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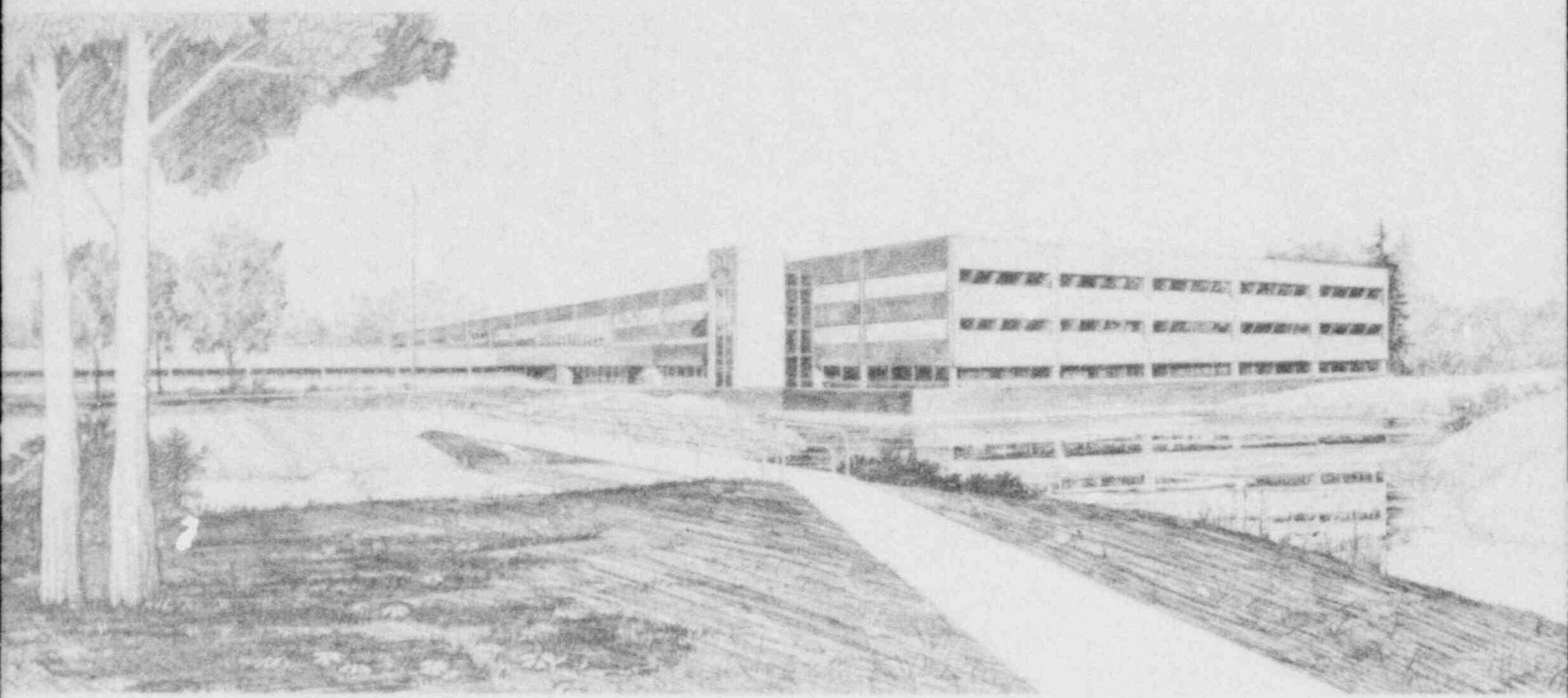

U.S. Department of Energy
Idaho Operations Office • Idaho National Engineering Laboratory

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research January—March 1980

April 1980

Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
8005290 566

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QUARTERLY TECHNICAL PROGRESS REPORT ON WATER REACTOR SAFETY PROGRAMS SPONSORED BY THE NUCLEAR REGULATORY COMMISSION'S DIVISION OF REACTOR SAFETY RESEARCH JANUARY—MARCH 1980

L. J. Ybarrondo, Director
Water Reactor Research

Published April 1980

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

ABSTRACT

Water reactor research performed by EG&G Idaho, Inc., during January through March 1980 is reported. The Semiscale Program conducted four small break loss-of-coolant tests to aid in evaluation of Semiscale system heat losses and provide data to the NRC for evaluation of small break analytical model capability to properly predict the effects of primary coolant pump opertion during a small break loss-of-coolant accident. The Loss-of-Fluid Test (LOFT) Experimental Program conducted the second nuclear experiment in its Small Break Test Series L3. The Thermal Fuels Behavior Program completed a power-cooling-mismatch/reactivity initiated accident test in the Power Burst Facility reactor, a fission gas release test 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 advanced computer codes (FRAPCON-2 and FRAP-T6) for predicting the steady state and transient behavior of light water reactor fuel rods. The Code Assessment and Applications Program characterized a data sample for fuel code assessment and initiated PWR and BWR analysis to identify and analyze accident sequences. Engineering Support Projects progressed in development of flow measurement instrumentation in the 3-D Experiment Project and in advanced instrumentation development, with the application of a heated differential thermocouple liquid level system.

PREFACE

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

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 (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)

SUMMARY

The Semiscale Program conducted four small break tests and continued analysis of these and previous small break tests. The Semiscale tests are conducted as part of an overall Nuclear Regulatory Commission (NRC) water reactor research program directed at improving and assessing analytical models used for evaluation of commercial pressurized water reactor (PWR) system transients. One test (Test S-SB-2A) was performed to evaluate a method for compensating for one of the scaling differences between Semiscale and commercial PWRs (the relatively large heat loss from the Semiscale system). An evaluation of the results of that test, compared to those from a previous test (Test S-SB-2) which did not include heat loss compensation by increased core power, indicates that increasing core power is a viable method of compensating heat losses until the core is uncovered, but not thereafter. The other three tests (Tests S-SB-P1, S-SB-P2, and S-SB-P7) were conducted as part of a revised Semiscale test series to aid the NRC in evaluating the best method of handling the primary coolant pumps during a small break loss-of-coolant accident.

The LOFT Experimental Program conducted Loss-of-Coolant Experiment (LOCE) L3-2 in the LOFT Facility. LOCE L3-2 was the third experiment in the Small Break Test Series and was designed to simulate a 1-inch-diameter break area in the primary system pipe in a commercial PWR. Analysis of the LOCE L3-2 data, now underway, is expected to result in better understanding of the thermal and hydraulic phenomena associated with the small break type of loss-of-coolant accident (LOCA), and it will provide the basis for development of analytical models that are used for licensing commercial PWRs. LOCE L3-2 was performed to determine how the primary coolant system responds during a small break when the break flow is nearly the same as the high pressure injection system flow. Although more mass exited the system than was anticipated, especially during the initial phase of the transient, the test objective was met. The cause of the additional mass loss is being investigated. An analysis of LOFT fuel module structural response was performed, in which data from LOFT large break experiments were compared with calculations from structural computer codes. This comparison was used to assess the capability of the codes for calculating core mechanical response under loss-of-coolant conditions in the LOFT Facility and to determine the actual mechanical response of the LOFT core. The comparison indicates that the measured response can be adequately calculated by the codes and that expected LOCE hydraulic forces will neither cause deformation of the LOFT fuel bundles nor disturb the normal gravity drop of the control rods.

The Thermal Fuels Behavior Program completed (a) the power-cooling-mismatch/reactivity initiated accident test (Test PR-1) in the Power Burst Facility, (b) a fission gas release test in the Halden Reactor in Norway, and (c) the second in a series of internal fuel rod fill gas composition tests with mixtures of xenon and helium in the Halden Reactor. Test PR-1 was performed to (a) evaluate test conditions leading to the onset of DNB and rewet for fresh fuel rods, (b) evaluate test conditions leading to the onset of DNB and rewet for rods with collapsed cladding,, (c) evaluate the potential for two-phase flow instabilities, and (d) evaluate the fuel pellet temperature distribution during low-energy reactivity initiated accident power excursions and provide additional data on collapsed, embrittled fuel and failure limits. The fission gas release test (Test FGRT-1) was performed to measure the release of xenon, krypton, and iodine from two LWR-type fuel rods during steady state operation at about 25 kW/m in the IFA-430 experiment. 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 Code Development and Analysis Program made progress in the development of computer codes for predicting the steady state and transient behavior of light water reactor fuel rods. This has involved developing a steady state code version with the Pacific Northwest Laboratory (PNL) and providing a link between a transient code version and the pressurized water thermal hydraulics code (TRAC) being developed at the Los Alamos Scientific Laboratory.

The Code Assessment and Applications Program characterized a data sample used for assessment of the FRAPCON-1 and FRAP/T5 fuel rod analysis programs. A new program was initiated to identify and analyze accident sequence of

events for both boiling water reactors and pressurized water reactors and to provide assistance to the NRC during commercial reactor transients such as occurred at Three Mile Island. A computer simulation and analysis of U.S. Standard Problem 9 was completed using the RELAP4/MOD6 computer code.

Engineering Support Projects comprises the 3-D Experiment Project and advanced instrumentation development. The 3-D Ecperiment 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. Instruments were also designed for the Japanese Slab Core Test Facility (SCTF). Advanced instrumentation development efforts were continued for two-phase fluid flow reference instrumentation, thermometry, radiation hardened optics, and in the use of a differential temperature liquid system in the PBF reactor.

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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 NRC in assessment and improvement of models for small break loss-of-coolant accidents. Test S-SB-2A was conducted to evaluate a method (augmentation of core power) to compensate for the atypically large heat losses in the Semiscale system. Results from analysis of that test are reported in Section 2.

In order to better aid the NRC in evaluating current licensing concerns, the Semiscale test program was revised to include six tests designed to provide data for evaluation of primary coolant pump operation during small breaks and a test to help evaluate the consequences of station blackout (loss of ac and dc power).

Tests S-SB-P1, S-SB-P2, and S-SB-P7, performed during this quarter, simulated a 2.5% break in the side of the broken loop cold leg piping with the pumps tripped at the time of initiation of core power decay, with the pumps running throughout the transient, and with the pumps running until a high system void fraction was achieved.

The remaining three tests to be performed to evaluate pump operation are similar to the three tests conducted, except that the break is in the hot leg pipe. Following the completion of these tests and the station blackout tests, the system will be modified to provide better small break simulation. These modifications will include a new, better scaled, intact loop pump and steam generator, insulation in the vessel, piping heat tracing, replacement of the electrical core simulator, and improved instrumentation.

2. PRELIMINARY ANALYSIS OF COMPENSATON FOR SYSTEM HEAT LC S BY AUGMENTED CORE POWER

T J. Fauble

Test S-SB-2A was conducted to evaluate a method (augmentation of core power) for compensating for large heat losses from the Semiscale system. To meet this objective, the initial conditions and operational parameters were, as nearly as possible, identical to those of a prior test (Test S-SB-2) which did not employ heat loss compensation. The conduct of Test S-SB-2 was reported in a previous quarterly report.[1] The major diference was that the transient core power was increased by an amount calculated to offset the energy lost from the primary system due to heat transfer through the piping and vessel walls. Test S-SB-2A, like Test S-SB-2, simulated a 2.5%[a] communicative break in the cold leg of a PWR. Test S-SB-2A was designed to simulate the sequence of events used in a code assessment PWR audit calculation,[2] as was Test S-SB-2. The sequence and timing of major events in Tests S-SB-2A and S-SB-2 are compared in Table 1.

Because of the relatively high surface area to volume ratio in the Semiscale system, heat losses from the system are significant. These heat losses are not typical of those in a PWR, in which heat loss to the environment is a negligible percentage of decay heat. To minimize the effects of the heat losses and thereby make the Semiscale small break test results more representative of a PWR small break LOCA, the transient core power was increased to offset the energy lost from the system due to heat losses.

Transient system heat losses from the Semiscale system during Test S-SB-2A, calculated using an experimental version of the RELAP4 computer code,[b] were used to establish the required core power augmentation during the test. However, once the core becomes uncovered, augmenting the core power does not directly offset system heat losses, but rather increases rod stored energy, causing atypically rapid heatup. Therefore, to avoid atypical temperature excursions, the power was reduced to the decay heat level when core uncovering was observed. The power was ramped down between 500 and 550 s after rupture, and thereafter no further attempt to compensate for heat losses was made. Figure 1 compares the core power decay profiles for Tests S-SB-2 and S-SB-2A.

Since the only significant difference between the operating conditions of Tests S-SB-2A and S-SB-2 was the augmented core power, all major changes in system behavior are attributed to this difference. The direct effect of the additional power was a slower pressure decay due to increased steam generation. The primary system pressure responses in Tests S-SB-2 and S-SB-2A are compared in Figure 2. The higher pressure in Test S-SB-2A caused the break flow rate to be somewhat higher, which resulted in a greater loss of primary coolant inventory early in the test. This led to more of the core being uncovered more rapidly in Test S-SB-2A. The slower depressurization in Test S-SB-2A also caused a delay in the initiation of accumulator injection relative to Test S-SB-2, which further promoted uncovering of the core. In Test S-SB-2, the core began to uncover about 600 s after rupture and only the upper 30 cm had uncovered by the time accumulator injection began to refill the core at about 660 s. In comparison, the core began to uncover about 440 s after rupture in Test S-SB-2A, and the upper half of the core had dried out by 730 s, when the accumulator began to refill the vessel. The highest cladding temperature reached in Test S-SB-2A was 805 K. No cladding temperatures higher than the initial values were observed in Test S-SB-2. In both tests, accumulator injection was sufficient to maintain the mixture level above the core until low pressure injection system flow was initiated, thus ensuring adequate core cooling. Although core power augmentation caused substantially more of the core to become uncovered, the overall system hydraulic behavior was not significantly affected. Both tests exhibited the same trends in loop flow rates and densities.

System hydraulic behavior was reasonably well calculated with the RELAP4/MOD7 code, but core mixture level was not accurately predicted, although the level trend was similar. As a result,

a. Percentage of total pipe flow area.

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

TABLE 1. Sequence of events in Tests S-SB-2 and S-SB-2A

Event	Time After Rupture (s)	
	Test S-SB-2	Test S-SB-2A
Rupture	0	0
Steam generator steam valves began closing	17.5	17.5
Core power decay initiated	21.0	20.5
Pump coastdown began	21.3	20.8
Steam generator feedwater valves began closing	26.0	25.5
System pressure reached hot leg saturation pressure	38.0	37.0
High pressure injection system flow initiated	46.0	45.5
Auxiliary feedwater flow initiated	80.8	80.2
Subcooled break flow ended	198	199
Augmented core power reduced	Not used	505 to 555
Accumulator injection initiated	645	725
Auxiliary feedwater terminated	1860	1860
Steam generator bleeding began	3955	3610
Low pressure injection system flow initiated	4300	4000
Test completed	4550	4400

the core heat transfer and resultant heatup after the core uncovered could not be predicted well. The predicted and actual core collapsed liquid levels for Test S-SB-2A are shown in Figure 3.

In conclusion, Test S-SB-2A showed that the core thermal behavior is sensitive to core power augmentation for the break size simulated, but that the overall system behavior is relatively unaffected. Consequently, core power augmentation appears to be an acceptable short-term means of compensating for atypical heat losses from the Semiscale system. However, core power augmentation cannot be used to offset system heat losses during a period in which the core is uncovered. The use of external pipe heaters is planned as an additional approach to reducing heat losses in future Semiscale tests.

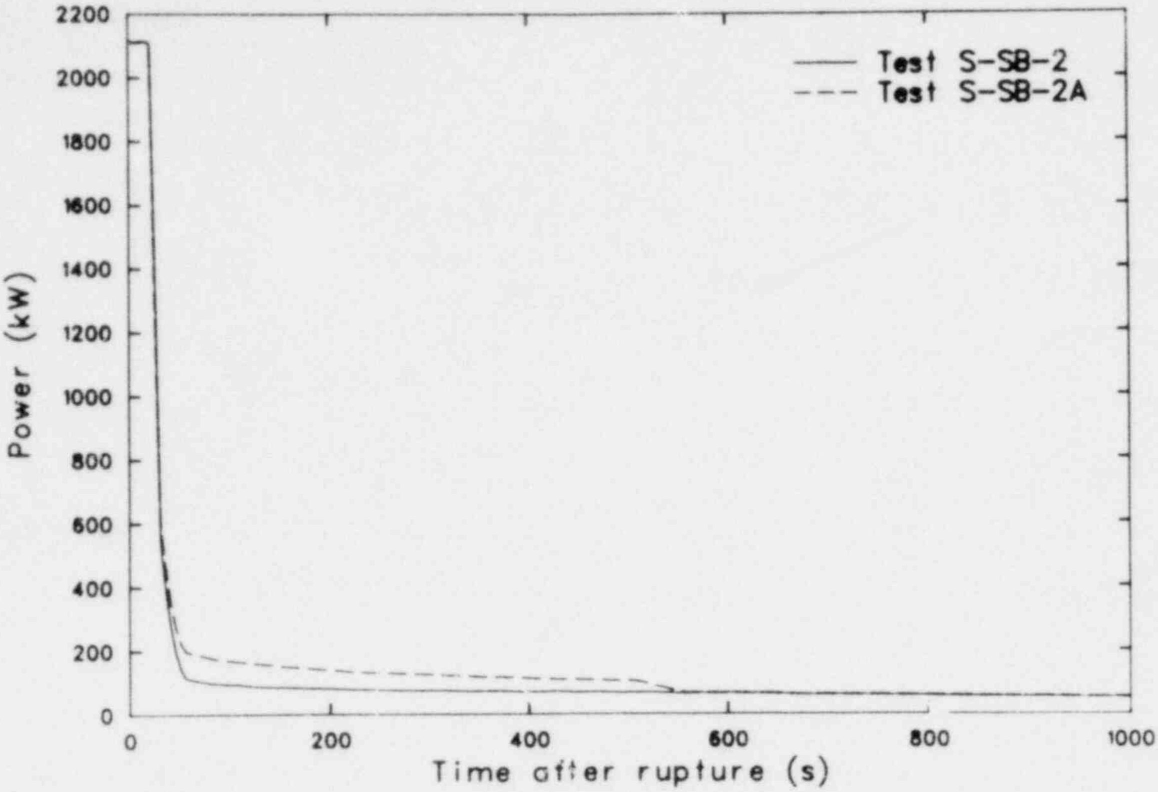

Figure 1. Core power decay in Semiscale Small Break Tests S-SB-2 and S-SB-2A.

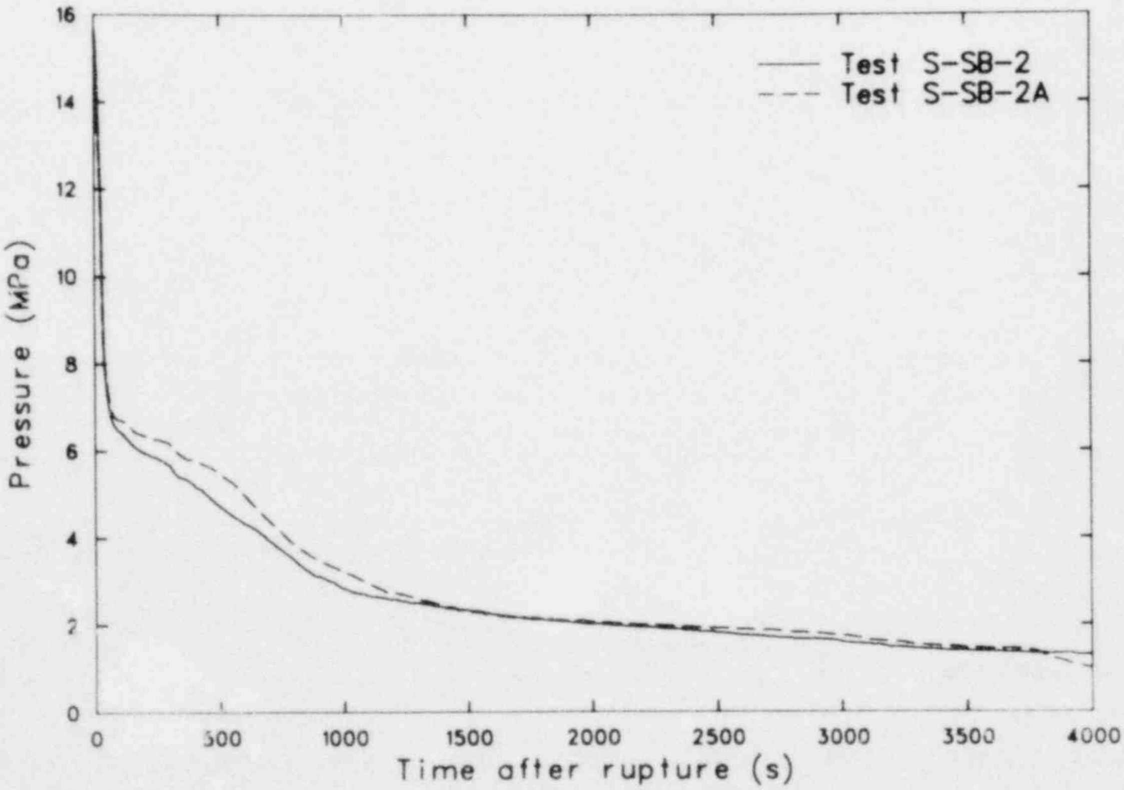

Figure 2. Primary system pressure response in Semiscale Small Break Tests S-SB-2 and S-SB-2A.

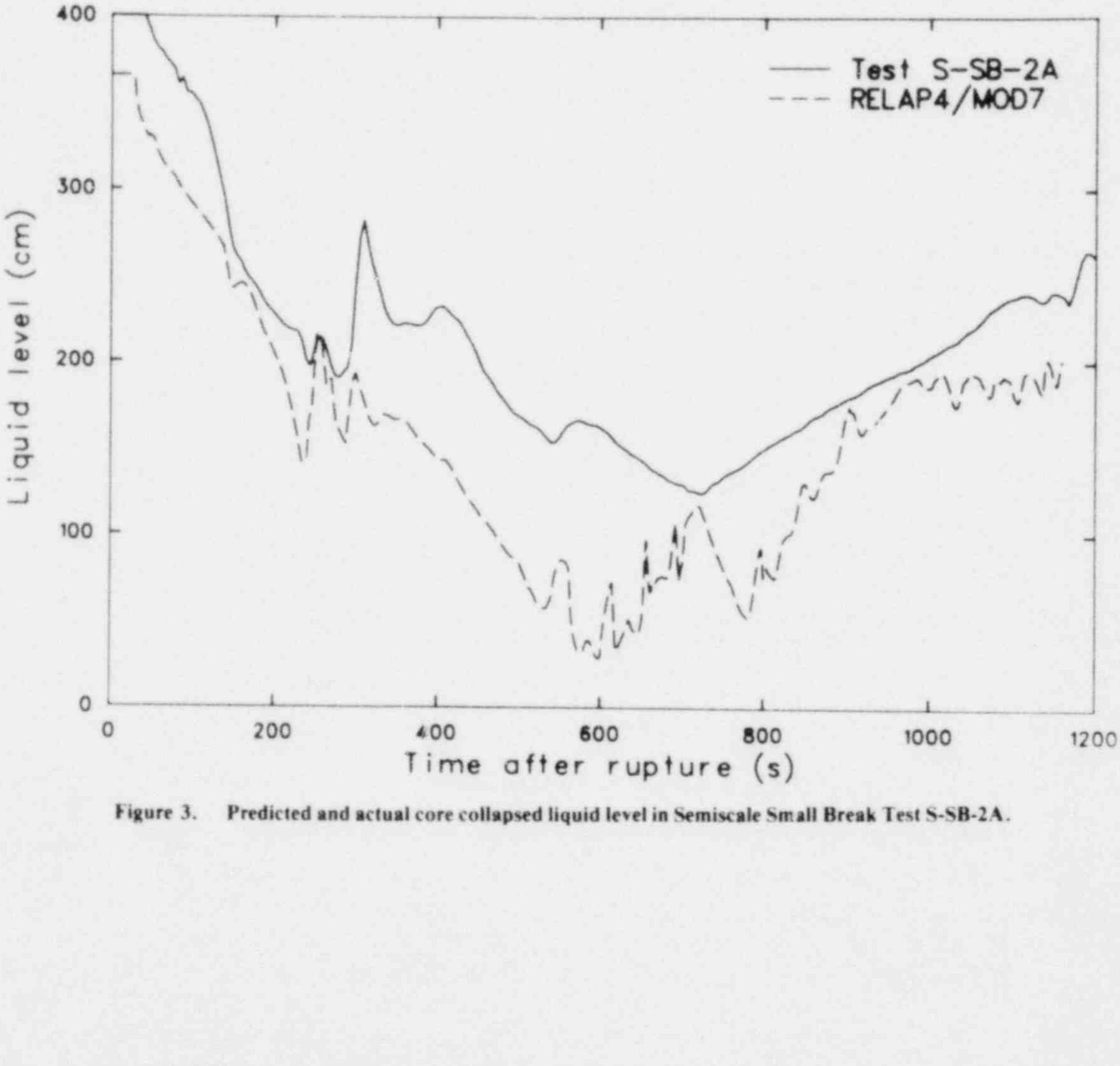

Figure 3. Predicted and actual core collapsed liquid level in Semiscale Small Break Test S-SB-2A.

II. LOFT EXPERIMENTAL PROGRAM

S. A. Naff, Acting Manager

The LOFT Experimental Program conducted Loss-of-Coolant Experiment (LOCE) L3-2 in the LOFT Facility.[3] LOCE L3-2, which was completed on February 7, 1980, was the third experiment in the Small Break Test Series (Series L3) and was designed to simulate a 1-inch-diameter break area in a primary system pipe in a commercial PWR. Analysis of the LOCE L3-2 data, now underway, is expected to result in better understanding of the thermal and hydraulic phenomena associated with this type of LOCA and to provide the basis for development and assessment of analytical models that are used for licensing commercial PWRs.

An objective of the Small Break Test Series is to investigate the primary coolant system response during a small break when the break flow is greater than high pressure injection system (HPIS) flow and when break flow is nearly the same as HPIS flow. The specific objective of LOCE L3-2 was to determine how the primary coolant system responds when the break flow is the same order of magnitude as the HPIS flow, as system pressure stabilizes late in the transient prior to the initiation of accumulator flow. Although, more mass exited the system than was anticipated, particularly during the initial phase of the transient, the test objective was achieved. The cause of the additional mass loss is being investigated.

An analysis of the LOFT fuel module structural response was performed in which data from LOFT large break Experiments L1-5, L2-2, and L2-3 were compared with calculations obtained from the structural computer codes WHAM,[4] SHOCK,[5] and SAP.[6] This comparison was used to assess the capability of the codes to predict core mechanical response under loss-of-coolant conditions in the LOFT Facility and to determine the actual mechanical response of the LOFT core. The comparison indicates that the measured response can be adequately calculated by the codes and that expected LOCE hydraulic forces will not cause residual deformation of the LOFT fuel bundles or disturb the normal gravity drop of the control rods.

1. LOFT NUCLEAR LOCE L3-2

J. H. Lineberger

LOFT LOCE L3-2 was designed to simulate a 1-inch-diameter break area in the primary system of a commercial PWR. LOCE L3-2 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 loss-of-coolant accident in a commercial PWR. A detailed description of the LOFT system is provided in Reference 3.

The LOFT system conditions at experiment initiation were: a maximum linear heat generation rate of 52.2 ± 3.7 kW/m (simulating the maximum expected in a commercial PWR, approximately 130% of nominal 100% power conditions in a PWR), an average temperature of 567 ± 3 K, a hot-to-cold leg differential temperature of 18 ± 3.5 K, a flow rate to system volume ratio of 61.4 ± 1.2 kg/m^3·s, and a system pressure of 14.85 ± 0.04 MPa.

The experiment started with the opening of the quick opening blowdown valve in the broken loop cold leg. Thirteen seconds later, the reactor scrammed on a low system pressure signal. The primary coolant pumps were tripped, immediately after the reactor scrammed, and coasted down. As planned, the operator intervened later in the transient by bleeding steam from the secondary system to increase the primary system depressurization rate. After the system fluid became subcooled and once the purification system limits were reached, the system was taken to a cold shutdown condition by the purification system, and the experiment was terminated. Table 2 contains the sequence of major events for LOCE L3-2, including predictions of the sequence.

The primary system pressure dropped rapidly, after the reactor scrammed, until about 200 s. At 200 s system pressure control passed from the pressurizer, which had emptied, to the reactor

TABLE 2. Chronology of events for LOCE L3-2

Event	Time After LOCE Initiation (s) LOCE L3-2 Data	RELAP5[a,b] Prediction	RELAP4[a,c] Prediction
Reactor scrammed	12.92 ± 0.10	94.0	45.8
Control rods reached bottom	14.98 ± 0.10	Not calculated	47.8
Primary coolant pumps tripped	16.90 ± 0.10	94.0	47.8
HPIS injection initiated	33.84 ± 0.10	127.0	88.0
Primary coolant pumps coastdown completed	35.0 ± 1.0	Not calculated	60
First indication in core of natural loop circulation	36.0 ± 2.0	Not calculated	Not calculated
Secondary coolant system auxiliary feed pump started (initial steam generator fill)	114.0 ± 1.0	154.0	112.6
Pressurizer emptied	136.0 ± 7.0	400.0	359.0
Upper plenum fluid reached saturation temperature (end of subcooled blowdown)	180.0 ± 1.0	450.0	440.0
End of subcooled break flow[d]	650 to 800	—	—
Secondary coolant system auxiliary feed pumps tripped (terminated initial steam generator fill)	1 878.0 ± 1.0	1 954.0	1 913.0
Secondary coolant system steam bleed initiated	4 118.0 ± 1.0	3 600.0	—
HPIS flow ≥ break flow	4 200.0 ± 10	4 350.0	—

TABLE 2. (continued)

Event	Time After LOCE Initiation (s) LOCE L3-2 Data	RELAP5[a,b] Prediction	RELAP4[a,c] Prediction
Accumulator injection initiated	5 029.0 ± 4.0	7 200.0	–
Primary system fluid became subcooled	8 200 ± 50	–	–
Purification system cooldown initiated[e]	12 300 ± 60	–	–
LPIS injection initiated	21 418 ± 5	–	–
Experiment completed[f]	23 350 ± 100	–	–

a. RELAP4 calculation terminated at 3600 s, RELAP5 at 7800 s.

b. The experimental RELAP4 code used was RELAP4/MODG, Version 92, (experimental version of RELAP4/MOD7), Idaho National Engineering Laboratory Configuration Control Number H00718B. The new object deck, which includes changes to correct known coding errors and to incorporate the LOFT steam valve control logic into the code, was RLP4G92LFT04, Idaho National Engineering Laboratory Configuration Control Number H01168IB.

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

d. Subcooled, critical break flow continued throughout the transient in RELAP4 and RELAP5 calculations.

e. From experiment log.

f. End of experiment is defined as time at which system temperature dropped below 366.5 K.

vessel as fluid in the hot regions of the core and upper plenum started to flash to vapor. The pressure control transition significantly moderated the depressurization rate until the rate increased again after 4000 s, when secondary steam bleeding was initiated. Measured and calculated system pressure histories are shown in Figure 4.

Natural loop circulation started just after the primary coolant pumps coasted down. By 2000 s natural loop circulation could no longer be measured. However, positive core outlet flow continued and the steam generator continued to function effectively as a heat sink for the system. Between 2000 and 8000 s a negative temperature gradient developed in the primary system coolant

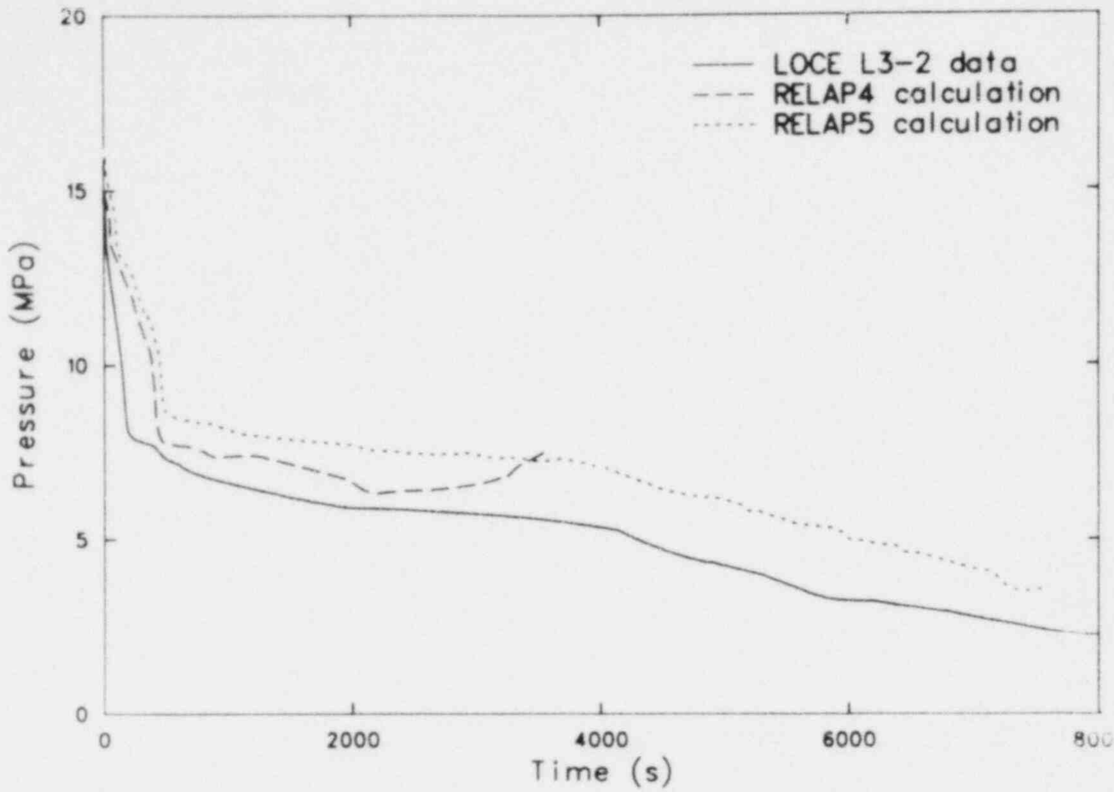

Figure 4. Comparison of calculated and measured primary system pressures for LOCE L3-2.

between the steam generator inlet and outlet. This negative temperature difference indicates that another mode of cooling, other than natural loop circulation, may have occurred. The other mode may have been reflux cooling.

By 4200 s HPIS flow equalled or exceeded the break flow (prior to accumulator initiation), and the net system fluid inventory started increasing. By 8500 s measureable natural loop circulation was restored as the fluid in the reactor vessel became subcooled, but long before the plant became liquid full. Significant fluid stratification in the system continued from that time until the end of the transient.

The combined effects of adequate system fluid inventory (the liquid level in the system did not fall below the bottom of the reactor vessel nozzles) and effective heat removal in the steam generator produced these results:

1. The fuel remained covered with fluid and cool
2. The system continued to depressurize throughout the experiment
3. As predicted by the RELAP5 calculation (Figure 4) the operator-initiated steam bleeding was effective in increasing the rate of system depressurization.

The break flow measured during the experiment is compared in Figure 5 with the flow calculated to leave the system through the break orifice. Measured break flow exceeded the calculated flows, particularly during the early portion of the transient. The system depressurization rate (Figure 4) was also underpredicted and the reactor scram and pressurizer emptying were calculated to occur much later than actually happended, as indicated in Table 2. Consequently, the mass exiting the system was also underpredicted.

The system break flow shown in Figure 5 was calculated from suppression tank liquid level measurements. The result was confirmed, early in the transient, by calculating system mass flow from pressurizer liquid level data. The result was confirmed later in the transient by comparing the time that the emergency core cooling flow exceeded the break flow with the measured fluid density in the cold legs.

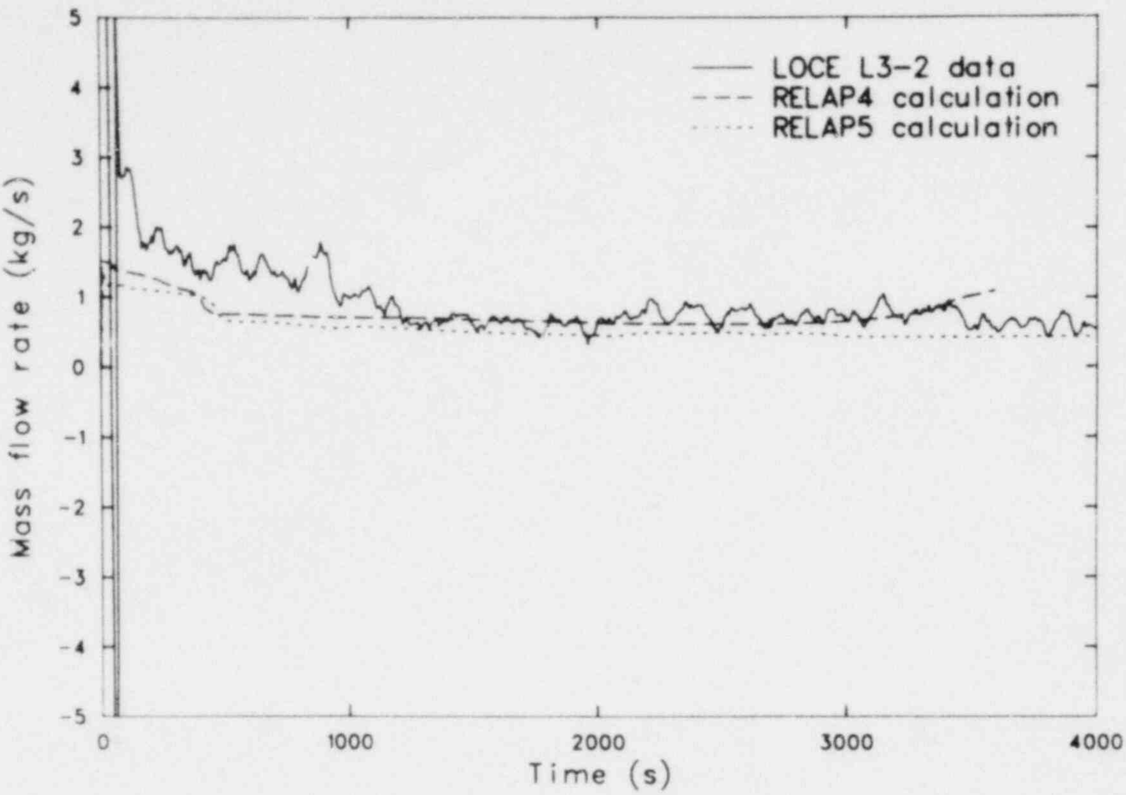

Figure 5. Comparison of calculated and measured break mass flow rate for LOCE L3-2.

The cause or causes of the excess mass flow during the experiment are being investigated. The two most probable causes, actively being pursued, are (a) flow from the system, other than through the break orifice, into the suppression tank; and (b) miscalculation of break flow through the break orifice.

Posttest analysis of the LOCE L3-2 data is continuing. Conclusions based on the results of analyses completed thus far include:

1. The core remained covered during the entire transient. No fuel rod damage resulted.

2. The steam generator was an effective heat sink throughout the experiment, even though natural loop circulation could not be measured for 6500 s, starting about 2000 s into the transient.

3. Another cooling mode may have occurred in the steam generator during the period that natural loop circulation could not be measured.

4. Fluid in the system became subcooled long before the plant became liquid full

5. Measurable natural loop circulation was reestablished as the vessel refilled and the system fluid became subcooled.

6. Secondary steam bleeding was effective in reducing primary system pressure.

7. HPIS flow equaled or exceeded break flow about the time secondary system steam bleeding was initiated.

8. The mass leaving the system early in the transient was significantly greater than anticipated.

9. Computer calculations predicted the dominant phenomena, in the proper time sequence, except for the larger-than-anticipated mass flow from the system.

2. LOFT FUEL MODULE LOCA DECOMPRESSION STRUCTURAL RESPONSE ANALYSIS

B. F. Saffell and M. L. Russell

An analysis of the LOFT fuel module structural response during the decompression phase of large break loss-of-coolant accident experiments has been performed. The analysis program included development of computer code models for pretest calculation of expected LOCE structural response and evaluation of the experiment data. The LOFT test data indicate that conventional fuel bundle structural analysis techniques are valid for predicting LOCA effects and that LOCA hydraulic forces do not cause residual deformation of the LOFT fuel bundles or disturb the normal gravity drop of the control rods.

The LOFT core consists of six instrumented fuel modules and three noninstrumented modules assembled in a 3 x 3 array configuration. Each module includes the fuel bundle (core section), the upper support structure, and the instrumentation penetration. The fuel bundles are modeled after a typical commercial 15 x 15 fuel-rod-array fuel assembly design, except that stainless steel guide tubes are used to improve column strength during blowdown loading. Also, the active fuel length is only 1.68 m due to reactor size constraints.

Three large break loss-of-coolant experiments have been conducted with the LOFT reactor core in place. The tests (LOCEs L1-5, L2-2, and L2-3) were all 200% cold leg break experiments. LOCE L1-5 was performed with no heat generation by the nuclear core. LOCE L2-2 was initiated at 50% reactor power (26.4 kW/m maximum linear heat generation rate), and LOCE L2-3 was initiated at 75% reactor power.

A combination of the WHAM[4], SHOCK[5], and SAP[6] computer codes was developed to analyze the fuel module mechanical response during the LOCE decompression.

Different and distinct input models of the LOFT internals are required fo ach analysis step. An hydraulic model of the pi ary coolant system is used as input to the WHAM computer code, an internals system structural model is input to the SHOCK code, and a detailed fuel bundle model is employed in SAP-IV.

A holddown spring in the SHOCK lumped mass model corresponds tc the holddown spring at the top of the LOFT center fuel module. Two linear variable differential transformers (LVDT) measure the axial displacement across the holddown spring. The displacement data are compared with the SHOCK displacement predictions for the subcooled blowdown portion (0 through 0.2 s) of LOCE L1-5 in Figure 6. The SHOCK model

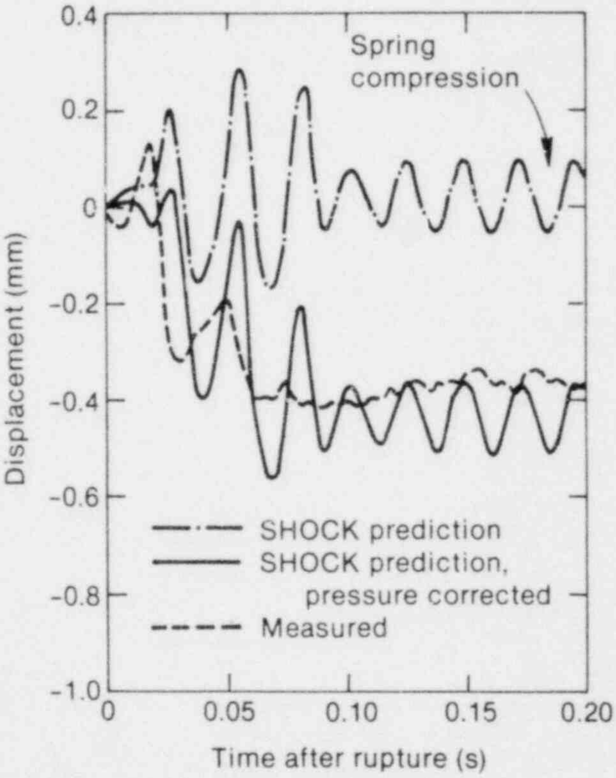

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Figure 6. Center fuel module holddown spring displacement during LOCE L1-5 subcooled blowdown (uncertainty analysis for displacement data given in Reference 7).

assumes the reactor head is stationary. Therefore, an experimentally determined pressure correction factor is applied to the SHOCK prediction to account for the actual relaxation of the head components during the rapid decompression. The

SHOCK code reasonably predicted the displacement frequency and the peak-to-peak displacement amplitude of the initial response cycle. The rapid attentuation of the measured displacement frequency is due to either or both of the following: (a) higher attenuation of the forcing function or (b) greater mechanical damping. The measurements of system decompression and fuel module axial motion show that LOCEs L2-2 and L2-3 created progressively less severe loads on the fuel modules during the decompression because the upper plenum and core regions were at higher initial temperature conditions.

The stresses on fuel bundle individual components and welds were analyzed using the SAP-IV computer model. For the LOCE L1-5 conditions, the maximum loads from the SHOCK analysis were applied to the SAP model. The resulting predicted stresses are a maximum of 60% of those allowed by the American Society of Mechanical Engineers Code. The actual LOCE L1-5 stresses are believed to have been lower than calculated since the displacement test data show the SHOCK loading inputs on the fuel to be conservatively high.

Structural deformation of the fuel bundle guide tubes would affect the control rod drop time envelopes during reactor scram. Individual control rod positions are monitored by use of magnetic reed switches located above the control rod drive motors. The control rods drop during the saturated phase of the decompression which occurs after the more violent subcooled decompression has ended. During the three large break LOCEs, all rods fell within the time envelope expected, if zero flow conditions existed. There were no indications of mechanical interaction caused by guide tube deformation or lateral fuel module motion.

The LOFT control rod downward motion is hydraulically stopped by a dashpot located in the control rod drive mechanism rather than by the normal PWR necked-down guide tube sections. A RELAP computer analysis was performed to determine the local guide tube coolant conditions during blowdown. The analysis indicates that sufficient liquid remains within the guide tubes during the control rod drop period to provide the hydraulic damping, if necked-down guide tubes were used.

III. 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. One test remains in this series.

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 Severe Fuel Damage 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 experiment 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.

During the past quarter, the Thermal Fuels Behavior Program completed (a) a power-cooling-mismatch/reactivity initiated accident test (Test PR-1) in the Power Burst Facility (b) a fission gas release test (Test FGRT-1) in the Halden Reactor in Norway and (c) the second in a series of internal fuel rod fill gas composition tests with mixtures of xenon and helium in the Halden reactor. Test PR-1 was performed to (a) evaluate test conditions leading to the onset of departure from nucleate boiling (DNB) and rewet for fresh fuel rods, (b) evaluate test conditions leading to the onset of DNB and rewet for rods with collapsed cladding, (c) evaluate the potential for two-phase flow instabilities, and (d) evaluate the fuel pellet temperature distribution during RIA power excursions and provide additional data on fuel rod failure limits. Test FGRT-1 was performed to measure the release of xenon, krypton, and iodine from two LWR-type fuel rods during steady state operation at about 25 kW/m in the IFA-430 experiment. 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.

1. PBF TESTING

P. E. MacDonald and R. K. McCardell

Test PR-1 was conducted and preliminary results were compiled in a quick look report; the results of the test, described in the following sections, are being analyzed. Results of Test RIA 1-1 were analyzed and a draft fuel behavior report completed. Experiment specifications were also prepared for Test OPTRAN 1-2.

Other accomplishments include the completion of the LOFT Lead Rod and LOC-3 postirradiation examinations and preparation for another series of blowdown tests (TC-2 Tests) to investigate thermocouple effects during a LOCA.

1.1 PCM Test Series—Test PR-1 Description

D. T. Sparks

To interpret the behavior of light water reactor fuel rods during postulated accident events requires an understanding of the phenomena and an ability to model the processes which dominate the fuel rod response during such events. To license a light water reactor requires that the applicant ensure either that adequate thermal margin exists to allay the consequences of such an accident scenario, or ensure, on the basis of an acceptable damage criterion, that significant damage would not occur. The Power-Cooling-Mismatch Test Series is being conducted in the Power Burst Facility (PBF) for the U.S. Nuclear Regulatory Commission to provide modeling and damage information on a spectrum of power-cooling imbalance events. Data from the test series will be used to help evaluate conservatisms in the current thermal margin criteria, and provide input data for development and assessment of computer models used to calculate fuel rod response under a range of transient conditions.

Test PR-1 was originally designed to provide fuel rod thermal response data under steady state and power oscillation conditions. The test objectives were subsequently expanded to include boiling transition and rewet information under PCM conditions and to include fuel temperature distributions during RIA power excursions. The preliminary results of Test PR-1 are described in this section.

1.1.1 Test Design. Test PR-1 was conducted with four BWR-type test fuel rods, identified as Rods 524-1, 524-2, 524-3, and 524-4. The active fuel length of each test rod was 0.914 m and the rod plenum volume was sized in proportion to the active fuel volume. Rods 524-1, 524-2, and 524-3 were backfilled with helium, and Rod 524-4 was backfilled with argon to allow comparison of the effect of fuel rod fill gas composition on fuel rod thermal response. The fuel densities of the four test rods were also varied to provide comparative data on the effect of fuel density during each phase of the test. The four test rods were contained in individual flow shrouds and symmetrically positioned within a test train in the PBF in pile tube.

Each test rod was instrumented with thermocouples to measure cladding surface temperature, fuel pellet centerline temperature, and off-center fuel temperature. In addition, Rod 524-4 was instrumented with cladding internal thermocouples to provide information on rewetting from film boiling conditions. The internal pressure and cladding elongation of each rod was also measured.

Coolant environmental conditions within each individual flow shroud were monitored using flowmeters and thermocouples. The coolant system pressure was also monitored within the Test PR-1 test train.

1.1.2 Test Conduct. Nuclear operation for Test PR-1 included (a) a period to evaluate fuel thermal response under steady state and power oscillation conditions; (b) a period to evaluate boiling transition, return to nucleate boiling, and the potential for two-phase instabilities; and (c) a period to assess fuel temperature distributions and to obtain rod failure information during RIA power excursions. Environmental conditions during each testing period were adjusted to support the objectives of each phase.

Steady state thermal response data were obtained during a two-segment power calibration and preconditioning period. The power calibration provided relationships between the test rod linear heat generation (from a system heat balance), thermal neutron flux (from self-powered neutron detectors), and PBF core power. The relationship between test rod power and neutron flux provided a method of determining test rod power when two-phase exit conditions, such as existed during the PCM transients, made an energy balance impractical. The preconditioning phase provided a period of operation to allow pellet cracking and relocation and an evaluation of these effects on thermal response.

Thermal response information during power oscillations was obtained by sinusoidally oscillating core power 20% at eight nominal power levels and by recording the relative phase lag between power and measured temperatures. Highly subcooled conditions were maintained during the power oscillations, with an inlet temperature of 478 K, system pressure of 7.17 MPa, and coolant volumetric flow rate of 0.5 L/s through each shroud.

The boiling transition and rewet test phase consisted of 23 flow reduction and 2 power increase PCM transients. Environmental conditions were varied between BWR (7.0 MPa) and PWR (15.5 MPa) system pressures at relatively constant inlet subcooling (14 K). The transients were conducted at test rod peak powers between 40 and 53 kW/m.

A total of three RIA power bursts were conducted with radial average fuel enthalpies at the peak elevation of 440, 523, and 754 J/g. The bursts were initiated from low power (200 W) at coolant conditions of 538 K inlet temperature, 6.45 MPa system pressure, and 0.107 L/s volumetric flow rate through each flow shroud. Prior to each RIA power burst, a short power calibration was performed to relate test rod power to reactor core power and control rod positioning.

1.2 Results of Test PR-1

D. T. Sparks

Thermal response data obtained during Test PR-1 complement similar data from previous gap conductance (GC) test series experiments. Since the Test PR-1 hardware was originally designed for a GC series experiment, instrumentation was optimally positioned for these measurements. The effect of fuel density variations in the helium filled test rods was expected to be small on the basis of previous test results. The minor variations noted in Test PR-1 are minimal and may be associated with centerline and off-center temperature measurement uncertainties. The effect of fill gas composition, however, was pronounced. The difference in measured fuel centerline temperatures between the helium filled test rods and the argon filled test rod are shown in Figure 7. For graphical clarity, the fuel centerline temperature data from the three Test PR-1 helium filled rods (Rods 524-1, 524-2, and 524-3) were averaged. Shown for comparison in Figure 7 are data from previous experiments on similar test rods (Rods GC 503 and GC 504). As expected, fuel temperatures in the argon filled rod were noticeably higher than the helium filled rods due to the lower thermal conductivity of argon.

Power-cooling-mismatch transients were conducted at system pressures between 7 and 15.5 MPa, with test rod peak powers between 40 and 53 kW/m. A total of seven flow reduction transients (each at constant test rod power) were conducted at low pressures, between 7 and 8 MPa, at a coolant inlet temperature of about 544 K. No discernible indications of boiling transition were observed. Either natural circulation was sufficient to preclude boiling transition, or the low temperature excursion associated with a high quality dryout transition was not detectable with the Test PR-1 instrumentation.

Eighteen PCM-type transients were conducted at system pressures between 13 and 15.5 MPa. The coolant inlet temperature at each pressure was adjusted to provide a nearly constant inlet subcooling (14 K). At least 13 of the transients resulted in detectable boiling transition on the rods. Rewet was induced by three methods: (a) increasing flow and decreasing power simultaneously, (b) increasing flow rate at constant power, or (c) decreasing power at constant flow

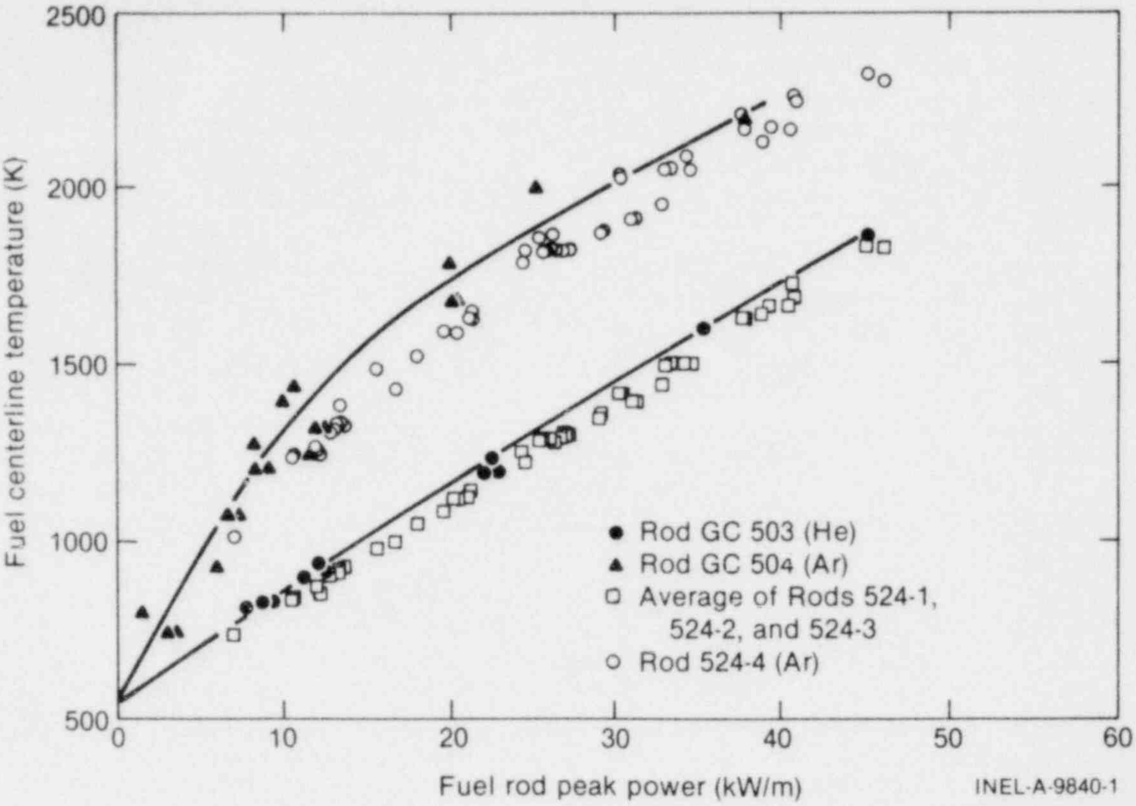

Figure 7. Composite plot of fuel centerline temperature measurements showing effect of fill gas composition (helium or argon) in rods having 0.22-mm initial diametral gap. Rod powers are averaged for the four Test PR-1 rods at the peak power elevation. Temperatures for helium filled Test PR-1 rods (Rods 524-1, 524-2, and 524-3) were averaged.

rate. One rod, Rod 524-1, failed during the boiling transition cycles. The rod likely failed due to embrittlement following extended high temperature operation.

Progressively severe RIA power excursions were performed at approximate peak fuel enthalpies of 440, 523, and 754 J/g. Reactor periods to attain these energies were 42.7, 8.7 and 6.2 ms, respectively. The measured fuel centerline and off-center temperatures during the first (lowest energy) power burst are shown in Figure 8. The reactor core power is also shown for reference. Pretest calculations indicated that the fuel temperatures should increase more rapidly than were measured during the two higher energy power bursts. The time delay may be associated with the thermocouple response time rather than an inherent delay in temperature increase. Film boiling was observed following each power burst, with measured cladding temperatures greater than 1650 K following the highest energy burst. Data from Rod 524-2 indicated rod failure during the final (highest energy) power burst.

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.

Fission product release was monitored during Test PR-1. In this experiment the release from a failed rod was measured under a variety of reactor maneuvers including power ramps, scrams, and DNB and RIA transients. The fission product detection system software was upgraded to expedite conversion of the spectral data to plots of isotope concentrations and inventories as a function of time.

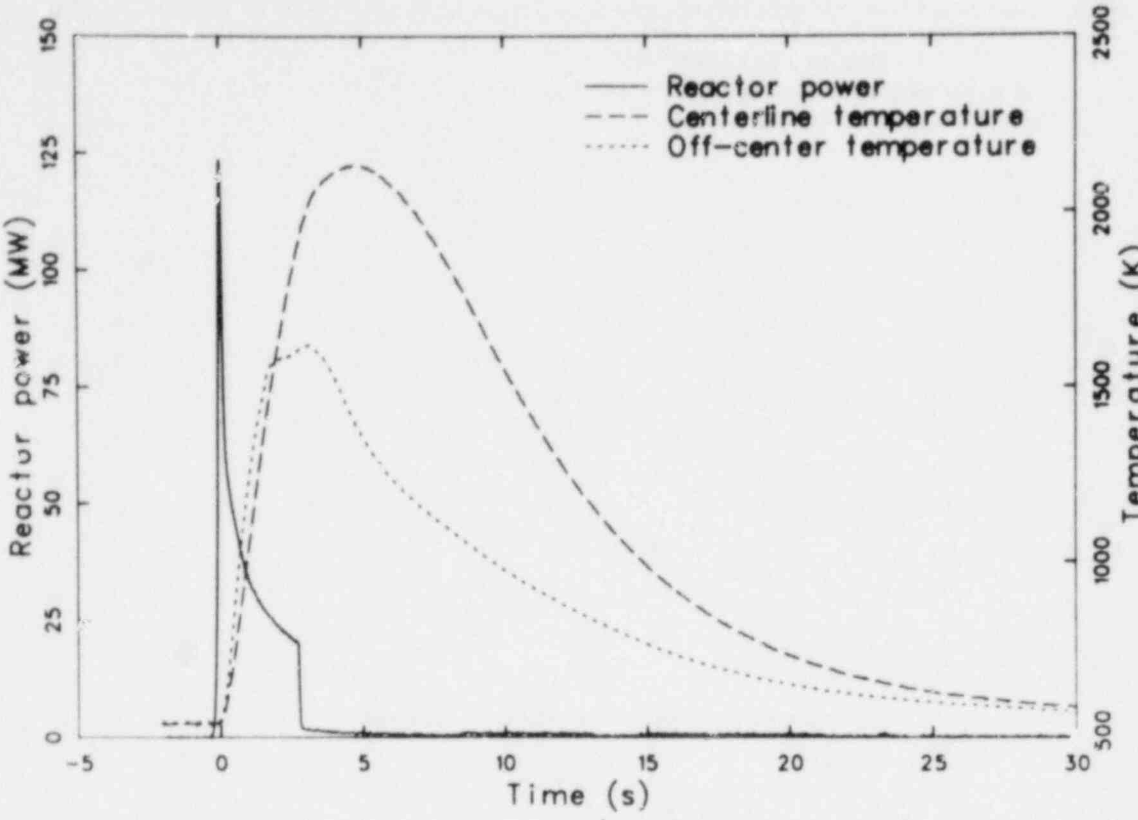

Figure 8. Fuel centerline and off-center temperatures measured during Test PR-1, Power Burst 1, for Rod 542-2.

Two tests were performed in the IFA-430 gas flow assembly in the Halden Reactor in Norway. The IFA-430 gas flow assembly consists of two LWR-type fuel rods connected to a gas supply which permits fill gases of various compositions and pressures to be introduced into the rods. Valves permit the rods to operate either with a static gas content or with gas flowing through the rods past a fission product monitor. With this assembly, Test FGRT-1, a fission gas release test, was performed to measure the release of xenon, krypton, and iodine during steady state operation at about 25 kW/m. Future gas release tests will be performed at different rod powers and at various fuel temperatures.

Also, in IFA-430, the second in a series of fuel thermal performance tests was run with a fill gas compostion of 90% helium and 10% xenon. In this test, fuel centerline temperatures were measured as a function of rod power and gas pressure. The results of the previous test were confirmed. These tests show a greater increase in gap conductance with gas pressure at this compostion than was calculated using the FRAP code.

Two rods with burnups of about 30 000 MWd/t were removed from the IFA-429 helium absorption-fission gas release experiment in the Halden Reactor. Postirradiation examination of these two rods and two rods previously removed with 10 000 MWd/t burnup is scheduled to begin later this fiscal year. An IFA-429 experiment update report was issued which highlights the significant data, including fission gas release, gathered through August 1979.

The model for the IFA-511 flow starvation tests was restructured to reflect changes in the loop piping. The model for the test train was formulated.

A topical report on fuel swelling due to retained fission gas in molten fuel during high temperature transients was published.[8]

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 advanced computer codes for predicting the steady state and transient behavior of light water reactor fuel rods was accomplished during the quarter. This effort has involved development of the FRAPCON-2 computer code in conjunction with the Pacific Northwest Laboratory (PNL) and development of the FRAP-T6 computer code. The FRAP-T6 effort has emphasized providing an efficient and accurate fuel model for linking with the TRAC-P thermal-hydraulic code. The results of the fuel code development progress are presented in the following sections.

1. FRAPCON-2 CODE DEVELOPMENT

G. A. Berna

FRAPCON-2 is a computer code being developed to predict the steady state response of light water reactor fuel rods during long-term burnup operation. Development of FRAPCON-2 is a joint effort of EG&G Idaho, Inc., and Pacific Northwest Laboratory which began with the development of the FRAPCON-1[a] code. FRAPCON-2 calculates the temperature, pressure, deformation and failure histories of a fuel rod as a function of the time-dependent fuel rod power and coolant boundary conditions. In addition, the code is designed to generate initial conditions for transient fuel rod analysis by either the FRAP-T5[9] or FRAP-T6 (currently under development) computer code.

FRAPCON-2, like its predecessor, FRAPCON-1, models basic phenomena including heat conduction through the fuel and cladding, elastic-plastic cladding deformation, fuel-cladding mechanical interaction (FCMI), fission gas release, fuel rod internal gas pressure, heat transfer between fuel and cladding, cladding oxidation, and heat transfer from cladding to coolant. The code contains all needed rod surface heat transfer coefficient correlations, water properties, and material properties.

Significant modeling improvements have been incorporated in FRAPCON-2 through the addition of the following subcodes: (a) MATPRO-11, Revision 1, a subcode which includes the latest fuel, cladding, and gas material properties; (b) the uncertainty analysis subcode which allows the user to estimate the uncertainty of code outputs as a function of known uncertainties in the code inputs; (c) FRACAS-II, a subcode which determines the state of stress, strain, and deformation in the fuel and cladding including creep, cracking, and hot pressing of the fuel; (d) AXISYM, a finite element subcoded used to calculate local cladding ridging strains during FCMI and which incorporates an elastic-plastic creep capability using plasticity relationships consistent with FRACAS-II; (e) the PELET mechanics analysis package developed by PNL; and (f) the ANS 5.4 fission gas release model developed by PNL. FRAPCON-2 includes preliminary advanced fuel relocation models which have been developed by both EG&G Idaho, Inc., and PNL for use with their respective mechanical models. In addition, the FAST/GRASS fission gas release model will be incorporated in FRAPCON-2 before code release.

a. FRAPCON MOD1/VER4, Idaho National Engineering Laboratory Configuration Control No. H00730IB.

As a result of the independent assessment of FRAPCON-1, certain additional models have been developed for FRAPCON-2. These are:

1. A model that couples the fuel density effects with fuel porosity and fuel conductivity. Studies showed that the FRAPCON-1 code overpredicted centerline fuel temperature in cases in which significant fuel densification occurred.

2. An improved model of the porosity and void volumes. Results of the independent assessment indicated FRAPCON-1 overpredicted rod internal gas pressure.

3. A refined crack healing model based on recent PBF data. Indications are that the value of the transition temperature used in FRAPCON-1 was too high.

4. A radial power profile model which is fuel burnup and enrichment dependent. The FRAPCON-1 radial power profile option was limited, and the contributors at PNL recommended replacement of the simplified model.

With the exception of the FAST/GRASS fission gas release model, all of the new models discussed previously have been incorporated in the FRAPCON-2 code. Before the code is released for general use, developmental assessment must be completed to check for correctness of the model additions and for any resulting unanticipated perturbations. This effort is in progress and will continue during the next quarter.

2. FRAP-T6 CODE DEVELOPMENT

L. J. Siefken

FRAP-T6 is a computer code being developed to predict the transient behavior of light water reactor fuel rods during any hypothesized accident, ranging from mild operational transients to design basis accidents such as the loss-of-coolant accident and the reactivity initiated accident. The code will calculate the variation with time of the significant fuel rod variables, including fuel and cladding temperatures, cladding hoop strain, cladding oxidation, and internal gas pressure. The code will calculate the uncertainties in the predicted fuel rod variables due to known uncertainties in fuel rod fabrication variables, material properties, and rod power and cooling. In addition, the code will be linked with the FRAPCON-2 code for initialization of burnup-dependent variables and with the MATPRO-11, Revision 1, subcode for determination of fuel, cladding, and gas material properties.

FRAP-T6 will have the following improvements with respect to its predecessor, FRAP-T5[9]: (a) dynamic dimensioning of all arrays, (b) simplified code input and input screening, (c) link with the fission gas release subcode FAST/GRASS, and (d) condensed FORTRAN coding and code storage to provide an efficient and accurate fuel model for linking with thermal-hyraulic codes such as TRAC-P and COBRA-IV. Additionally, improvements are planned for the gap conductance, fuel deformation, cladding ballooning, and cladding heat transfer models. An allowance will be provided for azimuthal variations of cladding surface heat transfer and axial variations of fuel pellet diameter.

The dynamic dimensioning and condensing tasks referred to above have been completed, and a simple link with certain thermal-hydraulic codes has been established. Problems encountered with the previously established link between RELAP4/MOD7 and FRAP-T5 have been corrected. The resulting version of FRAP-T6 has been transmitted to the Los Alamos Scientific Laboratory for inclusion in the TRAC-P code. This version of FRAP-T6 provides the TRAC-P code with efficient fuel models that have been independently assessed[10] and tested by many applications to the design and posttest analysis of LOFT and PBF experiments. The dynamic dimensioning allows modeling of an arbitrary number of fuel rods and arbitrary axial and radial fuel rod nodalization.

The condensed version of FRAP-T6 does not include models used for special or detailed fuel rod behavior analysis. The models excluded are (a) the HEAT-1 subcode for modeling heat conduction, (b) the HTRC subcode for modeling heat transfer at the cladding surface, (c) the GRASS subcode for modeling fission gas production and release, (d) the FRACAS-II subcode for modeling stress-dependent fuel deformation, (e) the FRAIL subcode for modeling fuel rod failure probability, and (f) the FRAP-T6 gas flow, plenum gas temperature, and uncertainty analysis option models. The major models included are (a) the MATPRO-11 subcode for material properties and modeling of cladding oxidation and annealing, (b) the FRACAS-I subcode for modeling cladding plastic deformation and pellet-cladding mechanical interaction, (c) the BALLOON-1 subcode for modeling cladding ballooning, (d) the gap pressure model, and (e) the gap conductance model. The link with the FRAPCON code for initialization of burnup-dependent variables was also retained.

A common block named FRAPC provides the communication between the TRAC-P code and FRAP-T6. The common block contains the following types of variables: (a) control variables which, for example, provide control of and time of calculations and printout of fuel rod state; (b) TRAC-P code computed variables, which are input to FRAP-T6 and specify the fuel and cladding temperature distribution and coolant pressure; and (c) FRAP-T6 computed variables, which are input to TRAC-P and specify the gas gap conductance and cladding outer diameter and an indicator of whether or not cladding failure has occurred.

The driver program for the linked TRAC-FRAP code will alternately execute the TRAC-P and FRAP-T6 codes. The TRAC-P code is executed first, and the coolant conditions and fuel rod temperature distribution at an advanced time are calculated. The TRAC-P code receives, through the FRAPC common block, the values of fuel rod gap conductance and cladding diameter calculated by FRAP-T6 at the previous time step. The FRAP-T6 code is then executed, and fuel and cladding deformation, gap conductance, and gas pressure are calculated. The FRAP-T6 code is given the fuel rod temperature distribution and coolant pressure at the advanced time by the FRAPC common block. Time is then advanced and the above process is repeated. The TRAC-P and FRAP-T6 codes are in separate overlays, which limits computer memory requirements to less than 160 000 octal words.

V. CODE ASSESSMENT AND APPLICATIONS PROGRAM

J. A. Dearien

The Code Assessment and Applications Program (CAAP) has a primary responsibility to the NRC for the assessment of thermal-hydraulic and fuel behavior analytical codes. Data obtained from experimental programs such as LOFT, Semiscale, and the Thermal Fuels Behavior Program are used to assess the results of code calculations. The purpose of code assessment is to provide a quantitative assessment of the computer programs being developed for the NRC. In support of code assessment activities, the NRC/Reactor Safety Research (RSR) data bank is being developed to facilitate the processing of experimental data and comparisons of the experimental data with calculations performed using the analytical codes. In addition to assessing codes, the CAAP is the technical advisor to the NRC on industry cooperative safety experimental programs. The purpose of this activity is to ensure that the 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. Technical assistance to the NRC continues to be performed in the audit of pressurized water reactor vendor safety calculations.

A new program has been added to the CAAP, the purposes of which are to identify and analyze accident or upset sequences of events for boiling and pressurized water reactors (BWRs and PWRs) and provide assistance to the NRC during commercial reactor transients such as occurred at Three Mile Island. These tasks are called BWR and PWR Analyses.

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

1. FUEL CODE ASSESSMENT DATA SAMPLE

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

The steady state data sample used for assessment of the FRAPCON-1[11] and FRAP-T5[10] fuel rod analysis programs was characterized. The data were separated into 12 measurement categories, representing data obtained during experiments with instrumented fuel rods and from postirradiation measurements of both instrumented and noninstrumented fuel rods. Distributions of the data within each category were analyzed according to rod design and operation parameters. Overall, data from about 700 fuel rods are contained in this data sample, representing rod temperature, strain, pressure, and corrosion measurements. Most of the data included in the sample thus far were obtained from unpressurized rods of low burnup.

Data scatter was analyzed for the two categories which have large sample sizes, fuel centerline temperature and rod internal pressure. All available data from rods of similar design and operation conditions were consolidated, and the spread or lack of reproducibility among the data sample was determined. The spread in the centerline temperature data is about 15% of the mean temperature value of the measurement range. Measurements obtained with a pressure-balance sensor show a spread of about 4.5% of the mean pressure value of the measurement range. The spread of the pressure transducer data is much larger—about 35% of the mean value—due to irradiation-induced drift of the instrument.

The need to expand the entire data sample and to continue data scatter analyses for other categories was identified. Categories of particular interest to code development and commercial reactor licensing activities include cladding radial strains and fission gas release.

2. BWR AND PWR ANALYSES

D. D. Christensen, C. D. Fletcher, A. C. Peterson, W. C. Phoenix, R. R. Schultz

BWR and PWR Analyses efforts will identify, analyze, and document accident and upset sequences, provide time frames for major events in the sequences, and form the capabilities of responding to a situation such as occurred at Three Mile Island.

Accidents, upset sequences, and event trees will be identified by either the NRC's Probabilities Analysis Staff (PAS) or by EG&G Idaho analysis. At first, initiating events such as turbine trips will be identified from actual commercial operating data and experience. An analysis of plant electrical and mechanical drawings and plant inspections will form the basis for further event trees. Probability values will be obtained from previous studies to help quantify the likelihood of the event sequences. The most likely areas will then be analyzed using thermal-hydraulics codes. The variation of parameters such as reactor power, pressure, and flow with time and times to major events, such as uncovering of the reactor core, will be documented. Documentation will describe the event tree probabilities and recommendations for reducing the probability of occurrence or consequences, or both.

Thermal-hydraulics analysis will use codes such as RELAP4/MOD6 and MOD7, RELAP5, and various versions of the TRAC code. PWR analysis is ongoing for a few plants; more computer code decks must be assembled so that accidents and upsets can be examined and so that any commercial BWR or PWR in the nation can be analyzed in a short time in case of an emergency. In order to minimize the effort and still be able to analyze each plant, all U.S. BWRs and PWRs have been grouped by similar characteristics such as manufacturer, core power level, number of fuel elements, number of primary coolant system loops, and generic types. As a result, only 15 PWR and 13 BWR decks and groups are required. Within each group a "target" plant was chosen. It will be the plant whose drawings will be used to assemble the decks; specific differences within plants in a group will be taken into account when and if others are analyzed.

Since manpower limits prevent all target plants from being examined simultaneously, two "focal" plants have been chosen for assembling the first decks. Zion Station Unit I has been selected as the focal PWR and Browns Ferry Unit I as the BWR.

A computer deck for Zion I has been assembled and is being used to study one transient called the station blackout, in which no source of electrical power is available from the electrical grid or the on-site emergency diesel generators. In this case, no large motor-driven pumps such as the emergency core cooling or motor-driven auxiliary feedwater pumps will be available, and this can leave the plant in a precarious position. A previously constructed standard BWR Mark 6 deck is being updated to study this same transient. Browns Ferry will also be studied, once the deck is assembled, from drawings and other information from the utility (Tennessee Valley Authority) and vendors (including General Electric). The study is expected to provide plant-specific calculations.

3. RELAP4/MOD6 BLIND PRETEST CALCULATION OF UNITED STATES STANDARD PROBLEM 9

D. M. Ogden

A computer simulation and analysis of U.S Standard Problem 9 (USSP9), using the RELAP4/MOD6 thermal hydraulic code,[a] was performed for the NRC. This problem contained two reflood separate effects tests. The purpose of the problem was to examine different reflood heat transfer and entrainment models in a forced reflood simulation. The prediction was termed "blind" because only the initial conditions for the test were known, not the results.

The experimental test facility for USSP9 was the Westinghouse 161-Rod Unblocked Bundle Test Facility, which is part of the jointly sponsored Westinghouse/NRC/EPRI FLECHT-SEASET Reflood Heat Transfer and Hydraulic Program[12]. The two tests chosen for the Standard Problem were Test 31701, a high flooding rate test, and Test 31805, a low flooding rate test.

The results of the computer simulation of Test 31701[b] are as follows:

1. The peak heater rod surface temperature of 1199 K occurred near the 1.68-m elevation at 10.5 s into the calculation.
2. Quenching due to dispersed flow occurred from the top down for elevations above 3.05 m.
3. The bundle was completely flooded 47 s into the calculation.

The results of the computer simulation of Test 31805[c] are as follows:

1. The peak heater rod surface temperature of 1402 K occurred near the 1.83-m elevation at 103 s into the calculation. This same elevation was quenched after the first 457 s of the calculation.
2. Quenching occurred from the top down for elevations above 2.44 m.
3. Quench times in the heater bundle ranged from a few to 800 s.

On the basis of the analysis of the calculated results, the conclusion reached is that the calculations for Tests 31701 and 31805 of Standard Problem 9 should be similar to experimental results with the following exceptions: the calculated results for heat flux, bundle exit quality, and bundle exit mass flow showed rapid oscillations and discrete spikes. These oscillations and spikes were thought to be caused by the discontinuous transition between heat transfer modes and the discrete nodalization of the heat slabs.

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a. RELAP4/MOD6, Idaho National Engineering Laboratory Configuration Control Number H007084B. Steam tables used are identified by Configuration Control Number H002011B.

b. Test 31701 input is identified by Idaho National Engineering Laboratory Configuration Control Number H00983B.

c. Test 31805 input is identified by Idaho National Engineering Laboratory Configuration Control Number H001083B.

VI. ENGINEERING SUPPORT PROJECTS

R. E. Rice, Acting 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 activities through the development of specialized measurement devices, and indirectly supports analytical efforts by allowing data to be gained in areas previously unmeasurable.

1. 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 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.

Progress this quarter was made 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.

Instruments for the Japanese Slab Core Test Facility (SCTF) were designed and initial stages of fabrication were begun. These instruments consist of liquid level detector systems, fluid distribution grids, gamma-ray densitometers, turbine meters, spool pieces, and drag disk flowmeters. Comprehensive interface discussions have been held with Japan Atomic Energy Research Institute (JAERI) personnel and precise interface responsibilities were agreed upon. Shipping and delivery dates have been established for the various instruments and installation schedules are being developed.

2. ADVANCED INSTRUMENTATION

Jay V. Anderson, Manager

2.1 Instrumentation Development

A 1.3-m optical probe for Semiscale upper plenum application has been completed; it is the longest probe for high temperature and high pressure use (620 K, 15.2 MPa) constructed at EG&G Idaho, Inc. Patent idea records have been filed on liquid level and valve leak detectors, and work is continuing to ready these devices, which are suitable for PWR environments, for applications in water reactor safety research programs. Developmental efforts were continued for two-phase flow reference instrumentation, low flow velocimetry, thermometry, radiation hardened optics, and steam generator measurements.

Heated/unheated differential temperature liquid level probes, as developed for the Thermal Fuels Behavior Program, have successfully been used in the Power Burst Facility in-pile tube. This liquid level detection system is described in the following section.

2.2 Heated Differential Thermocouple Liquid Level System

G. R. Coffin

A heated differential thermocouple liquid level system has been developed by EG&G Idaho, Inc., for the Thermal Fuels Behavior Program. Six systems, installed in the lower plenum and fuel shrouds of the Power Burst Facility (PBF) in-pile tube, were used for liquid level detection and subsequent control of reflood rates during testing in the PBF. These systems were successfully used in the PBF on the LOFT Lead Rod tests in March 1979 and on the TC-1 Test Series in December 1979. System description, operation, and system characteristics and advantages are discussed.

Apparatus used in the system is as shown in the diagram in Figure 9. The liquid level transducer is composed of a Chromel heating element and a Chromel-Alumel junction within a common sheath; the junction is connected differentially with a similar junction in an unheated thermocouple sheath. Each lead is connected to copper in the splice button such that a three-wire, copper conductor, 1.6-mm stainless steel sheathed, lead-out cable can be utilized. Heater power control is provided through the use of a three-mode controller and a power supply. An interface module internal to the three-mode controller provided the necessary wet or dry transducer (and other) indications to the PBF timing and control systems.

To provide the desired liquid level information, two sleeved (splash guard) liquid level transducers, as shown in Figure 10, were installed in the lower plenum, and unsleeved versions were installed at the bottom of each fuel rod. The sleeve eliminates a premature wet indication resulting from liquid froth in the lower plenum. Through the use of the lower system transducers, rapid reflood could be performed until the lower plenum was filled, at which time the flow rate was decreased for fuel rod reflood. The upper system transducers provided liquid level indications to the PBF data aquisition system at commencement of fuel rod reflood.

System operation with the transducer in air is that of a linear feedback control system which maintains the heated thermocouple at a desired temperature difference (setpoint) with respect to the reference or unheated thermocouple. This setpoint is adjustable in the three-mode controller and was set for 5 mV in this application. With the transducer in water, most of the heater power is dissipated in the water, causing a temperature difference less than the controller setpoint. This is because the power supply is current- and voltage-limited to provide the maximum power level consistent with transducer design. Thus, the transducer can have optimum response time and life expectancy. The wet-to-dry or dry-to-wet transition discriminator levels are individually set (infinitely adjustable between a known wet temperature difference and the setpoint) in the controller; 2.5 mV was chosen for this application. Resulting dry-to-wet response was 0.5 s and wet-to-dry response was 1 s. Response times half as long are obtainable and have been used. Ultimate response is limited by probe geometry and size, since commercially available controllers are suitable to 20 ms.

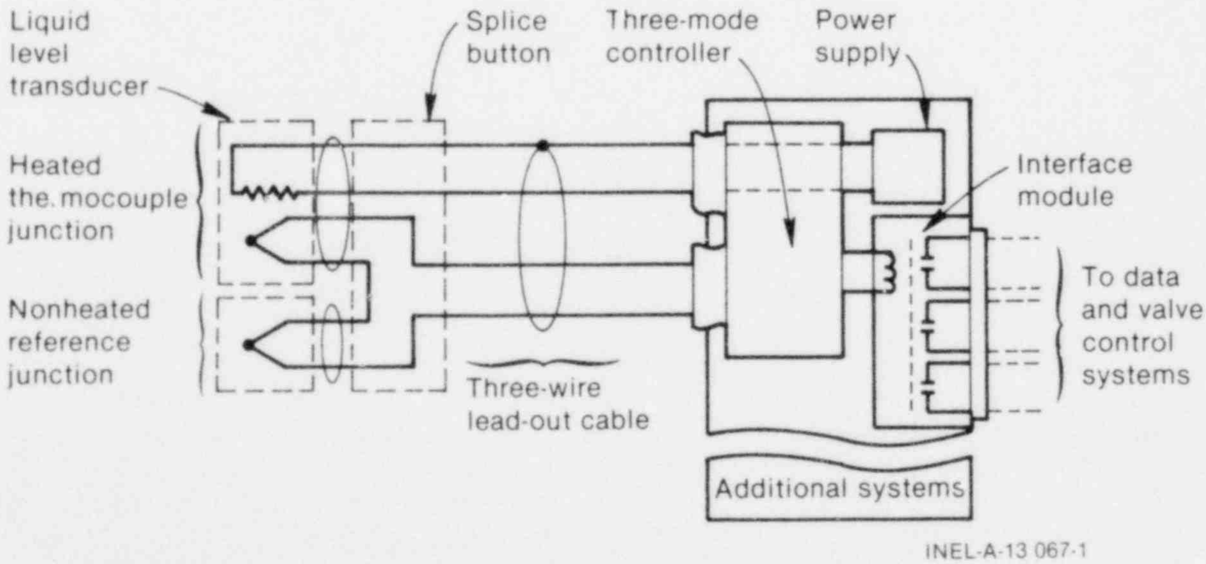

Figure 9. Diagram for heated differential thermocouple liquid level system.

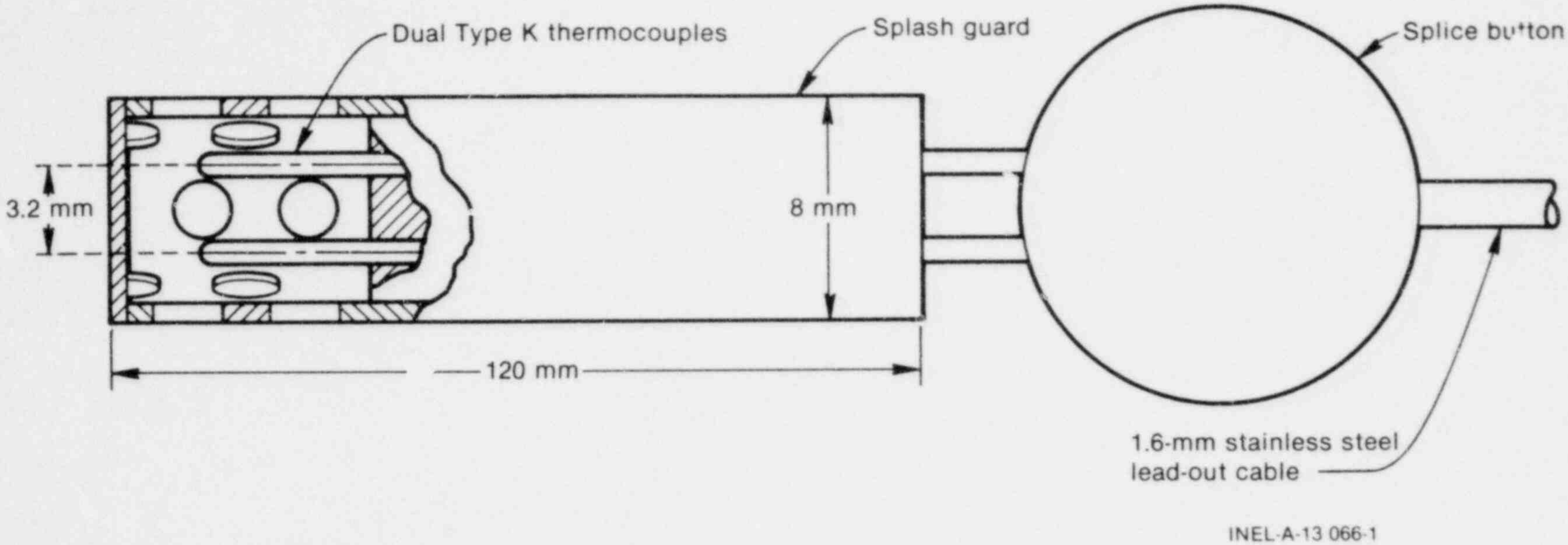

Figure 10. Sleeved liquid level detector transducer.

The advantages of such a system for the measurement of liquid level for nuclear core or other applications include (a) protection against heater burnup in water or steam, (b) materials are nuclear qualified, (c) response time to 20 ms is available and suitable for most applications, (d) the system is completely attainable with off-shelf equipment except for the liquid level transducer, and (e) adjustable discriminator points and latching or logic circuits, or both, provide indicators to external circuitry. A major disadvantage is the cost associated with this flexibility. Most applications would not require the capacity of the power supply or the flexibility and variety of adjustments available with this controller. Production lots of liquid level systems to a particular specification would be much less expensive. System testing was performed to determine reliability and durability.

Prior to qualification testing, tests were performed to establish setpoint, rate, and reset rate adjustments compatible with the liquid level transducer and program specifications. Qualification tests included (a) a 200-h life test with the liquid level transducer, splice button, and lead-out cable at 516 K, with the controller at setpoint; (b) a 200-h life test with the heater power at a predetermined level in excess of normal; and (c) wet-to-dry and dry-to-wet cycle tests for thermal shock life tests. These tests verified proper operation of the system prior to use in the PBF reactor. Nuclear irradiation tests were not performed, but the materials are common in the nuclear environments and are suitable for PWR applications.

Liquid level systems consisting of differentially connected heated and unheated thermocouples and commercially available equipment have been successfully used by the Thermal Fuels Behavior Program in the PBF reactor. The systems functioned as designed and as expected in several blowdown/reflood experiments. These systems are self-protecting and are constructed from materials compatible for PWR operation.

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