NUREG/CR-1400(2)
EGG-2048
Distribution Categories: R2, R3, and R4

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

L. J. Ybarrondo, Director
Water Reactor Research

Published July 1980

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555
Under DOE Contract No. DE-AC07-76IDO1570
FIN No. A6038

8008270260

ABSTRACT

Water reactor research performed by EG&G Idaho, Inc., during April through June 1980 is reported. The Semiscale Program conducted four small break loss-of-coolant tests and two station blackout tests that simulated loss of electrical power to the system. The Loss-of-Fluid Test (LOFT) Experimental Program conducted the first nuclear experiment in the Anticipated Transient Experiment Series. The Thermal Fuels Behavior Program completed a reactivity initiated accident test in the Power Burst Facility reactor, postirradiation examinations for loss-of-coolant accident (LOCA) Tests LOC-3 and LOC-5, a scoping test with a fission product measuring system in the Halden reactor in Norway, and the second in a series of internal fuel rod fill gas composition tests. The Code Development and Analysis Program progressed in the development of an advanced BWR thermal-hydraulics computer code (TRAC-B) and a fuel rod analysis subcode for calculating the shape of burst fuel rod cladding. The Code Assessment and Applications Program assessed the predictive capability of a fuel rod analysis computer code (FRAP-T5) to simulate fuel rod behavior under sinusoidal power variations, and performed a study of the potential responses of the Westinghouse Zion I PWR to a loss of offsite power. The 2D/3D Program contributed technology and advanced instrumentation to the multinational (U.S., Japan, and Germany) experimental program that investigates two- and three-dimensional phenomena in simulated PWR loss-of-coolant reflood tests.

FOREWORD

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

The Semiscale Program consists of a continuing series of small-scale, nonnuclear, thermal-hydraulic experiments having as their primary purpose the generation of experimental 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 relative capabilities, validity, and range of applicability of NRC-developed codes.

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

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

ANCR-1262 (April-June 1975)
ANCR-1296 (July-September 1975)
ANCR-NUREG-1301 (October-December 1975)
ANCR-NUREG-1315 (January-March 1976)
TREE-NUREG-1004 (April-June 1976)
TREE-NUREG-1017 (July-September 1976)
TREE-NUREG-1070 (October-December 1976)
TREE-NUREG-1128 (January-March 1977)
TREE-NUREG-1147 (April-June 1977)
TREE-NUREG-1188 (July-September 1977)
TREE-NUREG-1205 (October-December 1977)
TREE-NUREG-1218 (January-March 1978)
TREE-1219 (April-June 1978)
TREE-1294 (July-September 1978)
TREE-1298 (October-December 1978)
TREE-1299 (January-March 1979)
TREE-1300 (April-June 1979)
EGG-2003 (July-September 1979)
EGG-2012 (October-December 1979)
EGG-2031 (January-March 1980)

SUMMARY

The Semiscale Program conducted four small break tests and two station blackout tests during this quarter, and continued analysis of these tests and previous small break tests. The four small break tests consisted of three hot leg breaks (Tests S-SB-P3, S-SB-P4, and S-SB-P6), which were conducted to determine the influence of pump operation on Semiscale thermal hydraulic behavior during a small hot leg break loss-of-coolant accident (LOCA), and a repeat of Test S-07-10. The two station blackout tests were conducted to provide data for future test planning. The conversion to a new Semiscale system configuration (the Mod-2A system) was begun. The modified system will include a full-length intact loop steam generator, improved vessel insulation, piping heat tracing, replacement of the electrical core simulator, and improved instrumentation. Three previously conducted tests (Test S-SB-P1, S-SB-P2, and S-SB-P7) were evaluated to determine the influence of pump operation on Semiscale thermal-hydraulic behavior during a small, cold leg break LOCA. The results indicate that pump operation affects Semiscale thermal-hydraulic behavior during small break LOCAs, resulting in differences in primary system mass inventory and system mass distribution.

The LOFT Experimental Program conducted Experiment L6-5, the first experiment in the Anticipated Transient Experiment Series. It was designed to simulate a complete loss-of-feedwater to all steam generators in a commercial pressurized water reactor with U-tube steam generators. Data from Experiment L6-5 will contribute to the development of a data base that will be used to develop and assess analytical models that are used for licensing commercial PWRs and to plan more complex experiments in the LOFT Facility. The specific objective of Experiment L6-5 was to determine the response of the LOFT system to a secondary-system-initiated transient, in which all feedwater flow to the steam generator was stopped. The experiment objective was achieved.

The Thermal Fuels Behavior Program completed (a) a reactivity initiated accident test (Test RIA 1-4) in the Power Burst Facility (PBF), (b) the postirradiation examinations for loss-of-coolant accident Tests LOC-3 and LOC-5, previously performed in the PBF; (c) a scoping test with the Instrumented Fuel Assembly-430 (IFA-430) fission product measuring system in the Halden Reactor in Norway; (d) analysis of boiling transition, quench, and rewet phenomena during high pressure power-cooling-mismatch conditions with a variety of nuclear fuel rods tested in the PBF; and (e) analysis of the transient freezing of molten debris observed during a severe fuel failure experiment (Test RIA-ST-4) previously conducted in the Power Burst Facility. Test RIA 1-4 was the first reactivity initiated accident test with a fuel rod bundle (nine rods). It was performed at boiling water reactor hot-startup conditions, and will provide data to evaluate the present NRC licensing criteria for an RIA event in a light water reactor. The IFA-430 Scoping Test was performed to evaluate the capability for measuring the release rate of the long-lived ^{131}I isotope by monitoring its daughter product, ^{131}Xe, following reactor shutdown.

The Code Development and Analysis Program has made progress in the development of TRAC-BD1, a computer program for analysis of boiling water reactor systems. Results of calculations using a radiation heat transfer model show good agreement with test data from the Swedish Göta test loop. Improvements have been made in the development of the BALOON subcode, which is used in the calculation of transient fuel rod behavior to determine fuel rod shape at the point of failure.

The Code Assessment and Applications Program assessed the predictive capability of the FRAP-T5 code to simulate fuel rod behavior under sinusoidal power variations. Test predictions of two small break events in the General Electric Two-Loop Test Apparatus were completed using the RELAP4/MOD6 code. The effects of two entrainment models, relative to calculated mass inventory and heater rod temperature response, were determined and compared with experimental data from FLECHT-SEASET boil-off and reflood tests. The Westinghouse Zion I PWR was studied for potential responses to a loss of offsite power. The study developed an analysis technique which was then tested with selected event chains having relatively high probabilities of occurrence.

The 2D/3D Program continued its progress in contributing technology and instrumentation to the multinational (U.S., Japan, and Germany) experimental program that investigates two- and three-dimensional phenomena in simulated PWR loss-of-coolant reflood tests. Liquid level detectors and instrumented piping were delivered to Japan for use in the Slab Core Test Facility.

Advanced instrumentation completed (a) studies on gamma scattering, (b) a survey of potential acoustic water reactor applications, (c) a compilation of liquid level detection techniques, and (d) a review of commercially available low flow velocity measuring systems. Developmental activities completed include the heated thermocouple liquid level probe and the local ultrasonic densitometer.

CONTENTS

FIGURES

TABLES

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

I. SEMISCALE PROGRAM

L. P. Leach, 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.

1. PROGRAM STATUS

Program emphasis was directed at providing data and performing analyses to support the Nuclear Regulatory Commission (NRC) in assessment and improvement of computer models for small break LOCAs and station blackouts (loss of electrical power to the facility). Tests S-SB-P1, S-SB-P2, and S-SB-P7 were conducted earlier this year to evaluate the effect of pump operation on overall Semiscale Mod-3 system thermal-hydraulic response during a small break LOCA. Results from analyses of these three tests are reported in Section 2. Data from five small break tests [1,2,3] conducted in the Semiscale Mod-3 system have been recently published. Results from three of these small break tests were reported in earlier quarterly reports.[4,5]

Tests S-SB-P3, S-SB-P4, and S-SB-P6 were performed during this quarter. These tests simulated 2.5% breaks in the side of a PWR hot leg pipe with (a) the pumps tripped at the time of initiation of core power decay, (b) with the pumps running throughout the transient, and (c) with the pumps running until a high system void fraction was achieved. Two station blackout tests and a repeat of Test S-07-10 were also performed in this quarter. The results from the hot leg break and station blackout experiments are being analyzed and will be reported at a later date.

The completion of the two station blackout tests and a repeat of Test S-07-10 signifies the end of the planned tests for the Semiscale Mod-3 system and the beginning of a planned conversion to a new Semiscale system configuration (the Mod-2A system) that more closely models a PWR. The system modifications will include a new, full-length steam generator, improved vessel insulation, piping heat tracing, replacement of the electrical core simulator, and improved instrumentation.

2. PRELIMINARY ANALYSIS OF THE EFFECT OF PUMP OPERATION ON COLD LEG SMALL BREAK BEHAVIOR

S. E. Dingman

Tests S-SB-P1, S-SB-P2, and S-SB-P7 were conducted to evaluate the effect of pump operation on overall Semiscale system thermal-hydraulic behavior during a small, cold leg break LOCA. Of particular interest was the effect of pump operation on primary system mass inventory and system mass distribution. These objectives were met by varying the time for tripping the primary coolant pumps in the three tests. The pumps were tripped at a system pressure of 12.4 MPa in Test S-SB-P1 and at 3.28 MPa in Test S-SB-P7. Both of these tests were terminated at a system pressure of 1.42 MPa. The pumps were maintained at their initial speeds throughout Test S-SB-P2, which was terminated at a system pressure of 4.14 MPa. The initial conditions for Tests S-SB-P1, S-SB-P2, and S-SB-P7 were, as nearly as possible, identical to those of a prior test, Test S-SB-2.[3] These conditions closely approximate those expected in a full-sized commercial PWR operating at full load conditions. Tests S-SB-P1, S-SB-P2, and S-SB-P7 each simulated a 2.5% communicative break, which corresponds to about a 10-cm break in the side of a PWR cold leg pipe. Emergency core coolant (ECC) was provided by the high pressure injection system (HPIS) only.

The pressure suppression system was modified to provide a measurement of the total primary system mass discharged through the break for each test. The pressure suppression tank was disconnected from the pressure suppression header and was replaced with a condensate system that drained into a small catch tank. The total mass of fluid exiting the break was determined by weighing this catch tank before and after each test.

A comparison of the upper plenum pressure response for Tests S-SB-P1 (early pump trip) and S-SB-P7 (late pump trip) is shown in Figure 1. The depressurizations were relatively rapid during the subcooled portion of the transients. At about 35 s, the hot leg fluid reached saturation in both tests, resulting in decreased depressurization rates due to fluid flashing. Differences in the depressurizations between the two tests after 35 s were due to the effect of pump operation on fluid conditions near the break. With the early pump trip

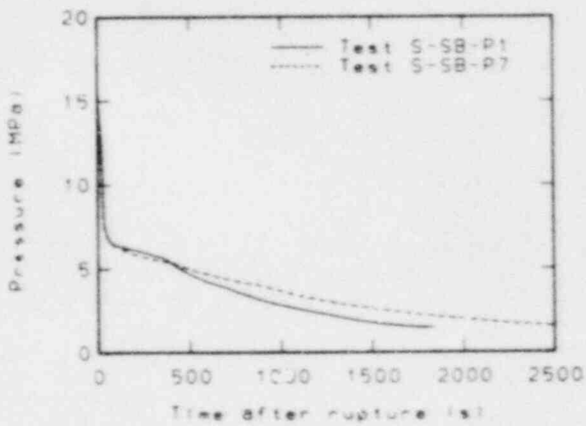


Figure 1. Comparison of upper plenum pressure response for Tests S-SB-P1 and S-SB-P7.

(Test S-SB-P1), liquid drained from the upper head and steam generators to the system low points while steam filled the upper portions, forming liquid seals in the intact and broken loop pump suctions. This early pump trip also tended to stagnate fluid in the cold leg, causing a collection of ECC, and thus, increased fluid subcooling near the break. With the pumps running, the hot leg fluid was circulated through the loops, preventing collection of ECC in the broken loop cold leg. The fluid near the break was thus less subcooled early in the transient, with the pumps running, resulting in a lower break flow rate.

After the upper head drained in Test S-SB-P1 (early pump trip), a steam flow path was established between the vessel upper head and the downcomer inlet annulus through the upper head bypass line. This steam flow produced a two-phase mixture at the break, and thus a decreased break flow rate. At about 360 s in the early pump trip test (Test S-SB-P1), the intact loop liquid seal cleared, delivering a high quality fluid to the break, and further decreasing the break flow rate. With the pumps running, the density near the break gradually decreased until the cold leg pipes were essentially full of steam at about 450 s. After this time, fluid conditions near the break were nearly identical (essentially steam) for the early and delayed pump trip tests, resulting in nearly identical depressurization rates and break flow rates.

The calculated primary system mass inventories for Tests S-SB-P1 and S-SB-P7 are compared in Figure 2. The primary system coolant inventory initially decreased more rapidly for the early pump trip than for the delayed pump trip because of the higher break flow rate, discussed previously. The greatest difference in primary coolant inventory between the two tests occurred at about 350 s, with about 10 kg more mass in Test S-SB-P7 (delayed pump trip). The difference in mass between the two tests decreased slightly, then remained approximately constant for the remainder of the transients due to similar conditions near the break. The total primary system coolant inventory was higher with the delayed pump trip than with the early pump trip throughout the tests.

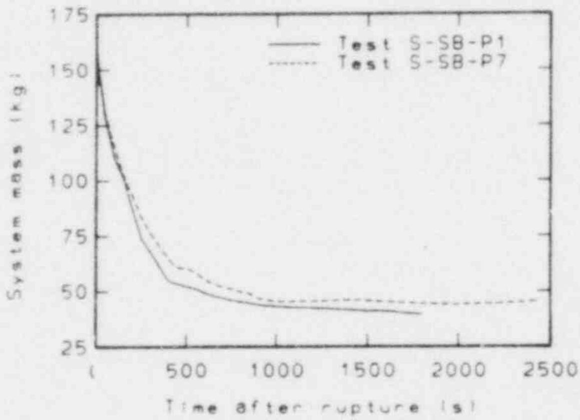


Figure 2. Comparison of primary system mass inventory for Tests S-SB-P1 and S-SB-P7.

Pump operation also influenced system mass distribution. With the pumps running, a significant amount of mass was distributed into the Semiscale loop piping in the early portion of the transient. As the pump head degraded, less fluid was circulated through the loops and the vessel mass began increasing relative to the mass in the loops. As indicated in Figure 3, the combined effect of the lower break flow and the redistribution of fluid to the vessel region resulted in more total mass in the vessel throughout the transient when the pump trip was delayed. In addition, more liquid was present in the hot legs with the pumps running than with an early pump trip. This excess liquid fell back into the vessel region at about 1150 s, when the delayed pump trip occurred in Test S-SB-P7. The resultant increased vessel mass, shown in Figure 3, provided additional core cooling potential.

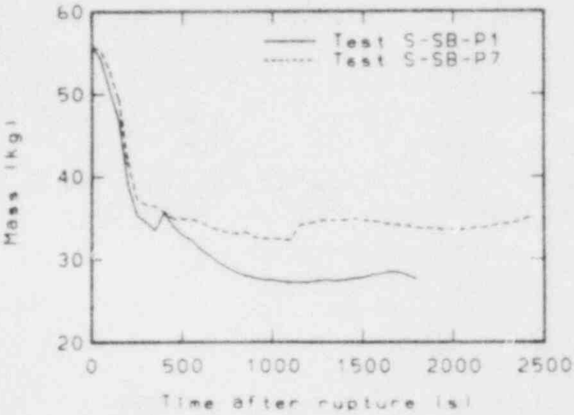


Figure 3. Comparison of vessel mass for Tests S-SB-P1 and S-SB-P7.

The vessel mixture level remained above the top of the core with the delayed pump trip throughout the transient, and thus, adequate core cooling was provided. With an early pump trip, however, the mixture level dropped below the top of the core at about 600 s, resulting in rod cladding temperature excursions (see Figure 4). The mixture level continued to drop until about 1050 s, uncovering about half the core. The mixture level then began rising due to increasing total system mass (due to ECC injection exceeding break flow) and liquid fallback from the steam generators and hot legs. This increasing mixture level provided enough additional core cooling to turn over the cladding temperatures. The maximum cladding temperature during this excursion was 837 K.

The results reported here should not be construed as representative of what would occur in a full-sized PWR. The objective of these tests

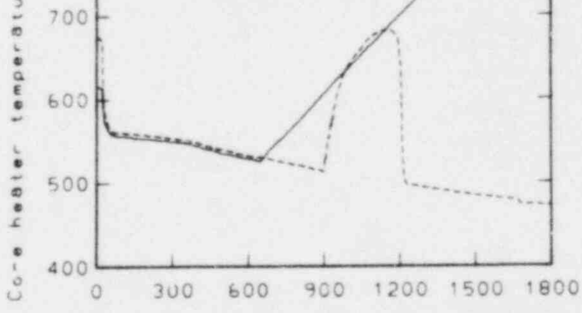


Figure 4. Cladding temperature response for Test S-SB-P1.

was to provide data for assessing the ability of codes to calculate the influence of pump operation on overall Semiscale system behavior. Therefore, while scaling distortions in Semiscale may produce results atypical of those that might occur in a full-sized PWR, the capability of codes to predict important phenomena occurring in Semiscale will provide insight into the capability of codes to predict similar phenomena in a full-sized commercial plant. In this way, Semiscale Program testing contributes to the experimental data base used to evaluate the adequacy of computer codes to define appropriate pump operation in the event of a small break LOCA in a commercial PWR.

II. LOFT PROGRAM DIVISION

C. W. Solbrig, Manager

The LOFT experimental program conducted Experiment L6-5 in the LOFT facility. Experiment L6-5, completed May 29, 1980, was the first experiment in the Anticipated Transient Experiment Series (Series L6) and was designed to simulate a complete loss of feedwater to all steam generators in a commercial PWR with U-tube steam generators. The data from Experiment L6-5 will contribute to the development of a data base that will be used (a) to develop and assess analytical models used for licensing commercial PWRs, and (b) to plan more complex experiments in the LOFT Facility.

The specific objective of Experiment L6-5 was to determine the response of the LOFT system to a secondary-system-initiated transient, in which all feedwater flow to the steam generator was stopped. The experiment objective was achieved.

1. LOFT NUCLEAR ANTICIPATED TRANSIENT EXPERIMENT L6-5

D. L. Reeder

LOFT nuclear Experiment L6-5 was designed to simulate a complete loss-of-feedwater to all steam generators in a four loop commercial PWR. Experiment L6-5 was conducted in the LOFT Facility. A detailed description of the LOFT system is given in Reference 6.

The LOFT system conditions at experiment initiation were: a maximum linear heat generation rate of 39.6 ± 3 kW/m (simulating the normal power level in a commercial PWR), an average temperature of 561.5 ± 3 K, a hot-to-cold leg differential temperature of 13.5 ± 4 K, a primary coolant system flow rate of 479.7 ± 6.3 kg/s, a primary system pressure of 14.79 ± 0.04 MPa, a secondary system flow rate of 20.6 ± 0.4 kg/s, a secondary system pressure of 5.58 ± 0.11 MPa, and a steam generator secondary side liquid level of 3.21 ± 0.06 m above the tube sheet. The primary coolant pumps were operating throughout the experiment.

The experiment was initiated by shutting off the steam generator feedwater pump. Twenty-four seconds later, the reactor was manually scrammed upon indication of low liquid level in the steam generator secondary side. The reactor was scrammed as planned when the indicated liquid level was ~ 380 mm above the top of the tubes. Manual scram was used because the LOFT system does not have an automatic scram on low steam generator secondary side liquid level. As planned, operator intervention was used to return the plant to a hot-shutdown condition. This was accomplished by initiating refill of the steam generator and adding makeup fluid to the primary system to maintain pressurizer level while the steam generator continued to remove heat. After the steam generator secondary liquid level was returned to its operating band, the system was in a hot-shutdown condition and the experiment was terminated. Table 1 contains the sequence of major events for Experiment L6-5, including predictions of the sequence.

The experiment proceeded as expected. The running pumps provided sufficient forced circulation to maintain the fuel rod cladding subcooled throughout the transient. The steam generator remained an effective heat sink, removing enough energy to prevent the primary coolant system from going solid and experiencing any large pressure excursions. When operator actions were taken as required, the system remained stable and responded as expected.

Calculations were made prior to the experiment with the RETRAN computer code of the first 200 s of the transient. The measured and calculated liquid level in the pressurizer and vessel pressure responses are shown in Figures 5 and 6.

Variations in pressurizer liquid level indicate changes in the primary-to-secondary energy balance. Neglecting pressurizer heater and spray effects, changes in pressurizer liquid level cause the pressurizer steam space volume to expand or contract, respectively, decreasing or increasing system pressure. Calculated and measured pressurizer liquid levels and system pressure are

Table 1. Chronology of events (experimental data versus experiment predictions) for LOFT nuclear Experiment L6-5

Event	Time after Experiment Initiation (s) Experiment L6-5 Data	RETRAN[a,b] Prediction
Feedwater pump tripped	0	0
Reactor scrammed	23.8 ± 0.10	23
MSCV[c] closed	35.4 ± 0.10	39
MSCV opened	768.2 ± 0.10	107[d]
MSCV closed	792.5 ± 0.10	146[d]
Charging initiated	929.9 ± 0.10	Not calculated
Steam gene :tor refill initiated	1050 ± 2.0	Not calculated
Charging terminated	1119.6 ± 0.10	Not calculated
MSCV bypass valve opened	2142 ± 2	Not calculated
MSCV bypass valve closed	2200 ± 2	Not calculated
Charging initiated	2374.2 ± 0.10	Not calculated
Charging terminated	2518.2 ± 0.10	Not calculated
Steam generator liquid level in narrow range	2720 ± 10	Not calculated

a. Calculation terminated at 200 s.

b. The computer code used was RETRAN-01-MOD002, Idaho National Engineering Laboratory Configuration Control Number H000986B. The input deck and results are stored under Idaho National Engineering Laboratory Configuration Control Numbers H001086B and H001186B, respectively.

c. MSCV = main steam control valve.

d. These large differences are due to leakage of the MSCV.

shown in Figures 5 and 6. In the calculations, the pressurizer fluid was assumed to be homogeneous and in thermal equilibrium.

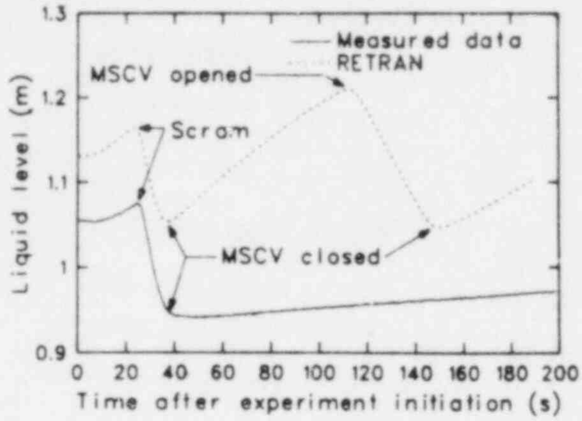


Figure 5. Measured and RETRAN calculated liquid level in pressurizer for Experiment L6-5.

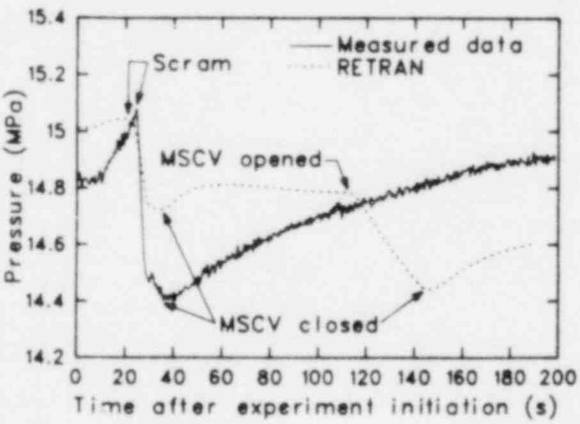


Figure 6. Measured and RETRAN calculated pressure in reactor vessel upper plenum for Experiment L6-5.

During the first 25s of the experiment, the percent compression of the steam space volume was more in the calculation than in the data, whereas the increase in system pressure was significantly greater in the data than in the calculation. The difference in behavior was due to the pressurizer heaters and pressurizer fluid nonequilibrium condensation effects. The difference between the data and calculated initial system pressures caused the heaters to be on initially in the experiment, but not in the calculations. The effect of the heaters was to increase system pressure. The thermal time constant of the pressurizer fluid and nonequilibrium condensation kinetic effects delayed condensation when the primary energy balance caused compression of the steam volume. This delay exaggerated the system pressure rise during the experiment. Nonequilibrium effects contributed to the differences between the experimental data and the calculated response from 25 to 120 s. From 25 to 40 s, nonequilibrium flashing delay allowed the experimental pressure to drop more than the calculated pressure. Overestimation of the effectiveness of the small continuous spray flow in the calculations, combined with the previously mentioned condensation delays, caused the differences in pressure responses between 40 and 120 s.

After 110s, the main steam control valve (MSCV) was calculated to open, but this was not observed in the experiment. It appeared that a small amount of leakage through the MSCV in the experiment vented sufficient steam to prevent the steam generator pressure control high pressure setpoint from being reached until later in the transient. For this reason, after 110s direct comparison of the data and the calculation yield large differences; however, the differences are as expected.

Postexperiment analysis of Experiment L6-5 is continuing. Conclusions based on the results of analyses completed thus far include:

1. The thermal response of the core followed the thermal response of the primary coolant. The cladding temperature remained below fluid saturation temperature during the entire transient. No core damage occurred.

2. The automatic plant control systems and operator actions were effective in returning the plant to a hot-shutdown condition.

3. The LOFT steam generator continued to cool the primary system throughout the transient and was an effective heat sink following a loss-of-feedwater transient.

4. During the operator-controlled recovery period, the reactor system remained stable and recovery was able to proceed without the occurrence of unforeseen events.

5. The calculation made prior to the experiment properly predicted the sequence of events and general magnitudes of the phenomena prior to the occurrence of the MSCV leakage.

6. Consideration of nonequilibrium effects in modeling the pressurizer should improve calculations for this type of transient.

III. THERMAL FUELS BEHAVIOR PROGRAM

H. J. Zeile, Manager

The objective of the Thermal Fuels Behavior Program (TFBP) 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. 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 programmatic tests in the PBF are divided into the following test series:

1. The Power-Cooling-Mismatch (PCM) Test Series
2. The Loss-of-Coolant Accident (LOCA) Test Series
3. The Reactivity Initiated Accident (RIA) Test Series
4. The Operational Transient (OPTRAN) Test Series
5. The Severe Fuel Damage (SFD) Test Series.

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

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

During the past quarter, the Thermal Fuel Behavior Program completed (a) a reactivity initiated accident test (Test RIA 1-4) in the PBF; (b) the postirradiation examinations for loss-of-coolant accident Tests LOC-3 and LOC-5 that had previously been performed in the PBF; (c) a scoping test with the IFA-430 fission product measuring system in the Halden reactor in Norway; (d) analysis of boiling transition, quench, and rewet phenomena during high pressure power-cooling-mismatch conditions with a variety of nuclear fuel rods tested in the PBF; and (e) analysis of the transient freezing of molten debris observed during a severe fuel failure experiment (Test RIA-ST-4). Test RIA 1-4 was the first reactivity initiated accident test with a fuel rod bundle (nine rods). It was performed at boiling water reactor hot-startup conditions, and will provide data to evaluate the present NRC licensing criteria for an RIA event in a light water reactor. The IFA-430 Scoping Test was performed to evaluate the capability for measuring the release rate of the long lived isotope ^{131}I by monitoring its daughter product, ^{131}Xe, following reactor shutdown.

1. PBF TESTING

P. E. MacDonald and R. K. McCardell

Test RIA 1-4, a nine-rod bundle test, was conducted and preliminary results were compiled in a quick look report and are briefly described in the following section. Results of the RIA Scoping Tests were issued,[7] results of Tests RIA 1-1 and RIA 1-2 were compiled, and the final draft of the PBF/LOFT Lead Rod Test Series Test Results Report[8] was completed.

Test RIA 1-1 was conducted with four PWR design test rods (two previously unirradiated and two irradiated to 5000 MWd/t) subjected to a radially averaged peak fuel enthalpy of approximately 1172 J/g, the present NRC licensing limit for commercial nuclear power reactors. Posttest examinations showed that both of the irradiated fuel rods had undergone extensive fuel and cladding melting, accompanied by massive fuel swelling, rod fragmentation, and fuel powdering. The molten fuel swelled and foamed out into the coolant channel, completely blocking the flow shroud. The two previously unirradiated rods also failed, but the flow shrouds were only partially blocked by debris, which consisted of cracked and crumbled fuel and cladding. Test RIA 1-2 was performed with four previously irradiated test rods (5000 MWd/t) subjected to a radially averaged peak fuel enthalpy of 775 J/g, the identified failure threshold. The Test RIA 1-2 posttest examination revealed that only one of the four rods failed, but that the failed rod remained relatively intact, with multiple longitudinal cracks in the central 60% of the rod.

The PBF/LOFT Lead Rod (LLR) Test Series was a series of four tests conducted in the Power Burst Facility with LOFT design fuel rods for the purpose of evaluating the extent of mechanical deformation that would be expected to occur to the LOFT fuel rods during the LOFT Power Ascension Test Series. The information presented in the LLR Test Results Report demonstrates that low pressure, LWR design fuel rods can withstand multiple loss-of-coolant transients without failure at power densities as high as 57 kW/m.

Other accomplishments during the reporting period include completion of the experiment operating specifications for Tests TC-3 and LOC-6, completion of experiment predictions for Test LOC-6 and Tests OPTRAN 1-1 and 1-3, issuance of the Test PR-1 Quick Look Report, presentation of the results of Test LOC-3 at the 1980 Thermal Reactor Safety Conference in Knoxville, Tennessee,[9] and preparation of an article presenting an assessment of LWR fuel damage during an RIA with comments on the adequacy of the present NRC design requirements, to be published in Nuclear Safety. This article will review the results obtained from early SPERT tests and provide comparisons with results from recent computer simulations and PBF tests.

1.1 Test RIA 1-4 Results
Z. R. Martinson

The rapid insertion of excess reactivity into an LWR core has long been recognized as an accident mechanism with the potential for failure of the fuel rod cladding. Extensive cladding failure and subsequent dispersal of fuel into the coolant could disrupt the core such that the postaccident capability for cooling the core would be significantly impaired. Light water reactors operated within the United States must be designed such that a worst-case RIA will not result in a radial average fuel enthalpy greater than 1172 J/g at any axial location in any fuel rod.

The pressurized water loop located in the Power Burst Facility enables reactivity insertion tests to be performed at simulated BWR hot-startup conditions with forced coolant flow. Test RIA 1-4, the first RIA fuel rod bundle test performed at BWR hot-startup conditions, was completed this quarter. The test objective was to provide data to evaluate the present NRC licensing criteria under near typical BWR coolant conditions and for a bundle of previously irradiated fuel rods. Brief descriptions of the test design, performance, and preliminary test results are presented in this section.

1.1.1 Test Design. Test RIA 1-4 was composed of a nine-rod bundle of PWR design (Saxton) fuel rods previously irradiated to burnups of about 5300 MWd/t. The 3 x 3 array of fuel rods was positioned within a zircaloy flow shroud by a series of four Inconel grid spacers on a pitch of 14.3 mm. Three of the fuel rods were instrumented with two cladding surface thermocouples each, and the other six fuel rods were uninstrumented. The nine-fuel-rod bundle, flow shroud, and instrumentation were positioned in the PBF in-pile tube by a support structure referred to as the test assembly. Test assembly instrumentation measured the temperature, pressure, and volumetric flow rate of the coolant, and the local neutron and gamma flux.

1.1.2 Test Conduct. Performance of Test RIA 1-4 consisted of a nonnuclear loop heatup, a nuclear power calibration and preconditioning period, shutdown for fuel rod and flux monitor replacement, a second nonnuclear heatup, and a single power burst. The power burst was conducted at near typical BWR hot-startup conditions of 538 K inlet coolant temperature, 6.45 MPa coolant pressure, 0.8 L/s inlet shroud coolant flow, and an initial zero fuel rod bundle power. The power burst had a reactor period (time required for the reactor power to increase by a factor of $\sim$2.72) of 2.8 ms and a peak reactor power of 37 000 MW. On the basis of the average of the preliminary energy deposition data, a total fuel energy of 1486 J/g was deposited at the axial flux peak in the corner fuel rods, 1350 J/g in the side fuel rods, and 1243 J/g in the center fuel rod. These total fuel energy values include 63 J/g UO_2 to account for the initial zero power fuel temperature of 538 K.

The preliminary (unqualified) online fuel bundle power and energy deposition data were used as input for a posttest FRAP-T5[a] calculation to determine the maximum fuel enthalpy for the power burst. The FRAP-T5 calculation accounts for heat transfer from the fuel pellets to the cladding and coolant during and after the power burst. According to the FRAP-T5 calculation, the test objective of a radially averaged peak fuel enthalpy at the axial flux peak of 1172 J/g was reached for the corner fuel rods about 18 ms after the time of peak power. A peak radial and axial fuel enthalpy of 1415 J/g was calculated for the corner rods. The peak radial and axial fuel enthalpy occurs near the fuel pellet outer surface at the axial flux peak. It should be noted that a fuel enthalpy of about 1118 J/g is required to reach incipient UO_2 melting, and about 1415 J/g is required for complete UO_2 melting.

The final value of the peak fuel enthalpy will be determined after the online data are qualified, the flux monitor wires are scanned, and fuel burnup results from a fuel rod exposed only to the power burst are completed.

1.1.3 Cladding Surface Temperature. Figure 7 illustrates the measured response of the cladding surface thermocouple at 0.59 m, 315 degrees, on a corner fuel rod in comparison to the pretest FRAP-T5 calculated cladding temperature following the power burst. Note that the measured cladding peak temperature was 1625 K at 0.57 s after peak power, as compared with a calculated peak temperature of 2100 K (zircaloy melting temperature) from 0.33 to 9 s. The thermocouple response also indicated that the time in film boiling was about 6.5 s, which was much shorter than the calculated film boiling time of about 30 s. The other cladding surface thermocouples displayed similar responses of lower measured than calculated temperatures, and film boiling times less than calculated.

1.1.4 Thermal-Hydraulic Response. A composite plot of the shroud coolant outlet temperature, shroud pressure, and shroud inlet flow rate is shown in Figure 8. A sharp increase in the coolant pressure up to ~8.4 MPa occurred due to the rapid heating and steam bubble formation of the coolant by the extremely high neutron and gamma flux during the power burst. The rapid pressurization within the shroud produced a flow excursion out both ends of the flow shroud. The initial flow excursion out the inlet of the shroud was fast, and then slowed as the pressure in the shroud decreased to the initial loop pressure (~ 6.45 MPa). Integration of the shroud inlet flow during the flow reversal indicates that about 30% of the coolant over the lower half of the active fuel rods was expelled from the shroud. Although there was no flowmeter located at the outlet of the shroud, the flow excursion out of the shroud outlet was probably comparable to that out the shroud inlet. The steam bubble in the fuel bundle cooled and gradually collapsed when the heat flux from the fuel rods was less than the heat being transferred out of the bundle. As the pressure pulse decreased to the initial system pressure, normal upward flow through the shroud was restored. The test rods were in stable film boiling at this time, as indicated by the cladding surface thermocouples. About 0.5 s was required to refill the flow shroud after the flow reversal.

Heat transfer from the fuel rods resulted in a shroud outlet coolant pressure increase to 8.0 MPa (not shown). The coolant outlet temperature increased to a maximum of 568 K (saturation temperature for 8.0 MPa). At 20 s, both shroud inlet flow measurements decreased to about 50% of the pre-power-burst flow rate, indicating a reduction in the coolant flow area within the flow shroud.

1.1.5 Preliminary Postirradiation Examination Results. A composite photograph of the fuel rod bundle after removal of the flow shroud in the hot cell is shown in Figure 9. Although all nine fuel rods failed as a result of the power burst, severe fuel and cladding fragmentation did not occur. Co-planar cladding swelling, which decreased the coolant flow area, is evident. Numerous cladding ruptures are evident on the eight peripheral fuel rods of the bundle. Most of these cladding ruptures are only evident on the portion of the cladding facing away from the center of the bundle. One cladding rupture was located on the inner side of a fuel rod toward the center fuel rod in the bundle. Figure 10 shows that the cladding rupture on the peripheral rod apparently caused cladding failure of the center rod due to cladding melting (possibly failure propagation). Further examinations will be performed to verify that the

a. FRAP-T5, Idaho National Engineering Laboratory Code Configuration Control Number H001183B.

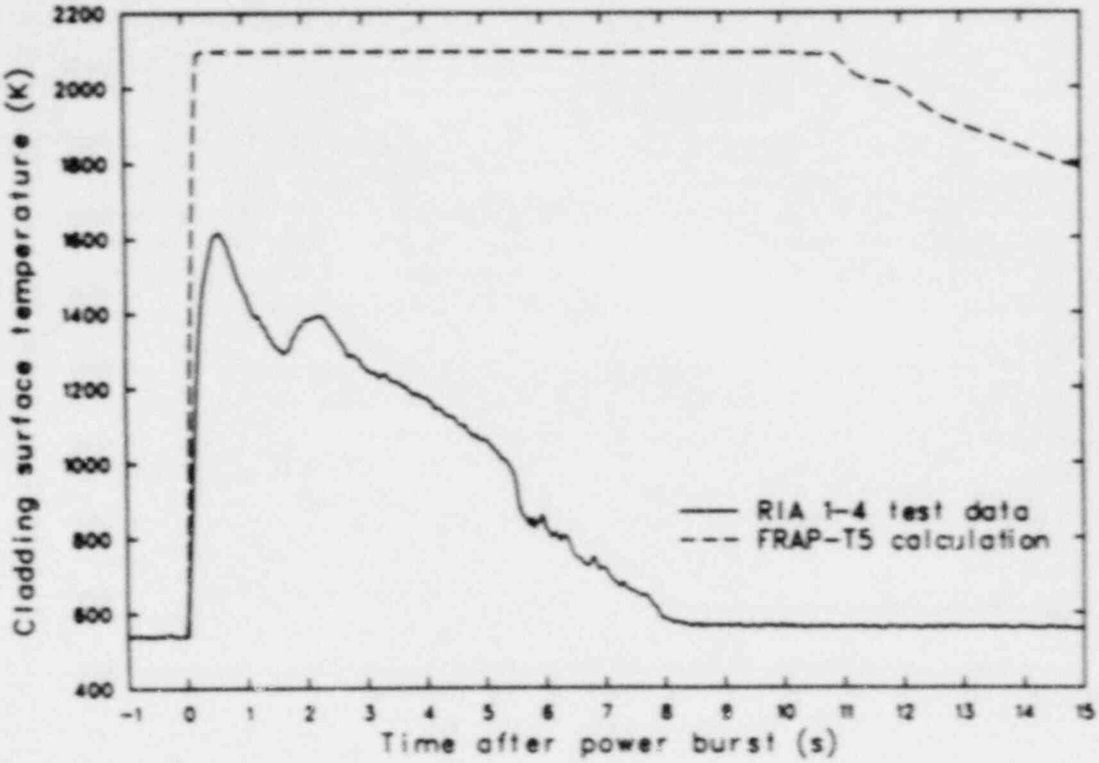


Figure 7. Response of cladding thermocouple at 315 degrees, 0.59-m elevation on the corner fuel rod and FRAP-T5 calculated cladding temperature following Test RIA 1-4 power burst.

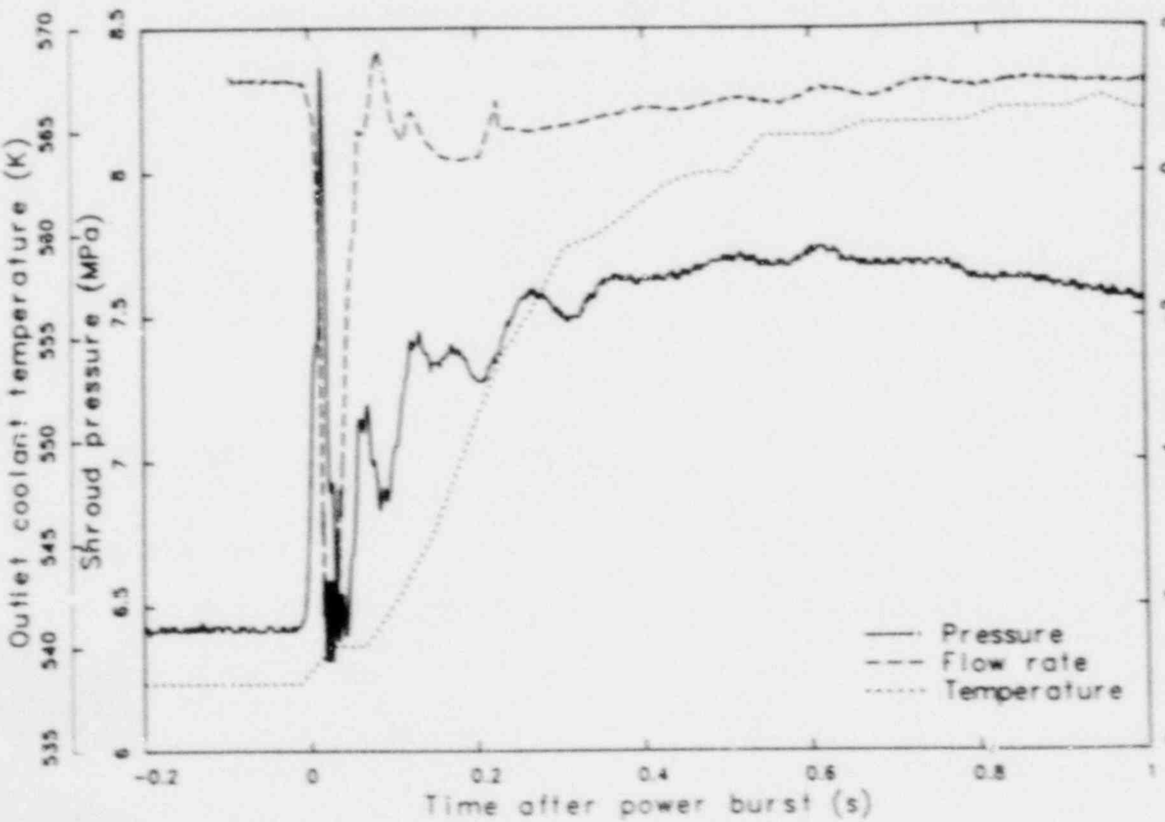


Figure 8. Shroud outlet coolant temperature, shroud pressure, and shroud inlet flow rate during Test RIA 1-4.

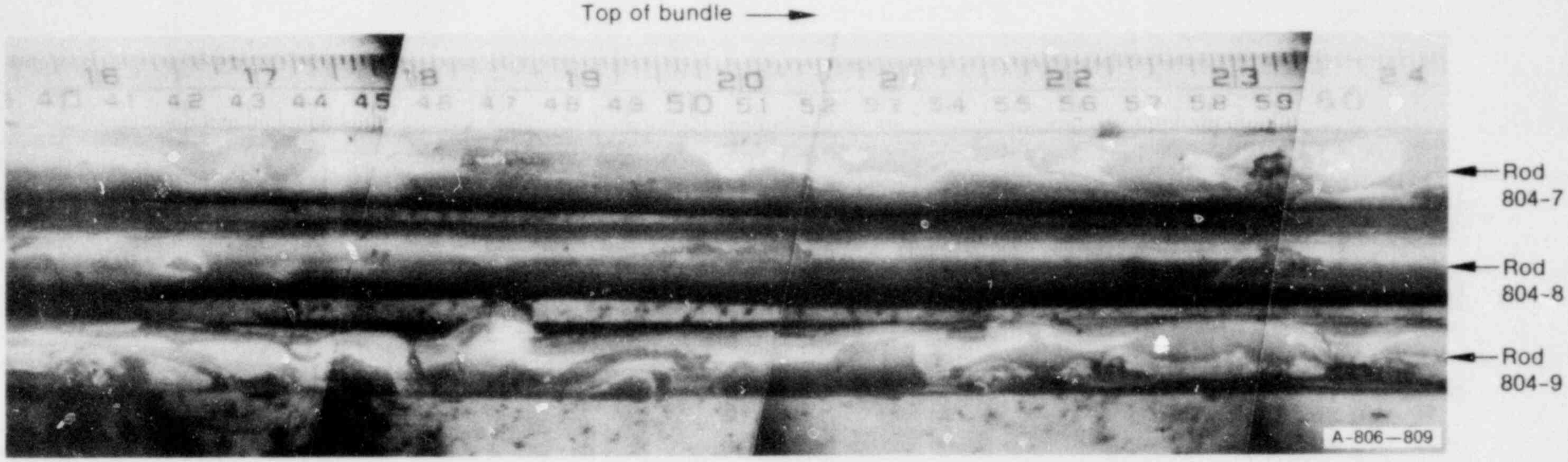


Figure 9. Composite posttest photograph of Test RIA 1-4 fuel rod bundle.

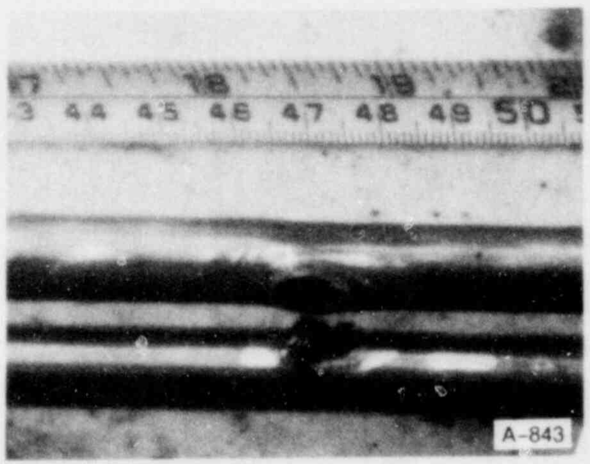


Figure 10. Posttest photograph of two fuel rods from Test RIA 1-4 indicating apparent fuel rod failure propagation.

center fuel rod failure was in fact due to fuel rod failure propagation. Several other peripheral rod cladding failures appear to have been caused by failure propagation from an adjacent fuel rod. Detailed metallurgical examination including characterization of the cladding swelling is planned for the entire fuel rod bundle.

2. 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.

Tests LOC-3 and LOC-5 postirradiation examinations were completed early in the quarter. The examinations revealed that the burst failures in Test LOC-3 occurred primarily at rod cladding temperatures in the $\alpha + \beta$ two-phase region, as had been planned, and the failures in Test LOC-5 occurred in the β-phase region, also according to plan. The three previously irradiated rods in these tests exhibited greater total circumferential elongations than three of the four unirradiated rods; however, the deformations were all within the scatter band of previously reported out-of-pile data. In addition to burst temperature, heating rate at burst varied among the rods and may account for some of the larger rod deformations observed in the irradiated rods relative to the unirradiated rods. Cladding circumferential temperature gradients, which are also known to strongly influence total circumferential elongation, are also being analyzed. An unirradiated rod in which cladding collapse had occurred prior to the loss-of-coolant blowdown, exhibited much larger total circumferential elongation compared with an unirradiated rod that failed in the same temperature region (~1300 K) but at a much more rapid heating rate (~300 versus ~10 K/s). The fuel fragments in the ballooned regions of irradiated rods were considerably smaller than those in unirradiated rods and were qualitatively in agreement with results from loss-of-coolant tests on irradiated rods performed in Germany.

Fuel rod behavior in Test PCM-5, in which a 3 x 3 array of nine previously unirradiated PWR-type rods was subjected to power-cooling-mismatch conditions, is described in a report[10] issued during the quarter. Rod powers ranged from 55 to 62 kW/m in this test, and seven of the nine rods operated in film boiling for times ranging from 45 to 690 s. Considerable fuel rod bow toward the center rod, which is the cool side of the peripheral rods, was observed in the film boiling regions. From structural analyses, the conclusion was reached that the peripheral rods were actually bowed away from the center rod toward the flow shroud during film boiling, and bow direction reversal occurred upon quenching from film boiling.

Instrumentation indicated that two of the Test PMC-5 rods failed during the test. One rod failed ~250 s after DNB on the rod, while it was quenching. Bow direction reversal in combination with thermal shock to the embrittled cladding probably caused rod failure. Considerable cladding loss occurred, with little fuel loss. Cladding remaining near the failed region was 80% reacted to oxide and oxygen-stabilized α zircaloy. The second rod failed ~500 s after DNB on the rod, while it was in stable film boiling, and continued to operate in film boiling for an additional ~190 s. The rod probably failed as a result of severe cladding embrittlement. Considerable cladding and fuel loss occurred. Cladding remaining in the failed region was completely reacted to oxide and oxygen-stabilized α zircaloy. No evidence of rod-to-rod failure propagation was found. A third rod broke into two pieces during posttest handling.

During the past quarter, the coolant activity during Test RIA 1-4 was monitored in the PBF. One of the most significant results obtained from these data was the identification of the long-lived isotopes, ^{134}Cs and ^{137}Cs, in the test loop coolant, simultaneously with the measurement of several short-lived isotopes. This result is encouraging to the long-term goal of fuel condition monitoring in power plants that will rely on identification of such mixtures.

The preliminary results of analysis of the first fission gas release test and the Xe/He fill gas tests in the IFA-430 gas flow assembly in the Halden reactor in Norway were presented at the American Nuclear Society Topical Meeting on Reactor Safety.[11,12] The preliminary analysis shows that the release of short-lived (<1 day) fission gas can be calculated to within a factor of ~2 using diffusion equations, and that the release of short-lived iodine (^{135}I) is on the same order as the release of short-lived xenon. Data from the Xe/He fill gas tests were compared with FRAP-T5 calculations and showed that, for fill gas pressures less than ~1.0 MPa and xenon concentration in the helium fill gas less than 10%, calculated temperatures were within a few percent of the measured data.

A scoping test was performed with IFA-430 and the results indicate that measurement of the release rate of the relatively long-lived ^{131}I isotope is possible by monitoring the ^{131}Xe daughter of ^{131}I following reactor shutdown. The technique will be applied to determine concentrations of ^{131}I in the fuel-cladding gap during steady state operation of the fuel. These tests are directed toward defining the amount of iodine in the fuel-cladding gap that could contribute to stress corrosion cracking of the cladding, as well as providing quantitative data for radioactive source term determination and the development of models for fission product release.

A report on fuel cracking and relocation of fuel pellets during initial nuclear operation, based on IFA-430 data, was published.[13]

A study summarizing boiling transition and quench and rewet phenomena during high pressure power-cooling-mismatch and irradiation effects testing within the PBF has been completed. Power-cooling-mismatch testing, including two-phase flow patterns, typical PCM boiling cycles, and an overview of worldwide PCM testing, are reviewed. Empirical correlations are derived to model experimentally observed boiling transition behavior, both as local and global phenomena. Comparison with several critical heat flux correlations indicates that the Combustion Engineering (CE-1) and Loss-of-Fluid Test (LOFT) correlations, in general, best model the boiling transition behavior observed during PCM testing within the PBF. Quenching and rewet behavior are also assessed. Empirical and analytical correlations are derived to predict the cladding surface temperatures at the onset of quench and rewet. Results indicate that quenching and rewetting are distinctly different phenomena, entailing different modes of heat transfer. The thermal-hydraulic conditions at the onset of film boiling are compared to the conditions at the onset of film boiling destabilization (quench). Results indicate that the return to nucleate boiling process (quench-rewet) proceeds via the same path as departure from nucleate boiling, with little or no hysteresis.

A second study[14] was performed to (a) analyze the transient freezing of molten debris (which is primarily a mixture of UO_2 fuel and zircaloy cladding) on the test shroud wall produced during a severe fuel failure experiment (designated RIA-ST-4), and (b) to assess the potential for shroud wall melting upon contact with the molten debris. A physical model was developed considering the conditions of finite wall thickness, continuous convective cooling at the wall outer surface, radiative cooling of the molten debris, temperature-dependent thermophysical properties, and internal heat generation in the debris. The analysis showed that the freezing of the debris was governed by transient conduction through the wall and radiative cooling of the debris. However, the initial debris temperature, the zircaloy volume ratio in the debris, internal heat generation, and the initial wall temperature strongly influenced the debris freezing and the potential for wall melting upon being contacted by the debris. The study indicates that treating the molten debris during the transient freezing process as a homogenous mixture of UO_2 and zircaloy, in which the effective thermophysical properties of the debris are functions of those of the constituents, is a reasonable assumption. The agreement between the calculations and the RIA-ST-4 experiment results strongly supports the validity of the physical model and the simplifying assumptions in the analysis.

IV. CODE DEVELOPMENT AND ANALYSIS PROGRAM

P. North, Manager

The Code Development and Analysis Program has a primary responsibility for the development of 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.

Development of an advanced boiling water reactor thermal-hydraulics code, TRAC-B, has reached the point that several models are complete and results can be demonstrated. The results from a radiation model test are presented in Section 1. An important item in transient fuel rod behavior is the determination of cladding failure and the fuel rod shape at the point of failure. Results of work in this area are presented in Section 2.

1. DEVELOPMENT OF THE TRAC CODE FOR BWR ANALYSIS

F. Aguilar, J. W. Spore, and R. W. Shumway

Substantial progress has been made on the development of TRAC-BD1, a computer program for the detailed analysis of a design basis LOCA in BWR systems.

The computer program is based on TRAC Version 22.8, which uses the two-fluid formulation on both one- and multi-dimensional components. Development of basic models for BWR components such as jet pumps, separator and dryers, control rod guide tube stored energy, and reactor trip on water level is complete. A critical flow model compatible with the two-fluid TRAC formulation has been implemented successfully, the recent ANS decay heat standard has been incorporated[15] and, finally, the CHAN component (see Reference 16 and the discussion below) has been improved and inserted into TRAC-BD1. All these development tasks have been performed in conformance with newly adopted code development and quality control procedures. Complete traceability for every change and new model is provided by the new procedures.

The CHAN component (i.e., the new model for BWR fuel assemblies) has been exercised extensively, including scoping analyses performed by the Thermal Fuels Behavior Program in support of the Severe Fuel Damage Test. Also, a radiation heat transfer test on the Swedish Göta Test Loop[17] has been modeled. These results are discussed subsequently.

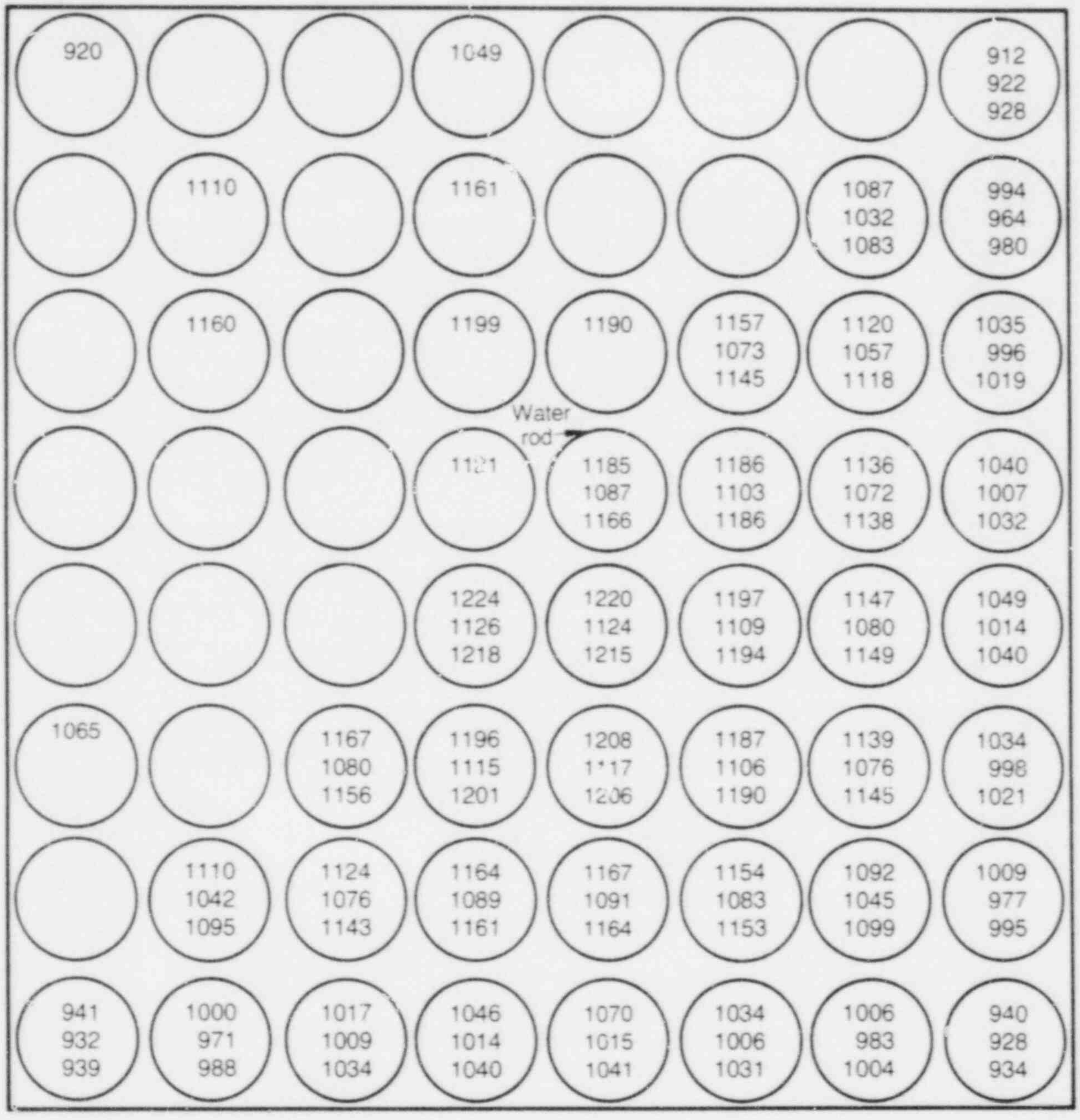
Göta Loop Test 27 is a test of an 8 x 8 electrically heated bundle immersed in a stagnant steam environment. The important heat transfer mechanism is rod-to-rod and rod-to-channel thermal radiation; thus, the experiment is an excellent test of the CHAN radiation heat transfer model.

The test was performed by turning on outer channel cooling water, rod bundle power, and then recording the rod temperatures as steady state was reached. The bundle power was symmetrical about a diagonal from the upper right corner to the lower left corner of Figure 11.

The TRAC-BD1 analysis was formulated such that rods lying along the diagonal were modeled individually. Off-diagonal rods were grouped or paired with the respective mirror image rods, so the calculation was perfomed with a total of 36 rod groups. The results of two analyses are provided in Figure 11.

The upper numbers in the figure are measured rod surface temperature, and the lower numbers are the TRAC-BD1 results. The first analysis assumes isotropic reflection (middle numbers in Figure 11), and the second assumes anisotropic reflection (i.e., preferential reflecting back to an emitting rod—bottom numbers in Figure 11). An emissivity of 0.7 for all surfaces has been assumed in both calculations. The average difference between measured and predicted rod temperatures is 54.6 K with isotropic reflection. The calculated

Top number - Data
Middle number - TRAC-BD1, isotropic reflection
Bottom number - TRAC-BD1, anisotropic reflection



INEL-A-15 790

$$\sum_{i=1}^{36} \left| T_{Data} - T_{TRAC} \right| / 36 \quad \begin{array}{l} = 54.6 \text{ K isotropic} \\ = 8.3 \text{ K anisotropic} \end{array}$$

Figure 11. Göta Loop Test 27 measured and calculated rod midplane temperatures (K).

temperature gradient from bundle center to edge is 100 K; the measured gradient is approximately 170 K. With the anisotropic assumption, the average difference in rod temperatures is 8.3 K, and the temperature gradient is also in good agreement with the measured value. These results point to the significance of anisotropic reflection, a conclusion supported by other investigators.[18,19] It should be noted that the average temperature discrepancy with the first analysis (isotropic reflection) can be improved by varying the rod and channel emissivities. However, it is not possible to simultaneously improve the prediction of the bundle temperature gradient. It should also be noted that the inclusion of anisotropic effects does not significantly affect execution time.

2. BALOON-2 SUBCODE DEVELOPMENT

D. L. Hagrman

BALOON-2 is being developed as a subcode of the FRAP-T6 fuel rod analysis code. The subcode predicts the shape of burst fuel rod cladding and is similar to its predecessor subcode, BALOON-1.[20] The principal differences between BALOON-2 and BALOON-1 are:

1. True stress[a] is used to calculate rod cladding time of failure in BALOON-2. BALOON-1 uses a correlation for engineering strain as a function of burst temperature to calculate rod cladding time of failure.

2. A perturbation approach is used to calculate local stress in BALOON-2.

3. Time step sizes used by the FRAP-T6 code are reduced in BALOON-2 if the true stress, temperature, and given time step size lead to large deformations during the given time step.

4. Cladding anisotropy and bending into an internal source of heat is considered by BALOON-2.

The most significant modification made in the development of BALOON-2 is the use of true stress to predict failure. This failure criterion is based on cladding burst tests for which the initial cladding dimensions, temperature at failure, pressures at failure, wall thickness at the failed region, and some means of estimating the axial and azimuthal radii of curvature at the burst region were reported. In all cases, the wall thickness measurements were accurate to no better than 10% and the azimuthal radii of curvature were estimated from circumference measurements by assuming a circular cross section at the moment of burst. The assumption that the cross section was circular at the moment of burst introduces significant systematic error. This systematic error has been approximately compensated for by multiplying the true stress at failure (obtained with an assumed circular cross section) by 0.6 when the true stress is used with a code that does not assume a circular cross section.

Figure 12 is a plot of the tangential component of true stress at failure obtained from data published by several authors[21-34]. Heating rates from a few kelvin per second to one hundred kelvin per second are represented and no systematic variation in failure stress with heating rate is present. A detailed discussion of the data and the reason for the nonsymmetric scatter about the best-fit line shown in Figure 12 is available in the MATPRO-11, Revision 1, handbook[35].

When BALOON-2 is used with FRAP-T6, pressure, temperature, and shape (midwall radius and wall thickness at each node) data are passed to the subcode. These data are used to calculate local stress at each node. If the temperature, stress, and FRAP-T6 time step size imply more than a 1% strain increment, the time step size is subdivided. Next, each node is tested to determine whether the local stress exceeds the failure stress. If failure does not occur, strain component increments are calculated and bending is estimated. When this is complete, the internal pressure, cladding temperatures, and cladding shape parameters are updated. The process is repeated until failure is indicated or until the FRAP-T6 time step is completed.

a. True stress is the force per unit cross-sectional area determined at the instant of measurement of the force.

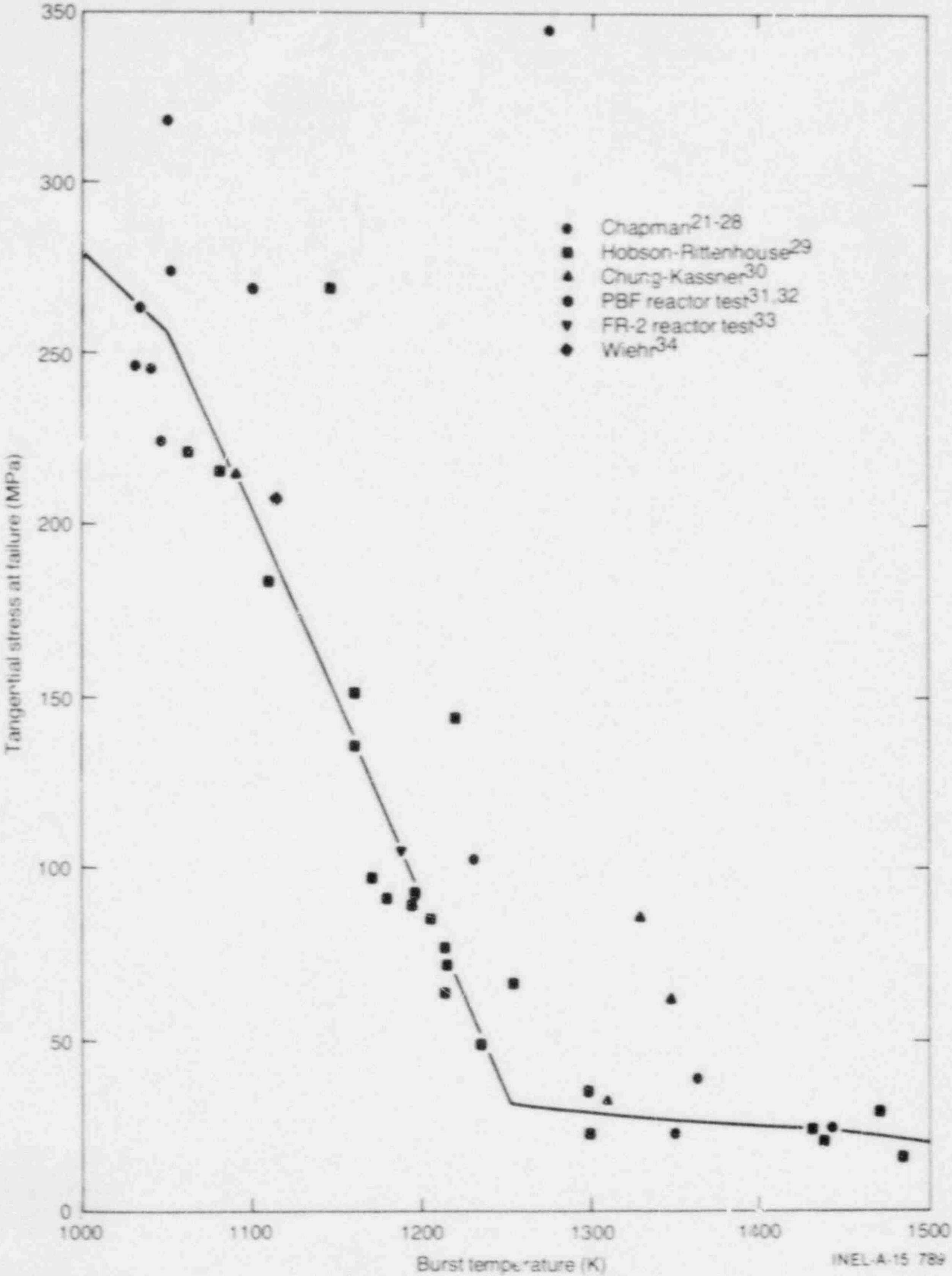


Figure 12. Local tangential stress at fuel rod cladding failure (burst) versus temperature, assuming a circular cross section at failure.

The BALOON-2 subcode is being tested by comparing shape predictions with data from the Nuclear Regulatory Commission's Multirod Burst Test Program[22] and by conducting parametric studies to determine the effect of temperature gradients and heating rates on cladding shape at burst. The principal conclusions to date are:

1. Parameters that affect cladding shape at burst interact—simple correlations such as shape versus burst temperature are not satisfactory in calculating cladding shape at burst.

2. Axial and circumferential temperature variations affect burst diameters in opposite senses.

3. Heating rates and burst temperatures both have a significant effect on burst diameters.

V. CODE ASSESSMENT AND APPLICATIONS PROGRAM

J. A. Dearien

The Code Assessment and Applications Program (CAAP) has a primary responsibility to the NRC for the assessment of thermal-hydraulic and fuel behavior analysis 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 analysis codes. In addition to assessing codes, the CAAP is the technical advisor to the NRC on industry cooperative safety experimental programs. The purpose of this activity is to ensure that data from these experimental programs are adequate for assessment of thermal-hydraulic codes. The CAAP is also assisting in the NRC Standard Problem Program in which computer code simulations of nuclear safety related transient tests are performed by participants using calculation techniques (computer codes) of their choice. This program is a cooperative effort among the NRC, U.S. reactor vendors, and the international nuclear community. The last program in which the CAAP is providing assistance to the NRC is the Severe Accident Sequence Analysis (SASA) task. The purpose of this program is to identify and analyze accident or upset sequences of events and to provide assistance during commercial reactor transients such as occurred at Three Mile Island.

The following sections summarize results from two code assessment tasks, one combined code assessment and technical support task, and a PWR analysis task.

1. FRAP-T5 RESPONSE TO SINUSOIDAL POWER VARIATIONS

G. B. Peeler, E. T. Laats, R. Chambers

The predictive capabilities of the FRAP-T5[20] transient fuel rod analysis program to simulate rod behavior under sinusoidal power variations were assessed. The primary objectives were to demonstrate where best-estimate model capabilities exist, to provide guidance for future model development, and to recommend the most appropriate input options to be used for modeling this scenario.

Code predictions were compared with experimental data from the Gap Conductance Test Series in the Power Burst Facility. BWR-type rods with a fuel density of 95% of theoretical density (10.97 g/cm^3) and initial pellet-cladding gap size varying from 0.94 to 3.4% were used.

Fuel centerline temperature response was predicted to be affected by decaying exponential terms that decreased with decreasing frequency. When gap closure was achieved, the shape of both the predicted and the observed temperature responses differed significantly from a sine wave. This wave shape change was related to the closing and opening of the pellet-cladding gap.

Overall, results indicated that FRAP-T5 reproduced the basic behavior traits observed in the experiments. The FRACAS-II deformation model, a subcode in FRAP-T5, was the most appropriate option for simulating this type of rod behavior.

2. RELAP4/MOD6 BLIND PRETEST CALCULATIONS OF THE TWO- LOOP TEST APPARATUS SMALL BREAK TESTS

R. J. Dallman

Computer simulations and analyses of the Two-Loop Test Apparatus (TLTA) Small Break Test 1 (Test 6431) and Test 2 (Test 6432), using the RELAP4/MOD6 thermal-hydraulic code,[a] were

a. RELAP4/MOD6, Idaho National Engineering Laboratory Configuration Control Number H007084B.

performed for the NRC. The purpose of the calculations was to examine the ability of RELAP4/MOD6 to calculate TLTA system thermal-hydraulic behavior during a small break.

Tests 6431 and 6432 were performed by the General Electric Company in the TLTA, which is located in San Jose, California. The tests are part of the jointly sponsored NRC/EPRI/General Electric Boiling Water Reactor Blowdown Emergency Core Cooling (BWR-BD/ECC) Program. Although the TLTA is scaled well to simulate typical BWR response during the blowdown phase of a postulated loss-of-coolant accident, the TLTA responses during refill and reflood phases are compromised by the scaling. For this reason, results from these tests were not considered to be totally representative of a BWR.

Test 6431 utilized a 0.318-cm-diameter orifice in the break line that came off the suction side of one recirculation loop. This represented a communicative break scaled to a 0.0046-m^2 break in a BWR/6. The ECC system was limited to the high pressure core spray (HPCS) delivering rated flow.

System thermal-hydraulic behavior for Test 6431 was calculated well using the RELAP4/MOD6 code. The steam dome pressure is shown in Figure 13. Causes for the differences between the calculation and data are attributed to poor break flow prediction and atypical steam condensation rates. Both the test data and calculation showed level recovery in the downcomer to occur at the same time. The simulated fuel bundle remained covered throughout the transient in both the test and the calculation, which resulted in steadily decreasing temperatures at all bundle elevations.

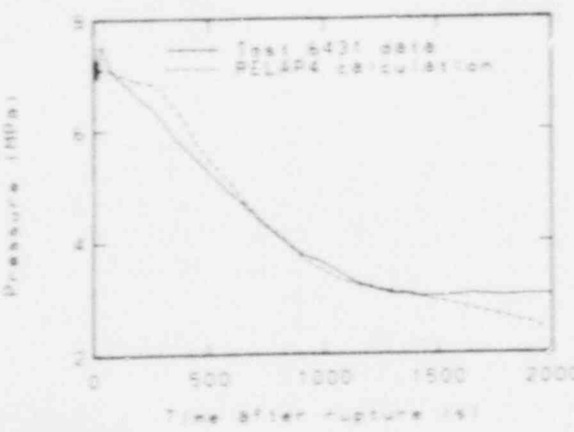


Figure 13. Comparison of calculated and measured steam dome pressure for Test 6431.

Test 6432 was designed to measure system response for the degraded condition of no HPCS. Operable ECC systems included the automatic depressurization system (ADS), the low pressure core spray (LPCS), and the low pressure coolant injection (LPCI). Steam dome pressure for this test is shown in Figure 14. Following ADS activation at 286 s, depressurization occurred much faster in the test than in the calculation. This caused the LPCS and LPCI initiations to be significantly delayed in the calculation. As a result, fluid inventory in the fuel bundle was reduced in the calculation, which led to a calculation of heatup at the upper bundle elevations. These temperatures turned over shortly after the LPCS and LPCI initiations, as the bundle fluid inventory recovered. Due to earlier LPCS and LPCI flow in the test, the measured test data indicate no uncovering or heatup of the fuel bundle.

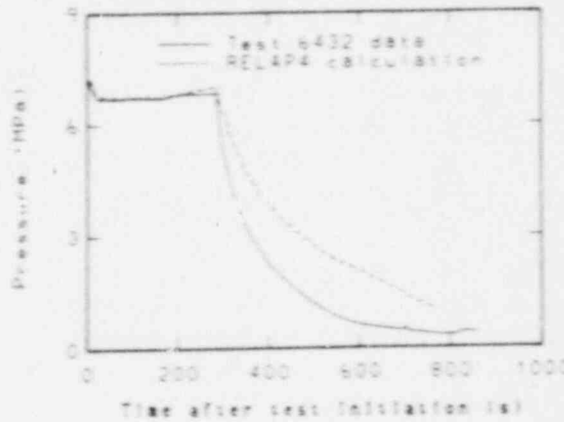


Figure 14. Comparison of calculated and measured steam dome pressure for Test 6432.

3. COMPARISON OF RELAP4/MOD6[a] and MOD7[b] CALCULATIONS WITH FLECHT-SEASET BOIL-OFF AND REFLOOD TESTS

D. M. Ogden and G. E. Wilson

The Full-Length Emergency Cooling Heat Transfer Separate Effects and Systems Effects Test (FLECHT-SEASET) Program, being conducted by Westinghouse Electric Corporation, is an ongoing effort to provide experimental data on heat transfer and two-phase flow behavior in simulated PWR geometries. These data are, and will be, used to assess computer codes. As part of the program, numerous reflood and three boil-off tests were conducted with the 161-rod Unblocked Bundle Test Facility. The purpose of these tests was to measure and characterize the thermal-hydraulic behavior of the system under calculated boundary conditions for hypothesized PWR loss-of-coolant accidents.

Test predictions and subsequent comparisons with data from one reflood and one boil-off test were completed as contributions to technical support to the NRC for industry cooperative programs. This work also serves as a part of the assessment effort of the RELAP4 series of thermal-hydraulic systems codes. Measured initial and boundary conditions were used in all calculations. The primary emphasis of the studies was to determine the effects of two different entrainment models on the calculation of heated bundle mass inventory and rod temperature response for both reflood and boil-off conditions.

Comparisons between the code calculations and the test data indicate the calculated rod temperature response was similar for both entrainment models when applied to each type of test. In general, the calculated rod peak temperatures compared favorably with the data at the low and middle bundle elevations, regardless of entrainment model or type of test. At the upper elevations, both entrainment models produced calculated rod peak temperatures higher than were measured for the boil-off test. The opposite was true for both entrainment models in the reflood test.

The bundle mass inventory was well calculated with the Heat Transfer Research Institute (HTRI) entrainment model contained in RELAP4/MOD7 for both the boil-off and reflood tests. The code results appeared to be within the cyclic range of the measured data throughout both tests. The Steen-Wallis entrainment model, contained in both RELAP4/MOD6 and MOD7 codes, calculated higher bundle mass inventories than measured throughout the boil-off test. The same was true for the reflood test during the last 450 s of the 600-s duration test.

Because the bundle mass inventory calculation was significantly improved with use of the HTRI entrainment model, and a similar improvement was not shown for the rod temperature response, it was concluded that the differences between code calculated and measured rod temperatures at higher elevations, are probably a result of the code heat transfer surface rather than a result of poor hydraulic simulation.

4. PWR ANALYSIS

C. D. Fletcher, S. J. Bruske, H. M. Delaney

The Westinghouse Zion I PWR was analyzed for responses to a loss-of-offsite-power event during full power operation. This initiating event was chosen because of a high frequency of occurrence and the possibility of severe core damage resulting from multiple failures. The analysis technique included a systematic development of a simplified event tree, the calculation of probabilities associated with the legs of the event tree, and thermal-hydraulic calculations of selected event tree sequences using the RELAP4/MOD7 computer code. The primary purpose of this effort was to establish and test this analysis technique.

---

a. RELAP4/MOD6, Update 4, Idaho National Engineering Laboratory Configuration Control Number C0010006.

b. RELAP4/MOD7, Idaho National Engineering Laboratory Configuration Control Number H013441B.

The event tree was developed for a loss of offsite power by considering the requirements, both on the hardware and the operator, for the plant to be normally recovered. The requirements are listed in chronological order across the top of the event tree shown in Figure 15. For each requirement, a decision is made; either 'yes' the requirement is met, or 'no' it is not. In this manner, a logic diagram of meaningful responses is generated, which may be used to identify the systems requiring detailed analysis. The normal recovery sequence is described by the top leg of the tree (all 'yes' decisions). The delta notation indicates a transfer to separate operator action event trees in which specific operator and system responses for the event sequence are identified.

A probability analysis of the event tree legs was performed. The information for this analysis was obtained from the Reactor Safety Study[36] rather than from an independent determination.

The event tree indicated which sequences were expected to result in severe consequences (core damage), and the probability analysis indicated which of those sequences were most likely to occur. The high risk sequences were a small break loss-of-coolant accident followed by a loss of offsite power, a failure of all diesel generators and the turbine-driven auxiliary feedwater system, and a failure to scram the reactor.

The final step of the study was to perform thermal-hydraulic analyses of selected sequences using the RELAP4/MOD7 computer code. The contribution of thermal-hydraulic analysis is twofold. First, it provides a confirmation of the logic and assumptions used in developing the event tree. Second, it provides a quantitative method to establish the timing of a sequence.

The thermal-hydraulic calculations performed and the major results of the analyses are summarized in Table 2.

The analyses performed to date are not a complete treatment of the loss-of-offsite-power event. As previously mentioned, probabilities were borrowed from previous studies and may not adequately represent best-estimate conditions. Furthermore, pertinent information about the control room layout, instrumentation, and the emergency operating procedures was not available. The development of the event tree was accordingly restricted. This effort did, however, demonstrate an analysis technique having considerable merit and providing a basis for further improvement.

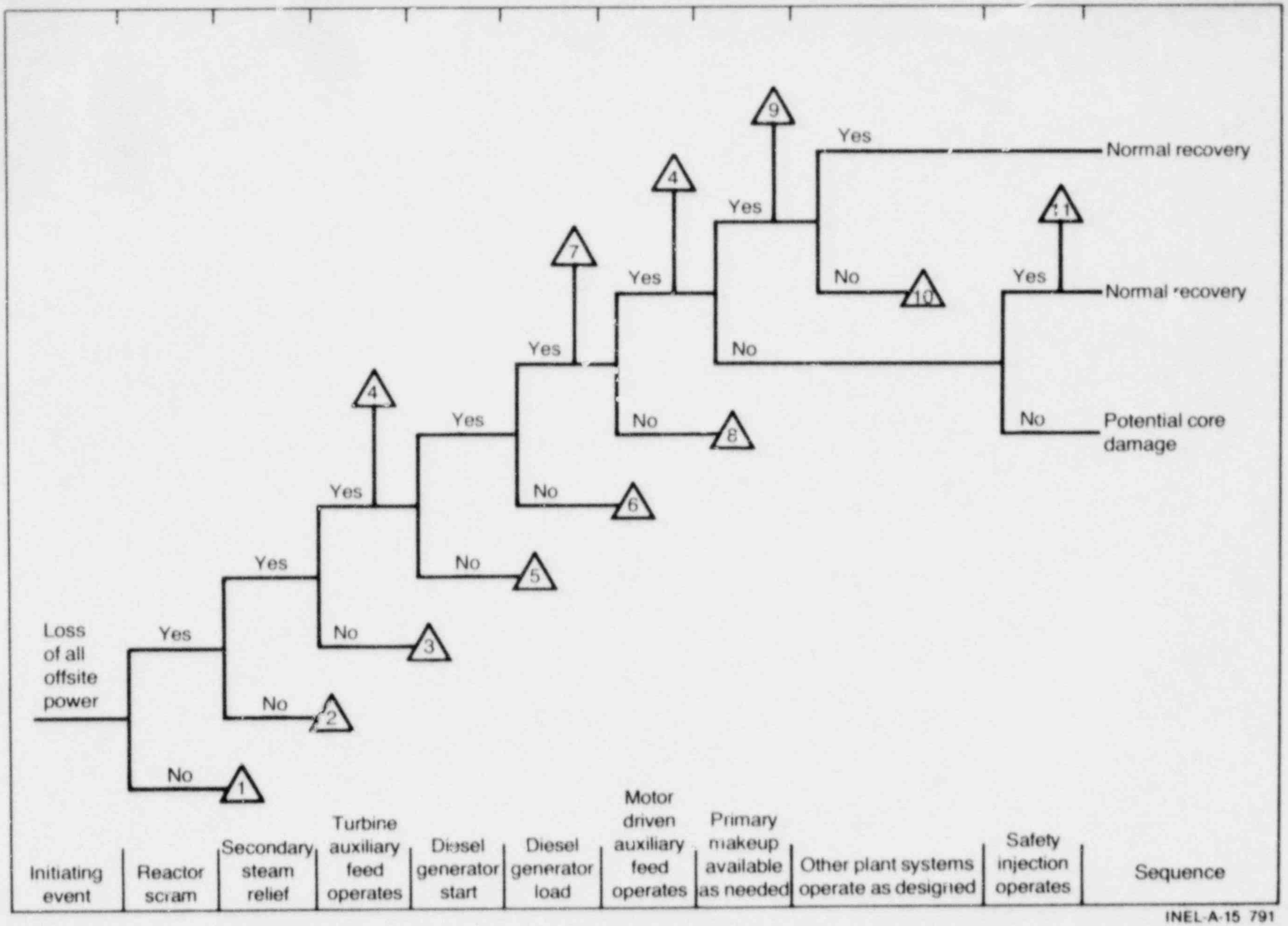


Figure 15. Simplified event tree for loss of offsite power—Westinghouse Zion 1 PWR.

Table 2. Results of PWR analysis for loss of offsite power

Exceptions to Normal Recovery Sequence	Results
None	Plant normally recovered
All diesel generators and turbine auxiliary feedwater fail to start	Core uncovering starts at 4700 s
All diesel generators fail to start; delayed turbine auxiliary feedwater start	Core uncovering is avoided if auxiliary feedwater is started by 3700 s
Turbine auxiliary feedwater fails to start; delayed diesel generator start	Core uncovering is avoided if diesel generator is started by 4300 s
Power-operated relief valve (PORV) sticks open	No core uncovering
Steam generator relief valve ruptured	No core uncovering
Failure to scram	Inconclusive because of computer code and model limitations
Offsite power lost following a stuck open PORV incident; failure of operator to manually restart safety injection	Core uncovering starts 2600 s after the loss of offsite power
Reactor coolant makeup fails	Core uncovering if makeup not available within 8 h

VI. 2D/3D PROGRAM
R. E. Rice, Manager

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

1. 2D/3D INSTRUMENT PROJECTS
J. B. Colson

The objectives of the projects are the experimental investigation of the refill and reflood phases of a postulated loss-of-coolant accident and development and assessment of computer codes suitable for describing such behavior. EG&G Idaho, Inc., is providing flow instrumentation for German and Japanese experiments, and design and analysis support to the NRC. In-core liquid level detectors were delivered to Japan and installed in the Slab Core Test Facility (SCTF). Plans for training Japanese personnel in the operation of SCTF instruments were completed.

2. ADVANCED INSTRUMENTATION
Jay V. Anderson, Manager

Studies have been completed on gamma scattering, a survey of potential acoustic water reactor applications, a compilation of liquid level detection techniques, and a review of commercially available low flow velocity measuring systems. Developmental activities completed include the heated thermocouple liquid level probe and the local ultrasonic densitometer as presented subsequently.

2.1 Local Ultrasonic Densitometer
A. E. Arave

A local ultrasonic densitometer (LUD) system has been developed by EG&G Idaho, Inc., for the water reactor safety research programs. Applications are for high radiation environments such as in nuclear cores, for space-constrained and corrosive environments such as in downcomers or in-pile tests, or for large volume applications such as steam generators or pressurizers where photon densitometry techniques are not suitable.

The system consists of a magnetostrictive transducer coupled to a sensor, interconnecting cable, and electronics as used for nuclear fuel centerline temperature measurements. Transducer components have been demonstrated to withstand temperatures to 1150 K, dependent on material and techniques used, and to withstand nuclear environments of 10^{18} nvt.

The magnetostrictive transducer generates an extensional wave that is converted to a sensor torsional wave at the mode conversion junction, as shown in Figure 16. Torsional wave propagation velocity in the sensor is a function of sensor material properties and the density of the medium surrounding the sensor. The torsional wave, reflected from the end of the sensor, is converted back to an extensional wave and detected by the magnetostrictive transducer. The change in propagation time, ΔT in Figure 16, varies depending on the density of the medium (a steam-water mixture).

The pressure housing that separates the magnetostrictive transducer from the high pressure and temperature environment is nonmagnetic stainless steel. Both the extensional wave transmission line and sensor are exposed to the medium, but only the sensor is sensitive to

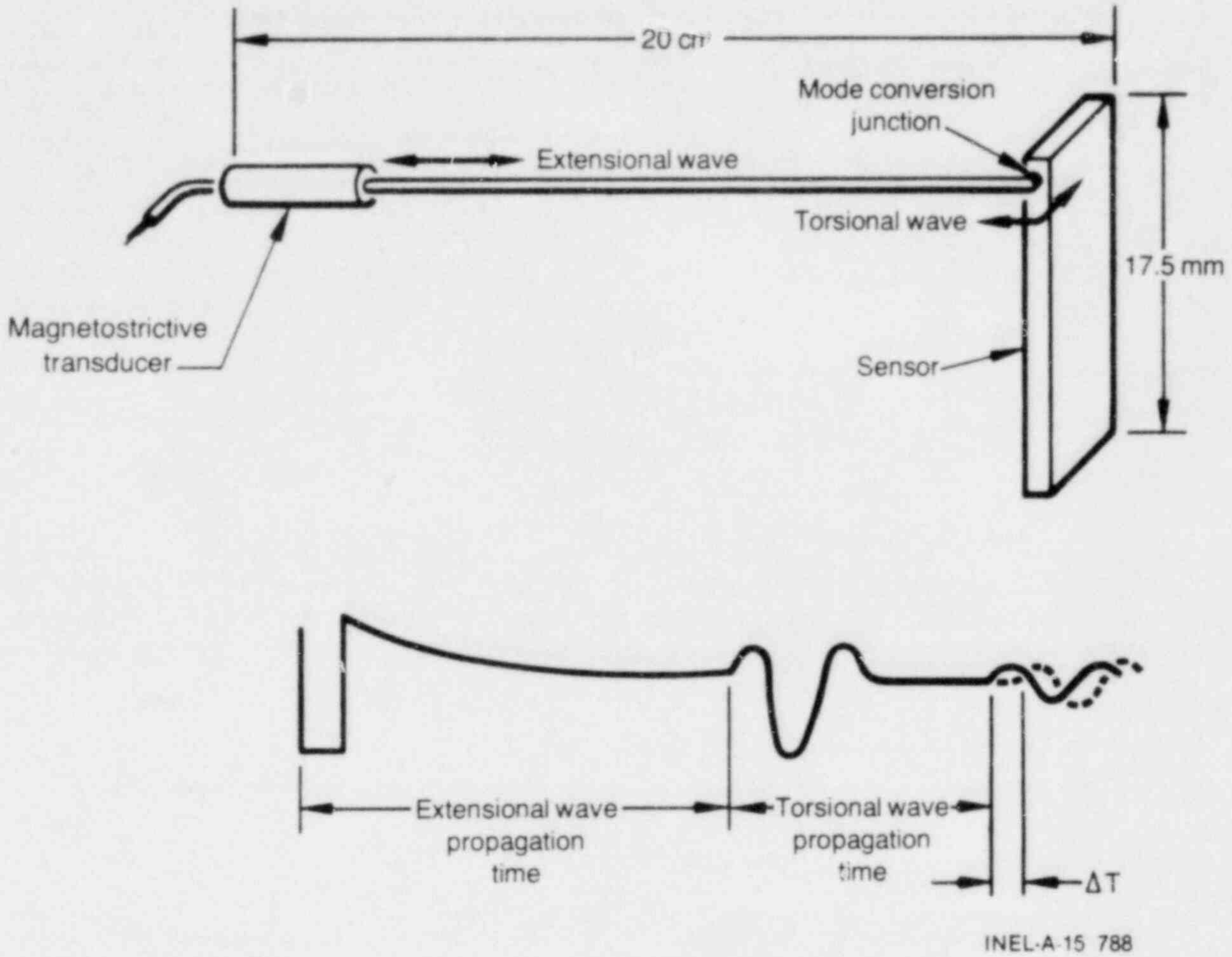


Figure 16. Local ultrasonic densitometer basic configuration.

density changes. Thus, a local density measurement results. No exposed ceramics are required.

Torsional wave propogation velocity is a function of sensor modulus of elasticity, which in turn is a function of temperature. This temperature dependence must be considered as part of the calibration model, which relates the total propagation time and the density of the medium under transient temperature conditions.[37,38] Incaloy 902 (Elinvar), as used in the LUD, has been shown to have one tenth the temperature sensitivity of previously used sensor materials. Both stainless steel and Elinvar were tested in autoclave corrosion tests for 500 h at 610 K. Oxide buildup and corrosion was insignificant. Proof-of-principle tests were conducted under steady state and transient conditions. Figure 17 is a picture of the spool piece containing three LUDs, as used in blowdown tests at the Water Reactor Research Test Facility. The spool piece was located upstream of a quick opening valve, and the blowdown results were compared to a three-beam gamma densitometer.

The three LUD responses are shown in Figure 18. Note that the upper sensor is first to indicate voiding, the center second, and the bottom last in the horizontal pipe. The lower sensor also senses liquid in the bottom of the pipe later in the blowdown (80 to 130 s). Figure 19 is the chordal average density at the top of the pipe and contains the same information as the top LUD response. The apparent increase in density (50 to 100 s) determined by the LUD is due to the velocity dependence, above 3 m/s, of the intrusive measurement. Lower velocities will be experienced in reactor applications, for which the sensor was developed and has primary use.

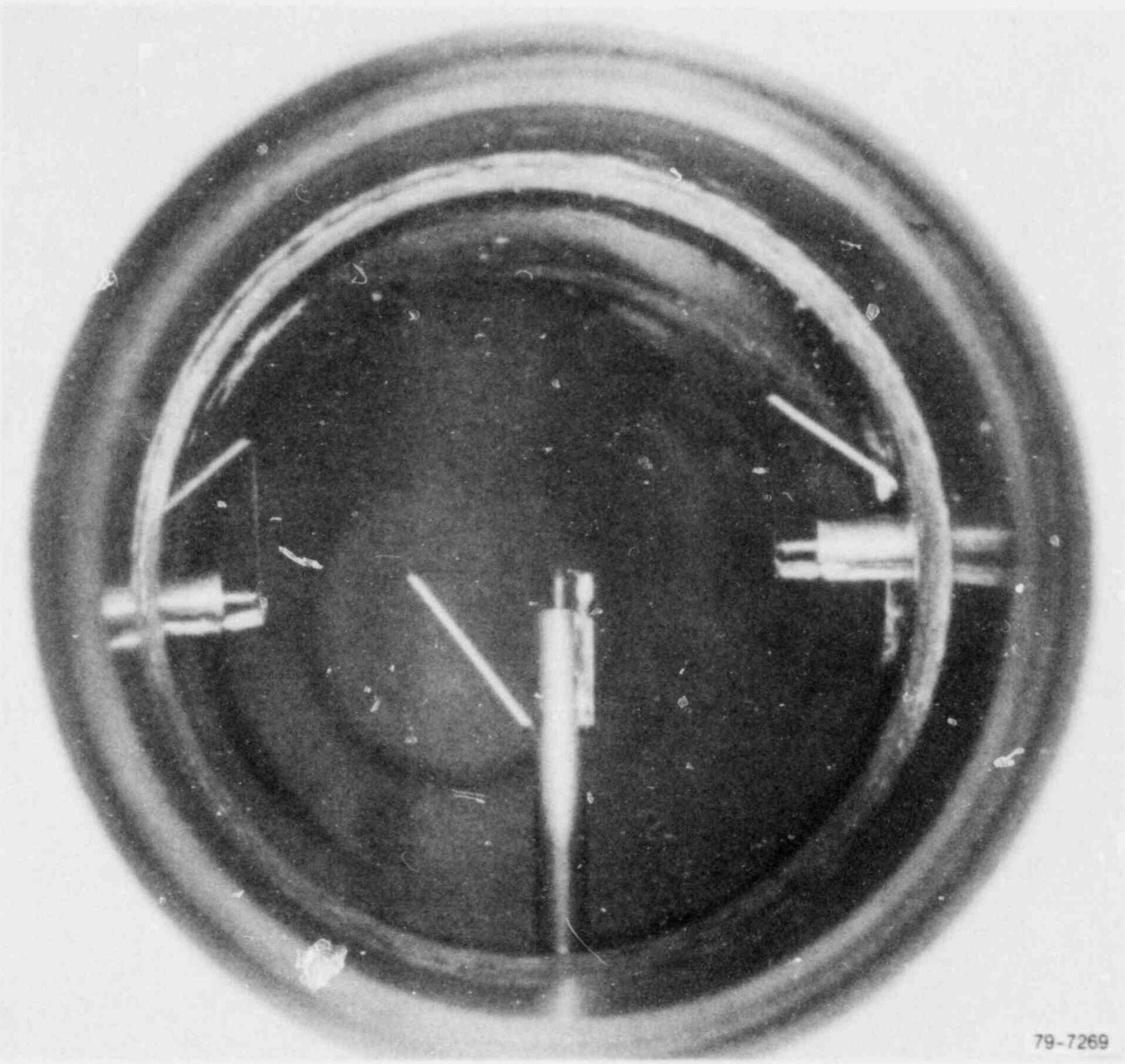


Figure 17. Local ultrasonic densitometer sensors in spool piece for transient steam-water tests.

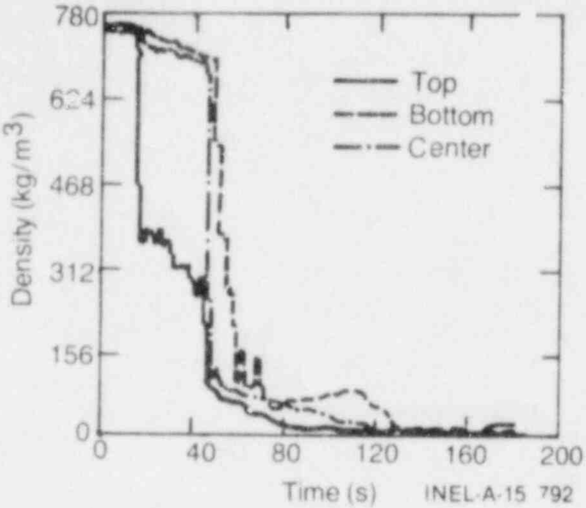


Figure 18. Response of LUDs during transient steam-water tests.

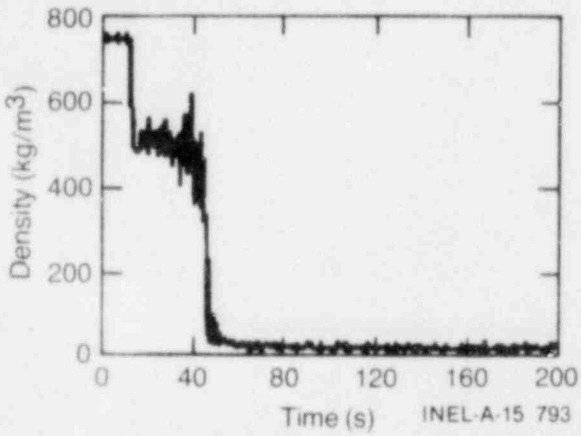


Figure 19. Top gamma densitometer data.

The use of ultrasonics has been demonstrated as a useful technique for determination of density in locations not accessible to photon densitometers. The LUD is small enough to be located in or on the fuel bundle support tube or other structure to monitor local density in the core region of a nuclear reactor. Development of the LUD has advanced the state of the art in the use of ultrasonics for the measurement of two-phase density.

VII. REFERENCES

1. K. E. Sackett and M. N. Arevalo, *Experiment Data Report for Semiscale Mod-3 System Small Break Test Series (Tests S-SB-4 and S-SB-4A)*, NUREG/CR-1293, EGG-2021, April 1980.

2. K. E. Sackett, *Experiment Data Report for Semiscale Mod-3 System Small Break Test S-07-10 (Baseline Test Series)*, NUREG/CR-1456, EGG-2035, June 1980.

3. D. H. Miyasaki, *Experiment Data Report for Semiscale Mod-3 System Small Break Test Series (Tests S-SB-2 and S-SB-2A)*, NUREG/CR-1459, EGG-2038, June 1980.

4. EG&G Idaho, Inc., *Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, October-December 1979*, NUREG/CR-1203, EGG-2012, January 1980, pp. 2-3.

5. EG&G Idaho, Inc., *Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, January-March 1980*, NUREG/CR-1490, EGG-2031, April 1980, pp. 2-5.

6. D. L. Reeder, *LOFT System and Test Description (5.5-ft Nuclear Core 1 LOCEs)*, NUREG/CR-0247, TREE-1208, July 1978.

7. R. S. Semken et al., *Reactivity Initiated Accident Test Series, RIA Scoping Tests Fuel Behavior Report*, NUREG/CR-1360, EGG-2024, April 1980.

8. D. J. Varacalle, Jr., et al., *PBF/LOFT Lead Rod Test Series Test Results Report*, NUREG/CR-1538, EGG-2047, July 1980.

9. T. R. Yackle et al., "An Evaluation of the Thermal-Hydraulic Response and Fuel Rod Thermal and Mechanical Deformation Behavior During the Power Burst Facility Test LOC-3," *1980 ANS/ENS Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 8-11, 1980.*

10. D. K. Kerwin, *Test PCM-5 Fuel Rod Materials Behavior*, NUREG/CR-1430, EGG-2033, May 1980.

11. R. W. Miller et al., "Influence of Gas Pressure and Composition on Fuel Temperature Observed in the USNRC/Halden Assembly IFA-430," *1980 ANS/ENS Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 8-11, 1980.*

12. A. D. Appelhans et al., "Fission Gas Release in LWR Fuel Measured During Nuclear Operation," *1980 ANS/ENS Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 8-11, 1980.*

13. A. D. Appelhans et al., *Cracking and Relocation of UO_2 Fuel During Initial Nuclear Operation*, NUREG/CR-1425, EGG-2032, May 1980.

14. M. S. El-Genk, *A Study of Molten Debris Freezing and Wall Erosion in the RIA-ST-4 Experiment*, NUREG/CR-1072, EGG-2030, April 1980.

15. American Nuclear Society, *Decay Heat Power in Light Water Reactors*, ANSI/ANS-5.1-1979, August 1979.

16. EG&G Idaho, Inc., *Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, October-December 1979*, NUREG/CR-1203, EGG-2012, January 1980, pp. 19-21.

17. L. Nilsson, L. Gustafson, and R. Hayer, *Experimental Investigation of Cooling by Top Spray and Bottom Flooding of a Simulated 64-Rod Bundle for a BWR*, Part 2, Studsvik/RL-78/59 Norhar S-046, June 30, 1978.

18. J. G. M. Anderson and C. L. Tien, "Radiation Heat Transfer in a BWR Fuel Bundle Under LOCA Conditions," *1979 ASME Winter Annual Meeting, New York, December 2-7, 1979*, ASME 79-WA/XX-00.

19. C. L. Tien et al., "Surface Radiation Exchange in Rod Bundles," *Transactions of the ASME, 101*, 1979, p. 378.

20. L. J. Siefken et al., *FRAP-T5: A Computer Code for the Transient Analysis of Oxide Fuel Rods*, NUREG/CR-0840, TREE-1281, June 1979.

21. R. H. Chapman, *Multirod Burst Test Program Quarterly Progress Report for April-June 1977*, ORNL/NUREG/TM-135, December 1977.

22. R. H. Chapman et al., *Effect of Creep Time and Heating Rate on Deformation of Zircaloy-4 Tubes Tested in Steam with Internal Heaters*, NUREG/CR-0345, ORNL/NUREG/TM-245, October 1978.

23. R. H. Chapman, *Multirod Burst Test Program Quarterly Progress Report for April-June 1976*, ORNL/NUREG/TM-74, January 1977.

24. R. H. Chapman, *Multirod Burst Test Program Progress Report for July-December 1977*, NUREG/CR-0103, ORNL/NUREG/TM-200, June 1978.

25. R. H. Chapman, *Multirod Burst Test Program Progress Report for January-March 1978*, NUREG/CR-0225, ORNL/NUREG/TM-217 August 1978.

26. R. H. Chapman, *Multirod Burst Test Program Quarterly Progress Report for January-March 1976*, ORNL/NUREG/TM-36, September 1976.

27. R. H. Chapman, *Multirod Burst Test Program Quarterly Progress Report for October-December 1976*, ORNL/NUREG/TM-95, April 1977.

28. R. H. Chapman, *Multirod Burst Test Program Quarterly Progress Report for January-March 1977*, ORNL/NUREG/TM-108, May 1977.

29. D. O. Hobson and P. L. Rittenhouse, *Deformation and Rupture Behavior of Light-Water Reactor Fuel Cladding*, ORNL-4727, October 1971.

30. H. M. Chung and T. F. Kassner, *Deformation Characteristics of Zircaloy Cladding in Vacuum and Steam Under Transient-Heating Conditions: Summary Report*, NUREG/CR-0344, ANL-77-31, July 1978.

31. D. W. Croucher, *Behavior of Defective PWR Fuel Rods During Power Ramp and Film Boiling Operation*, NUREG/CR-0283, TREE-1267, February 1979.

32. T. F. Cook, S. A. Ploger, and R. R. Hobbins, *Postirradiation Examination Results for the Irradiation Effects Test IE-5*, TREE-NUREG-1201, March 1978.

33. E. H. Karb, "Results of the FR-2 Nuclear Tests on the Behavior of Zircaloy Clad Fuel Rods," *6th NRC Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, November 7, 1978.*

34. K. Wiehr and H. Schmidt, *Out-of-Pile Versuche zum Aufblahvorgang von Zirkaloy-Hullen Ergenbnisse aus Vorversuchen mit verkurzten Brennstabsimulatoren*, KfK 2345, October 1977.

35. D. L. Hagrman, G. A. Reymann, and R. E. Mason, *MATPRO-11, Revision 1: A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior*, NUREG/CR-0497, TREE-1280 (Revision 1), February 1980.

36. U.S Atomic Energy Commission, *Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400, August 1974.

37. A. E. Arave and E. Fickas, "Progress Report on LOFT Ultrasonic Density Detector for Fuel Inlet Blowdown Measurements," *USNRC Review Group Meeting on Two-Phase Flow Instrumentation, Rensselaer Polytechnic Institute, Troy, New York, March 13-14, 1978*, NUREG/CP-0006.

38. A. E. Arave, "Ultrasonic Densitometer Development," *26th International Instrumentation Symposium, New Orleans, Louisiana, September 26-28, 1979*.