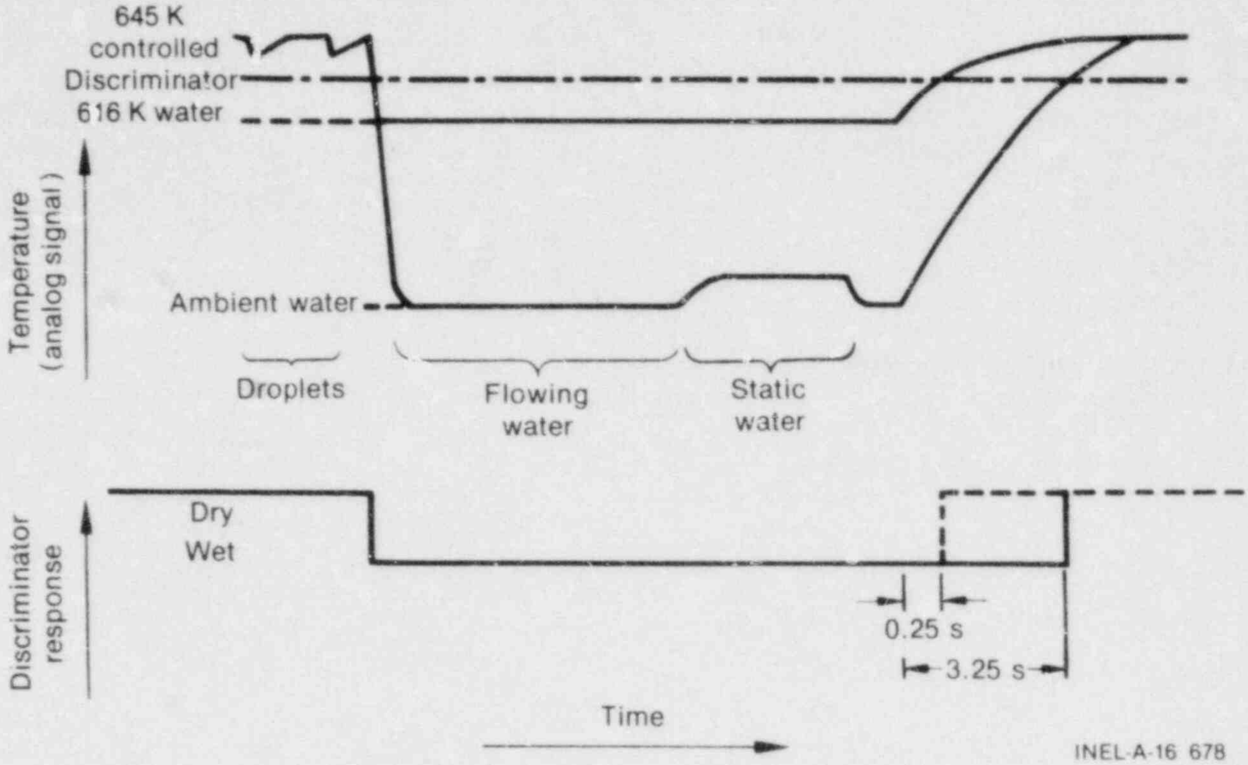

Figure 21. Liquid level system discriminator response.

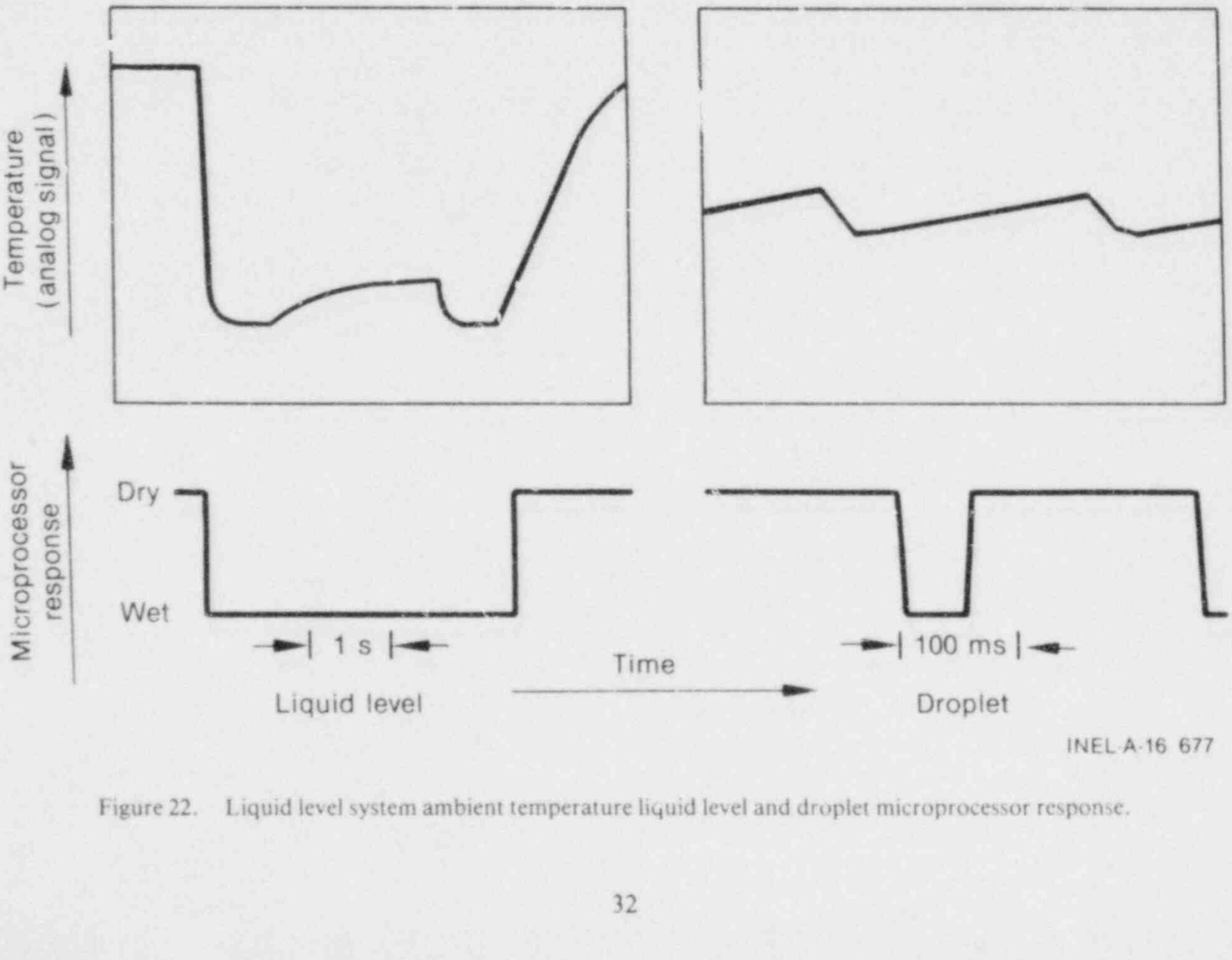

Figure 22. Liquid level system ambient temperature liquid level and droplet microprocessor response.

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016

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