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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 OCTOBER-DECEMBER 1979

L. J. Ybarrondo, Director
Water Reactor Research

Published January 1980

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ABSTRACT

Water reactor research performed by EG&G Idaho, Inc., during October through December 1979 is reported. The Semiscale Program conducted the first three tests in the Semiscale small break test series to investigate the phenomena that occur in a small break loss-of-coolant experiment. The Loss-of-Fluid Test (LOFT) Experimental Program performed and reported results of the first in a series of small break tests in the LOFT nuclear test reactor. The Thermal Fuels Behavior Program completed (a) three loss-of-coolant accident (LOCA) tests in the Power Burst Facility, (b) scoping tests with a fission product measurement system, and (c) the first in a series of internal fuel rod fill gas composition tests (with xenon and helium mixtures) in the Halden reactor in Norway. The Code Development and Analysis Program progressed in the development of advanced computer codes, including checkout of the BEACON/MOD3 containment analysis code and development of a capability in the TRAC code for analysis of LOCA transients in a boiling water reactor. The Code Assessment and Applications Program performed calculations in support of the NRC review of the Three Mile Island accident and performed review of various reactor vendors' small break analyses. The 3-D Experiment Project made progress in its instrument project for a test facility in Japan. Advanced instrumentation progress shows the viability of a tomographic technique for imaging two-phase fluid flows using a nine-beam densitometer.

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 following programs: 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 Engineering Support Projects.

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). The test program includes a series of nonnuclear (without nuclear heat) LOCEs, a series of low-power nuclear LOCEs, and a series of high-power nuclear LOCEs.

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 experimental data. Statistical evaluations of both the analytical and experimental results are part of the assessment process. Assessment results serve to inform the scientific community interested in reactor safety of the relative capabilities, validity, and range of applicability of NRC-developed codes.

Engineering Support Projects encompasses the 3-D Experiment Project and water reactor research advanced instrumentation. The 3-D Experiment Project 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 (for April-June 1975)
ANCR-1296 (for July-September 1975)
ANCR-NUREG-1301 (for October-December 1975)
ANCR-NUREG-1315 (for January-March 1976)
TREE-NUREG-1004 (for April-June 1976)
TREE-NUREG-1017 (for July-September 1976)
TREE-NUREG-1070 (for October-December 1976)
TREE-NUREG-1128 (for January-March 1977)
TREE-NUREG-1147 (for April-June 1977)
TREE-NUREG-1188 (for July-September 1977)
TREE-NUREG-1205 (for October-December 1977)
TREE-NUREG-1218 (for January-March 1978)
TREE-1219 (for April-June 1978)
TREE-1294 (for July-September 1978)
TREE-1298 (for October-December 1978)
TREE-1299 (for January-March 1979)
TREE-1300 (for April-June 1979)
EGG-2003 (for July-September 1979)

SUMMARY

The Semiscale Program conducted the first three tests in the Semiscale small break test series. Tests S-SB-4, S-SB-4A, and S-SB-2 were conducted in the Semiscale Mod-3 test facility to investigate the thermal-hydraulic phenomena resulting from a small break loss-of-coolant experiment (LOCE) and to evaluate the capability of available computer codes to calculate those phenomena. The Semiscale small break test series is part of the overall water reactor safety research effort directed toward assessing and improving the analytical capability of computer codes to accurately predict the behavior of a pressurized water reactor (PWR) during a postulated small break loss-of-coolant accident (LOCA). This research effort also includes small break tests in the Loss-of-Fluid Test (LOFT) Facility and a series of computer code calculations as an audit of PWR small break calculations performed by reactor vendors. The Semiscale small break test series has been designed to be compatible with LOFT small break tests and the PWR audit calculations. Tests S-SB-4 and S-SB-4A were conducted as counterpart tests to LOFT LOCE L3-1, with initial conditions and system geometry closely matching those in LOFT. Test S-SB-2, the third test performed, was conducted to provide data for a small break LOCE in a PWR in which the break flow rate was greater than the high pressure injection system flow rate. The results from the small break tests will be used primarily as part of an experimental data base for the development and assessment of analytical techniques used to predict the behavior of a PWR under LOCA conditions.

The LOFT Experimental Program conducted and reported the preliminary results of the first nuclear experiment in the small break loss-of-coolant test series. This experiment was conducted at the typical PWR maximum licensed linear heat generation rate. Preliminary results show that system behavior was about as expected for a small break in which the break flow exceeds the high pressure safety injection system flow. The emergency core cooling systems were effective in keeping the core covered with water throughout the gradual depressurization.

The Thermal Fuels Behavior Program completed (a) the Loss-of-Coolant Accident (LOCA) Blowdown Tests TC-1, LOC-5B, and LOC-5C in the Power Burst Facility; (b) the scoping tests with the IFA-430 fission product measurement system in the Halden reactor in Norway; and (c) the first in a series of internal fuel rod fill gas composition tests with mixtures of xenon and helium in the Halden reactor. The TC-1 tests were performed to determine if externally mounted cladding surface thermocouples significantly affect the cladding heatup and subsequent quench following a loss-of-coolant accident. The objective of Tests LOC-5B and LOC-5C was to determine the effect of prior irradiation and internal rod prepressurization on light water reactor (LWR) fuel rod behavior when cladding temperatures in the beta-phase of zircaloy (>1245 K) are reached during a LOCA. The fission product measurement system scoping tests were performed to determine the operational limits and optimum parameter setpoints for fission gas and iodine release measurements. In the fill gas composition tests, the thermal performance of two LWR-type fuel rods was measured as a function of internal pressure and gas composition.

The Code Development and Analysis Program emphasized development of advanced computer codes. The BEACON/MOD3 containment analysis computer code was in the process of developmental checkout during the reporting period. A set of boiling water reactor (BWR) system component models was added to the TRAC code to form the basic capability for analysis of BWR LOCA transients, TRAC-BDO. Results of pretest calculations performed with developmental versions leading to TRAC-BDO indicate the code can be used to execute a BWR calculation.

The Code Assessment and Applications Program performed audit and sensitivity calculations with an experimental version of the RELAP4 code to support NRC review of the Three Mile Island (TMI) accident and Westinghouse Electric Corporation, Combustion Engineering, Inc., and Babcock & Wilcox Company small break safety analyses of PWRs. The independent assessment of FRAP-T5, the fifth version of a transient fuel rod analysis code, was completed.

Engineering Support Projects comprises the 3-D Experiment Project and advanced instrumentation development. The 3-D Experiment Project

efforts have been directed toward completion of instrument projects for the Cylindrical Core Test Facility (CCTF) located in Japan. Instruments delivered over the past year have now been made operational and have provided data from several of the CCTF experiments. Advanced instrumentation development efforts completed include high count rate electronics and initial software for tomographic densitometers, a local ultrasonic densitometer, and a thermocouple liquid level probe. Proof-of-principle prototype tests that were performed on a nine-beam tomographic densitometer and associated reconstructive algorithms have shown the viability of the tomographic technique for imaging two-phase fluid flow.

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SEMISCALE PROGRAM

D. J. Olson, 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) during an operational transient and during the depressurization (blowdown) and emergency cooling phase of small and large pipe break LOCAs.

The current Semiscale test facility is a two-loop configuration identified as the Mod-3 system. One loop is scaled to represent three of four operating loops in a commercial PWR. The other loop (the broken loop) represents the fourth PWR loop and has rupture disks that, when broken, cause the Mod-3 system to decompress (blow down) into the pressure suppression system that simulates the backpressure in a reactor containment system. The Mod-3 system has a simulated reactor pressure vessel that contains a 25-rod, full-length (3.66-m) electrically heated core simulator, a full-length upper head and upper plenum, and an external downcomer. The Mod-3 system broken loop has, like the intact loop, an active pump and steam generator. The Mod-3 system is designed with the capability to investigate the influence of upper head emergency core coolant (ECC) injection on the core thermal hydraulics.

Three experiments were conducted in November and December 1979, the first tests in the Semiscale small break test series. This test series is being conducted in the Semiscale Mod-3 facility to investigate the thermal-hydraulic phenomena resulting from a small break loss-of-coolant experiment (LOCE) and to evaluate the capability of available computer codes to predict those phenomena. The Semiscale small break test series is part of the overall water reactor safety research effort directed toward assessing and improving the analytical capability of computer codes to accurately predict the behavior of a pressurized water reactor (PWR) during a postulated small break loss-of-coolant accident (LOCA). This research effort also includes small break tests in the Loss-of-Fluid Test (LOFT) Facility and a series of computer code calculations performed by the Code Assessment and Applications Program as an audit of PWR small break calculations performed by reactor vendors. The Semiscale small break test series has been designed to be compatible with LOFT small break tests and the PWR audit calculations.

Test S-SB-4, the first test conducted and a counterpart to LOFT LOCE L3-1, had initial fluid conditions and system geometry that matched as closely as possible those in LOFT. Emergency core coolant flow rates and volumes were scaled from LOFT values by the ratio of Semiscale primary coolant volume to LOFT primary coolant volume. Semiscale system modifications included shortening the intact loop pump suction leg (to better represent the elevation differences between the cold leg centerline and the bottom of the pump suction leg in LOFT) and placing a valve between the broken loop pump discharge and the break assembly to simulate the noncommunicative break configuration in LOFT. The break size was $0.0613\ cm^2$, which was volume scaled from LOFT and represented a 4-inch-diameter break in the cold leg of a full-sized PWR.

Basic differences exist between Semiscale Mod-3 and LOFT. The Mod-3 system includes a

3.66-m electrically heated core and an active broken loop. The LOFT system has a 1.68-m nuclear core and has resistance simulators for the broken loop pump and steam generator. The LOFT system uses bypass lines to establish initial conditions in the broken loop, whereas initial broken loop fluid conditions cannot be established in the Mod-3 system without operating the broken loop.

In addition to system geometry differences between Semiscale and LOFT, other scaling influences exist that can affect the transient behavior of Semiscale relative to LOFT or a full-sized PWR. Since heat losses from the primary system are difficult to quantify, but can significantly affect system response, the second test in the series was conducted to determine the sensitivity of system behavior to the augmentation of core power to offset system heat losses. Therefore, test conditions and the system configuration for the second test, Test S-SB-4A, were identical to those of Test S-SB-4, except that additional power was applied to the core.

The results from Test S-SB-4 indicate the occurrence of a relatively slow but continuous depressurization during LOFT LOCE L3-1. The sequence of events including the initial voiding of the upper plenum, the loop seal blowout, and the gradual refilling of the system by a relatively low accumulator injection rate in Semiscale indicate the occurrence of a similar sequence of events in LOFT. Although bulk boiling in the core and voiding of fluid in the upper plenum and upper core regions may occur in LOFT, the results from Semiscale Test S-SB-4 indicated that substantial uncovering of the core was an unlikely event for LOFT LOCE L3-1.

Atypically high heat losses from the primary system in Semiscale are expected to have a substantial influence on system response relative to LOFT. The relatively high heat losses in Semiscale tend to increase the rate of system depressurization. Therefore, if significant behavioral differences are found to have occurred between LOFT LOCE L3-1 and Semiscale Test S-SB-4, they will most likely have been due to a slower depressurization in LOFT and a delay in the initiation of the sequence of events which occurred in Semiscale.

In Test S-SB-4A, additional power was applied to the core between 40 and 660 seconds after rupture to approximately compensate for system heat losses. The increased steam generation caused by the increased power resulted in a higher system pressure and larger break flows after 300 seconds in Test S-SB-4A relative to Test S-SB-4. The higher pressure caused a delay in accumulator injection which occurred about 770 seconds into the transient. However, when the reduction in power was initiated at about 610 seconds after rupture, the system pressure began to decrease relatively quickly, and after 885 seconds the pressure in Test S-SB-4A fell below that in Test S-SB-4.

The additional power applied to the core in Test S-SB-4A caused significant voiding in the core and upper plenum. Dryout in Test S-SB-4A began to occur in the upper third of the core at 510 seconds, and within 100 seconds dryout had occurred throughout most of the upper half of the core. The resulting cladding temperature rise in the upper half of the core was nearly adiabatic. A core power decrease to the normal decay power was initiated about 610 seconds after rupture, and the maximum cladding temperature occurred within 20 seconds. The power reduction allowed some cooling of the core to occur, but the core level did not recover and quenching did not begin until after accumulator injection was initiated. The entire core was quenched by 850 seconds. The low pressure injection system (LPIS) injection was initiated by 2240 seconds after rupture and the test was terminated approximately 600 seconds later.

The results of Test S-SB-4A indicate that, initially, heat losses can be made up by applying additional power to the core. However, after the core becomes uncovered, the additional power contributes to an atypically rapid rise in heater rod temperatures.

The third test conducted in the Semiscale small break test series was Test S-SB-2. The primary objective of this test was to provide data for a small break loss-of-coolant experiment in which the break flow rate was greater than the high pressure injection system (HPIS) flow rate. The break location and size were the same as in the two earlier tests, but the system configuration and operation were arranged to simulate a commercial PWR rather than the LOFT system. Initial and boundary conditions for the test were representative of typical pressurized water reactor operating conditions. Ambient temperature (300 K) emergency core coolant was injected into both the intact and broken loop cold legs using accumulator, high pressure injection, and low pressure injection systems.

As planned, the break flow rate for Test S-SB-2 was greater than the HPIS flow rate, resulting in a continuous system depressurization. Beginning at about 30 seconds after rupture, bulk boiling occurred in the core and upper plenum regions. However, the core remained covered throughout the test, and the cladding temperatures followed a few degrees above the fluid saturation temperature. Blowout of the pump suction seal occurred at about 270 seconds after rupture, and the intact loop accumuator injection began at about 650 seconds. LPIS flow was initiated at about 4200 seconds, and the test was terminated at 4700 seconds.

The results from the small break tests will be used primarily as part of an experimental data base for the development and assessment of analytical techniques used to predict the behavior of a PWR in the event of a LOCA. Because of the potential for scaling distortions in the Semiscale system data, the test results cannot be assumed to be typical of either a PWR or LOFT. The results can, however, be used as a means of identifying phenomena which may be important to the understanding of small break accidents.

LOFT EXPERIMENTAL PROGRAM
L. P. Leach, Manager

The LOFT Experimental Program was involved in preparation for and successful conduct of Loss-of-Coolant Experiment (LOCE) L3-1 in the LOFT system.[1] LOCE L3-1, which was conducted on November 20, 1979, was the first nuclear experiment[a] in the Small Break Test Series L3 and simulated a single-ended shear of a small (4-inch diameter) primary system pipe in a large PWR. Preliminary analysis has shown that each of the experimental objectives was achieved.

Analysis of the LOCE L3-1 data, currently underway, will result in additional understanding of the thermal and hydraulic phenomena associated with this type of LOCA and, together with the results from the NRC experimental program, will provide the basis for development and assessment of analytical models that are used for licensing commerical PWRs.

Experimental Conditions and Conduct of LOFT Nuclear LOCE L3-1
J. P. Adams

LOFT LOCE L3-1 simulated a single-ended shear break of a small (4-inch diameter) pipe in the primary system of a large PWR. LOCE L3-1 was conducted in the LOFT Facility, the extensively instrumented nuclear test system designed to reproduce, both in sequence and approximate magnitude, the thermal and hydraulic phenomena expected during a LOCA. A detailed description of the LOFT system is provided in Reference 1.

The LOFT system conditions at experiment initiation were: a maximum linear heat generation rate of 51.7 ± 1 kW/m (simulating the maximum expected in a commercial PWR, approximately 130% of nominal 100% power conditions in a PWR), an average temperature of 564 ± 3 K, a core differential temperature of 20 ± 4 K, a flow rate to system volume of 62.0 ± 2 kg/m^3·s, and a system pressure of 15.02 ± 0.03 MPa.

a. LOCE L3-0, the first experiment in the L3 series, was nonnuclear.

The experiment started by scramming the reactor. After all control rod bottom lights came on (at approximately -0.97 seconds), the quick-opening blowdown valve in the broken loop cold leg was opened (defined as time = 0 seconds) and the primary coolant pumps were tripped off. The pumps coasted down under the influence of a flywheel generator (simulating the coastdown of a pump in a PWR) until they reached 12.4 Hz (at 19 ± 1 seconds), at which time they were disconnected from the motor generator. The sequence of major events for LOCE L3-1 is included in Table 1, including predictions of the sequence.

LOCE L3-1 Results and Preliminary Analysis
J. P. Adams

Primary system depressurization was characterized by a rapid drop to system fluid saturation pressure. Between 5 and 60 seconds, the depressurization rate decreased in two distinct steps as the system pressure decreased to the cold leg fluid saturation pressure. At about 300 seconds, the depressurization rate increased. (Measured and calculated system pressure is shown in Figure 1.) Examination of additional experimental instrumentation (liquid level detectors, fluid densitometers, and thermocouples) indicates that at this time the liquid level decreased below the level of the break orifice (located in the midplane of the broken loop pipe), allowing higher quality fluid to flow out the break.

Accumulator flow was initiated at about 634 seconds, gradually increasing the system liquid level. By 1000 seconds, the liquid level apparently increased to the level of the break orifice, causing denser fluid to exit the break and the depressurization rate to abruptly decrease by approximately a factor of three.

The depressurization rate remained constant until approximately 1750 seconds, at which time it again abruptly decreased by approximately a factor of three. This time coincides with the initiation of the injection of nitrogen, a noncondensible gas,

TABLE 1. CHRONOLOGY OF EVENTS FOR LOCE L3-1

Event	Time After LOCE Initiation (s) RELAP4/MOD7 Prediction	Time After LOCE Initiation (s) LOCE L3-1 Data
Reactor scrammed	a	-2.15
Control rods bottomed	a	-0.97
Quick-opening blowdown valve opened	0	0
Primary coolant pumps tripped	a	0.040 ± 0.01
High pressure injection system initiated	3	4.6 ± 0.5
Pressurizer emptied	24.5	17 ± 1
Pump coastdown completed	14	19 ± 1
System pressure reached hot leg saturation pressure	32	19.5 ± 0.5
Subcooled, choked break flow ended	168	46.4 ± 0.5
Steam generator feed initiated	60[b]	75 ± 1
Accumulator flow initiated	950	633.6 ± 0.5
Nitrogen from accumulator entered system	a	1741 ± 1
Steam generator feed stopped	1860[b]	1875 ± 1
Steam generator steam bleeding initiated	a	3622.5 ± 1
Low pressure injection system initiated	a	4240 ± 1
Test completed	a	4368 ± 1

a. Not calculated.

b. Used as part of input data.

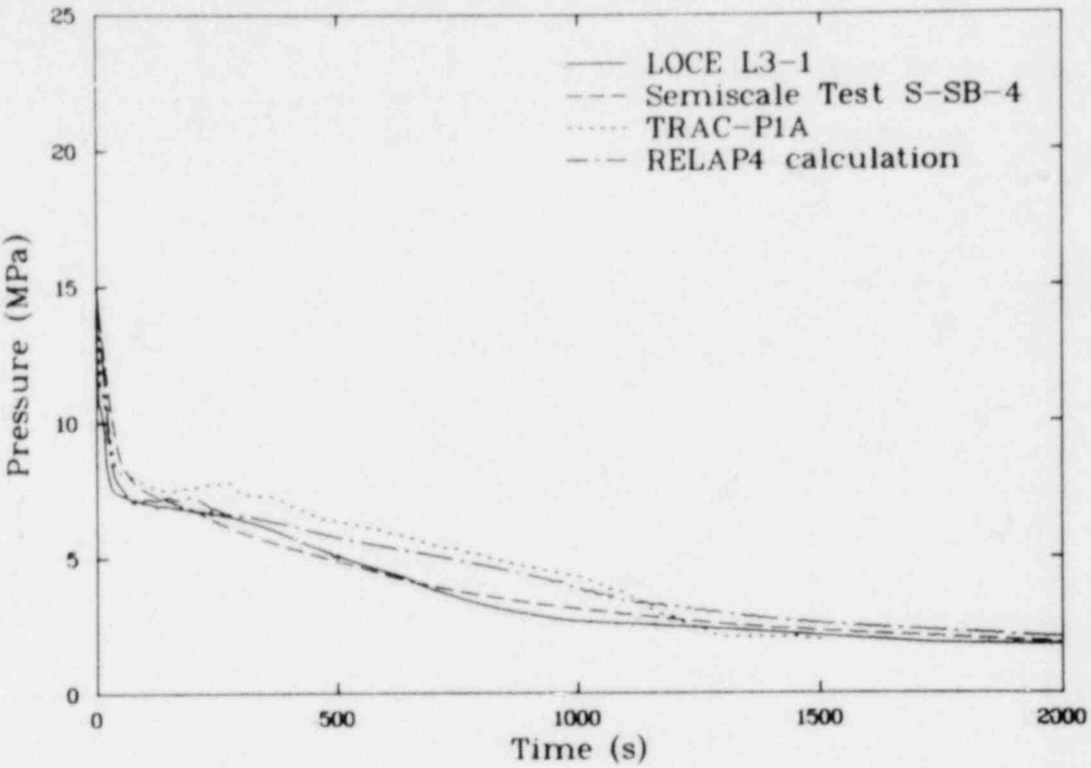

Figure 1. Comparison of calculated and measured primary system pressure for LOCE L3-1 with primary system pressure for the LOCE L3-1 counterpart test (Test S-SB-4) performed in the Semiscale test system.

from the accumulator into the system. Apparently, the cold nitrogen thermally expanded upon injection, causing the depressurization rate to decrease.

The pressure continued to decrease at approximately the same rate until 3622 seconds, at which time steam generator steam bleeding was initiated in an attempt to increase the depressurization rate. Steam bleeding was continued for 100 seconds with, however, no detectable effect of the steam bleeding on the primary side pressure. This was possibly due to the fact that the secondary side pressure remained above the primary side pressure throughout the steam bleeding operation. The primary side depressurization rate remained approximately constant until the LPIS initiation pressure was reached at 4240 seconds. The test was terminated at 4368 seconds.

Loop circulation (defined as positive circuit flow from the lower plenum, through the core to the upper plenum, out the reactor vessel hot leg nozzle to the intact loop, around the intact loop, into the reactor vessel cold leg nozzle, and down the downcomer to the lower plenum) decreased as the primary coolant pumps coasted down and forced convection ceased. At this time (approximately 2[illegible] seconds after blowdown initiation), a continuation of the loop circulation without forced circulation was indicated by the reactor vessel fluid temperature distribution and by flow and momentum transducers. At approximately 80 seconds, the intact loop hot leg started to void, and loop circulation was therefore interrupted.

After approximately 150 seconds, the steam generator was a heat source for the system, as secondary pressure exceeded primary system pressure. After this time, the driving head for the nonforced loop circulation was no longer available; heat transfer between the secondary and primary systems was apparently not significant due to the combined effects of stagnant primary flow and primary system voiding.

Departure from nucleate boiling was not detected at the fuel cladding surfaces, indicating that the core remained covered throughout the transient. This is confirmed by data from the reactor vessel liquid level detectors, which indicated that the minimum liquid level in the vessel was greater than 1 m above the top of the core.

Circulation of fluid within the core was apparently also a cooling mechanism. Cooler fluid from the accumulator entered the core through the lower plenum, decreasing the lower plenum fluid temperature and increasing the reactor vessel liquid level. The mixing of the cooler accumulator fluid with the core fluid during this time was apparently caused by circulation within the core.

An overlay of measured system pressure with predictions of LOCE L3-1 using an experimental version of RELAP4[a] and TRAC-PIA, as well as system pressure from the Semiscale LOFT Counterpart Test S-SB-4, is shown in Figure 1. As shown in Figure 1, the major transient trends were predicted by the calculations. In addition, the results of the Semiscale counterpart test (Test S-SB-4) follow the trends of LOCE L3-1 data, indicating that Semiscale data can be used to predict LOFT experiment results.

LOCE L3-1 provided data for the first time on a nuclear reactor undergoing a small break loss-of-coolant accident. The continuing analysis of LOCE L3-1 data will further the understanding of this type of accident. Conclusions based on the results of analyses completed thus far include:

1. The test objectives were met. The principal test variables (temperature, pressure, density, etc.) were measured and are available for assessment of computer code predictions. As indicated in Figure 1, the LOCE L3-1 results were predicted by both the pretest calculations and Semiscale test data.

2. The core remained covered with saturated or subcooled water during the entire transient.

3. After forced convection ceased at 20 seconds, loop circular flow continued until approximately 80 seconds after blowdown initiation. After 150 seconds, the steam genenerator became a heat source to the primary system; therefore, the possibility of thermal gradient natural circulation no longer existed.

4. Steam bleeding operations late in the transient were not effective in reducing primary system pressure because (a) the steam generator primary side was voided, which reduced primary to secondary heat transfer, and (b) steam bleeding operations were not continued long enough to reduce secondary pressure (temperature) below that of the primary system.

5. Although accumulator injection initially causes a slight increase in depressurization rate, it appears to have subsequently reduced the primary system depressurization rate, thus effectively elongating the transient. This is postulated to have been caused by first increasing the water level above the break orifice, and then by thermal expansion and pressurization of the accumulator nitrogen upon injection into the system.

6. The location of the break orifice (relative to the broken loop liquid level) affected the depressurization rate, resulting in a higher depressurization rate when the liquid level was below the break orifice, and lower when the level was above the break orifice.

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a. The experimental RELAP4 code used was RELAP4/MODG, Version 92, a preliminary version of 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 RLP4G92LFT03, Idaho National Engineering Laboratory Configuration Control Number H010084B.

THERMAL FUELS BEHAVIOR PROGRAM
H. J. Zeile, Manager

The objective of the Thermal Fuels Behavior Program is to provide experimental data for the development and assessment of computer codes used to calculate the behavior of typical power reactor fuel rods under normal, off-normal, and accident conditions. In the pursuit of this objective, a closely integrated program of experimentation and analysis is performed.

The experimental portion of the program is concentrated on the testing of single fuel rods and small clusters of fuel rods in the Power Burst Facility (PBF). The PBF mission is the completion of approximately 40 high-priority tests selected to obtain fuel rod behavior data under a wide variety of operating conditions and hypothesized accident sequences. The programmatic tests in PBF are divided into different test series. Three of the test series—Irradiation Effects, Gap Conductance, and PBF/LOFT Lead Rod—have been completed. The current series of tests are grouped as follows:

1. The Power-Cooling-Mismatch (PCM) Test Series provides in-pile experimental data on the behavior of PWR-type fuel rods during a decrease in coolant flow or during a slight overpower condition. Two tests remain in this series, one of which is a combined PCM/reactivity initiated accident type of test.

2. The Loss-of-Coolant Accident (LOCA) Test Series measures the response of both irradiated and unirradiated fuel rods during each major phase of a variety of LOCA situations. Two tests remain in this series.

3. The Reactivity Initiated Accident (RIA) Test Series determines threshold energy limits of incipient fuel rod failure and prompt fuel dispersal for test environments typical of power reactor conditions. Four tests remain in this series.

4. The Operational Transient (OPTRAN) Test Series will evaluate fuel behavior under severe operational transient conditions. Four tests with irradiated rods are planned in this series.

5. The Small Break LOCA/Flow Starvation Test Series will evaluate fuel behavior under severe flow starvation conditions. This test series is in the planning stage.

The TFBP also participates in in-pile testing of instrumented fuel assemblies (IFA) performed in the Halden reactor in Norway. The tests in Halden are generally long-term irradiations (two to three years) to provide data on irradiation effects on the steady state behavior of fuel rods.

The IFA-429 experiment is being used to measure the thermal performance and internal rod pressure of LWR-type fuel rods as a function of operating power and burnup. The present burnup of 30 000 MWd/t will be extended to the 50 000 MWd/t range.

The IFA-430 test is providing measurements of thermal performance, rod internal gas flow resistance, and fission product release in LWR-type fuel rods as a function of operating power and burnup. The effects of fill gas composition and pressure on fuel thermal performance are also measured.

The IFA-511 Test Series is intended to provide comparisons of the response of nuclear and electrically heated rods tested under reflood conditions using identical initial thermal-hydraulic conditions. Testing with nuclear rods has begun, and tests with two types of electrically heated rods are scheduled.

Program Status

During the past quarter the Thermal Fuels Behavior Program completed (a) the Loss-of-Coolant Accident (LOCA) Blowdown Tests TC-1, LOC-5B, and LOC-5C in the Power Burst Facility; (b) the scoping tests with the IFA-430 fission product measurement system in the Halden reactor in Norway; and (c) the first in a series of internal fuel rod fill gas composition tests with mixtures of xenon and helium in the Halden reactor. The TC-1 tests were performed to determine if externally mounted cladding surface thermocouples significantly affect the cladding heatup and subsequent quench following a loss-of-coolant accident. The objective of Tests LOC-5B and LOC-5C was to determine the effect of prior irradiation and internal rod prepressurization on light water reactor fuel rod behavior when cladding temperatures in the beta-phase of zircaloy (>1245 K) are reached during a LOCA. The fission product measurement system scoping tests were performed to determine the operational limits and optimum parameter setpoints for fission gas and iodine release measurements. In the fill gas composition tests in Halden, the thermal performance of two LWR-type fuel rods was measured as a function of internal pressure and gas composition. The following sections describe (a) PBF testing and (b) program development and evaluation activities during the quarter.

PBF Testing

P. E. MacDonald and R. K. McCardell

Tests LOC-5B and LOC-5C were conducted and preliminary results were compiled in a quick look report. The results of the TC-1 tests, described in the following section, were analyzed. Results of a nine-rod bundle test, Test PCM-5, were analyzed and published.[2] Experiment specifications were prepared for Tests OPTRAN 1-1 and RIA 1-7.

Other accomplishments included preparation for another series of blowdown tests (TC-2) to investigate thermocouple effects, and preparations for a combined PCM/RIA test (Test PR-1) and for a nine-rod bundle test (Test RIA 1-4), including completion of the test trains for these two tests.

PBF TC-1 Test Series

T. R. Yackle and M. E. Waterman

The behavior of light water reactors (LWRs) following a postulated loss-of-coolant accident must conform to criteria specified in the Code of Federal Regulations. To assure that the behaviors of both the cooling system and the nuclear core are correctly understood and properly modeled, in-pile experiments are being conducted in the LOFT Facility and Power Burst Facility (PBF) at the Idaho National Engineering Laboratory by EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission. The LOFT Facility was designed to represent the behavior of an entire large pressurized water reactor (PWR) during a postulated LOCA. The PBF-LOCA program is one of several PBF programs providing in-pile information on the behavior of nuclear fuel rods subjected to normal, off-normal, and accident conditions.

The fuel rods in these and similar test programs are instrumented with externally mounted cladding surface thermocouples. The cladding surface thermocouple measurements are used to assess computer calculations of the cladding temperature response and to evaluate and interpret the core thermal-hydraulic behavior. However, these surface thermocouples may act as cooling fins and influence the cladding temperature and fuel rod behavior.

The results of the TC-1 series (Tests TC-1A, TC-1B, TC-1C, and TC-1D) are discussed in this section. The TC-1 tests were conducted in the PBF to specifically evaluate the influence of cladding surface thermocouples on the thermal-mechanical behavior of nuclear fuel rods during PBF LOCA conditions. The PBF tests were as similar as possible to the LOFT L2 Test Series[1] so that the results could be extended to the LOFT Experimental Program.

Experiment Design. The TC-1 series was performed with four LOFT-type fuel rods contained in individual flow shrouds and symmetrically positioned within a test train in the PBF in-pile tube in an environment similar to the LOFT test environment.

Two of the rods, Rods 02 and 03, were each instrumented with four LOFT cladding surface thermocouples. The other two rods, Rods 01 and 04, were not instrumented with external thermocouples. In addition to the cladding surface thermocouples, each of the fuel rods was instrumented with a cladding axial extension transducer at the bottom, and interior cladding and fuel thermocouples located near the high power region of each rod. All of the internal thermocouples were fitted in slots at the outside of the fuel pellets. Some of the thermocouple junctions were welded directly to the inside cladding surface, and the remaining junctions (of the fuel thermocouples) were fitted in holes near the outside surface of the fuel pellets.

Experiment Conduct. The TC-1 series consisted of four separate LOCA tests (Tests TC-1A, TC-1B, TC-1C, and TC-1D) performed in the PBF. The four tests were performed in the following phases: preconditioning (Test TC-1A only), power calibration, decay heat buildup, blowdown and reflood, quench and cooldown. The four TC-1 LOCA transients were performed with the same initial conditions of inlet temperature, system pressure, coolant flow rate, and rod peak power, which were 600 K, 15.5 MPa, 0.8 l/s and 49.5 kW/m, respectively.

The preconditioning phase was conducted to allow fuel pellet cracking, restructuring, and pellet-cladding mechanical interaction to stabilize.

The power calibration phase provided data to intercalibrate the test rod powers with the reactor power. The power calibration phase was performed during the preconditioning phase of Test TC-1A and during the decay heat buildup phase of Tests TC-1B, TC-1C, and TC-1D. After the calibrations were performed, the desired fuel rod powers were achieved by monitoring the reactor power.

The decay heat buildup phase established approximately 78% of the maximum possible decay heat in the fuel rods just prior to blowdown. This was accomplished by increasing the reactor power by 1 MW per minute to 14.5 MW (which resulted in fuel rod peak powers of 49.5 kW/m), and then holding the power at that level for 90 minutes before initiating the blowdown phase.

The TC-1 LOCA transient was divided into blowdown and reflood phases. During blowdown, the test train and LOCA system were depressurized from PWR conditions to atmospheric conditions within 25 seconds. The blowdown was conducted to simulate the LOFT L2 Nuclear Test Series conditions in the PBF. The PBF reactor power was automatically controlled during blowdown to achieve a fuel rod cladding peak temperature between 850 and 1000 K, as was measured in the LOFT L2 tests. The PBF blowdown valves were also automatically cycled between 5 and 12 seconds during blowdown to force a two-phase liquid slug from the lower plenum past the fuel rods. This two-phase slug was planned to simulate the blowdown quench that occurred in the LOFT L2 tests. The slug period lasted 2, 4, and 6 seconds for Tests TC-1A, TC-1B, and TC-1C, respectively. Test TC-1D was a repeat of Test TC-1C.

The reflood phase was also designed to simulate the LOFT L2 test conditions in the PBF. Prior to reflood, the PBF core power was manually controlled to establish cladding temperatures between 900 and 1000 K. Reflooding of the test train commenced 100 seconds after initiation of blowdown in each of the tests. The reflood was performed by injecting coolant at a temperature of 311 K into the in-pile tube (IPT) upper head, down the center hanger rod, and into the plenum volume beneath the lower particle screen.

Results of TC-1 Tests.

The TC-1 Test Series was designed to subject the four test rods to thermal- hydraulic conditions that were as similar as possible to the LOFT L2 tests. The thermal-hydraulic conditions for Test TC-1C are summarized by the fuel rod coolant flow response as measured at the shroud outlet and inlet.

The outlet volumetric flow rates, as measured by upper turbine flowmeters, rapidly decreased from the initial 0.8 l/s to 0 at the initiation of blowdown. A comparison of the outlet volumetric flows for Rods 01, 02, and 03 shows that the flow was essentially zero early in blowdown, as expected. (The Rod 04 upper turbine flowmeter did not function.) Between 5 and 11 seconds, one hot leg valve was opened and the cold leg valve was closed to force a two-phase slug of liquid from the lower plenum and downcomer past the fuel rods. The resultant two-phase slug reached a peak flow rate of 1.6 l/s during this period. Reflood was initiated at 100 seconds with a brief period of high flow reflood (1.6 l/s) to fill the lower plenum, followed by low flow reflood (0.95 l/s) until the rods rewet. The outlet volumetric flow was positive during reflood and oscillated as reflood water quenched the cladding. Cladding rewet was completed at about 125 seconds.

The inlet volumetric flow for each flow shroud became negative as blowdown was initiated and remained negative during the early stages of blowdown as liquid exited the flow shrouds. Cycling of the hot and cold leg blowdown valves between 5 and 11 seconds resulted in the two-phase slug period with a minimum flow rate of

1.6 l/s, similar to the shroud outlet. Blowdown was completed at 22 seconds by reopening the large cold leg blowdown valve and rapidly completing the depressurization. The shroud inlet volumetric flow increased at this time as the rate of expansion of steam within the flow shrouds increased. The volumetric flow was essentially zero between 50 and 100 seconds in all shrouds. After 100 seconds, the flow oscillated as reflood water entered the flow shrouds.

The response of the four shroud inlet volumetric flows during blowdown was similar for Rods 01, 03, and 04. However, the Rod 02 inlet turbine flowmeter measured a significantly larger flow in Tests TC-1B, TC-1C, and TC-1D. This response has been determined to result from leakage of fluid from the upper plenum into the Rod 02 shroud. Results from Rod 02 will not be considered in determining thermocouple effects during the blowdown portion of each test.

The thermal and mechanical response of Rod 01 (a rod with two internal cladding thermocouples and one fuel thermocouple, but no external cladding thermocouples) is presented in Figure 2 for Test TC-1C. Prior to initiation of blowdown, the fuel and internal cladding temperatures were approximately 1030 K and 810 K, respectively. During the initial second of blowdown, both the fuel and internal cladding temperatures gradually decreased until critical heat flux (CHF) occurred at 0.8 seconds and then rapidly increased until the blowdown two-phase slug flow occurred. A momentary rewetting of the cladding occurred at about 1.5 seconds, reducing the internal cladding temperature to approximately 800 K during Test TC-1C and to 775 K during Test TC-1D. The fuel temperature reached a maximum of approximately 1135 K at 3 seconds. The internal cladding temperature continued to increase until 5 seconds, the time at which the two-phase slug was introduced, causing both the fuel and cladding temperatures to decrease to about 960 K at 10 seconds. Both the fuel and cladding temperatures remained at approximately 960 K until 22 seconds, at which time the core power was intentionally adjusted to raise the cladding temperature to approximately 1000 K prior to initiation of reflood.

The thermal and mechanical response of Rod 03 (a rod with both external and internal cladding thermocouples) is presented in Figure 3. Results are shown for internal fuel and cladding thermocouples, one external cladding thermocouple, and cladding displacement. The internal and external cladding temperatures decreased, as blowdown was initiated, until CHF. Following CHF, external cladding and internal fuel temperatures reached 900 and 1020 K, respectively, before the slug period. During the period of two-phase slug flow (5 to 11 seconds), the cladding temperatures gradually decreased until 11 seconds. Fuel and cladding temperatures increased after 11 seconds and stabilized at about 980 K prior to reflood. Rod axial extension (cladding displacement) corresponded, in general, with the thermocouple response.

General observations are presented on the basis of Rod 03 data. The time of CHF as measured by the external thermocouples on Rod 03 was between 2 and 3 seconds; significantly later than the time of CHF on Rod 01. Apparently, the physical presence of the external thermocouples delayed CHF and influenced the surface heat transfer. Secondly, both internal and external thermocouples responded in a similar fashion during the two-phase slug flow period. The external thermocouples provided a good measurement of the rod thermal response as the high quality vapor flowed past the thermocouples. Test results from Rods 02 and 04 are not shown, but results obtained from Rod 04 are similar to those from Rod 01. The temperatures from Rod 02 were found to be somewhat lower than the temperatures from Rod 03, and this is thought to be due to leakage in the shroud of Rod 02. This aspect of the test is presently being reviewed.

The surface thermocouples also apparently enhance cladding quench during reflood. The reflood phase is characterized by: cladding temperature turnaround (in which the temperature gradually decreased as steam with entrained liquid passed the thermocouples), quench (in which the cladding temperatures rapidly decreased as the quench front passed the thermocouple junctions), and finally, rewet with liquid contacting the cladding surface. For the 4 cm/s reflood rate of the TC-1 tests, no significant difference in the cladding temperature turnaround was measured for any of the fuel rods. The cladding generally cooled at the same rate during the period for rods with and without surface thermocouples. However, the quench time and subsequent rewet time varied

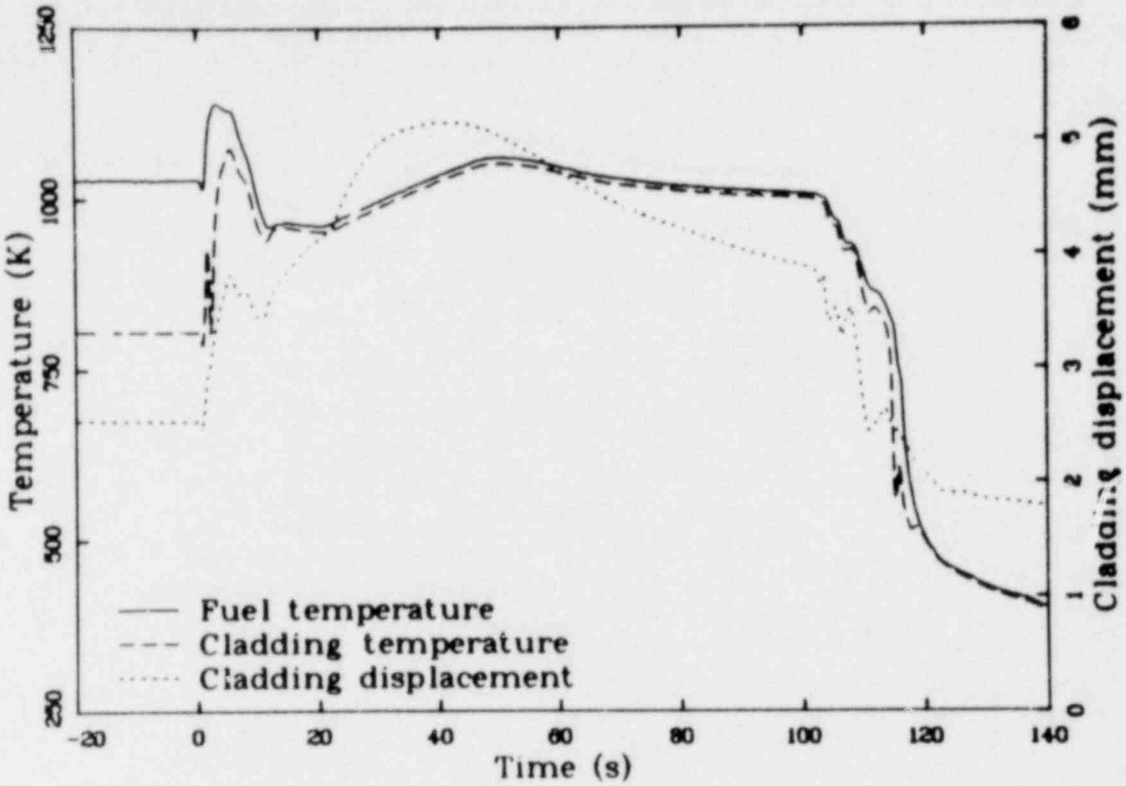


Figure 2. Internal rod temperatures and axial extension (cladding displacement) for Rod 01 during Test TC-1C.

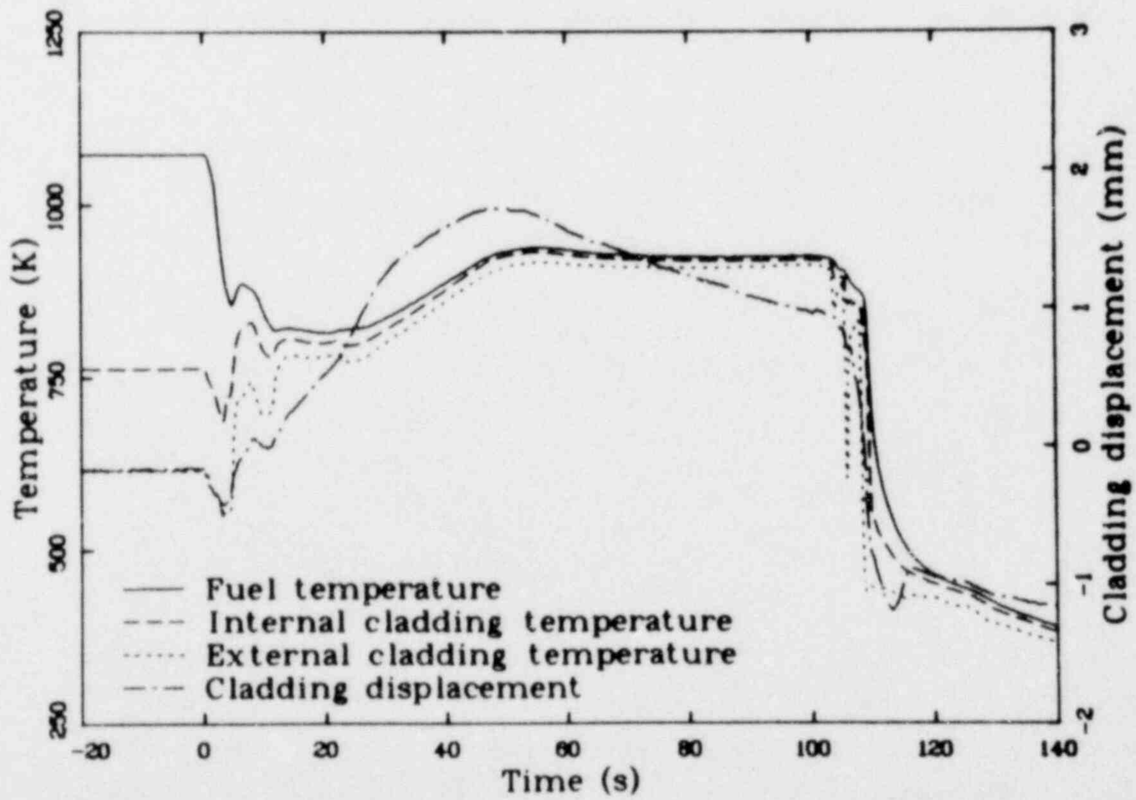


Figure 3. Internal rod and cladding surface temperatures and axial extension (cladding displacement) for Rod 03 during Test TC-1C.

significantly, as fuel rods with surface thermocouples quenched between 3 and 12 seconds before the other rods. Additionally, some external thermocouples momentarily quenched and reheated during reflood prior to the actual rod quench.

An attempt was made in the TC-1 tests to simulate the LOFT L2 test conditions with a two-phase coolant slug forced past the fuel rods early in blowdown. Generally, all the TC-1 thermocouples measured rod cooling during the two-phase slug flow period, but the cladding did not quench as measured during the LOFT L2 tests. The two-phase slug apparently consisted of high quality vapor that would not rapidly quench the TC-1 rods. To properly simulate LOFT L2 test conditions, the liquid content of the two-phase slug should be increased (to a quality of 15 to 30%). A means of attaining these conditions in PBF has been established, and a second test series, TC-2, is planned to investigate thermocouple effects during a blowdown quench.

Program Development and Evaluation

P. E. MacDonald and R. R. Hobbins

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.

Summary of Progress

The destructive examination of fuel rods from the LOFT Lead Rod and the LOC-3 tests have been initiated. The postirradiation examination for Test RIA 1-2 has been completed.

The first of a series of fuel thermal performance tests as a function of fill gas composition and pressure has been conducted in the Halden reactor. Fuel centerline temperatures were measured at fill gas pressures from 0.1 to 5.1 MPa for He, 90% He/10% Xe, and 95% He/5% Xe fill gas compositions in two LWR-type fuel rods in the Instrumented Fuel Assembly 430 (IFA-430) experiment in the Halden reactor in Halden, Norway. The data generally confirm the effects of gas pressure composition on gap conductance as modeled in fuel behavior codes for pressures less than about 1.0 MPa; significant differences occ at higher pressures.

The results of scoping tests with the IFA-430 fission product measurement system to determine the operational limits and optimum parameter setpoints for fission gas and iodine release measurements are described in the following section.

IFA-430 Fission Product Measurement System Scoping Test Results

A. D. Appelhans

The Thermal Fuels Behavior Program is conducting tests to determine the release of fission gases from UO_2 during irradiation. These tests are being performed using instrumented fuel assemblies in the Heavy Boiling Water Reactor (HBWR) in Halden, Norway.

A fission product measurement system (FPMS) has been installed on Instrumented Fuel Assembly 430. This system is being used to obtain on-line data to quantify the release of Xe, Kr, and ^{135}I from typical light water reactor fuel pellets during irradiation.

The IFA-430, a multipurpose assembly designed to study axial gas flow characteristics, fuel thermal behavior, and fission gas release, is composed of four, 1.28-m-long, LWR-type fuel rods containing 10% enriched UO_2 fuel pellets. Two of the rods in the assembly, termed gas flow rods, are used for measuring fission product release. These two rods each have a fuel centerline thermocouple and three axially spaced pressure sensors and are connected to a gas flow system which is used to sweep the fission gases out of the fuel rods and to the FPMS. Quantitative determination of the radioactive fission gases in the sample stream is made with a gamma ray spectrometer consisting of a hyper-pure Ge detector and microcomputer-based data acquisition and reduction system.

Two scoping tests have been performed with the FPMS. These tests have shown that the FPMS can provide quantitative data on the release of Xe, Kr, and ^{135}I. Scoping Test 1 was designed to determine the operating limits and optimum parameter setpoints for measurement during nuclear operation. During these tests, the release of Xe and Kr fission gases was measured. Scoping Test 2 was performed to evaluate the feasibility of measuring the ^{135}I that plates out in the fuel rod and piping system. This measurement is used to estimate the ^{135}I release rate from the fuel during steady state nuclear operation.

Scoping Test 1, Xe and Kr Release. During Scoping Test 1, the release rates of Xe and Kr were measured at full power steady state reactor conditions (∿12 MW) using He sweep gas flow rates through the rod of 1.4, 2.2, and 3.5 l/min. The fuel rod average linear heat rate was ∿25 kW/m during the tests.

The isotopes of Xe and Kr were measured in a continuous flow mode and the release rates are shown in Table 2. Of the isotopes listed in Table 2, ^{135}Xe, ^{138}Xe, ^{85m}Kr, ^{87}Kr, and ^{88}Kr have all been used by other researchers to characterize fission gas release from UO_2, and are therefore comparable with the results of the IFA-430 tests. The direct measurement of the short half-lived isotopes ^{139}Xe, ^{140}Xe, ^{89}Kr, and ^{90}Kr has not been possible in previous experiments and application of these data to overall fission gas release characterization will have to be developed.

To compare the results of the IFA-430 fission gas release tests with the results of other researchers, the release-to-birth (R/B) ratio of the isotopes is used. To obtain an estimate of the R/B ratios, the birth rates for ^{138}Xe and ^{88}Kr were calculated using

$$B_x = Y_x f$$

where

B_x = birth rate of isotope x (numbers of x per second)

Y = cumulative fission yield for isotope x

f = fission rate.

The R/B ratio for ^{88}Kr, at an average rod linear heat rate of ∿25 kW/m, was 4.6×10^{-6} and for ^{138}Xe was 8.9×10^{-7}. The average fuel surface temperature for these measurements was ∿750 K, at which temperature recoil is expected to be the predominant mode of release. Friskney and Turnbull[3,4] reported R/B ratios for 1.2-mm-diameter spheres of 8.6×10^{-4} for ^{88}Kr, and 3.6×10^{-5} for ^{138}Xe at fuel temperatures of ∿1050 K. Turnbull's data also showed that the R/B ratios for these two isotopes are relatively linear for fuel temperatures in the range of 1050 to 1300 K. By performing a least squares fit to Turnbull's data and extrapolating to 750 K, the expected R/B ratios for a small sphere should be

TABLE 2. ISOTOPES AND RELEASE RATES[a] OF Xe AND Kr MEASURED WITH IFA-430 FPMS IN A CONTINUOUS MODE

Isotope	Half Life (s)	Release Rate (atom/s)
$^{135}Xe^c$	3.27×10^4	—
$^{135m}Xe^b$	9.36×10^2	2.4×10^7
^{137}Xe	2.30×10^2	3.8×10^7
^{138}Xe	8.48×10^2	5.6×10^7
^{139}Xe	4.04×10^1	3.3×10^7
$^{140}Xe^c$	1.36×10^1	—
^{85m}Kr	1.61×10^4	4.4×10^7
^{87}Kr	4.58×10^3	6.2×10^7
^{88}Kr	1.02×10^4	1.6×10^8
^{89}Kr	1.90×10^2	3.2×10^7
^{90}Kr	3.23×10^1	3.0×10^6
$^{91}Kr^c$	8.57	—

a. At a rod average linear heat rate of 25 kW/m.

b. Not corrected for neutron capture.

c. Isotope is measureable, but release rate was not calculated for the scoping test.

3.4×10^{-5} for ^{138}Xe and 4.7×10^{-4} for ^{88}Kr. Therefore, the R/B ratio for the fuel pellets was smaller by a factor of ∿50 than the ratio for the 1.2-mm spheres; this may be due to the difference in the surface-to-volume (S/V) ratios. The S/V ratio for a 1.2-mm sphere is 5, whereas for an intact IFA-430 fuel pellet having a radius of 5.4 mm and a length of 12.7 mm, the S/V ratio is 0.16. The R/B ratios normalized by the S/V ratios for fuel pellets and spheres are shown in Table 3. The estimated error in the R/B ratio for the spheres is ± 20%, and for the fuel pellets is ± 50%.

TABLE 3. R/B RATIOS NORMALIZED BY S/V RATIOS FOR 1.2-mm SPHERES AND IFA-430 FUEL PELLETS

Isotope	R/B S/V Sphere	R/B S/V Pellet
^{138}Xe	7×10^{-6}	5×10^{-6}
^{88}Kr	9×10^{-5}	4×10^{-5}

This highly simplified comparison shows that the release rate per unit surface area from the IFA-430 fuel pellets is of the same order of magnitude as for the small spheres.

Of particular interest in the measurement of fission gas release is the determination of the predominant release mechanism. The three mechanisms for release of fission gases from UO_2 are diffusion, knockout, and recoil. Carroll[5,6] has shown that the predominant release mechanism can be determined from the ratio of the release rates of two isotopes which have the same diffusion coefficient. Recoil release is proportional to the surface area and fission rate; thus, the ratio of the release rates for two isotopes is proportional to the ratio of the cumulative yields of the two isotopes. Diffusion release is proportional to the fission yield of the isotopes and inversely proportional to the square root of their decay constants (for short half-lived parents). A preliminary analysis of the IFA-430 data indicated that the Xe release was predominantly due to recoil, whereas the Kr release showed no conclusive trend. The diffusion coefficient[3] of Br, the precursor of Kr, at these temperatures is ∿10^3 times that of I, the Xe precursor; this may be a cause of the apparent difference in the behavior of these two gases.

Scoping Test 2, ^{135}I Release. The objective of Scoping Test 2 was to evaluate the feasibility of measuring the ^{135}I that plates out in the fuel rod and piping system during operation. This measurement, taken after the reactor has been operating at a constant power for a predetermined time and then scrammed, allows the steady state iodine release from the fuel to be estimated.

Measurement of the iodine that has plated out in the system is accomplished by measuring the daughter of ^{135}I, ^{135m}Xe. The production/decay rate of ^{135}I at a constant power reaches 99% of equilibrium value after 6.5 half lives (43 hours), and it is assumed that the release rate also comes to equilibrium during this time period. If all of the ^{135}I released from the fuel plates out in the system, the plateout concentration of ^{135}I will reach equilibrium such that the amount decaying is equal to the amount being deposited. Measurements by Bannister et al.,[7] have shown that for temperature gradients similar to those in the IFA-430 experiment, essentially all of the iodine will be trapped in the piping or fuel rod; this is in agreement with the on-line measurements which show no iodine present in the gas stream at the detector station. During steady state operation, the equilibrium ^{135}I decay rate (from plated out ^{135}I) results in an effective ^{135m}Xe production rate, which, combined with the ^{135m}Xe released from the fuel as a gas, makes up the total ^{135m}Xe measured release rate. Scramming the reactor stops the production of I and Xe from fission, and the rapid cooling of the fuel effectively halts any further diffusion of I and Xe out of the fuel. Following scram, the ^{135m}Xe production comes only from decay of the ^{135}I plated out in the system; thus, measurement of the ^{135m}Xe present after scram provides an indirect measurement of the equilibrium ^{135}I release.

To determine if this measurement was possible, the reactor was scrammed from 98% of full power and the ^{135m}Xe in the sweep gas stream was measured for seven hours following the scram. Figure 4 shows the normalized ^{135m}Xe release rate as a function of time after reactor scram. Since the reactor had not been operating at constant power for a long enough period prior to scram to achieve an equilibrium ^{135}I plateout condition, the data are only qualitative and have been

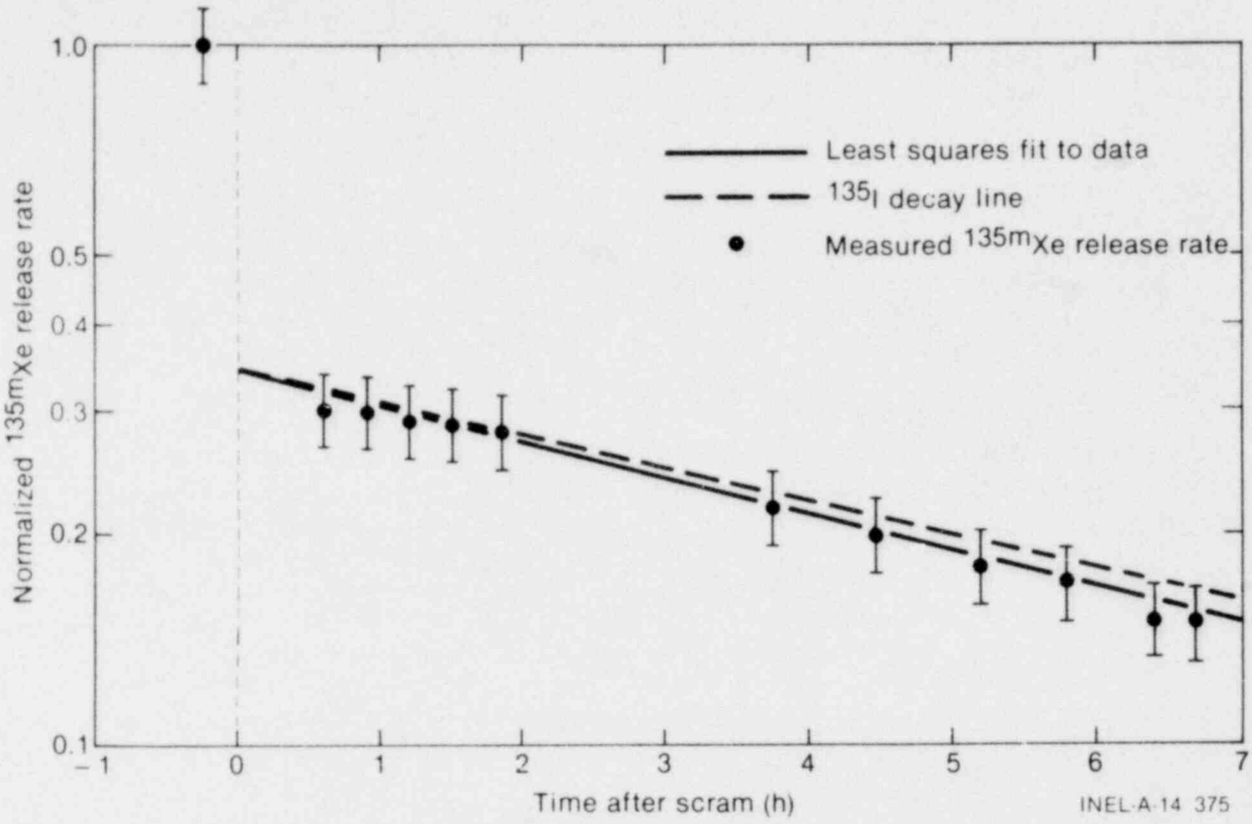


Figure 4. Postscram release of ^{135m}Xe resulting from the decay of ^{135}I plated out during full power operation. The data have been normalized to the steady state ^{135m}Xe release rate just prior to reactor scram. The least squares fit line of the data is compared with the decay line expected for ^{135}I. Extrapolation to time zero shows that $\sim$35% of the ^{135m}Xe measured prior to scram was due to ^{135}I decay, and that $\sim$65% was actually released from the fuel as Xe gas. (Note: data are only qualitative since ^{135}I had not reached equilibrium prior to scram.)

normalized to the ^{135m}Xe release rate at steady state conditions prior to scram. Plotted with the data are a least squares fit line for the data and the calculated decay line for ^{135}I. To determine the percent of the ^{135m}Xe being released from the fuel and the percent due to the decay of ^{135}I during full power operation, the least squares fit line is extrapolated to time zero (scram). For the scoping test data, this shows that 35% of the ^{135m}Xe being measured at full power prior to scram was due to the decay of ^{135}I plated out in the system, and 65% of the ^{135m}Xe was being released from the fuel as Xe gas. Thus, the ^{135}I release rate at full power is $\sim$35% of the measured ^{135m}Xe release rate.

Conclusions. A fission product measurement system has been installed at the Halden reactor to provide data on the release of xenon, krypton, and iodine in the IFA-430 fuel rods. Two scoping tests have been performed, showing that the system can provide data to quantify the release of Xe, Kr, and ^{135}I.

The predominant short-lived isotopes of Xe and Kr are measurable on-line at steady state nuclear operating conditions. The ^{85}Kr and ^{133}Xe isotopes are not measurable on-line, but possible alternate measurement techniques are being considered.

Estimates of the release-to-birth ratio have been made using a simplified model for the birth term. The measured data compare well with data of other researchers when the data are normalized to the surface-to-volume ratio of the fuel.

Measurement of the iodine release rate can be made by measuring the ^{135m}Xe released in the decay of ^{135}I following reactor scram. The measurements made in Scoping Test 2 showed that the technique can be used for the IFA-430 system and that the release rate of ^{135}I is of the same magnitude as ^{135m}Xe and ^{138}Xe. This technique also allows the ^{135m}Xe release rate to be decoupled from the ^{135}I precursor contribution to the measured ^{135m}Xe release rate at power.

CODE DEVELOPMENT AND ANALYSIS PROGRAM

P. North, Manager

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 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.

During this last quarter, program emphasis was on development of advanced codes. The containment analysis program, BEACON/MOD3, was in the process of developmental checkout. A description of the code and results of analysis performed as part of the code checkout are presented. Substantial progress was made toward establishing a capability for the analysis of boiling water reactor LOCA transients in the TRAC code. A set of boiling water reactor (BWR) system component models has been added to the TRAC code to form the basis of TRAC-BDO. A typical BWR plant model was constructed and results of preliminary test calculations indicate the code will execute a BWR calculation. The channel component model and the test calculations are described.

BEACON/MOD3 Development and Assessment Program

C. R. Broadus and J. F. Lime

BEACON/MOD3 is an advanced, best-estimate computer code designed to calculate thermal-hydraulic phenomena associated with nuclear reactor containments. The BEACON code is described and results of an analysis of the Battelle- Frankfurt Test D-3 are presented.

Code Description and Status

The BEACON code is currently under development at the Idaho National Engineering Laboratory for the U.S. Nuclear Regulatory Commission. BEACON is an advanced, best-estimate containment analysis tool designed to analyze a wide range of containment-related problems and phenomena. It is a versatile, user-oriented program which may be used for a wide range of containment system problem applications. The code can be used, for example, to model complete containment systems, to perform isolated or separate effects component studies, or to examine in detail complex nonhomogeneous, nonequilibrium flow behavior.

BEACON utilizes a two-dimensional, two-component, two-phase numerical method which solves the six field equations of the two phases using a coupled, explicit-implicit Eulerian finite-difference computational procedure. The two phases are coupled by interphasic mass, momentum, and energy exchange functions. BEACON capabilities include specified flow boundaries; obstacle cell modeling; mass, momentum, and energy source modeling; and a generalized heat structure model.

An extensive developmental checkout of BEACON/MOD3 is being performed by using BEACON to analyze various containment problems such as the Battelle-Frankfurt test series.

Battelle-Frankfurt Test D-3 Analysis

One of the tests being used to check out BEACON/MOD3 is the Battelle-Frankfurt Test D-3.[8] A schematic of this test setup is shown in Figure 5. This test consisted of essentially three subcompartments connected in series with a steam blowdown into the first compartment (Room 6). The blowdown flow then passed from Room 6 through Room 4 and exhausted into the relatively large volume of Room 9.

To analyze Test D-3 with BEACON[a], Rooms 4 and 6 were modeled as two-dimensional mesh regions and the intercompartment connections were modeled as one-dimensional regions. The large volume of Room 9 was treated as a lumped-parameter region. The blowdown flow was input as a source in Room 6 and the exposed walls and equipment were treated as film-covered, heat-conducting structures.

Figures 6 through 8 show the short-term results of the analysis. At this time, the flow distribution is still being established and a pressure wave is

a. BEACON/MOD3F is the version of the code used in the analysis.

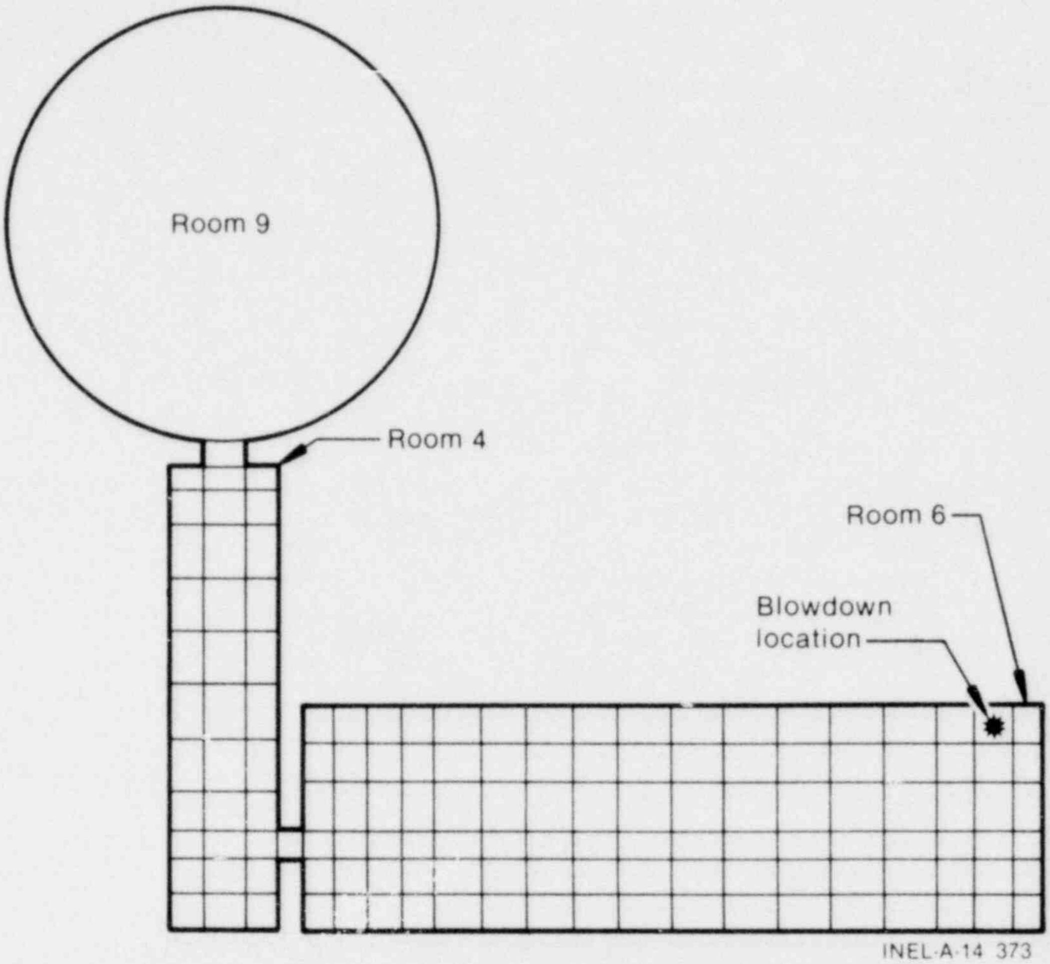


Figure 5. Schematic of Battelle-Frankfurt Test D-3.

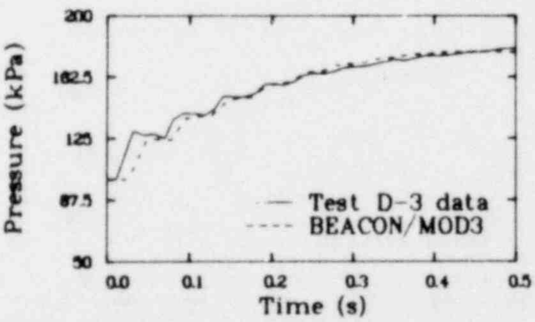


Figure 6. Comparison of calculated and measured pressure response in Room 6 of Battelle-Frankfurt Test D-3.

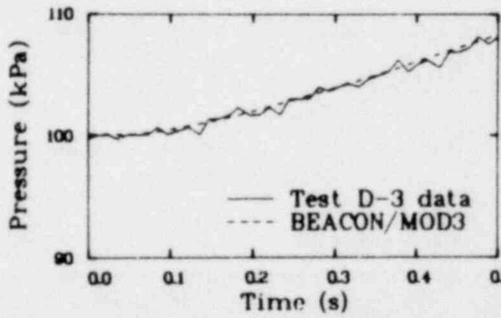


Figure 8. Comparison of calculated and measured pressure response in Room 9 of Battelle-Frankfurt Test D-3.

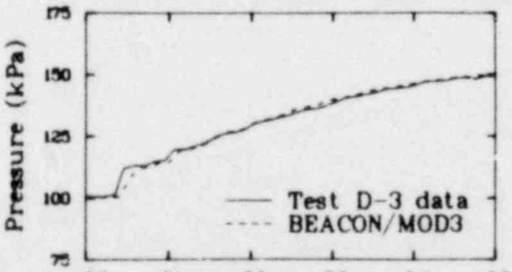


Figure 7. Comparison of calculated and measured pressure response in Room 4 of Battelle-Frankfurt Test D-3.

traveling through the room away from the break. The effects of this wave can be seen in Figure 6, which shows a comparison between the BEACON-calculated pressure transient and the measured pressure transient in Room 6. The measuring point is located at the opposite end of the room from the blowdown location. A delay in the pressure response is therefore experienced, corresponding to the time required for the pressure wave to travel the length of the room. After this time, the pressure increases in steps as the wave traverses the room and eventually dies out.

The frequency of the BEACON-calculated pressure oscillations is the same as and in phase with the measured data. The calculated pressure peaks are not as pronounced as the data due to a numerical diffusion of the pressure wave. Figures 7 and 8 compare the short-term pressure transients in Rooms 4 and 9 with the BEACON calculations. Again, the BEACON calculations compare well with the data.

Development of the TRAC Code for BWR Analysis

F. Aguilar

Substantial progress has been made toward creating a basic capability in the TRAC code for the analysis of LOCA transients in BWR systems. This basic BWR capability, TRAC-BDO, is being developed from TRAC-P1A.[9] A set of models describing certain BWR components (jet pump, separator, and dryer) and important hydrodynamic phenomena (level swell and countercurrent flow) has been supplied by the General Electric Company and implemented into a developmental version of the TRAC-P1A code. An important new TRAC component called CHAN (for channel) has been developed to enable realistic modeling of BWR core heat transfer and coolant flow.

A typical BWR/6 plant has been modeled with the proposed BWR modeling scheme to confirm that the TRAC solution algorithm will converge. The standard TRAC code VESSEL component is the basic element of the BWR model. However, an unusually large number of VESSEL connections must be made in the BWR model, causing concern about the numerical stability of the BWR scheme. The new CHAN component and the BWR/6 test calculation are described in the following sections.

The CHAN (Channel) Component

M. M. Giles and J. W. Spore

The CHAN component is an extension of the standard TRAC-code PIPE component and simulates a BWR rod bundle and canister assembly. CHAN components will be used to represent all fuel channels of a BWR and are connected across the usual core region of a conventional VESSEL component. The connections to the vessel are made with standard VESSEL sources. Three-dimensional core flow in the bypass region between channels is calculated with the usual VESSEL hydrodynamics. Convective heat transfer between channels and bypass coolant is also modeled by the CHAN component.

A typical BWR/6 model might contain six CHAN components, as shown in Figure 9. Each CHAN component represents a large number of actual fuel rod bundles, and the flow through each CHAN is assumed to be one-dimensional. Within each CHAN component, radiation and conduction heat transfer are calculated for a number of rod groups specified by the user. The core schematic in Figure 9 is only intended to show how components are connected and how the bypass region is partitioned in a three-dimensional sense. The figure does not accurately represent the flow areas of the channel and bypass regions, since the bypass region in a BWR is only 13% of the total core volume.

The total heat transfer within a CHAN component is determined by calculating the heat transfer from an average rod bundle and multiplying by the number of bundles lumped together in the CHAN component. The heat transfer modes provided by the CHAN component are

1. Conduction heat transfer in fuel rods and channel wall
2. Convective heat transfer from fuel rods and channel wall during blowdown and reflood
3. Radiation heat transfer from surface to surface, surface to steam, and surface to droplets.

Existing TRAC-P1A models were used for the conductive and convective heat transfer modes. For the radiation heat transfer mode, a diffuse gray body model with [illegible] and droplet participation was developed. The radiation heat transfer model is similar to those in References 10 and 11. The major differences between the TRAC-BDO radiation model and the NORCOOL[10] and MOXY-SCORE[11] models are the calculation of

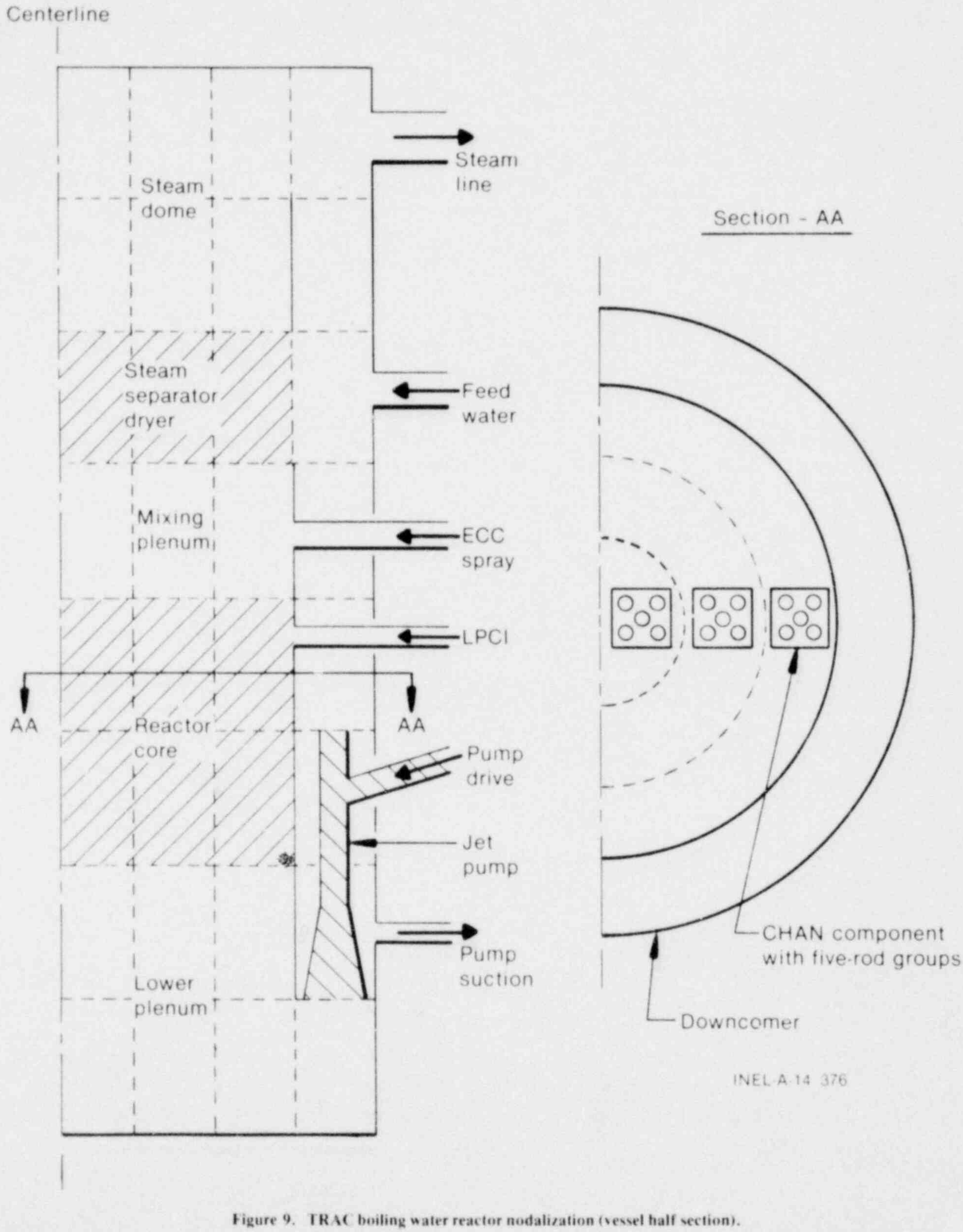


Figure 9. TRAC boiling water reactor nodalization (vessel half section).

steam emissivity and absorptance and the determination of the radiant energy absorbed by steam and droplets. A falling-film and bottom-flood quench front propagation model for fuel rods and the inner channel wall is presently being implemented into the CHAN component.

BWR/6 Test Case

Y. S. Chen and R. W. Shumway

Test calculations have been made with the TRAC-BDO code to determine its capability to simulate BWR operation. The BWR test calculations have two goals. One goal is to compare TRAC-BDO performance with a one-dimensional, homogeneous BWR calculation performed with the RELAP4 code. A more immediate goal, however, is to confirm that the large number of source terms required to model a BWR core with CHAN components and to model core sprays, recirculation lines, etc., does not cause convergence problems. This numerical assessment effort has proceeded in parallel with the CHAN component development. Consequently, the initial BWR calculations have been performed with a core model built with PIPE rather than CHAN conponents. The hydrodynamics of the core model are the same, regardless of whether PIPE or CHAN components are used.

In the vessel noding schematic shown in Figure 9, the core is radially divided into high, average, and low power zones. Outside the low power zone is the downcomer region. The core PIPES have no interior fuel rods, but power is input to the PIPE walls. The present BWR model contains most of the essential features required for LOCA analysis of a typical plant.

The fluid conditions for all TRAC components were estimated on the basis of a 100% power condition of the reactor. The model was allowed to run in a transient mode for 9 seconds with boundary conditions that represent normal plant operation. Steady state was achieved, and the final results agree well with plant operational parameters such as jet pump flow, lower plenum subcooling, core flow and pressure drop, separator pressure drop, steam dome pressure, and steam exit flow rate. No numerical stability problems were observed. The running time was reasonable (2 ms per cell per time step) when compared with that of a typical PWR TRAC run.

A second demonstration run was conducted with the components filled with low pressure steam. A break was initiated in the pump inlet pipe and the ECC systems were then activated. The run was executed successfully and without major difficulty. These preliminary test cases indicate that TRAC-P1A, with incorporation of updates by the General Electric Company and EG&G Idaho, Inc., can be used to execute a BWR calculation.

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 NRC's technical advisor 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 (codes) of their choice. This program is a cooperative effort among the NRC, U.S. reactor vendors, and the international nuclear community. Technical assistance to the NRC continues to be performed in the audit of pressurized water reactor vendor safety calculations.

The following sections summarize results from audit activities and a code assessment task.

NRC Audit Calculations

An additional series of independent calculations using an experimental version of RELAP4 was performed to extend the NRC review of the Three Mile Island (TMI) accident and Westinghouse Electric Corporation, Combustion Engineering, Inc., and Babcock & Wilcox Company small break safety analyses of PWRs. At the direction of the NRC, postulated accidents with power operated relief valves (PORVs) stuck in the open position, and small cold leg and hot leg breaks were simulated. The system nodalizations and user input options used in the RELAP4 computer models were based on guidelines developed in the assessment of RELAP4/MOD6.

TMI Sensitivity Calculations

C. B. Davis

The accident of March 28, 1979, at the Three Mile Island Unit 2 nuclear steam supply system was simulated using an experimental version[a] of the RELAP4 computer code. The purpose of this calculation was to provide a basis for a parametric study of the TMI accident and associated operator actions. The parametric study consisted of four sensitivity calculations that were used to investigate the potential effects of selected sets of operator actions on an accident. The sensitivity of the calculated results to the high pressure injection (HPI) flow, closure of the pressurizer relief valve, and the trips of the reactor coolant pumps at two different times was investigated.

Base Case Calculation. The calculation simulating the TMI accident agreed reasonably well with available measurements from the accident. Figure 10 shows the calculated and measured system pressure in the primary coolant system. The time of the feedwater trip corresponds to a time of zero seconds on Figure 10. The pressure, which was initially about 14.8 MPa, increased rapidly after the feedwater trip. The pressurizer relief valve opened (and stuck open) when the pressure exceeded 15.65 MPa. The reactor was scrammed at 6 seconds in the calculation when the pressure exceeded 16.34 MPa. The pressure then decreased until 250 seconds, at which time the mixture level reached the open relief valve in the top of the pressurizer. The volumetric flow out of the system decreased when liquid exited through the relief valve and the pressure increased until 498 seconds, at which time auxiliary feedwater began to flow into the steam generators. The primary system then depressurized until 950 seconds, at which time the primary coolant had cooled nearly to the steam generator secondary temperature. The primary pressure followed the secondary pressure for the remainder of the calculation. The agreement of the calculations with the measurements

a. RELAP4/MOD7, Version 87, Idaho National Engineering Laboratory Configuration Control Number C0010007.

(Figure 10) was similarly good for most of the available measurements. Thus, the calculation was judged adequate to serve as a base case for conducting sensitivity studies to evaluate alternate operator actions.

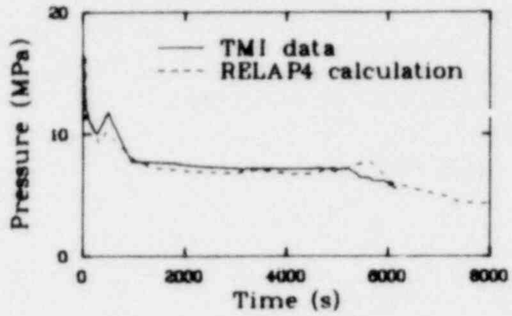


Figure 10. Comparison of calculated and measured primary coolant pressure during the TMI accident.

Effect of Throttling HPI. The results of this calculation show that the system became liquid solid at about 1400 seconds after the feedwater trip when the HPI was not throttled. A heatup of the reactor core was not calculated. This calculation indicates that if the HPI had not been throttled during the TMI accident, core heatup and damage would have been avoided.

Effect of Time of Closing a Stuck Open Power Operated Relief Valve. The results of this calculation indicate that if the PORV was closed at 1500 seconds, the reactor coolant pumps would not have to be tripped at 4380 seconds due to a large volume of voids in the primary system as occurred in the TMI accident. The operation of the pumps assures effective core cooling and heat removal through the steam generators. This calculation indicates that an early closure of the PORV would have prevented the core heatup and damage that occurred in the TMI accident.

Effect of Stopping One Reactor Coolant Pump in Each Loop. These calculations indicate that core heatup would have been mitigated, and possibly eliminated, if one pump in each loop had remained operating at 4380 seconds after the loss of feedwater flow, rather than tripping both pumps in one loop.

Effect of Tripping the Reactor Coolant Pumps at Reactor Scram. The results of this calculation indicate that uncovering of the core would have occurred if all four reactor coolant pumps were tripped at scram, but the heatup occurred later than in the base case calculation. Thus, the core heatup was mitigated when the reactor coolant pumps were tripped off at reactor scram.

Westinghouse Electric Corporation Audit Calculations

C. D. Fletcher

A series of fifteen calculations has been performed on a Westinghouse Electric Corporation PWR using an experimental version[a] of RELAP4. The study was completed with calculations using hot leg circular break diameters of 0.1270, 0.1016, and 0.0762 m and cold leg circular break diameters of 0.1270 and 0.0762 m.

The results of the calculations for small diameter hot leg breaks indicate that uncovering of the core did not occur for any of the breaks investigated. No core uncovering was calculated for the cold leg break of 0.0762 m. The core uncovering calculated for the 0.1270-m-diameter cold leg break resulted in a less severe core heatup than occurred for the 0.1016-m cold leg break previously reported.[12] Therefore, the results of this analysis indicate that the worst small break size and location in a Westinghouse PWR would be a 0.1016-m-diameter break in the cold leg.

Combustion Engineering, Inc., Audit Calculations

C. B. Davis

Postulated accidents that might be initiated by power operated relief valves being stuck open in the pressurizer of a Combustion Engineering, Inc., pressurized water reactor were investigated using an experimental version[a] of RELAP4. Two calculations were performed to analyze accidents initiated by either one or two PORVs stuck open in the pressurizer.

In the calculation for the single PORV, the relief valve was assumed to stick open at zero seconds. The pressure dropped rapidly in response

a. RELAP4/MOD7, Version 92, Idaho National Engineering Laboratory Configuration Control Number H007184B.

to the stuck open relief valve and, at about 1200 seconds, stabilized above the steam generator secondary pressure. The high pressure injection flow prevented the mixture level from dropping below the hot leg; therefore, core uncovering or heatup was not calculated.

The overall calculated system response for the study with two PORVs stuck open was similar to the response obtained with one PORV stuck open. However, the timing of events occurred somewhat earlier in the transient for the larger break.

Babcock & Wilcox Company Audit Calculations

C. A. Dobbe

Postulated accidents with break areas of 0.0065 and 0.00093 m^2 in the cold leg pipe of a Babcock & Wilcox Company PWR were simulated.

The analysis of the 0.0065-m^2 break showed that partial core uncovering occurred. However, the initiation of accumulator flow refilled the core prior to any significant cladding surface temperature increase over the primary system saturation temperature.

The calculation with a 0.00093-m^2 break indicated that a potential for core uncovering existed. The calculation was terminated with the primary system pressure at 8000 kPa and the reactor vessel mixture level 1 m above the top of the core and decreasing at a rate of approximately 0.3 m/h. Therefore, in about 3-1/2 hours of transient time, the core may uncover.

These calculations may result in a somewhat more severe condition than would occur with present administrative controls. Present controls require maintenance of the secondary side mixture level at 95% of the normal operating level instead of at 50% of the normal operating level. This additional heat sink may be sufficient to initiate and sustain natural circulation, which would prevent a core heatup.

Independent Assessment of the Transient Fuel Rod Analysis Code FRAP-T5

E. T. Laats, G. B. Peeler, N. L. Hampton, and R. Chambers

The fifth version of the transient fuel rod analysis program, FRAP-T5,[13] has been independently assessed for the Nuclear Regulatory Commission. The primary objectives of this assessment were to demonstrate the capabilities of a best-estimate model and to provide guidance for improved model development.

FRAP-T5 calculations were compared with in-pile measurements and postirradiation examination data. Overall, FRAP-T5 exhibited better calculational accuracy than the previously assessed code, FRAP-T4.[14] Significant results from this study are summarized as follows.

1. For PWR system conditions, the onset of critical heat flux (CHF) was most accurately calculated when the LOFT or CE-1 CHF correlation was used. For BWR low coolant flow system conditions, FRAP-T5 did not acceptably calculate the onset of CHF and should not be used for BWR low flow system conditions.

2. During a reactor shutdown event, the fuel centerline temperature was overpredicted at the initiation of the event, and the rate of centerline temperature decrease during the shutdown was overestimated, but the equilibrium centerline temperature following shutdown was well characterized.

3. During a reactivity initiated accident, the fuel centerline peak temperature was well predicted. The rate of rod temperature decrease was undercalculated; however, the measured conditions (but not the time) at the termination of film boiling were accurately calculated. Use of the FRACAS-I/effective fuel conductivity/pellet relocation option resulted in the most accurate calculation during this transient.

4. The calculated centerline temperature at the onset of a LOCA was within measurement uncertainty. The rate of centerline temperature decrease during the blowdown was overcalculated; however, the calculated rate of cladding surface temperature increase was close to the observed rate.

5. The temperature and pressure at cladding burst were very accurately calculated at low (α-phase) temperatures. At higher temperatures corresponding to the $\alpha+\beta$ and β phases, the cladding burst

temperature and pressure were overestimated due to the strength of the cladding being overestimated.

6. The most realistic FRAP-T5 results, in terms of accuracy and computation time, were obtained using the following options: (a) the FRACAS-I deformation model and (b) fuel pellet relocation with the FRAPCON-1 effective conductivity model.

ENGINEERING SUPPORT PROJECTS

R. D. Wesley, Manager

Engineering Support Projects includes the 3-D Experiment Project and advanced instrumentation development. The 3-D Experiment Project contributes technology and instrumentation to a multinational (U.S., Japan, and Germany) experimental program that investigates two- and three-dimensional phenomena in simulated pressurized water reactor loss-of-coolant and reflood recovery experiments. Advanced instrumentation efforts directly support all water reactor research activities through the development of specialized measurement devices and indirectly support analytical efforts by allowing data to be gained in areas previously unmeasurable.

3-D Experiment Project

R. E. Rice, Manager

The objectives of the project are the experimental investigation of the refill and reflood phases of a postulated loss-of-coolant accident and the development and assessment of computer codes suitable for describing such behavior. EG&G Idaho, Inc., is providing flow instrumentation for German and Japanese experiments and design and analysis support to the NRC.

Project focus this quarter has been directed toward completion of instrument projects for the Cylindrical Core Test Facility (CCTF) located in Japan. Instruments delivered over the past year have now been made operational and have provided data from several of the CCTF experiments. The instruments developed by EG&G Idaho, Inc., that are used in the CCTF include: 15 liquid level detector systems (for use in the CCTF downcomer, core, lower plenum, and upper plenum), 8 spool pieces (used in the primary loop piping), and 4 drag disks (used in the downcomer).

Advanced Instrumentation

J. V. Anderson, Manager

Development efforts completed include high count rate electronics and initial software for a tomographic densitometer for use in the LOFT test support facility, a local ultrasonic densitometer, and a heated/unheated thermocouple liquid level probe for use in the Thermal Fuels Behavior Program. Proof-of-principle prototype tests that were performed on a nine-beam tomographic densitometer and associated reconstructive algorithms (as provided under subcontract with the University of Utah) have shown the viability of the tomographic technique for imaging two-phase fluid flow; results are presented in the following sections.

Application of Reconstructive Tomography for Imaging Two-Phase Flow

J. R. Fincke

The viability of imaging two-phase flows using the technique of algebraic reconstructive tomography (ART) has been demonstrated by reconstruction of density measurements of Plexiglas test objects (phantoms). The technique is intended as a reference for the calibration of other two-phase density measuring devices as well as to provide information needed to characterize the basic nature of steady state two-phase flow. The details of a nine-beam prototype system on three-inch pipe and reconstructions from actual data on Plexiglas phantoms are presented.

The investigation considered application of ART to quantitative measurements of two-phase density distributions that might occur in a reactor coolant pipe under loss-of-coolant conditions. The hardware developed to accomplish this is intended for use on steady state experiments simulating the types of flows expected. Due to the dynamic nature of the flow field and the finite scan times dictated by the hardware, the results obtained will represent a time-averaged density distribution. The scan times are therefore chosen to be long enough to accurately represent the true time-averaged density.

The tomographic densitometer is similar in theory and operation to standard densitometers. However, several features make this density measuring system unique. The nine-beam system has the ability to be rotated about the pipe center so that an infinite number of views can be obtained. The radiation detector is operated in the pulse mode and, through the use of special electronics, the system may be operated at very high count rates (in excess of 1×10^6 counts per second to obtain low statistical error). The electronics is stablized against drift by a photopeak locking technique.

The radiation detector is a commercial sodium iodide crystal photomultiplier tube detector operated in the pulse mode. This commercial detector, a Hardshaw 4S4 with a 2.54 x 2.54-cm crystal, is coupled to an electronic package designed and built by EG&G Idaho, Inc.

The source used was a 0.5 Ci americium source with 60 keV gamma radiation as the principal emission energy. The source active area exposed to the detectors is 0.635 x 1.54 cm and is collimated into a fan beam, the total angle of which is 58 degrees; the nine detectors have individual collimators. The entire source, collimator, and detector array rotates about the pipe center on two ball bearings. Future modifications will include gearing and a stepping motor to provide computer-controlled scanning.

Results of Densitometer Measurements

J. R. Fincke

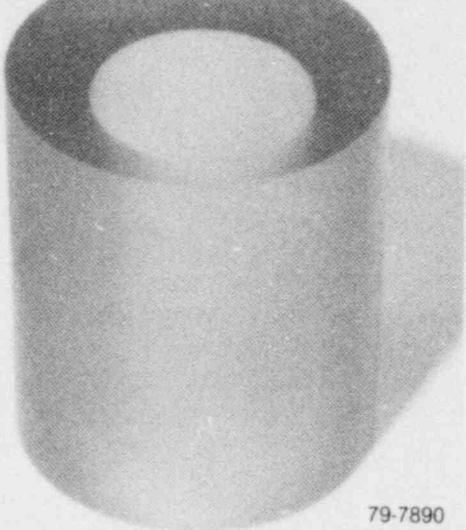
Plexiglas phantoms (test objects which simulate two-phase flow density distributions) were used to test the capabilities of the tomographic densitometer for measuring density. Figure 11 shows two phantoms; one a photograph of an eccentric bubble Plexiglas phantom, and the other, a stratified Plexiglas phantom, along with the density reconstructions of each on 12 x 12 grids using 21 views over 180 degrees. The reconstructions were obtained using the ART algorithm. The distortions are due to the finite size of the individual beams.

Figure 12 is a plot of the cross-sectional average density from the reconstruction from actual data versus the known phantom average density for eight phantoms that were tested. The solid line in the figure is the line of perfect agreement. The root mean square (RMS) average error is defined by

$$\mathrm{RMS} = \left[\frac{1}{M}\sum_{i=1}^{M}(\bar{\rho}_p - \bar{\rho}_i)^2\right]^{1/2}$$

where $\bar{\rho}_p$ is the phantom average density and $\bar{\rho}_i$ is the average density of the reconstruction. The RMS error is 0.0185 for the reconstructions from actual data.

The results of performing reconstructions of phantoms indicate that even with limited numbers of data values, typically less than 10 detectors and 15 views with a 12 x 12 grid, the technique can provide useful two-phase flow data. Estimated RMS average density accuracy is $\pm 2\%$ of full scale. The accuracy of the density distribution obtained depends on flow regime. The technique is a significant advancement over current techniques in that it provides model-independent phase distribution information.



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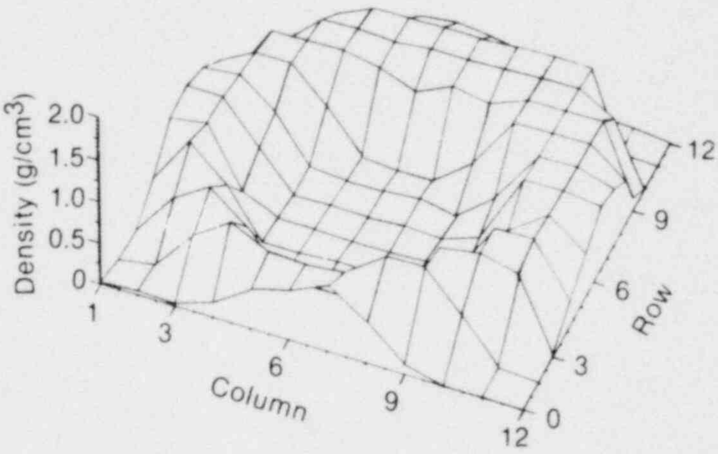

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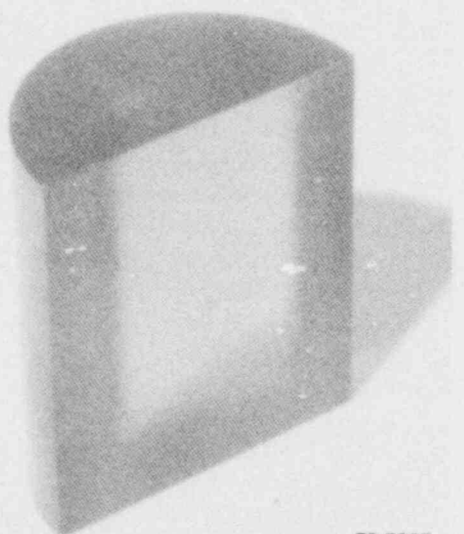

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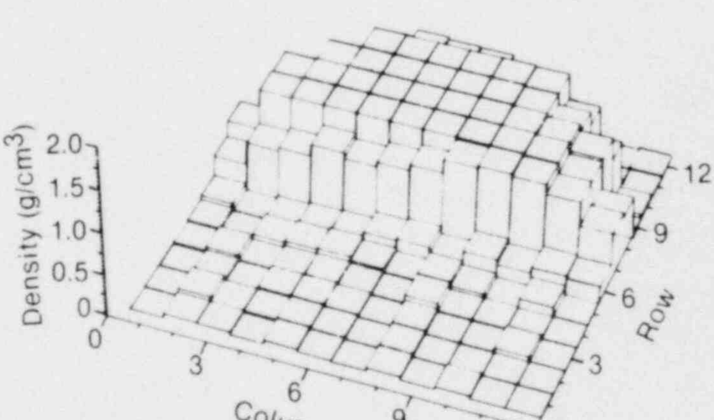

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Figure 11. Photograph (top left) of eccentric Plexiglass bubble phantom and reconstruction (top right) from densitometer measurement data and ART algorithm. Stratified Plexiglass phantom is shown at bottom left, and its density reconstruction from measurement data is shown at bottom right.

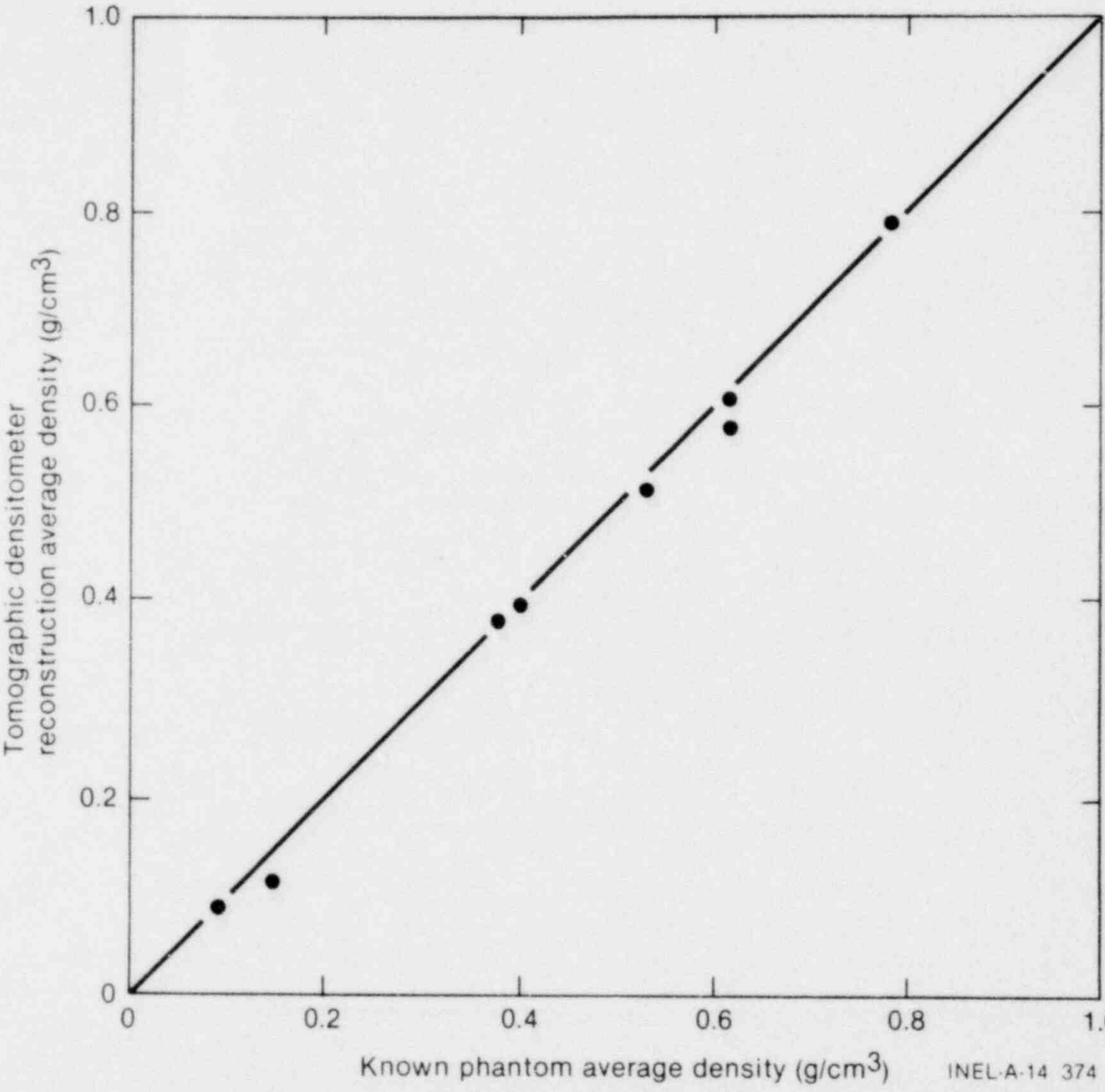

Figure 12. Comparison of tomographic densitometer reconstruction average density with known phantom average density for eight phantoms tested.

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