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for U.S. Nuclear Regulatory Commission

**IRRADIATION EFFECTS TEST SERIES
SCOPING TEST 1
TEST RESULTS REPORT**

W.J. QUAPP C.M. ALLISON
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September 1977

IDAHO NATIONAL ENGINEERING LABORATORY

ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

IDAHO OPERATIONS OFFICE UNDER CONTRACT EY-76-C-07-1570

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Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161
Price: Printed Copy ~~\$6.00~~ 7.25; Microfiche \$3.00

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SCOPING TEST 1
TEST RESULTS REPORT

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Distributed Under Category:
NRC-3
Water Reactor Safety Research
Fuel Behavior

IRRADIATION EFFECTS TEST SERIES
SCOPING TEST 1
TEST RESULTS REPORT

By:

W. J. Quapp
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EG&G IDAHO, INC.

Date Published - September 1977

PREPARED FOR THE
U.S. NUCLEAR REGULATORY COMMISSION
AND
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
IDAHO OPERATION OFFICE
UNDER CONTRACT NO. EY-76-C-07-1570

ABSTRACT

This report^[a] describes the results of the first scoping test in the Irradiation Effects Test Series conducted by the Thermal Fuels Behavior Program, which is part of the Water Reactor Research Program of EG&G Idaho, Inc. The research is sponsored by the United States Nuclear Regulatory Commission. This test used an unirradiated, three-foot-long, PWR-type fuel rod. The objective of this test was to thoroughly evaluate the remote fabrication procedures to be used for irradiated rods in future tests, handling plans, and reactor operations. Additionally, selected fuel behavior data were obtained. The fuel rod was subjected to a series of preconditioning power cycles followed by a power increase which brought the fuel rod power to about 20.4 kW/ft peak linear heat rating at a coolant mass flux of 1.83×10^6 lb/hr-ft². Film boiling occurred for a period of 4.8 minutes following flow reductions to 9.6×10^5 and 7.5×10^5 lb/hr-ft². The test fuel rod failed following reactor shutdown as a result of heavy internal and external cladding oxidation and embrittlement which occurred during the film boiling operation.

[a] This report was previously published (TFBP-TR-110, January, 1976) with a limited internal distribution. This publication is made under the distribution selected for final reports on tests in the NRC Fuel Behavior Program.

SUMMARY

This report describes the results of the first test in the Irradiation Effects Test Series, which is part of the Thermal Fuels Behavior Program being conducted at the Idaho National Engineering Laboratory by EG&G Idaho Inc., to define the behavior of fuel rods during transient reactor operating conditions.

This test, IE Scoping Test 1, was conducted on October 30, 31, and November 1, 1975, in the Power Burst Facility reactor at the Idaho National Engineering Laboratory (INEL). The test used an unirradiated, archival Saxton^[a] UO_2 fuel rod (designated Rod IE-001) supplied by Westinghouse Electric Corporation, which was modified at the INEL to contain an internal fuel centerline thermocouple, an internal gas pressure transducer, and cladding surface thermocouples.

This scoping test was conducted to evaluate methods and procedures for remote fabrication of irradiated fuel rods, test instruments, handling plans and reactor capability prior to the beginning of the test series. In addition to the scoping nature of the test, fuel behavior data were acquired which will be part of a larger data base on fuel rods operating under various steady-state and transient reactor conditions to be used for development and verification of fuel behavior codes.

The test conduct included a preconditioning period, during which time the fuel rod performance was nominally similar to that experienced during normal low power operation in a power reactor. Following the preconditioning period, a power ramp brought the fuel rod power to about 20.4 kW/ft peak linear heat rating. After 78 minutes of operation at this condition, a flow reduction transient was initiated to induce film boiling. After a flow reduction from 1.83×10^6 to 9.6×10^5 lb/hr-ft², film boiling was indicated by a fuel rod length change and temperature excursions on four of the eight cladding surface thermocouples located

[a] The Saxton Reactor was designed by Westinghouse Electric Corporation for the USAEC. The reactor was a small, prototype, pressurized water reactor.

23.8 and 27.8 inches above the bottom of the fuel rod. After approximately two minutes, the flow was reduced to 7.5×10^5 lb/hr-ft².

Following this reduction, the film boiling zone was observed to extend downward on the fuel rod to the 19.8-inch thermocouple location based on the thermocouple readings. The fuel rod experienced film boiling for a total of 4.8 minutes over the 23.8 to 27.8-inch region. The region of the fuel rod around the 19.8-inch thermocouple was in film boiling for 1.4 minutes.

The test was terminated by a relatively fast run-in of the control rods from a reactor power of 25 MW to 1 MW and a scram. Approximately 80 seconds following the shutdown, a temporary reduction in the coolant flow through the test shroud was observed. This flow reduction was presumed to be associated with fuel rod failure. Almost simultaneously, radiation monitors in the reactor building started to rise, indicating the presence of radioactive material circulating through the loop, and, therefore, fuel rod failure.

Fuel rod failure was confirmed during the removal of the fuel rod and shroud from the test train. The fuel rod was heavily oxidized over the 18.75 to 32.0-inch region. A portion of the fuel rod from the 21.5 to 23.75-inch region was broken in the shroud and fell into the canal during the handling process. The fuel rod failed as a result of the thermal and hydrostatic stresses during cooldown on the heavily oxidized and completely embrittled cladding.

Comparisons of the experimental data against the results of the FRAP-T1 fuel behavior code using the as-run conditions give reasonable agreement for fuel centerline temperature and cladding elongation prior to fuel-cladding mechanical interaction. Cladding surface temperature comparisons were reasonably good only for one welded junction thermocouple. Comparison of the fuel rod internal pressure was not possible due to failure of the transducer.

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1. INTRODUCTION

The Irradiation Effects (IE) Test Series is part of the Nuclear Regulatory Commission (NRC) Fuel Behavior Program^[1], defining the behavior of fuel rods during abnormal reactor conditions. The irradiation effects experiments will be performed in conjunction with similar tests on unirradiated fuel rods, and are designed to delineate the burnup factors that may contribute to fuel rod failures. This test series is directed toward satisfying one of the major objectives of the Fuel Behavior Program -- the development of verified analytical models for predicting the behavior of irradiated fuel rods. Detailed discussions of the irradiation effects and fuel behavior data to be evaluated in this test series are presented in the Irradiation Effects Experiment Requirements Report^[2].

To evaluate the behavior of irradiated fuel rods, techniques must be developed to modify irradiated fuel rods to include adequate instrumentation. This required remote removal of the upper end cap and the remote rewelding of a new end cap containing instrumentation connected to 40-foot long stainless steel instrument leads. The instrumented fuel rod must then be inserted into a flow shroud. A more detailed description of the hardware is contained in Section 2. Following these activities, the assembled rods must be shipped from the hot cells to the reactor, loaded into the test train and then loaded into the reactor without damaging the delicate instruments and leads.

To assure a successful completion of the IE tests in this series, two scoping tests were scheduled prior to the actual beginning of the test series^[2,3]. The objectives of IE Scoping Test 1 were:

- (1) To evaluate the methods and procedures to remotely recondition and instrument an irradiated fuel rod included such items as:
 - (a) the zircaloy to stainless end cap transition, (b) remotely made end cap to cladding weld, and (c) the gas pressurization and remote scaling process.

- (2) To evaluate the performance of the test train and fuel rod instrumentation under actual operating conditions.
- (3) To evaluate the facility capability to provide the required power and flow transients.
- (4) To generate fuel behavior data on an unirradiated fuel rod similar in design to the irradiated fuel rods to be used in later tests.

IE Scoping Test 1 used an unirradiated Saxton UO_2 fuel rod to facilitate manual repair or handling if it became necessary. This fuel rod was assembled in the hot cell mock-up area to evaluate the remote assembly equipment and procedures.

The nominal fuel rod design parameters are given in Section 2. The fuel rod was tested in the Power Burst Facility (PBF) reactor under the same conditions and with nominally the same instrumentation as planned for the balance of the test series. The tests consisted of a preconditioning phase of nearly 24 hours followed by a power ramp and a flow reduction transient to induce film boiling. A complete discussion of the test conduct is provided in Section 3 and test results are provided in Section 4.

The results of the postirradiation visual examinations are presented in Section 5. Additional postirradiation examination results are reported separately^[4].

In Appendices A and B, an assessment of the instrumentation and system errors are presented. In Appendix C, the techniques used for thermal hydraulic power calibration and axial flux profile determination are discussed. The preirradiation fuel rod characterization data are contained in Appendix D. Appendix E contains additional test data not directly related to the fuel behavior data, such as the coolant temperatures and pressures.

The authors recognize that the data provided on the time history plots in this report do not have sufficient resolution to be used by others for model development or verification. Upon request, such data can be provided in digital form.

2. EXPERIMENT DESCRIPTION

The principal components used in the conduct of this experiment include the test train, the flow shroud assembly, the fuel rod, and various instrumentation. The test hardware and instrumentation were designed to provide experimental measurement of irradiated fuel rod behavior during nuclear operation.

2.1 Test Train

Figure 1 depicts the test train assembly mounted in the in-pile tube (IPT). The two system pressure transducers are shown in the upper view of this figure. As seen in the lower view of this figure, the fuel rod was centered in the position denoted by the 36.00-inch dimension. The linear variable differential transformer (LVDT) and turbine flow-meter instruments were located at the lower end of the test train assembly.

The test train and additional instrumentation are illustrated in Figure 2. The two cross-sectional views at the top of the figure show the relative positioning of the inlet and outlet coolant temperature measurement devices. These instrument clusters were mounted on each end of the test train in locations as shown in the lower diagram of Figure 2. The positions of a few of the fuel cladding surface and fuel centerline temperature measurement devices are shown in the center section of the lower view of Figure 2. The LVDT, turbine flowmeter, orifice plate and flow screen assembly are also depicted in this figure.

2.2 Flow Shroud

The flow shroud was securely positioned in the PBF in-pile tube, as shown in Figure 1. The flow shroud was fabricated from zircaloy-4 material and had a nominal inside diameter of 0.760 inch. The fuel rod outside diameter was 0.391 inch. The resultant hydraulic and equivalent heated diameters were 0.369 and 1.086 inch, respectively.

Fig. 1 IE Scoping Test 1 experiment assembly and instrumentation shown installed in the PBF in-pile tube.

Fig. 2 IE Scoping Test 1 fuel rod and instrumentation as installed in the flow shroud.

2.3 Fuel Rod

An unirradiated, 9.5 wt% enriched, Saxton Rod 955 was utilized for the experiment. The fuel rod cladding was constructed of zircaloy-4 and had a mean outside diameter of 0.391 inch and an overall length of 38.167 inches. The 34.632-inch long UO_2 fuel stack consisted of 56 fuel pellets with a 0.3361-inch mean outside diameter. Detailed fuel pellet and fuel rod cladding characterization data are given in Appendix D. The fuel rod was designated Rod IE-001.

Modification of the original fuel rod was necessary to accommodate the required test instrumentation. The original top end cap of the fuel rod was removed and 23 of the 56 fuel pellets were drilled to provide a centerline hole, with a nominal diameter of 0.077 inch, for placement of a fuel centerline thermocouple. The fuel rod was refitted with an instrumented end cap, which was welded into position as shown in Figure 2. The fuel rod was evacuated, backfilled with helium for void volume measurement and subsequently backfilled with a mixture of 76% helium and 24% argon to a pressure of 375 psig. This gas mixture was chosen on the basis of an analysis that indicated the thermal conductivity of the gas mixture closely matched the thermal conductivity of the fission product gases found in irradiated Saxton fuel rods. The fuel rod cladding was further modified by laser welding two titanium sheathed thermocouples to the outside cladding surface. These thermocouples were located at 19.8 and 27.8-inch elevations (from the bottom of the fuel rod). In the test assembly, the fuel rod was held rigidly at the top and was free to expand axially downward.

2.4 Instrumentation

Instrumentation was provided to monitor fuel rod response and coolant conditions in the flow shroud during the test. A detailed listing of the transducer calibrations and error analysis for the test instruments is given in Appendix A.

2.4.1 Test Train Instrumentation. The instrumentation associated with the test train consisted of the following:

- (1) Two pressure transducers were employed to measure the pressure of the coolant during the test. One KAMAN Model 1910, 5,000 psig and one 10,000 psig transducer were used. Both transducers were located above the fuel rod in the flow loop, as shown in Figure 1.
- (2) Two turbine flowmeters were located at the entrance of the test flow shroud to measure coolant flow through the shroud. The flowmeters were mounted in a series configuration, as shown in Figure 3. Each flowmeter was fitted with a different type of bearing arrangement. The rotor of one flowmeter was mounted on stainless steel ball bearings while the other flowmeter utilized a graphite journal bearing.

Prior to entering the flowmeter, the coolant was directed through a series of screens which act as flow straighteners; the fluid then flows through the flowmeter turbine and enters the bottom of the test shroud via an orifice plate. These components are shown in Figures 2 and 4. Flow measurements were made by utilizing a magnetic pickup coil and a pulse rate indicator that monitored the turbine velocity.

- (3) Two sets of calibrated copper-constantan (Type T) differential thermocouple pairs were used to measure the coolant temperature rise across the test flow shroud.
- (4) One set of calibrated platinum resistance thermometers (RTD) were configured as a differential pair to measure the coolant temperature rise across the test shroud.
- (5) Two chromel-alumel (Type K) thermocouples were positioned at the inlet and outlet of the test flow shroud to measure the coolant inlet and outlet temperatures.

Fig. 3 Turbine flowmeters, inlet thermocouples, and RTD cluster shown as mounted on the bottom end of the test train.

Fig. 4 Expanded layout of the linear variable differential transformer, coolant flow orifice plate, turbine flowmeter, and coolant flow screen assembly.

These temperature measurement devices were mounted in clusters at each end of the test flow shroud, as shown in Figure 2. Both of the aforementioned thermocouple types utilized magnesium oxide for insulating the thermocouple wires from their respective sheaths. A photograph of the inlet temperature measurement cluster can be seen in Figure 3.

- (6) One linear variable differential transformer was positioned at the bottom of the fuel rod to measure changes in fuel rod length (see Figure 4).
- (7) Five 4-inch-long cobalt self-powered neutron detectors (SPNDs) were axially distributed on the test train assembly (see Figures 5 and 6). These devices were employed to measure the neutron axial flux profile in the IPT during the test. The data provided by these instruments, in conjunction with coolant temperature differential and flow rate measurements, were used to determine fuel rod power (see Appendix C). The five detectors were mounted on the test train with centers at 5.48, 11.64, 17.80, 23.96 and 30.12-inch elevation (from the bottom of the fuel stack).
- (8) A cobalt wire was mounted axially in the test train to monitor neutron flux in the IPT. A posttest gamma scan of the wire yielded the integrated power flux profile over the duration of the test.

2.4.2 Fuel Rod Instrumentation. The following transducers were furnished to detect the response of the fuel rod during the experiment:

- (1) One 2000 psig strain post type pressure transducer was used to monitor the fuel rod internal pressure. This transducer was mounted in the instrumented end cap at the upper end of the fuel rod as indicated in Figure 2.

Fig. 5 Self-powered neutron detectors as mounted on support bar.

Fig. 6 View of a section of the test train, as assembled for insertion into the IPT. The spring loaded thermocouple mountings are shown at their respective positions on the outside of the flow shroud. The relative position of the self powered neutron detectors are shown at their location in the test assembly.

- (2) Six grounded junction, beryllium oxide insulated tungsten-rhenium (W5%Re/W26%Re) alloy wire, zircaloy sheathed thermocouples (TC) were used to measure the cladding surface temperature^[a]. These thermocouples were spring-loaded against the outer surface of the fuel cladding with an approximate three-pound preload. The thermocouples were located as specified in Table I.

TABLE I
IE SCOPING TEST 1 CLADDING SURFACE
SPRING-LOADED THERMOCOUPLE LOCATIONS

TC No.	Elevation from Bottom of Fuel Rod (inches)	Azimuthal Orientation
1	19.8	180°
2	19.8	0°
3	23.8	270°
4	23.8	90°
5	27.8	180°
6	27.8	0°

These thermocouples were held in position by mountings on the outer wall of the flow shroud. The positioning of these devices appears in Figures 2 and 6.

- (3) Two swaged, grounded junction, chromel-alumel (Type K), magnesium oxide insulated thermocouples were mounted directly on the fuel rod. The titanium sheath of the thermocouples was separated from the chromel-alumel junction by a tantalum sleeve. The thermocouples were laser welded into position at the 19.8 and 27.8-inch elevations. These instruments, denoted as Thermocouples 7 and 8, respectively, had a nominal azimuthal

[a] A top view of the fuel rod is provided in Figure 7 to give the orientation of the fuel rod with respect to the thermocouples.

SPRING-LOADED THERMOCOUPLE

Thermocouple	Axial Location	Circumferential Location
TC-1 and TC-2	19.8-in. (0.50 m)	0, 180°
TC-3 and TC-4	23.8-in. (0.60 m)	90, 270°
TC-5 and TC-6	27.8-in. (0.71 m)	0, 180°

WELDED-JUNCTION THERMOCOUPLE

Thermocouple	Axial Location	Circumferential Location
TC-7	19.8-in. (0.50 m)	335°
TC-8	27.8-in. (0.71 m)	155°

ANC-A-6761

Fig. 7 Thermocouple orientation with respect to upper end cap.

location of 335° and 155° , respectively. The sheath at the junction of the thermocouples was spaded prior to attachment to the cladding surface. This was done to minimize the cross sectional area of the thermocouple junction protruding out into the flow stream. Additional laser welds of the sheath to the fuel rod were used to secure the thermocouple cables to the fuel rod (see Figure 8).

- (4) One calibrated tungsten-rhenium (W5%Re/W26%Re) alloy wire, hard-fired beryllium oxide insulated, tantalum sheathed thermocouple was provided to measure the fuel centerline temperature. This thermocouple, inserted from the top of the fuel stack through drilled fuel pellets, was located 23.8-inches above the bottom of the fuel rod as shown in Figure 2.

Fig. 8 Chromel-Alumel cladding surface thermocouple shown as welded into position on the fuel rod cladding at the 19.8-inch location.

3. EXPERIMENT CONDUCT

The experiment was conducted during a 31 hour period and consisted of the following phases: (a) power calibration, (b) preconditioning, (c) power ramp, and (d) flow reduction. The phases of the experiment are shown on Figure 9 (in this report, the time scale for all plots is time from the beginning of nuclear operation). Each phase and the major activity conducted during that phase is discussed below. The key experimental results on fuel behavior are included in this section. The remainder of the results are presented in Appendix E.

Average linear heat ratings were calculated using a thermal balance as described in Appendix C. The relationship between average neutron flux and peak linear heat rating versus the driver core power is shown in Figure 10. Peak linear heat ratings, as used in the remainder of the section, were found by applying an axial peaking factor, obtained from physics calculations, of 1.32. The actual peaking factor varies slightly with rod power because of a shift or skew in the axial flux profile due to changes in driver core control rod position. This was indicated by the axially distributed self-powered neutron detectors. Almost all of the effects of the skew in the profile are limited to low rod powers (Figure 11). Above 8 kW/ft (3 kW/ft for the upper half of the rod), the axial flux profile does not change appreciably. It is difficult to separate the effects of skew in the profile because of the scatter in the data due to difficulties with the SPND amplifiers. However, the larger uncertainty limits in the axial flux profiles at the lower end of the rod can be attributed mainly to the effects of axial flux skew at lower power levels.

Fig. 9 Reactor power, inlet flow, and inlet temperature for IE Scoping Test 1.

Fig. 10 Average flux from neutron detectors and peak rod power from thermal hydraulic measurements versus core power.

Fig. 11 Axial flux profile for various rod peak powers.

3.1 Power Calibration Phase

During the first phase of operation, a series of heat balances was conducted at several combinations of inlet flow rate and reactor power. Data from this period were used to calibrate the neutron detectors against the thermal hydraulically determined power to be used in the event of flowmeter or differential thermocouple failure or a system condition where a thermal balance was not possible. As seen in Figure 10, the average neutron detector signal varies linearly during the power calibration part of the test with respect to rod power, expressed in terms of linear heat rating. This was verified during the remaining phases of the test.

Power calibration data were recorded in three power cycles identified as Cycles 1, 2, and 3. The power calibration consisted of nominal peak linear heat rating steps of 2, 4, 7, and 10 kW/ft at three different flow rates: 4, 9, and 13 gpm (6.08×10^5 , 1.37×10^6 and 1.98×10^6 lb/hr-ft²). The power was increased and decreased at approximately 0.25 kW/ft/min and held at each power level for 20 minutes. As shown in Figure 9, all cycles were completed without difficulty except for the first cycle at 4 gpm. During this cycle, a reactor scram occurred at approximately 1.08 hours into the test due to a loss of facility power. Operation resumed about 1 hour later starting with the second cycle.

The nominal inlet temperature during this period was 590°F with a deviation of up to $\pm 5^\circ\text{F}$, as shown in Figure 9.

Since the power calibration phase also preconditions the fuel rod, the measurements concerning fuel rod behavior were equally important during this period. There were four main measurements: (a) fuel centerline temperature, (b) rod internal pressure, (c) cladding surface temperature, and (d) cladding elongation. The results for the rod internal pressure measurement are considered invalid, since this transducer exhibited a strong drift during the course of the experiment. Measurement results for the fuel centerline temperature and cladding elongation are shown in Figure 12 along with the system measurements of

Fig. 12 Fuel centerline temperature at 23.8-inch elevation, cladding length, inlet flow and reactor power during power calibration.

inlet flow and reactor core power. The cladding surface temperature measurements are shown in Figures 13 and 14 along with inlet temperature.

3.2 Preconditioning Phase

The primary purpose of preconditioning the fuel rod is to allow for the development of fuel restructuring and initial pellet-cladding mechanical interaction (PCI). These processes occur to varying extents depending on the fuel rod power level and operating history. The preconditioning phase (including the previous power calibration) does not, of course, simulate the effects of burnup experienced by a fuel rod whose reactor residence time is measured in months or years. Nevertheless, some effects of operation are present and the preconditioning phase does permit quasi-equilibrium fuel behavior data to be acquired.

The preconditioning phase was conducted in a manner similar to the power calibration, and data were recorded in three power cycles identified as Cycles 4, 5, and 6. These power cycles each consisted of power steps at 2, 4, 7, and 10 kW/ft at a nominally constant flow rate of 13 gpm (1.85×10^6 lb/hr-ft²), with ramp rates of approximately 4 kW/ft/hr. The inlet temperature was increased to 620°F and then held approximately constant for this portion of the test. Two reactor shutdowns occurred during this period causing only a minor disruption in test conduct.

The measurement results for fuel rod behavior are presented in Figures 15, 16, and 17.

3.3 Power Ramp Phase

Following the preconditioning period, the rod linear heat rating was increased from 7.4 to 20.4 kW/ft in 1.6 minutes at a rate of approximately 8.1 kW/ft/min. The power was held constant for approximately 78 minutes before the flow reduction phase was started. The other system conditions were the same as for the preconditioning phase, with a nominal inlet temperature of 620°F.

Fig. 13 Cladding surface temperature at the 19.8-inch elevation and coolant inlet temperature during the power calibration phase of IE Scoping Test 1.

Fig. 14 Cladding surface temperatures at the 23.8 and 27.8-inch elevations during power calibration phase of IE Scoping Test 1.

Fig. 15 Fuel centerline temperature at the 23.8-inch elevation, cladding length, inlet flow, and reactor power during the preconditioning phase of IE Scoping Test 1.

Fig. 16 Cladding surface temperature at the 19.8-inch elevation and coolant inlet temperature during the preconditioning phase of IE Scoping Test 1.

Fig. 17 Cladding surface temperatures at the 23.8 and 27.8-inch elevations during the preconditioning phase of IE Scoping Test 1.

Immediately following the power ramp, the fuel centerline thermocouple failed at an indicated temperature of 3900°F. Measurement results of fuel rod behavior are shown in Figures 18, 19, and 20. Figure 18 shows the fuel centerline temperature measurement (up to the time of failure) cladding length change, inlet flow, and the variation in relative core power. Note the strong fuel-cladding mechanical interaction during this power ramp. During the period at maximum power, the cladding elongation decreased by about 25 mils, indicating fuel creep which reduces the tensile forces on the cladding. The cladding surface temperatures are shown in Figures 19 and 20. Figure 19 shows that the spring-loaded thermocouples at the 19.8-inch elevation indicated a temperature increase of less than 15°F while the laser welded thermocouple (TC 7) indicated a change of more than 30°F. Another interesting point is that the laser welded thermocouple (TC 7) appeared reasonably independent of the change in inlet temperature, while both spring-loaded thermocouples showed a slight variation as the inlet temperature changed. The welded junction thermocouple (TC 8) at the 27.8-inch elevation appeared to have formed a false junction and was measuring only coolant temperature.

3.4 Flow Reduction Phase

Following the power ramp and steady-state operation (about 78 minutes at 20.4 kW/ft) a flow reduction of about 0.73 gpm/minute (approximately 6.2 gpm in 8.5 minutes), was initiated which was approximately 6%/minute of full flow (approximately 13 gpm) as shown in Figure 21. The thermocouple response during the flow reduction is shown in Figures 22 and 23. Film boiling was first observed on thermocouple TC 4 near the 23.8-inch elevation (see Figure 22). Film boiling was indicated by both thermocouples at the 23.8-inch elevation while the LVDT indicated an abrupt rod length change. A slight temperature increase was also indicated by the 19.8-inch welded thermocouple at that time. The initial temperature excursions appeared to be small, therefore, the flow reduction was continued from approximately 6.76 to 5.6 gpm (9.6×10^5 to 7.95×10^5 lb/hr-ft²) in three steps. Continued flow reduction caused

Fig. 18 Fuel centerline temperature at the 23.8-inch elevation, cladding length, inlet flow and reactor power during the power ramp phase of IE Scoping Test 1.

Fig. 19 Cladding surface temperature at the 19.8-inch elevation and coolant inlet temperature during the power ramp phase of IE Scoping Test 1.

Fig. 20 Cladding surface temperatures at the 23.8 and 27.8-inch elevations during the power ramp phase of IE Scoping Test 1.

Fig. 21 Rod internal pressure, cladding length, inlet flow and reactor power during the flow reduction phase of IE Scoping Test 1. Film boiling begins at about 31.1 hours.

Fig. 22 Cladding surface temperatures at the 19.8 and 23.8-inch elevations during the flow reduction phase of IE Scoping Test 1. Temperature excursions were caused by film boiling.

Fig. 23 Cladding surface temperatures at the 19.8 and 27.8-inch elevations during the flow reduction phase of IE Scoping Test 1. Temperature excursions were caused by film boiling.

the film boiling to spread axially along the rod, as calculated during the pretest analysis using FRAP-T1^[5].

Film boiling initially occurred in the region from the 23.8-inch to the 27.8-inch thermocouples. Film boiling did not occur at the 19.8-inch elevation until additional flow reduction had occurred. This region was in film boiling for about 1.4 minutes. The results from all the cladding surface temperature measurements are shown in Figure 24 along with flow rate, cladding length change, and relative rod power as indicated by SPND3.

After the rod operated about 4.8 minutes in film boiling, the reactor was shut down. Approximately 80 seconds after shutdown, a sharp, temporary reduction in flow from 6.0 to 4.5 gpm was indicated by the inlet flowmeter. As shown in Figure 21, fuel rod failure was also indicated by an abrupt change in the internal pressure transducer output, even though it had drifted during the test to the point where the absolute value was meaningless. Within seconds following that event, fuel rod failure was indicated by plant radiation monitors in the loop cubicle.

Fig. 24 Neutron detector 3, all cladding surface temperatures, inlet flow and cladding length change during the flow reduction phase of IE Scoping Test 1. Film boiling begins at about 31.1 hours.

4. EXPERIMENTAL RESULTS

4.1 Comparison of Test Data and the Posttest Analysis

A posttest analysis of the fuel rod behavior was accomplished via FRAP-T^[a] computer code calculations and the results of these calculations were compared against experimental measurements. The parameters of interest were: (a) the fuel centerline temperature at the 23.8-inch elevation (above the bottom of the fuel rod), (b) the cladding surface temperature at the 19.8, 23.8 and 27.8-inch elevations (above the bottom of the fuel rod), and (c) the cladding length change. A comparison of experimental data and analysis is provided in Figures 25 through 48. Due to instrument failure, a comparison of the internal pressure transducer cannot be made. Nominal values for the as-run measured parameters, such as, coolant flow history, coolant inlet temperature, system pressure, fuel rod power and the axial flux profile were used as input for the FRAP-T1 analysis. These input values were obtained from experimental data as presented in Section 3 and Appendix E of this report. Computer input for the physical dimensions of the fuel rod cladding and fuel stack were obtained from the fuel rod characterization data (Appendix D). The fuel rod axial power profile used in the analysis was obtained from a gamma scan of the cobalt wire located in the test train, and is presented in Table II. The raw data from the flux wire was smoothed by a Fourier series fit. Additional pertinent code input included the selection of the W-3^[7] and Groeneveld tubes and annulus^[8] CHF and post CHF heat transfer correlations.

In Figures 25 through 39, plots of cladding surface temperature, fuel centerline temperature, and cladding elongation versus average fuel rod power are presented for the power calibration portion of the test. The symbols in each figure represent the experimental data. The solid line shows the analytical results from the FRAP-T1 calculations for a particular parameter versus average or local fuel rod power as appropriate.

[a] The analytical results were obtained by using FRAP-T1, MOD-02, EXPO4A in combination with the material properties code MATPRO MODPP2^[6].

Fig. 25 Steady-state cladding surface temperature at the 19.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 26. Steady-state cladding surface temperature at the 23.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 27 Steady-state cladding surface temperature at the 27.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 28 Steady-state fuel centerline temperature at the 23.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 29 Steady-state cladding elongation for the power calibration phase of IE Scoping Test 1.

Fig. 30 Steady-state cladding surface temperature at the 19.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 31 Steady-state cladding surface temperature at the 23.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 32 Steady-state cladding surface temperature at the 27.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 33 Steady-state fuel centerline temperature at 23.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 34 Steady-state cladding elongation for the power calibration phase of IE Scoping Test 1.

Fig. 35 Steady-state cladding surface temperature at the 19.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 36 Steady-state cladding surface temperature at the 23.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 37 Steady-state cladding surface temperature at the 27.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 38 Steady-state fuel centerline temperature at the 23.8-inch elevation during the power calibration phase of IE Scoping Test 1.

Fig. 39 Steady-state cladding elongation for the power calibration phase of IE Scoping Test 1.

Fig. 40 Steady-state cladding surface temperature at the 19.8-inch elevation during the preconditioning phase of IE Scoping Test 1.

Fig. 41 Steady-state cladding surface temperature at the 23.8-inch elevation during the preconditioning phase of IE Scoping Test 1.

Fig. 42 Steady-state cladding surface temperature at the 27.8-inch elevation during the preconditioning phase of IE Scoping Test 1.

Fig. 43 Steady-state fuel centerline temperature at the 23.8-inch elevation during the preconditioning phase of IE Scoping Test 1.

Fig. 44 Steady-state cladding elongation during the preconditioning phase of IE Scoping Test 1.

Fig. 45 Cladding surface temperature at the 19.8-inch elevation for the transient flow reduction phase of IE Scoping Test 1.

Fig. 45 Cladding surface temperature at the 23.8-inch elevation for the transient flow reduction phase of IE Scoping Test 1.

Fig. 47 Cladding surface temperature at the 27.8-inch elevation for the transient flow reduction phase of IE Scoping Test 1.

Fig. 48 Cladding elongation for the transient flow reduction phase of IE Scoping Test 1.

TABLE II

IE SCOPING TEST 1 AXIAL PROFILE^[a] FROM
THE GAMMA SCAN OF THE COBALT FLUX WIRE

<u>Elevation</u>	<u>Normalized Flux</u>
(inches above the bottom of the fuel rod)	(<u>calculated flux</u>) (<u>average flux</u>)
0.0	0.6258
0.9	0.6538
1.9	0.6897
3.4	0.7781
4.9	0.8939
6.4	1.016
7.9	1.125
9.4	1.203
10.9	1.244
12.4	1.248
14.4	1.226
15.9	1.192
17.4	1.157
18.9	1.132
20.4	1.118
21.9	1.108
23.4	1.093
24.9	1.061
26.4	1.006
27.9	0.9285
29.4	0.8363
30.9	0.7454
32.9	0.6745
34.6	0.6382

[a] The reader is reminded of the observed shift in the axial flux profile as derived from the SPND measurements and as discussed in this section. The cobalt wire profile is, of course, a time-power averaged result.

These calculated values are for steady state conditions. Three sets of plots are provided, one for each coolant flow rate. Figures 25 through 29 are for a nominal flow rate of 4 gpm, Figures 30 through 34 are for a nominal flow rate of 9 gpm and Figures 35 through 39 are for a nominal flow rate of 13 gpm. Slight variations in the coolant inlet temperature and flow rate are partially responsible for the scatter in the data.

Experimental data and posttest analytical results for the preconditioning portion of the test are presented in Figures 40 through 44. During the preconditioning period, the coolant flow rate varied from approximately 12-13 gpm. Since the FRAP-T1 calculations were performed for a constant inlet flow of 13 gpm, only test data with a nominal flow of 13 gpm is presented. As in the previous set of plots, symbols represent data and the calculated values are presented as a solid line. Multiple data points for each transducer are provided on each of these plots. Each data point represents the steady-state mean value of a parameter for a particular average or local fuel rod power. Spread in the dependent variable, for a particular power level, can again be partially attributed to variations in the nominal inlet flow rate and temperature, as well as fuel rod cladding surface oxidation, and fuel restructuring, e.g., densification and cracking.

The fuel centerline temperature data in Figures 28, 33, 38 and 43, is in reasonably good agreement within the calculated values. The output of the laser welded cladding surface thermocouple (TC 7), located at the 19.8-inch elevation is also in agreement with the posttest calculated values for the power calibration (See Figures 25, 30 and 35). As depicted in Figures 27, 32, 37 and 42, the welded cladding surface thermocouple (TC 8) indicates an unrealistically low temperature throughout the test, hence the output of this transducer should be disregarded. The measurements of cladding surface temperatures provided by the spring-loaded thermocouples at the 19.8-inch elevation above the bottom of the fuel rod are consistently lower than the temperature values provided by the welded thermocouples at the same location, while the welded thermocouple TC 7 agrees with the posttest calculated values.

Therefore, it can be concluded that there is a substantial systematic error in cladding surface temperature measurements by the spring-loaded thermocouples.

In Figures 29, 34, and 39, the cladding elongation values are presented. In reducing this data from the test, an offset was subtracted from the initial LVDT reading for each power calibration cycle. The value of the offset was determined at zero rod power prior to each power cycle. This value was then subtracted from the remainder of the data points of that particular cycle. The analytical values were handled in a similar manner; i.e., the axial growth of the fuel rod at zero average fuel rod power was taken to correspond to zero elongation. These procedures allowed for a comparison between the experimental data and FRAP-T1 calculated values for cladding elongation.

The cladding elongation during the preconditioning phase is shown in Figure 44. The sequence in which the data were recorded is shown by the direction of the arrows. The null position for the linear variable differential transformer apparently shifted over the duration of the preconditioning phase. It is believed that this offset is due to drift in the electronics.

Plots of cladding surface temperature and cladding elongation versus time for the transient flow reduction phase are depicted in Figures 45 through 48. Again, both FRAP-T1 calculations and experimental data are presented. The FRAP-T1 calculations for the flow reduction phase were initiated at approximately 30.94 hours and were terminated at approximately 31.17 hours. An average fuel rod power of 16.32 kW/ft (peak/average was 1.248) and the actual coolant flow rate history (Figure 21) were used in the FRAP-T1 calculations. Critical heat flux (CHF) occurred at a reduced flow rate of approximately 6.2 gpm (8.8×10^5 lb/hr-ft²) in the FRAP-T1 analysis as compared to 6.76 gpm (9.6×10^5 lb/hr-ft²) in the test. This lower flow rate results in a slight time delay between the test data and the FRAP-T1 analysis as is apparent in the figures. As indicated in Figures 46 and 47, the occurrence of film boiling precipitates large temperature excursions at the 23.8

and 27.8-inch elevations. The predicted cladding surface temperatures are in excess of those measured by either the welded or spring-loaded cladding surface thermocouples. CHF did occur at the 19.8-inch elevation during the test, but was not calculated to occur at this location, as shown in Figure 45. The response of the linear variable differential transformer and the calculated elongation are shown on Figure 48. The initial response of the calculated cladding growth at the onset of CHF is similar in slope to the measured data, but the analytical value continues to increase in magnitude past the point where the measured cladding elongation decreases. It is probable that the core of the LVDT jammed at some time during the film boiling transient and did not register further elongation.

4.2 Discussion of Test Results

The test results for this experiment consisted of two types of data: (a) the pre-film boiling or quasi-equilibrium data and (b) the film boiling or transient data. The error bands on the instruments and data system are presented in Appendices A and B.

4.2.1 Pre-Film Boiling. For this period of the test, the fuel rod behavior during each of the six power cycles (Figure 9) and the power ramp is discussed. The data from the internal pressure transducer is not presented due to excessive and uncorrectable transducer drift.

(1) Cladding Length Change. The changes in the fuel rod length as measured by the LVDT are plotted against rod average power in Figure 49. During the first power cycle, the fuel is in weak contact with the cladding at effectively zero power, causing the slope of the length change to be greater than that calculated for free thermal expansion which is also shown in Figure 49.

During the second power cycle, the fuel rod behavior is more conventional. The pellet-cladding mechanical interaction begins at about 4 kW/ft. In Figure 49, the slope of the cladding length change prior to 4 kW/ft is seen to be approximately the same as the free thermal expansion.

Fig. 49 Cladding length change versus rod average power during each of the six power cycles and the power ramp of IE Scoping Test 1. Pellet-cladding mechanical interaction (PCI) is seen to occur at lower power levels. All cladding strain is elastic. The maximum strain (35 mils) occurs in the second cycle.

At the peak power of about 8 kW/ft (rod average) the total fuel rod length growth is about 35 mils. During operation at constant power, the cladding length is seen to shorten by about 10 mils, probably due to fuel creep in response to the cladding stress on the fuel stack. During the power decrease portion of Cycle 2, PCI persists to about 6 kW/ft before the cladding length decrease takes on the same slope as the free thermal expansion. The cladding length decrease (2 or 3 mils) seen at the end of Cycle 2 is probably an artifact (drift) of the data system.

In all power cycles (Figure 49), the linear heat rating at the onset of PCI increases until Cycle 7 when it reaches about 7.5 kW/ft. During Cycle 7, it can be seen (Figure 49) that the rate of length increase above approximately 12 kW/ft reduces to zero and becomes negative above 14 kW/ft. The rod was held at an average power of 15 kW/ft for about 78 minutes during which time the length decreased by approximately 25 mils due to fuel creep. The change in the slope from positive to negative is associated with the high fuel temperatures and softening of the central core of the fuel pellets. The weaker fuel pellet stack is not capable of forcing additional axial strain in the cladding and the tensile stress in the cladding tends to deform the fuel stack.

The smooth curve drawn through the data in Cycle 7 (Figure 49) represents the average trend. The scatter in the data caused by variations in the power ramp is associated with the negative reactivity feedback from the driver core Doppler coefficient during rapid manual power increases.

(2) Fuel Centerline Temperature. The fuel centerline temperature versus the rod local power for each power cycle and the power ramp is shown in Figure 50. The local power was generated by applying a constant peaking factor (1.085), determined from the cobalt wire scan data, to the thermal hydraulically determined fuel rod average linear heat rating. According to the data in Figure 11, the axial flux shape at the thermocouple is close to constant. As shown in Section 4.1, the comparison with FRAP-T1 calculations was reasonably good.

Fig. 50 Fuel centerline temperature versus rod local power during each of the six power cycles and the power ramp phase of IE Scoping Test 1.

A small amount of hysteresis is seen in the temperature data during Cycles 2,4,5, and 6. At the same power (~ 9.5 kW/ft) of Cycles 2, 3, 4, and 6, the temperature increased 120°F , from 2520 to 2640°F over the operational period. This effect is due to fuel pellet cracking and/or densification of the fuel resulting in a larger diametral gap. This fuel has been shown to densify in out-of-pile sintering tests at 3270°F ^[9].

(3) Cladding Surface Temperature. Cladding surface temperatures measured by the laser welded thermocouple (TC 7) were generally good. As seen in Section 4.1, the results agreed quite closely with FRAP-T1 calculations. There are two main features that were readily apparent from this comparison. First, the slope of measured cladding surface temperature versus power was consistently slightly lower than the FRAP-T1 calculations. This was caused by a fin effect, locally cooling the rod, and/or any contact resistance between the thermocouple and the cladding. The second feature was the variability of the measurement, particularly at the zero power intercept. This was caused by an inlet temperature variation from 587 to 595°F during the test. In the forced convection regime when the coolant was still subcooled at the elevation of the thermocouple, the coolant temperature variation showed up as a variation of the cladding surface temperature.

The spring-loaded thermocouples (TC 1 through TC 6) indicated much lower temperatures than were calculated or were indicated by TC 7 (at the same elevation). As shown by the wide-spread thermocouple data in Section 4.1, there was a probable variability in the contact resistance between the different spring-loaded thermocouples, resulting in different slopes of temperature versus power.

4.2.2 Film Boiling. The LVDT and the cladding surface thermocouples indicated the onset of film boiling. All spring-loaded thermocouples were determined to be good indicators of local film boiling, but the measurement errors were very large (approximately 1000°F). The welded junction thermocouple at the 19.8-inch elevation (TC 7) indicated about 1900°F , which was later determined to be within 100°F of the metallurgically determined temperature^[4]. The same type of thermocouple at the 27.8-inch elevation did not function at all.

5. POSTIRRADIATION EXAMINATION RESULTS

Only the postirradiation visual examination results are presented in this report. Metallurgical examination of the fuel and cladding, cladding micro-hardness testing, examination of oxide penetrations in the cladding and oxide thickness, microprobe analysis, and burnup samples are presented in Reference 4.

Figure 51 shows the center of the fuel rod with the broken segment of fuel missing. Spalled oxide can be seen over the entire segment down to about the 18.75-inch elevation, the probable lower boundary of film boiling. The junction of TC 7 can be seen on the 335° view of Figure 51. The highest recorded temperature of this thermocouple was about 1900°F. TC 8 can be seen in the 155° view. No response was observed from this thermocouple during film boiling, probably as a result of prior thermocouple failure. Additional views of the fuel rod are shown in Figures 52 to 57. In Figures 52 and 54, the upper boundary of heavy oxide and, therefore, probably film boiling, can be seen to have occurred near the 32-inch elevation. In Figures 55 and 57 a smooth black oxide is seen to cover the top and bottom ends of the rod.

Fig. 51 IE Scooping Test 1, Rod IE-001 with spalling oxide and broken fuel segment shown at 155°, 245°, and 335° orientations.

Fig. 52 Upper portion of Rod IE-001, IE Scoping Test 1, showing spalled oxide up to about 31.5 inches (viewed from 245° orientation).

Fig. 53 Upper portion of Rod IE-001, IE scoping Test 1, showing spalled oxide up to about 32.0 inches (viewed from 155° orientation).

Fig. 54 Top of Rod IE-001, IE Scoping Test 1, showing black oxide above film boiling region (viewed from 180° orientation).

245° Orientation

Fig. 55 Lower portion of Rod IE-001, IE Scoping Test 1, showing sharp break in heavy oxide formation at about 18.75 inches (viewed from 245° orientations).

155° Orientation

Fig. 56 Lower portion of Rod IE-001, IE Scoping Test 1, showing sharp break in heavy oxide formation at about 18.75 inches (viewed from 155° orientation).

Start of Fuel Column - 0.4 in. (10 mm) from Bottom of Rod

180° Orientation

Fig. 57 Bottom of Rod IE-001, IE Scoping Test 1, showing smooth black oxide.

6. CONCLUSIONS

The objectives of this experiment, as defined in Section 1, were satisfied. Postirradiation examination of the fuel rod and the welded junction thermocouples has provided additional understanding of the test results^[4].

In addition to satisfying the scoping aspects of the test objectives, this test provided data on PCI and fuel centerline temperature during the power calibration and preconditioning phases of the test. Comparison of the test results and the posttest FRAP-T1 analysis was reasonably good and gave reasonable confidence in the code's ability to predict fuel behavior.

Of primary significance is the rod behavior during and following 4.8 minutes of continuous film boiling operation. The rod integrity was maintained until after termination of nuclear operation and no disruptive forces were associated with the failure which occurred a few minutes after shutdown.

7. REFERENCES

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APPENDIX A

INSTRUMENTATION CALIBRATION DATA

This appendix provides an overview of the calibration functions and error bands of the transducers used in IE Scoping Test 1. The equations given in this appendix for converting transducer output to engineering units are the same equations that were utilized in the reduction of the test data. It is important to note that these error bands do not reflect the systematic measurement error associated with the transducer installation. This additional error is particularly evident in the case of the spring-loaded thermocouple which is apparently affected by high contact resistance between the thermocouple tip and the fuel cladding surface. Generally, this additional error was small for the other instruments.

Where appropriate, the instrument calibration data, the calibration function which was used for test data reductions, and the 95% confidence interval estimate for the instrument calibration equations are presented as plots in this section. The calibration equations are presented in a tabular form in Table A-1. The calibration errors and the data system errors are jointly evaluated in Appendix B to provide a total system error over the range of each transducer.

The symbols on each graph represent data points. In the case of a transducer which was not calibrated, e.g., some of the thermocouples, the tabular data from standard calibration tables are presented as symbols. The solid line on each plot is the functional relationship for the data points of each transducer. The dashed lines are the 95% confidence interval estimate of the dependent variable for a particular independent variable.

The criterion for the 95% confidence interval estimate for those transducers that had a linear calibration equation was established using the following relation:

TABLE A-1

FUNCTIONS FOR IE SCOPING TEST 1 TRANSDUCERS

Measurement	Transducer	Calibration Equation	Range for which Calibration was derived
System Pressure	5000 psi Strain Post Type Pressure Transducer	$\text{psig} = 7.552 \times 10^2 + 1.000(\text{mV})^{[a]} + 1.693(\text{T})^{[b]}$	0 to 5000 psig
Coolant Inlet Flow	Turbine Flowmeter (Journal Bearing)	$\text{gpm} = 2.111 \times 10^{-2} (\text{mV}) + 0.261$	0 to 20 gpm
Coolant Temperature Differential	Copper-Constantan Thermocouple Pair (Type-T) DTC-1	$^{\circ}\text{F} = 3.068 \times 10^1 (\text{mV})$	600 to 650°F
Coolant Temperature Differential	Copper-Constantan Thermocouple Pair (Type-T) DTC-2	$^{\circ}\text{F} = 3.072 \times 10^1 (\text{mV})$	600 to 650°F
Coolant Temperature Differential	Inlet and Outlet RTD Pair	$^{\circ}\text{F} = 5.018 \times 10^{-1} (\text{mV})$	580 to 660°F
Cladding Surface Temperature	W5%Re/W26%Re Cladding Surface Thermocouple (Spring-loaded) No. 1, 2, 3, 4, 5, 6	$^{\circ}\text{F} = 1.942 \times 10^2 + 9.988 \times 10^1 (\text{mV}) - 7.813 \times 10^{-1} (\text{mV})^2 + 2.858 \times 10^{-2} (\text{mV})^3$	500 to 4000°F

TABLE A-1 (Continued)

Measurement	Transducer	Calibration Equation	Range for which Calibration was Derived
Cladding Surface Temperature	Chromel-Alumel Thermocouple (Welded) No. 7, 8	$^{\circ}\text{F} = 1.533 \times 10^2 + 4.518 \times 10^{-1}(\text{mV})$ $- 1.585 \times 10^{-1}(\text{mV})^2$ $+ 2.961 \times 10^{-3}(\text{mV})^3$	570 to 2500°F
Fuel Centerline Temperature	W5%Re/W26%Re Thermocouple	$^{\circ}\text{F} = 2.072 \times 10^2 + 1.072 \times 10^2(\text{mV})$ $- 8.542 \times 10^{-1}(\text{mV})^2$ $+ 2.832 \times 10^{-2}(\text{mV})^3$	400 to 4000°F
Coolant Inlet and Outlet Temperature	Chromel-Alumel Thermocouple	$^{\circ}\text{F} = 1.609 \times 10^2 + 4.310 \times 10^1(\text{mV})$	580 to 660°F
Cladding Elongation	Linear Variable Differential Transformer	$\text{mil} = 1.24(\text{mV})$	-170 to + 500 mil

[a] mV = millivolts

[b] T = coolant temperature

$$L_i = Y_i \pm t(1 - \alpha/2) S_{y \cdot x} \left[\frac{1}{n} + 1 + \frac{n(X_i - \bar{X})^2}{n\sum X^2 - (\sum X)^2} \right]^{1/2} \quad (A-1)$$

where

L_i = is the confidence interval estimate

Y_i = calculated value of the dependent variable for the regression equation

$t(1-\alpha/2)$ = value from a t distribution for $n-2$ degrees of freedom

α = level of confidence (0.95 was used here)

$S_{y \cdot x}$ = standard error of the regression of y on x

n = number of points

X = independent variable

$\bar{X}$ = mean value of the independent variable

X_i = value of the independent variable for which the error estimate applies.

The evaluation of L_i for several X_i values establishes the 95% confidence interval estimate.

For those transducers which had functional relationships that were non-linear or were multivariate, the 95% confidence interval estimate is given by:

$$L_i = Y_i \pm t(1 - \alpha/2)E \quad (A-2)$$

where E is the standard error for the multivariate or polynomial relationship, and the other terms are as defined previously. The standard error term contains cross-terms of the coefficients and independent variables. The above procedures were extracted from Reference A-1.

For those transducers that were not calibrated, the 95% confidence interval estimate is assumed to be specified by the accuracy limits as established by a manufacturer or an ANSI standard. The type of error estimate used for each transducer is specified below.

Test Train Instrumentation

- (1) The output of the 5000 psig strain post type system pressure transducer was a function of both transducer case temperature and system pressure. Hence, a linear multivariate calibration curve was established for this instrument. Two plots of transducer output versus pressure are given. One curve is given for each temperature at which the transducer was required to function during the test. Figure A-1 shows the transducer response for a coolant temperature of 590°F while Figure A-2 is for 620°F.
- (2) The response of the 10,000 psig pressure transducer was not considered reliable and a calibration is not shown here. This deduction resulted from a comparison of the pressure readings yielded by this instrument as compared to the pressure indications given by both the 5000 psig pressure transducer and the loop system pressure Heise gauge.
- (3) As mentioned in Section 3 of this report, one of the two installed flowmeters did not function during the test, therefore, only the calibration for the operational flowmeter is given. The calibration for this flowmeter was performed for a different inlet and outlet geometry than was used during the test. Previous experience, obtained by comparing calibration data for various flowmeters, has shown that a linear shift in

Fig. A-1 5,000 psig strain post type system pressure transducer. Calibration for coolant temperature of 590°F.

Fig. A-2 5,000 psig strain post type system pressure transducer. Calibration for coolant temperature of 620°F.

the calibration curve can be used to account for the variation in flow geometry. The calibration for the graphite journal bearing flowmeter, used for this test, was obtained as described above. The 95% confidence interval estimate for this calibration reflects the uncertainties resulting from the above calibration procedure. This calibration is presented in Figure A-3.

- (4) Both of the copper-constantan differential temperature measurement thermocouple pairs were calibrated. The results of these calibrations and the 95% confidence interval estimates are given in Figures A-4 and A-5. The calibration data were compiled for an inlet temperature of 600°F and a temperature differential of 40°F.
- (5) The platinum resistance thermometer (RTD) temperature differential pair were used but not calibrated as a differential pair. Therefore the 95% confidence interval estimate of the RTD pair was taken to be the square root of the sum of the squares of the accuracy for each RTD. Figures A-6 and A-7 show a functional relationship resulting from a linear regression fit of tabulated data for each RTD. These functions were developed from data that ranged from 580°F to 660°F. The error bands in each of these figures reflects the 0.5 percent of indicated reading accuracy specification as provided by the manufacturer. Figure A-8 shows the regression equation used for the differential temperature calibration. The combined accuracy limits for the RTD pair are also shown as dashed lines on this figure.
- (6) The chromel-alumel inlet and outlet thermocouples were not calibrated. The confidence intervals shown in Figure A-9 reflect an error limit of 0.375 percent of the indicated reading as specified by ANSI temperature measurement standards.

Fig. A-3 Coolant flow calibration function for graphite journal bearing turbine flowmeter.

Fig. A-4 Differential temperature measurement calibration for copper-constantan thermocouple pair DTC-1.

Fig. A-5 Differential temperature measurement calibration for copper-constantan thermocouple pair DTC-2.

Fig. A-6 Platinum resistance thermometer (RTD) No. 2312 calibration curve.

Fig. A-7 Platinum resistance thermometer (RTD) No. 2317 calibration curve.

Fig. A-8 Platinum resistance thermometer (RTD) temperature differential pair calibration.

Fig. A-9 Chromel-Alumel inlet and outlet thermocouple calibration.

- (7) The output of the linear variable differential transformer (LVDT) was sensitive to the transducer case temperature, the lead wire resistance, and the electronic settings on the signal conditioner used. The transducer that was utilized for this experiment was not calibrated for case temperature with the same lead wire and electronics that were encountered during the test.

The procedure followed in determining a calibration function for the LVDT is briefly described below. Initially the transducer output was monitored for various core displacements at two case temperatures, 79°F and 650°F. Then, at a later date, the transducer was calibrated with a different set of lead wires and electronics; i.e., those that were used for the test. The case temperature for this calibration was 76°F.

The data for these three calibrations are presented as symbols

in Figure A-10. A linear least squares fit was then obtained for each of the calibrations above. The change was determined in slope of the calibration function that occurred for the LVDT when the case temperature was increased, i.e., for the initial calibration. Next, it was necessary to determine a calibration function for the transducer that was applicable at elevated case temperatures and for the lead wires and electronics that were associated with the test. The above change in slope that occurred for the case temperature increase was assumed to apply to the LVDT calibration performed on the test train where the new lead wires and electronics were used. This function is plotted as a solid line in Figure A-10. Note that in obtaining this calibration function it was assumed that the change in slope of calibration, due to case temperature variation, was not affected by the implementation of different lead wires and electronic hardware.

Fig. A-10 Linear variable differential transformer calibration.

Due to the nature of the linear variable differential transducer calibration, an estimate of the 95% confidence interval is not deemed appropriate and is not presented here. The calibration procedure outlined above is not considered optimum, but practical limitations prevented correcting the deficiency in time for the test.

Fuel Rod Instrumentation

- (1) The output of the 2000 psig fuel rod internal pressure transducer drifted off scale during the test.
- (2) The spring-loaded, cladding surface thermocouples of tungsten-rhenium (W5%Re/W26%Re) alloy wire were not calibrated. The error bands and a polynomial output function for these transducers are given in Figure A-11. The error bands for these transducers were generated using the manufacturer's accuracy specification of 1.0 percent of the indicated reading.
- (3) The chromel-alumel thermocouples that were welded to the cladding surface were not calibrated. Again, the error bands associated with the output of these transducers reflect an ANSI temperature measurement standard of 0.375 percent of the indicated reading. The output function for these thermocouples is given in Figure A-12.
- (4) The data trace from the tungsten-rhenium (W5%Re/W26%Re) alloy fuel centerline thermocouple during the zero power portions of the test (i.e., isothermal conditions throughout the loop), indicates that the fuel centerline thermocouple reading was approximately 40°F higher than the inlet coolant temperature. The offset of the centerline thermocouple was consistent throughout the test, when the fuel rod power was zero. The higher reading was not due to fuel rod heat storage effects, since there were several phases of the test where the fuel rod

Fig. A-11 Spring-loaded tungsten, tungsten-rhenium (W5%Re/W26%Re) cladding surface thermocouple calibration.

Fig. A-12 Welded chromel-alumel (type-K) cladding surface thermocouple calibration.

power was zero for long periods of time. Also, there was evidence of this temperature discrepancy prior to nuclear operations.

Recognition of the above problem precipitated a critical review of the fuel centerline thermocouple calibration procedures. A calibration of the fuel centerline thermocouples was performed prior to their placement in the fuel rod. The thermocouples were calibrated over a range of temperatures from ambient to 4000°F in approximately 400°F increments. The calibration procedure required that the transducer be cycled through the calibration temperature range a sufficient number of times to obtain a transducer output that was repeatable within 1 percent of reading. This stabilization usually requires 3 to 4 cycles. The final cycle, after repeatability was established, was recorded as the calibration of the thermocouple. The cause of this shift in the thermocouple output is not known, but was assumed to be associated with metallurgical changes in the stranded wires that made up the thermoelements.

A polynomial fit to data from the standard calibration table yields temperature lower than the function obtained from the above calibration procedure at the millivolt output. This difference was approximately 40°F for a nominal temperature of 600°F (the temperature range of the inlet flow during the test). The discrepancy between the two functions was not a constant value. Plots of both functions are shown in Figure A-13. The calibration function deviates from the function derived from the standard data by a maximum of approximately 150°F at 4000°F.

Fig. A-13 Tungsten, tungsten-rhenium (W5%Re/W26%Re) fuel centerline thermocouple calibration.

The above discrepancy has been reviewed and is understood to be caused by the differences in the thermal environment of the entire thermocouple and extension wires between its calibration and the actual usage in the reactor. According to Moffat's theory^[A-2] the thermocouple output voltage is proportional to the temperature difference between the measuring junction and the reference junction. If the thermocouple materials (including extension leads) are homogeneous from one end to the other, the voltage output versus temperature can be determined from standard tables. If the thermocouple materials are not uniform, the entire thermocouple including the extension leads must be calibrated in a temperature gradient similar to the one in which it will be used.

Due to a direct result of the changes in the thermoelectric properties of this thermocouple caused by calibration, this thermocouple circuit is inhomogeneous over its length. The

thermocouple consisted of two regions of different material. One region extends from the thermocouple tip to a location 4 to 6 inches from the tip. This 4 to 6 inch location is where the thermocouple exited the high temperature zone of the calibration furnace. The second region begins at the end of the first (4 to 6 inches from the tip) and extends to the reference junction. A schematic of the thermocouple and typical temperature gradients are shown in Figure A-14. The second region consisted of material whose thermoelectric properties were not heat affected during calibration. The temperature indicated by the thermocouple circuit corresponded to the temperature determined by using the standard calibration tables when the entire circuit inside the in-pile tube was isothermal and the gradient existed across the boundary of the in-pile tube.

During nuclear operation, a gradient which is equivalent to the fuel centerline temperature existed in the tungsten-rhenium portion of the thermocouple which extends to about 10.8-inches from the tip (the top of the fuel stack). The gradient during nuclear operation was different from the gradient during calibration and thus neither function, standard or actual calibration, was the true calibration function for the as-used environment. Both functions are shown in Figure A-13.

Cause for decalibration of the fuel centerline thermocouple was attributed to either shunting of the beryllium oxide insulation or a change in the thermoelectric potential of the tungsten-rhenium wires. The mechanism for either or both of these occurrences was the vapor transport and desposition of metal vapors inside the thermocouple sheath. Gas phase transport of rhenium from the rhenium rich leg of the thermocouple to the other leg, or vapor deposition of tantalum from the sheath onto the thermoelement wires are phenomena that would cause the observed decalibration. Another possibility was that these metallic vapors become deposited in the beryllium

Fig. A-14 Schematic of installed fuel centerline thermocouple and typical temperature profiles for nuclear and pre-nuclear operations.

oxide insulation resulting in a decreased resistance between the thermocouple wires and the sheath. Impurities that were either not removed before, or introduced during, thermocouple fabrication could also contribute to degradation of the resistivity of the beryllium oxide. Both, theoretical and experimental evidence for these phenomena are readily found in the literature^[A-3, A-4, A-5, A-6]. Assuming one of these mechanisms was responsible for the decalibration, it was expected that the measurement error of the fuel centerline thermocouple would be less than 10 percent of reading^[A-3]. Also, due to the nature of the decalibration (i.e., a reduction in the millivolt output at a given temperature), the reported temperatures which were determined using the calibration function (Figure A-13) already partially compensated for the expected decalibration mechanisms.

An investigation was initiated to resolve and correct the deficiencies regarding the fuel centerline temperature measurement for future tests.

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APPENDIX B

SYSTEM ERROR ANALYSIS

In Appendix A of this report the calibrations for the test instrumentation were presented. Included were estimates of the uncertainty in each calibration. Two additional sources of error are: (a) measurement errors, i.e., the difference between the Standards Laboratory calibration of the transducer and the as-used calibration of the transducer/measurand system, and (b) decalibration and drift of the electronic systems used to acquire, record, and reduce the data.

Measurement errors ("a" above), while known to be extreme in some cases (especially cladding temperatures during critical heat flux transients), are not included in the error analysis of Appendix A due to lack of appropriate data.

In this appendix, uncertainties in the data acquisition, recording, and reduction electronic systems ("b" above) are presented. These are then combined with the standard calibration uncertainties of Appendix A to arrive at a first order estimate of the accuracy of the experimental data.

All data for the IE Scoping Test 1 were recorded on the Power Burst Facility Surveillance System (SS). The SS is an FM multiplexed data recording system which records four data signals on each of 13 tracks of a 14 track magnetic tape. A standard time code (IRIG-A) is recorded on the fourteenth track. Center frequencies for the signals on each track are 23.125, 41.875, 60.625, and 79.375 kHz; the frequency deviation for each carrier is ± 5.0 kHz.

Each data channel on the SS consists of a low level amplifier (LLA) with a gain variable in discrete steps from 1.0 to 1000 and a voltage controlled oscillator (VCO) with center frequency set to one of the four values listed above. A few of the data channels also include signal

conditioning or transducer excitation equipment such as digital countrate meters, constant current sources, and thermocouple reference junctions. The outputs of four VCOs, each with a different center frequency, are transmitted over a single landline from the reactor building to the control room and recorded directly on one of the 14 tracks of an analog tape.

For data reduction, the 14 track magnetic tape is played back at a speed eight times faster than the recording speed (60-inches/s playback versus 7.5-inches/s record). The tape track containing the data channel to be reduced is fed to a universal demodulator which selects the frequency band of interest and converts the FM signal to a time-varying voltage. This signal is then digitized at a selected digitizing interval and stored in a 2000 digital point memory. After the memory has been filled, the 2000 points are written on an IBM compatible digital magnetic tape.

From this brief discussion of the Surveillance System and data reduction process, several sources of error in the final reduced data can be identified:

- (1) Variations in the gain and offset of the low level amplifiers
- (2) Variations in the center frequency and frequency deviation of the voltage controlled oscillators
- (3) Decalibration of the data reduction equipment over long periods of time
- (4) Resolution of the analog to digital converter
- (5) Drift, decalibration, loading or other problems with signal conditioning or transducer excitation equipment.

Of these sources of error, drifts in the low level amplifiers and voltage controlled oscillators have been found (in the past) to be the most severe; therefore, tests have been performed to evaluate these drifts.

The results of a previous test [B-1] indicated that the combined drift of the LLA and VCO resulted in a total uncertainty of $\pm 1.5\%$ of span^[a] for those channels with gains up to 300, and $\pm 4.5\%$ of span for channels with gains of 1000.

Since that evaluation was performed, most of the LLAs in the SS have been replaced with new equipment. Therefore, another test was performed to evaluate the system. As in the previous test, a high quality DC voltage source was used to check the calibration of the system over a period of several days. Various DC voltages were input and the outputs were recorded on tape and digitized as described above.

The results of these tests show that the drift is independent of amplifier gain and is due, therefore, to the voltage controlled oscillators. This drift is long term, occurring over a period of 12 to 24 hours; thus the results apply directly to tests in PBF, which typically last for this period of time.

The standard deviation of any measured output voltage is

$$S_V = \left[\left(V \frac{S_G}{G} \right)^2 + (G S_{V_z})^2 \right]^{1/2} \quad (B-1)$$

where

S_V = standard deviation of output

V = measured output

G = gain

[a] Here, span refers to the difference between the maximum and minimum signals for which the channel is set up to record.

S_G = standard deviation of gain

S_{V_z} = standard deviation of zero offset

In particular it was found that

$$S_G/G = 0.65\%$$

$$GS_{V_z} = 2.7 \text{ millivolts}$$

With a total span on the output of 1000 millivolts on any channel, this expression can be simplified to

$$\frac{S_Y}{Y_{\max} - Y_{\min}}(\%) = \left[\left(\frac{Y - Y_{\min}}{Y_{\max} - Y_{\min}} \right)^2 \times (0.65\%)^2 + (0.27\%)^2 \right]^{1/2} \quad (\text{B-2})$$

where

Y = measured signal in engineering units

S_Y = standard deviation of Y

$Y_{\max}$ = maximum output signal

$Y_{\min}$ = minimum output signal

$Y_{\max} - Y_{\min}$ = span of output

This expression is evaluated for various values of Y in Table B-I.

TABLE B-I

UNCERTAINTY IN SURVEILLANCE SYSTEM OUTPUT SIGNALS

$Y - Y_{min}$	S_Y	$2 S_Y$
(% of Span)	(% of Span)	(% of Span)
0	0.27	0.54
25	0.32	0.64
50	0.42	0.84
75	0.56	1.12
100	0.70	1.40

The errors of Table B-I may now be combined with the transducer uncertainties to estimate the first order uncertainty of the experimental data. The results are presented in Table B-II. The "SS Error" and the "Cal. Error" are the 2 sigma uncertainties of the Surveillance System and the calibration respectively. The "Total Error" is the square root of the sum of the squares of these two errors.

TABLE B-II

SUMMARY OF UNCERTAINTY IN EXPERIMENTAL DATA

Transducer: Differential thermocouple No. 1 & 2 (Copper-Constantan)
 Data Channel Minimum: 0°F
 Data Channel Maximum: 31°F

Output (% of Span)	Output (°F)	SS Error (°F)	Cal. Error (°F)	Total Error (°F)
0	0	0.2	0.3	0.4
25	7.8	0.2	0.3	0.4
50	15.5	0.3	0.3	0.4
75	23.3	0.3	0.3	0.4
100	31	0.4	0.3	0.5

Transducer: Differential Platinum resistance Thermometer
 Data Channel Minimum: 0°F
 Data Channel Maximum: 50°F

Output (% of Span)	Output (°F)	SS Error (°F)	Cal. Error (°F)	Total Error (°F)
0	0	0.3	4.2	4.2
25	12.5	0.3	4.2	4.2
50	25	0.4	4.2	4.2
75	37.5	0.5	4.2	4.2
100	50	0.7	4.2	4.2

TABLE B-II (Continued)

Transducer: System Pressure (5000 psi)

Data Channel Minimum: 250 psi

Data Channel Maximum: 5250 psi

Output (% of Span)	Output (psi)	SS Error (psi)	Cal. Error (psi)	Total Error (psi)
0	250	30	30	60
25	1500	30	30	40
50	2750	40	30	50
75	4000	50	40	60
100	5250	70	50	90

Transducer: Cladding Surface Thermocouple (Chromel-Alumel)

Data Channel Minimum: 580°F

Data Channel Maximum: 1430°F

Output (% of Span)	Output (°F)	SS Error (°F)	Cal. Error (°F)	Total Error (°F)
0	580	5	5	7
25	790	5	6	8
50	1010	7	8	11
75	1220	9	9	13
100	1430	12	11	16

TABLE B-II (Continued)

Transducer: Cladding Surface Thermocouple (W5%Re/W26%Re)^[a]
 Data Channel Minimum: 580°F
 Data Channel Maximum: 3800°F

Output (% of Span)	Output (°F)	SS Error (°F)	Cal. Error (°F)	Total Error (°F)
0	580	17	9	19
25	1390	21	14	25
50	2190	27	22	35
75	3000	33	30	45
100	3800	45	38	59

Transducer: Inlet Flowmeter
 Data Channel Minimum: 0.3 gpm
 Data Channel Maximum: 21.4 gpm

Output (% of Span)	Output (gpm)	SS Error (gpm)	Cal. Error (gpm)	Total Error (gpm)
0	0.3	0.1	1.0	1.0
25	5.6	0.1	1.0	1.0
50	10.9	0.2	1.0	1.0
75	16.1	0.2	1.0	1.0
100	21.4	0.3	1.0	1.0

TABLE B-II (Continued)

Transducer: Fuel Centerline Thermocouple (W5%Re/W26%Re)
 Data Channel Minimum: 200°F
 Data Channel Maximum: 3900°F

<u>Output</u> <u>(% of Span)</u>	<u>Output</u> <u>(°F)</u>	<u>SS Error</u> <u>(°F)</u>	<u>Cal. Error</u> <u>(°F)</u>	<u>Total Error</u> <u>(°F)</u>
0	200	20	[b]	[b]
25	1130	24	[b]	[b]
50	2050	31	[b]	[b]
75	2980	38	[b]	[b]
100	3900	52	[b]	[b]

Transducer: Coolant Temperature Thermocouple (Chromel-Alumel)
 Data Channel Minimum: 581°F
 Data Channel Maximum: 667°F

<u>Output</u> <u>(% of Span)</u>	<u>Output</u> <u>(°F)</u>	<u>SS Error</u> <u>(°F)</u>	<u>Cal. Error</u> <u>(°F)</u>	<u>Total Error</u> <u>(°F)</u>
0	581	0.5	2.2	2.3
25	603	0.6	2.3	2.4
50	624	0.7	2.3	2.4
75	646	0.9	2.4	2.6
100	667	1.2	2.5	2.8

TABLE B-II (Continued)

Transducer: Cladding Linear Variable Differential Transformer
 Data Channel Minimum: 0 mils
 Data Channel Maximum: 744 mils

<u>Output</u> <u>(% of Span)</u>	<u>Output</u> <u>(mils)</u>	<u>SS Error</u> <u>(mils)</u>	<u>Cal. Error</u> <u>(mils)</u>	<u>Total Error</u> <u>(mils)</u>
0	0	4	[b]	[b]
25	186	5	[b]	[b]
50	372	6	[b]	[b]
75	558	8	[b]	[b]
100	744	10	[b]	[b]

[a] See Appendix A for discussion of additional sources of measurement errors.

[b] Undefined, see Appendix A.

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APPENDIX C

POWER CALIBRATION - TECHNIQUES AND RESULTS

Thermal Hydraulic Calibration Technique

The determination of fuel rod power was based upon the fact that, under equilibrium conditions, the power generated in the fuel rod is equal to the energy storage in the coolant flowing past the rod. A basic formulation for fuel power can then be obtained by writing mass and energy balances for the coolant as idealized in the system shown in Figure C-1.

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Fig. C-1 Schematic of flow and heat addition for calculation of fuel rod power IE Scoping Test 1.

The mass balance of the coolant system is idealized assuming the mass coming into the inlet of the system is equal to the mass leaving the system^[a].

Mathematically, this balance is expressed as:

$$\dot{m} = \rho_{in} V_{in} = \rho_{out} V_{out} \quad (C-1)$$

where

$\dot{m}$ $\equiv$ coolant mass flow rate past the rod

ρ $\equiv$ density

V $\equiv$ volumetric flow rate.

The subscripts "in" and "out" refer to the inlet and outlet conditions, respectively.

Considering only terms of a relative magnitude of 0.01% or more^[b], the energy balance is expressed as:

$$Q_{in} = \dot{m} (h_{out} - h_{in}) + Q_{out} \quad (C-2)$$

where

Q_{in} $\equiv$ heat added to coolant from the fuel rod

[a] This is not true in a strict sense since there are leakage paths past the spring-loaded cladding surface thermocouples (into the bypass region) and possibly through attachment slots in the LVDT housing. An experiment evaluating the leakage past the thermocouples was performed. The results from this experiment indicate that the maximum leakage that might be expected is less than 5% of the total flow rate.

[b] For the range of conditions expected for the present test hardware, the first term $\dot{m} (h_{out} - h_{in})$, accounts for 97% of the energy storage, Q_{out} the remaining 3%. The contribution due to kinetic and potential energy, frictional losses, etc., is less than 0.01%.

$h \equiv$ coolant enthalpy

$Q_{out} \equiv$ heat loss through the flow shroud.

The first term on the right hand side of Equation (C-2), $\dot{m} (h_{out} - h_{in})$ can be expressed in another form which is readily applicable to power calculations. This is:

$$\dot{m} (h_{out} - h_{in}) = \dot{m} C_p (T_{out} - T_{in}) \quad (C-3)$$

where

$C_p \equiv$ coolant specific heat

$T \equiv$ coolant bulk temperature.

The remaining term, Q_{out} , can be calculated using a simple electrical analog heat transfer model as shown in Figure C-2. Using this model, Q_{out} can be written as

$$Q_{out} = \frac{T_{out} - T_{in}}{2R} \quad (C-4)$$

where R is calculated as

$$R = \frac{1}{2\pi L} \left[\frac{1}{r_a h_a} + \frac{1}{r_b h_b} + \frac{\ln(r_b/r_a)}{K_s} \right] \quad (C-5)$$

where

$r_a, r_b \equiv$ inner and outer radius of the flow shroud

$L \equiv$ effective length of the heat transfer region

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Fig. C-2 Geometric and electric analogy model for heat loss through the flow shroud of Rod IE-001.

$h_a, h_b \equiv$ convective coefficients for forced convection on the inner and outer surface of the shroud

$K_s \equiv$ shroud thermal conductivity.

Briefly then, rod power is calculated using two main equations:

$$\dot{m} = \rho V \quad (C-6)$$

and

$$Q_{in} = \dot{m} C_p (T_{out} - T_{in}) + \frac{T_{out} - T_{in}}{2R} \quad (C-7)$$

In the following sections, the application of these equations and the results obtained from them are discussed.

Application of Technique

Reduced data from several different measurements are used as input or to calculate an input for equations C-6 and C-7. These include:

- (1) Inlet temperature
- (2) System pressure
- (3) Volumetric flow rate
- (4) Coolant differential temperature.

System conditions such as inlet temperature and system pressure are used to determine the properties of the coolant, such as C_p , ρ , etc. In particular, C_p is calculated at the average coolant temperature, while ρ is calculated at the inlet temperature. These properties are calculated using the ASTEM program^[C-1] which solves the 1967 ASME equations of state for water. The measurements of volumetric flow rate and coolant differential temperature are then used directly for V (with appropriate unit conversion) and $T_{out} - T_{in}$ to complete the solution.

Flux Integral Technique

The flux integral technique depends upon the correlation between the combined output from a set of axially distributed SPNDs to the thermal hydraulically determined fuel rod power. In this method the relative neutron axial flux profile is first determined using a sine function fitted to the output of the SPNDs. This function of SPND output versus axial elevation is integrated, normalized and then compared to the thermal hydraulic rod power. The following sections show the equations used and the application of the technique to reduced experimental data.

Formulation of Equations

The first step in the technique is to represent the axial flux profile using a sine function in the form:

$$\phi(X) \propto ND(X) = \sum_{n=1}^N B_n \sin \frac{n\pi X}{L} \quad (C-8)$$

where

$\phi(X)$ $\equiv$ neutron flux profile

$ND(X)$ $\equiv$ profile in terms of detector output

L $\equiv$ effective length of the axial flux profile, where the flux extrapolates to zero at each end. This has been determined experimentally using cobalt flux wires to be about 40 to 44-inches. The latter (44-inches) has been used to derive the flux profiles of Figure 11.

X $\equiv$ axial elevation measured from bottom of active fuel region

$B_n \equiv$ coefficient

$N \equiv$ number of detectors.

The coefficients, B_n , are determined through the simultaneous solution of

$$\begin{aligned} ND(X_1) &= \sum_{n=1}^N B_n \sin \frac{n\pi X_1}{L} \\ &\vdots \\ ND(X_n) &= \sum_{n=1}^N B_n \sin \frac{n\pi X_n}{L} \end{aligned} \quad (C-9)$$

where

$ND(X_n) \equiv$ detector signal at X_n .

Using the representative profile, $ND(X)$, test rod power is evaluated as

$$\text{Power} = \frac{K \int_a^b ND(X) dx}{b - a} = \frac{K \sum_{n=1}^N \frac{L}{n\pi} B_n \cos \frac{n\pi X}{L} \Big|_b^a}{b - a} \quad (C-10)$$

where

$K \equiv$ constant of proportionality,

$a \equiv$ bottom of active fuel region,

$b \equiv$ top of active fuel region.

At a low enough test fuel rod power level to preclude two phase flow at the outlet of the experiment, a comparison is made between power determined by the thermal hydraulic technique and the flux integral technique. In this comparison, a best fit for K is determined in the form:

$$K = \frac{\text{Power (Thermal Hydraulic)}}{\frac{\int_a^b ND(X) dX}{b - a}} \quad (C-11)$$

Thus, power can be determined directly using the neutron detectors; even at high test fuel rod powers where the thermal hydraulic technique is no longer valid, due to the presence of voids in the coolant exiting the test space.

In addition, local power along the rod can be determined. This is given by

$$LHR(X) = K \cdot ND(X) \quad (C-12)$$

where

$LHR \equiv$ local power.

Application of Technique

The application of the flux integral technique is straight forward. The effective length is determined experimentally using cobalt flux wire data. Then the reduced data for each detector along with their locations is used to solve the representative flux profile. The proportionality constant K is then determined using the steady-state results from the power calibration portion of the test. The effects of void fraction and high fuel temperature on the reliability of this technique have previously been addressed.

Power Calibration Results

Average thermal hydraulic rod power (or linear heat rating when expressed in terms of kW/ft) was calculated from (a) coolant inlet temperature thermocouples, (b) coolant inlet differential temperature thermocouple pair, (c) inlet flowmeter, and (d) 500 psig system pressure transducer.

An axial peaking factor of 1.32, which was obtained from physics calculations, was used to convert from the average linear heat rating to a peak linear heat rating. Data from the five SPNDs shows the instantaneous axial flux profile shifts as a function of power due to changes in driver core control rod position. Thus, the actual peaking factor also varies with power. However, for average rod linear heat ratings in excess of 8 kW/ft, the peaking factor is approximately 1.32. This is due to the fact that at high powers the normalized axial flux profile remains constant. In fact, for heat ratings in excess of 3 kW/ft, the normalized flux profile on the upper half of the rod changes minimally. Thus, local power at each of the centerline and cladding surface thermocouple locations can be calculated using a constant times average power for most of the test.

The time-integrated axial power profile was obtained from a gamma scan of a cobalt flux wire mounted in the hardware. The peaking factor from the flux wire scan was 1.26, which is lower than the instantaneous factor due to the cumulative effects of the variation in the skew in the profile. The important characteristics of this profile, shown in Figure C-1, are listed in Table C-1.

TABLE C-1

CHARACTERISTICS OF THE INTEGRATED AXIAL FLUX PROFILE
OBTAINED FROM GAMMA SCAN OF THE COBALT FLUX WIRE

<u>Characteristic</u>	<u>Value</u>
Peak to average	1.26
Peak location	12.45 inches from bottom of fuel stack
Left Offset Zero ^[a]	3.09 inches
Right Offset Zero ^[b]	6.06 inches

[a] The distance from the bottom of the fuel stack to the point where the flux is extrapolated to zero.

[b] The distance from the top of the fuel stack to the point where the flux is extrapolated to zero.

Uncertainty in Power Calibration Results. Uncertainties in the average linear heat ratings as presented in the text were calculated using the error estimates for the data (flow rate, ΔT , etc.) presented in Appendix B. This was done using a standard error propagation technique. Rod power is:

$$RP = F (G, \Delta T, T, P) \quad (C-13)$$

where

$G \equiv$ flow rate

$\Delta T \equiv$ coolant temperature rise across the experiment

$T \equiv$ inlet temperature

$P \equiv$ system pressure.

Then

$$\sigma_{RP}^2 \left(\frac{(\Delta RP)^2}{\Delta G} \sigma_G^2 + \left(\frac{\Delta RP}{\Delta(\Delta T)} \right)^2 \sigma_{\Delta T}^2 + \left(\frac{\Delta RP}{2T} \right)^2 \sigma_T^2 + \left(\frac{\Delta RP}{\Delta P} \right)^2 \sigma_P^2 \right) \quad (C-14)$$

The results are shown in Table C-II.

The uncertainties in the normalized axial flux profile, as shown in Figure 10, were calculated at various points on the sine function fit of the SPND signals. Thus, this uncertainty includes effects of skewing at power levels from 0 kW/ft up to the maximum reached during the test plus effects of any drift in the SPND amplifiers propagated through the sine fit. Since most of the tests were run at average linear heat ratings greater than 3 kW/ft, almost all of the uncertainty in the upper half of the rod is due to difficulties in the SPND amplifiers.

TABLE C-II
UNCERTAINTIES IN ROD POWER^[a]

Rod Power ^[b] (kW/ft)	Total Error ^[c] (kW/ft)
0	0.58
4.22	0.58
8.40	0.58
12.62	0.57
16.79	0.59

[a] At system condition of: pressure, 2150 psia; inlet temperature, 620°F; and flow rate, 10 gpm.

[b] The rod powers shown in this table correspond to the coolant ΔT shown in Appendix B.

[c] Based on uncertainties in transducer calibration, data acquisition, and propagation of these uncertainties in the thermal balance. The 95% confidence interval estimate corresponding to the indicated rod power is given by: Rod Power $\pm$ Total Error.

REFERENCES

- C-1 K. V. Moore, ASTEM - A Collection of Fortran Subroutines to Evaluate the 1967 ASME Equations of State for Water/Steam and Derivatives of these Equations, ANCR-1026 (October 1971).

APPENDIX D

FUEL ROD CHARACTERIZATION DATA

The pretest fuel rod characterization data for IE Scoping Test 1 is presented in this appendix. These data are reported: (a) to facilitate a comparison of fuel rod physical characteristics between IE experiments and (b) to provide input for the analytical models used in the prediction of fuel rod behavior during nuclear operations. The data were obtained from the unirradiated fuel rod in the as-built configuration.

The overall fuel rod and cladding data are contained in Table D-I. The 95% population loaded value refers to the dimensional range of a parameter within which 95% of the measurements were found. The void volume measurement includes the void found in both the fuel rod and the instrumented end cap.

In Table D-II, the fuel rod cladding inside diameter is listed in increments over the length of the fuel rod which contains the fuel stack. An air gauge was used to obtain these diametral measurements. The accuracy of these measurements is ± 0.0001 inch. Table D-III lists the fuel pellet characterization data. The pellet length measurements listed in this table were obtained by measuring the length between the upper and lower dish shoulder of each pellet. Both the diametral and length measurements were performed using a Bausch and Lomb optical gauge, Model BR-25. The measurements are accurate to ± 0.0002 inch. An analytic balance, accurate to ± 1 milligram was used to provide pellet mass data.

Figure D-1 provides a plot of both the fuel rod cladding inside diameter as a function of axial location (as taken from Table D-I) and the average diameter of each fuel pellet in the fuel stack. The distance between these two plots represents the diametral gap between the fuel rod cladding and the fuel stack over the entire length of the fuel rod.

TABLE D-I

OVERALL PREIRRADIATED FUEL ROD AND CLADDING DATA
FOR FUEL ROD IE-001

Finished Tube Data:

Mean O/D	0.3910 inch (0.9931 cm)
Mean I/D	0.3446 inch (0.8753 cm)
Ovality (Max (ID/Min ID)	1.002
Overall Length	38.167 inch (96.944 cm)

Mechanical Properties of Fuel Rod Cladding^[a]

Yield Tensile Strength

At Room Temperature	82,900 psi (5.72×10^8 nt/m ²)
At 725F (385C)	56,400 psi (3.89×10^8 nt/m ²)

Ultimate Tensile Strength

At Room Temperature	112,100 psi (7.73×10^8 nt/m ²)
At 725F (385C)	71,000 psi (4.90×10^8 nt/m ²)

Fuel Data;

Enrichment	9.5% ²³⁵ U
Mean Geometric Density	10.067 gram/cm ³ (92%)
95% Population Loaded	10.067 + 0.041 - 0.027 gram/cm ³
Mean Diameter	0.3361 inch (0.8536 cm)
95 Population Loaded	0.3361 + 0.0013 - 0.0018 inch (.8536 + 0.0005 - 0.0007 cm)
Stack Mass	489.1125 gram
Measured Stack Length	34.6257 inch (87.9493 cm)

Dish Dimensions

Dish Chord	0.260 inch (0.660 cm)
Dish Depth	0.0135 inch (0.0343 cm)

Grain Size

ASTM Micrograin Size	No. 13.2
Grain Diameter	1.46×10^{-4} inch (3.7×10^{-4} cm)
Center Hole Length	14.8713 inch (37.7731 cm)
Plenum Length	3.042 inch (7.7267×10^{-2} cm)
Measured Void Volume	11.8 ± 0.2 cm ³

[a] Data Supplied by Westinghouse Electric Corporation.

TABLE D-II

FUEL ROD CLADDING DATA FOR SAXTON TYPE, ZIRCALOY-4 FUEL ROD IE-001

TUBE INSIDE DIAMETER MEASUREMENTS						TUBE OUTSIDE DIAMETER MEASUREMENTS					
Distance From Bottom of Rod		Inside Diameter		Distance From		Inside Diameter		Distance From		Outside Diameter	
(inches)	(cm)	(0°)	(90°)	(inches)	(cm)	(0°)	(90°)	(inches)	(cm)	(0°)	(90°)
1	2.5	0.8755	0.8755	19	48.3	0.8750	0.8750	0.6	1.4	0.9931	0.9931
2	5.1	0.8753	0.8750	20	50.8	0.8750	0.8750				
3	7.6	0.8750	0.8745	21	53.3	0.8753	0.8745	11.6	29.3	0.9931	0.9928
4	10.1	0.8753	0.8750	22	55.9	0.8750	0.8750				
5	12.7	0.8755	0.8750	23	58.4	0.8753	0.8753	23.6	59.8	0.9936	0.9926
6	15.2	0.8758	0.8758	24	61.0	0.8750	0.8748				
7	17.8	0.8753	0.8753	25	63.5	0.8748	0.8748	35.6	90.3	0.9931	0.9931
8	20.3	0.8750	0.8748	26	66.0	0.8750	0.8750				
9	22.9	0.8753	0.8750	27	68.6	0.8755	0.8755				
10	25.4	0.8748	0.8743	28	71.1	0.8758	0.8750				
11	27.9	0.8750	0.8753	29	73.7	0.8760	0.8745				
12	30.5	0.8750	0.8750	30	76.2	0.8755	0.8745				
13	33.0	0.8755	0.8753	31	78.7	0.8753	0.8745				
14	35.6	0.8753	0.8750	32	81.3	0.8750	0.8750				
15	38.1	0.8748	0.8748	33	83.8	0.8753	0.8753				
16	40.6	0.8755	0.8753	34	86.4	0.8755	0.8753				
17	43.2	0.8750	0.8750	35	88.9	0.8760	0.8750				
18	45.7	0.8753	0.8755								

TABLE D-III

PHYSICAL CHARACTERISTICS OF THE SAXTON TYPE, 92% THEORETICAL DENSITY, 9.5% ENRICHED,
DISHED END, UO₂ FUEL PELLETS FOR FUEL ROD IE-001

Pellet ^[a] No.	Diameter (cm)				Length (cm)			Mass (gram)	Geometric Density (gram/cm ³)	Immersion Density (gram/cm ³)
	Top	Center	Bottom	Average	0°	90°	Average			
1	0.8539	0.8539	0.8539	0.8539	1.5707	1.5705	1.5706	8.9231	10.051	10.112
2	0.8542	0.8542	0.8537	0.8540	1.5839	1.5855	1.5772	9.0226	10.073	
3	0.8537	0.8537	0.8537	0.8537	1.5545	1.5565	1.5658	8.8356	10.056	
4	0.8534	0.8534	0.8534	0.8534	1.5560	1.5583	1.5609	8.8257	10.039	
5	0.8534	0.8534	0.8534	0.8534	1.5616	1.5631	1.5613	8.8636	10.045	10.104
6	0.8545	0.8542	0.8537	0.8541	1.5580	1.5580	1.5580	8.8572	10.052	
7	0.8539	0.8539	0.8539	0.8539	1.5700	1.5679	1.5640	8.9320	10.079	
8	0.8537	0.8537	0.8537	0.8537	1.5636	1.5636	1.5636	8.9323	10.112	
9	0.8537	0.8539	0.8539	0.8538	1.5654	1.5621	1.5645	8.9061	10.076	10.104
10	0.8537	0.8537	0.8539	0.8538	1.5629	1.5657	1.5637	8.9000	10.071	
11	0.8532	0.8537	0.8539	0.8536	1.5591	1.5634	1.5614	8.8753	10.063	
12	0.8537	0.8537	0.8534	0.8536	1.5641	1.5606	1.5628	8.8604	10.040	
13	0.8537	0.8537	0.8537	0.8537	1.5728	1.5761	1.5678	8.9384	10.050	10.102
14	0.8534	0.8537	0.8537	0.8536	1.5834	1.5845	1.5756	9.0392	10.102	
15	0.8539	0.8539	0.8539	0.8539	1.5847	1.5847	1.5847	9.0157	10.066	
16	0.8537	0.8537	0.8537	0.8537	1.5804	1.5809	1.5825	8.9787	10.056	
17	0.8542	0.8542	0.8537	0.8540	1.5603	1.5591	1.5597	8.8591	10.049	10.089
18	0.8534	0.8534	0.8537	0.8535	1.5547	1.5512	1.5572	8.8230	10.064	
19	0.8532	0.8534	0.8537	0.8534	1.5596	1.5560	1.5584	8.8897	10.108	
20	0.8534	0.8537	0.8534	0.8535	1.5560	1.5565	1.5563	8.8367	10.058	
21	0.8542	0.8539	0.8534	0.8538	1.5857	1.5878	1.5867	9.0336	10.073	10.089
22	0.8537	0.8539	0.8539	0.8538	1.5900	1.5936	1.5918	9.0795	10.091	
23	0.8539	0.8542	0.8542	0.8541	1.5880	1.5867	1.5874	9.0413	10.070	
24	0.8532	0.8534	0.8534	0.8533	1.5370	1.5410	1.5390	8.7368	10.056	
25	0.8542	0.8545	0.8545	0.8544	1.5987	1.5987	1.5987	9.1104	10.070	10.089
26	0.8534	0.8534	0.8534	0.8534	1.5499	1.5502	1.5500	8.7859	10.040	
27	0.8542	0.8542	0.8539	0.8541	1.5578	1.5545	1.5561	8.8449	10.050	
28	0.8529	0.8534	0.8537	0.8533	1.5702	1.5710	1.5706	8.9301	10.071	
29	0.8539	0.8537	0.8537	0.8538	1.5606	1.5621	1.5614	8.8810	10.070	10.101
30	0.8539	0.8539	0.8539	0.8539	1.5321	1.5324	1.5323	8.7490	10.101	
31	0.8534	0.8534	0.8534	0.8534	1.5903	1.5865	1.5884	9.0635	10.106	
32	0.8539	0.8542	0.8539	0.8540	1.5895	1.5893	1.5894	9.0474	10.070	
33 ^[b]	0.8542	0.8542	0.8539	0.8541	1.5725	1.5725	1.5725	8.4930		10.089
34	0.8534	0.8534	0.8534	0.8534	1.6005	1.5987	1.5996	8.6528		
35	0.8534	0.8534	0.8534	0.8534	1.5649	1.5631	1.5640	8.3964		
36	0.8529	0.8529	0.8527	0.8528	1.6700	1.6637	1.6619	8.3456		
37	0.8529	0.8529	0.8529	0.8529	1.5735	1.5768	1.5752	8.5016		10.089
38	0.8532	0.8532	0.8532	0.8532	1.5695	1.5685	1.5690	8.4466		
39	0.8534	0.8534	0.8534	0.8534	1.5814	1.5814	1.5814	8.5209		
40	0.8529	0.8529	0.8529	0.8529	1.5552	1.5552	1.5555	8.3930		
41	0.8532	0.8532	0.8534	0.8533	1.5908	1.5908	1.5908	8.6166		10.089
42	0.8532	0.8534	0.8534	0.8533	1.5596	1.5568	1.5582	8.4093		
43	0.8534	0.8534	0.8532	0.8533	1.5794	1.5715	1.5754	8.5242		
44	0.8545	0.8545	0.8542	0.8544	1.5903	1.5911	1.5907	8.5969		
45	0.8532	0.8532	0.8532	0.8532	1.5664	1.5621	1.5643	8.3140		10.089
46	0.8529	0.8539	0.8539	0.8538	1.5883	1.5862	1.5873	8.5823		
47	0.8534	0.8534	0.8534	0.8534	1.5928	1.5872	1.5900	8.6120		
48	0.8529	0.8532	0.8532	0.8531	1.5837	1.5837	1.5837	8.5560		
49	0.8539	0.8539	0.8537	0.8538	1.5679	1.5682	1.5680	8.4344		10.089
50	0.8534	0.8534	0.8534	0.8534	1.5636	1.5664	1.5650	8.4731		
51	0.8532	0.8534	0.8537	0.8534	1.6685	1.6606	1.6690	8.4698		
52	0.8532	0.8537	0.8537	0.8535	1.5918	1.5857	1.5888	8.5902		
53	0.8529	0.8529	0.8529	0.8529	1.5646	1.5624	1.5635	8.4639		10.089
54	0.8529	0.8529	0.8529	0.8529	1.5524	1.5542	1.5533	8.3357		
55	0.8524	0.8532	0.8532	0.8529	1.5733	1.5728	1.5730	8.4604		
56	0.8537	---	0.8542	0.8540	1.5702	1.5738	1.5720	8.5160		

[a] Pellets are numbered from bottom to top of fuel stack.

[b] Pellets from this point up are center-drilled.

All measurements, with the exception of mass and volume, were obtained using devices that are scaled for U.S. Customary Units.

Fig. D-1 Fuel pellet-cladding diametral gap profile for fuel Rod IE-001. Data were obtained at ambient temperatures and represents the as-built configuration.

APPENDIX E

ADDITIONAL TEST RESULTS DATA

Included in this appendix are the experimental data which were necessary for the interpretation of the test results (system pressure, coolant temperatures, etc.) but not of primary interest with respect to fuel behavior.

For each of the four periods of operation as discussed in Section 3, the coolant inlet and outlet temperatures, the coolant differential temperatures, the neutron detector results, and the system pressure are presented in Figures E-1 through E-16.

Fig. E-1 Coolant inlet and outlet temperature during the power calibration phase of IE Scoping Test 1.

Fig E-2 Uncorrected coolant differential temperature during the power calibration phase of IE Scoping Test 1.

Fig. E-3 Uncorrected self-powered neutron detector output during the power calibration phase of IE Scoping Test 1.

Fig. E-4 System pressure and rod internal pressure during the power calibration phase of IE Scoping Test 1.

Fig. E-5 Coolant inlet and outlet temperature during the preconditioning phase of IE Scoping Test 1.

Fig. E-6 Uncorrected coolant differential temperature during the preconditioning phase of IE Scoping Test 1.

Fig. E-7 Uncorrected self-powered neutron detector output during the preconditioning phase of IE Scoping Test 1.

Fig. E-8 System pressure and rod internal pressure during the preconditioning phase of IE Scoping Test 1.

Fig. E-9 Coolant inlet and outlet temperature during the power ramp phase of IE Scoping.

Fig. E-10 Uncorrected coolant differential temperature during the power ramp phase of IE Scoping Test 1.

Fig. E-11 Uncorrected self-powered neutron detector output during the power ramp phase of IE Scoping Test 1.

Fig. E-12 System pressure and rod internal pressure during the power ramp phase of IE Scoping Test 1.

Fig. E-13 Coolant inlet and outlet temperature during the flow reduction phase of IE Scoping Test 1.

Fig. E-14 Uncorrected coolant differential temperature during the flow reduction phase of IE Scoping Test 1.

Fig. E-15 Uncorrected self-powered neutron detector output during the flow reduction phase of IE Scoping Test 1.

Fig. E-16 system pressure and rod internal pressure during the flow reduction phase of IE SCoping Test 1. Internal pressure transducer functioned well enough to show the rod failure.

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