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**IRRADIATION EFFECTS TEST SERIES
TEST IE-3
TEST RESULTS REPORT**

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October 1977

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DEPARTMENT OF ENERGY

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TEST IE-3
TEST RESULTS REPORT

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THERMAL FUELS BEHAVIOR PROGRAM
EG&G IDAHO, INC.

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IRRADIATION EFFECTS TEST SERIES

TEST IE-3

TEST RESULTS REPORT

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EG&G IDAHO, INC.

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ABSTRACT

This report describes the results of Test IE-3, the fifth test in the Irradiation Effects Test Series conducted under the Thermal Fuels Behavior Program of EG&G Idaho, Inc. for the Nuclear Regulatory Commission. The objectives of this test were to: (a) determine the behavior of irradiated fuel rods subjected to a rapid power increase during which the possibility of a pellet-cladding mechanical interaction failure is enhanced and (b) determine the behavior of these fuel rods during film boiling following this rapid power increase. Test IE-3 used four, 0.97-m long pressurized water reactor type fuel rods fabricated from previously irradiated fuel. The fuel rods were subjected to a preconditioning period, followed by a power ramp to 69 kW/m at a coolant mass flux of 4920 kg/s-m^2 . After a flow reduction to 2120 kg/s-m^2 , film boiling occurred on the fuel rods. One rod failed approximately 45 seconds after the reactor was shut down as a result of cladding embrittlement due to extensive cladding oxidation. Data are presented on the behavior of these irradiated fuel rods during steady-state operation, the power ramp, and film boiling operation. The effects of a power ramp and power ramp rates on pellet-cladding interaction are discussed. Test data are compared with FRAP-T3 computer model calculations and data from a previous Irradiation Effects test in which four irradiated fuel rods of a similar design were tested. Test IE-3 results indicate that the irradiated state of the fuel rods did not significantly affect fuel rod behavior during normal, abnormal (power ramp of 20 kW/m per minute), and accident (film boiling) conditions.

SUMMARY

Test IE-3 of the Thermal Fuels Behavior Program Irradiation Effects Test Series is part of the United States Nuclear Regulatory Commission Fuel Behavior Program being conducted by EG&G Idaho, Inc., to define the behavior of fuel rods during transient operating conditions and to provide data for verification of computer models. Test IE-3 was conducted in November, 1976 in the Power Burst Facility reactor at the Idaho National Engineering Laboratory. Four pressurized water reactor type Saxton fuel rods, supplied by Westinghouse Electric Corporation, were tested^[a]. The fuel rods, Rods IE-015, IE-016, IE-017 and IE-018, had burnups of 11,060, 8,550, 14,780 and 15,860 MWd/tU respectively. The upper end caps in the 0.97-m long fuel rods were removed and replaced with end caps containing a pressure transducer. The fuel rods had an active fuel length of 0.89 m and were prepressurized to 2.6 MPa with an argon and helium gas mixture that had a thermal conductivity similar to fission gases removed from irradiated Saxton fuel rods.

The objectives of this test were to: (a) determine the behavior of irradiated fuel rods subjected to a rapid power increase during which the possibility of a pellet-cladding mechanical interaction failure is enhanced, (b) determine the behavior of these fuel rods during film boiling following this rapid power increase and (c) evaluate the effect of a small variation in initial (preirradiated) gap size on pellet-cladding mechanical interaction. In addition to these objectives, the test was intended to provide new experimental data for comparison with data obtained from a previous Irradiation Effects test in which four irradiated fuel rods of a similar design were tested.

The test consisted of 30 hours of preconditioning power cycles at rod peak powers not exceeding 35 kW/m. This preconditioning period was

[a] The Saxton Reactor was designed by Westinghouse Electric Corporation for the United States Atomic Energy Commission. The reactor was a small, prototypic pressurized water reactor.

followed by a power ramp from an average fuel rod peak power of 22 kW/m to 69 kW/m, at a ramp rate of 20 kW/m per minute, to evaluate the potential for a pellet-cladding mechanical interaction induced cladding failure. The fuel rods were held at this high power for one hour, and then subjected to a flow reduction to induce film boiling. Film boiling was first detected at a flow rate of $430 \text{ cm}^3/\text{s}$ (2120 kg/s-m^2) and then flow reduction was continued to $280 \text{ cm}^3/\text{s}$ (1380 kg/s-m^2). Steady-state, film boiling conditions were maintained for about one minute, after which the reactor was rapidly shut down.

No fuel rod failures occurred during approximately one minute of film boiling operation. However, one fuel rod, Rod IE-015, failed about 45 seconds after the reactor had been shut down. Preliminary post-irradiation examination showed evidence that cladding oxidation, fuel swelling, cladding collapse, fuel rod bowing and molten fuel resulted from the film boiling operation. The test results also demonstrated that the fast power ramp did not cause fuel rod failure. The high rate of power increase caused a greater amount of cladding elongation than was observed for fuel rods in Test IE-1, which had a similar design and tested rods under similar conditions, but at a ramp rate of 3.2 kW/m per minute.

The fuel rod internal pressure data in conjunction with calculated estimates of gas inventory were used to estimate the fission gas release from the fuel rods during different phases of the test. During the one hour steady-state operation at 69 kW/m fuel rod peak power, 31 and 2 percent of the original inventory of fission gases were released from the fuel in Rods IE-016 and IE-017, respectively. During film boiling, fission gas release from the film boiling region of the fuel stack was 68 and 6 percent of the original inventory of that region for Rods IE-016 and IE-017, respectively. The difference in gas release between Rods IE-016 and IE-017 is attributed to the fuel temperatures. Rod IE-016 operated at higher fuel temperatures and had a higher molten fuel volume fraction during film boiling than Rod IE-017.

Although the amount of fission gas released from the 0.242-m long film boiling region in Rod IE-016 was significant, the total fuel rod internal pressure increase during film boiling due to both thermal effects and fission gas release was only 20 percent. For Rod IE-017, the total measured pressure increase during film boiling was 6 percent. The fission gas released during high power and film boiling operations was not great enough to cause large fuel rod internal pressure increases which might result in a loss of cladding integrity.

The steady-state and transient data from this experiment were compared with the FRAP-T3 fuel rod behavior computer code calculations to evaluate the analytical capabilities of the fuel rod code and to assist with interpretation of the data. The as-run test and fuel rod variables were used in the calculations. Calculated steady-state and measured fuel rod internal pressures during the preconditioning power cycles (steady-state) of the test agreed within 10%. Calculated values of the increase in cladding elongation during transient film boiling operation were within 12% of the increase measured.

Comparison of the results of this test with results from previous IE tests indicates that the irradiated state of the fuel rods did not significantly affect fuel rod behavior during normal, abnormal (power ramp of 20 kW/m per minute), or accident (film boiling) conditions.

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IRRADIATION EFFECTS TEST SERIES

TEST IE-3

TEST RESULTS REPORT

1. INTRODUCTION

The Irradiation Effects (IE) Test Series is part of the Thermal Fuels Behavior Program being conducted at the Idaho National Engineering Laboratory by EG&G Idaho, Inc. The research is sponsored by the Nuclear Regulatory Commission and is part of their Fuel Behavior Program to define behavior of irradiated fuel rods during transient reactor conditions^[1]. The Irradiation Effects experiments are being performed in conjunction with similar tests on unirradiated fuel rods and are designed to delineate the burnup factors that may influence fuel rod behavior in abnormal or accident conditions. The IE 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 to be evaluated in this test series are described in the Irradiation Effects Experiment Requirements document^[2].

Test IE-3 was conducted to investigate the behavior of irradiated fuel rods which had small variations in both diametral gap size and irradiation history. The data of interest include:

- (1) Steady-state cladding elongation, fuel rod internal pressure, and cladding surface temperature at various fuel rod powers.
- (2) Pellet-cladding mechanical interaction (PCI) during power ramps.
- (3) Fuel rod behavior during film boiling, at high fuel rod powers (approximately 69 kW/m peak). (These data on irradiated fuel rods, when compared with data for unirradiated fuel rods

tested under similar conditions, permit an evaluation of the influence of the irradiation state of the fuel and cladding on fuel rod behavior during film boiling).

Specific objectives of this test were:

- (1) To determine the behavior of irradiated fuel rods subjected to a rapid power increase during which the possibility of a PCI failure is enhanced
- (2) To determine the behavior of the irradiated fuel rods during film boiling following this rapid power increase
- (3) To evaluate the effect of a small variation in initial (pre-irradiated) gap size on PCI.

In addition to these objectives, the test was intended to provide new experimental data for comparison with data obtained from a previous Irradiation Effects Test, Test IE-1, in which four irradiated rod of similar design were tested.

Four fuel rods were tested simultaneously in the Power Burst Facility at the same nominal power and flow history. The 0.97-m long pressurized water reactor (PWR) type fuel rods were preirradiated in the Saxton reactor to burnups ranging from 8,550 to 15,860 MWd/tU. A complete description of the fuel rods and test train design is given in Section 2. The test consisted of a preconditioning period of 30 hours followed by a power ramp, a one-hour high power steady-state period, and a flow reduction transient to induce film boiling. Discussion of the test conduct is provided in Section 3. Section 4 contains the experimental results. The preliminary results of the postirradiation visual examinations (PIE) are presented in Section 5. The balance of the PIE results will be reported separately after completion of the examinations^[3]. Calculated and measured fuel rod behavior data are compared and presented in Section 6. Discussion and conclusions based on the test results in Section 4 and the comparisons made in Section 6 are

provided in Section 7. In Appendix A, an assessment of the instrumentation and data system uncertainties is presented. Pretest fuel rod characterization data are contained in Appendix B. Appendix C contains additional test data and a discussion of the methods used to reduce the test data. The results of a detailed power calibration are presented in Appendix D.

2. EXPERIMENT DESCRIPTION

The Power Burst Facility (PBF) consists of an open-tank reactor vessel, a driver core region with an active length of 0.91 m, a central flux trap region containing an in-pile tube (IPT), and a loop coolant system to provide an environment typical of pressurized water reactor system conditions. The re-entrant IPT, which encloses the test space, has inlet and outlet connections for loop coolant flow located at its upper end above the driver core. The coolant flow enters the top of the IPT and is directed downward on the outside of a flow tube which surrounds the test assembly. At the bottom of the IPT, the coolant flow reverses direction and flows up through the test assembly and out the IPT outlet.

Four PWR type fuel rods, previously irradiated in the Saxton reactor^[a] were tested in the PBF IPT. The fuel rods were 0.97-m long with a 0.89-m active fuel length. All four fuel rods contained 12.5 wt% $^{235}\text{UO}_2$ fuel with a theoretical density (TD) of 94% prior to Saxton irradiation. Each fuel rod was symmetrically positioned in the reactor test space and contained in a separate flow shroud which provided hydraulic and thermal isolation of each rod from the other rods tested and permitted the four single fuel rods to be tested simultaneously.

Each fuel rod and flow shroud assembly was instrumented to monitor fuel rod behavior during nuclear operations. The test train, flow shrouds, fuel rods, the instruments associated with the test assembly and each fuel rod are described in the following subsections.

2.1 Test Train

The test train hardware was designed to symmetrically position four fuel rods in the PBF reactor test space. The test train hardware installed in the PBF in-pile tube is shown in Figure 1. A fuel rod

[a] The Saxton Reactor was designed by Westinghouse Electric Corporation for the United States Atomic Energy Commission. The reactor was a small, prototypic pressurized water reactor.

Fig. 1 Four-rod experiment hardware assembly and instrumentation shown installed in the Power Burst Facility in-pile tube.

mounted in a flow shroud with a full complement of instruments is depicted in Figure 2. A cross-sectional view of the test train assembly presented in Figure 3 defines the relative location of each flow shroud, the cobalt flux wires, and the self-powered neutron detector (SPND) support tubes. Figure 4 shows the relative positioning of the SPNDs in the test assembly.

2.2 Flow Shrouds

The zircaloy-4 flow shrouds had a nominal inside diameter (ID) of 16.3 mm and a wall thickness of 3.15 mm. Resultant hydraulic and equivalent heated diameters were 6.39 and 16.9 mm, respectively. The four flow shrouds were positioned in the in-pile tube as shown in Figure 1.

2.3 Fuel Rods

The four irradiated fuel rods used for this test were of a similar design. The fuel rods, designated as Rods IE-015, IE-016, IE-017 and IE-018, were approximately 0.97-m long (not including the instrumented end caps) and had a nominal active fuel stack length of 0.89 m. Rods IE-015 and IE-016 had pre-PBF test irradiated fuel stack lengths of 0.69 m to which additional unirradiated fuel was added to the top of the fuel column to increase the total fuel stack length in these fuel rods to approximately match the 0.89-m fuel column length of Rods IE-017 and IE-018. During handling, the original 0.69 m of irradiated fuel became redistributed to a 0.72-m length, apparently as a result of the fragmented fuel pellets in the upper 0.05 m of the fuel stack becoming less densely packed. Therefore, only 0.17 m of fresh fuel (eleven, 15.2-mm long pellets) were added to each stack. A gamma scan indicated that redistribution of the irradiated fuel stacks did not result in any gaps larger than approximately 4 mm. The fuel pellets (12.5 wt% $^{235}\text{UO}_2$, 94% TD) contained in each fuel rod were of a dished end design. Fuel pellet data for these fuel rods are given in Appendix B.

The rods were clad with zircaloy-4 tubing having a nominal outside diameter (OD) of 9.93 mm and a wall thickness of 0.56 mm. Based on

EGG-B-834

Fig. 2 Test IE-3 instrumented fuel rod shown mounted in its flow shroud.

INEL-A-3017

Fig. 3 Test IE-3 Cross-sectional view of Power Burst Facility in-pile tube showing relative location of each flow shroud, cobalt flux wire, and self-powered neutron detector support tube. The zero degree (0°) orientation for each flow shroud is toward the center of the in-pile tube.

EGG-A-1448

Fig. 4 Orientation of the self-powered neutron detectors (SPND) for Test IE-3.

pretest dimensional characterization, the mean diameter in the central region of Rod IE-015 had increased by approximately 0.03 mm as a result of cladding creep and high fuel rod internal pressure during service in the Saxton reactor. Rod IE-016 experienced a negligible change in diameter. A decrease in cladding diameter of approximately 0.05 mm occurred during Saxton irradiation of Rods IE-017 and IE-018 as a result of their initial internal pressure of 0.1 MPa. The nominal design fuel-cladding gaps was 0.27 mm for Rods IE-015 and IE-016 and 0.22 mm for Rods IE-017 and IE-018. However, as a result of these dimensional changes in the cladding during irradiation in Saxton, the fuel-cladding gaps at the peak power location were estimated to be 0.30, 0.27, 0.17, and 0.17 mm for Rods IE-015, IE-016, IE-017, and IE-018, respectively. The pre-PBF test fuel rod diametral measurements and additional cladding characterization data are contained in Appendix B. Additional fuel rod information is summarized in Table I and Reference 4.

The original end caps were removed and replaced by new upper end caps containing pressure transducers. The fuel rods were then back-filled with a gas mixture of 77.7% helium and 22.3% argon to about 2.6 MPa. The thermal conductivity of this mixture closely approximates the calculated conductivity of the fission gases removed from irradiated Saxton fuel rods.

2.4 Instrumentation

Instrumentation was provided to measure fuel rod behavior and coolant conditions in each flow shroud during the test. A listing of the transducer calibration and measurement uncertainties for the instruments is given in Appendix A. The test instrumentation was divided into two classifications; test train instruments and fuel rod instruments.

2.4.1 Test Train Instrumentation. The instruments associated with the test train are the following:

- (1) One 0 to 69-MPa and one 0 to 20-MPa strain post type pressure transducer were used to measure the system coolant pressure

TABLE I

CHARACTERISTICS OF FUEL RODS USED IN TEST IE-3

Parameter	Rod			
	IE-015	IE-016	IE-017	IE-018
Saxton Rod Number ^[4]	721	724	888	890
Average Burnup MWd/tu ^[4]	11,060	8,550	14,780	15,860
Fabricated Fuel Density (%TD)	94	94	94	94
Initial Fill Gas/Pressure (MPa)	He/13.2	He/9.03	He/0.10	He/0.10
Original (unirradiated) Pellet Diameter (mm)	8.48	8.48	8.53	8.53
Fill Gas for PBF Test (at time of assembly)	77.7% He - 22.3 Ar	77.7% He - 22.3 Ar	77.7% He - 22.3 Ar	77.7% He - 22.3 Ar
Pressure for PBF Test (MPa) (at time of assembly)	2.56	2.65	2.68	2.6 ^[a]
Pressure prior to PBF Test (MPa)	2.36 ^[b]	2.46 ^[b]	2.18 ^[c]	2.62
Measured Void Volume (cm ³)	9.2 ^[d]	9.6 ^[d]	6.4	6.4

[a] This value was the design fill pressure (actual value not recorded).

[b] Fuel rod temperature differences could account for part of the difference between this pressure and the pressure obtained at time of fuel rod assembly (a 20 K difference in fuel rod temperature results in a 0.17-MPa change).

[c] Part of the decrease could be accounted for by fuel rod temperatures, but most of the pressure change is attributed to transducer drift.

[d] A larger void volume in these fuel rods is due primarily to the less densely packed fuel (and consequently fewer pellets) in these two rods (see Section 2.3).

during the tests. Both were positioned in the flow loop above the fuel rod assembly, as shown in Figure 1.

- (2) Each flow shroud was fitted with a turbine flowmeter to measure coolant flow through the shroud as shown in Figures 2 and 5.
- (3) A copper-constantan (Type T) differential thermocouple pair attached at the bottom and top of each flow shroud was installed to measure coolant temperature rise.
- (4) Two magnesium oxide insulated, Chromel-Alumel (Type K) thermocouples were positioned near the common inlet of the four flow shroud assemblies to measure the coolant inlet temperature. The coolant outlet temperature of each flow shroud was monitored with a similar Type K thermocouple.
- (5) A linear variable differential transformer (LVDT) was positioned at the bottom of each fuel rod to measure changes in fuel rod cladding length (Figure 5).
- (6) Two support tubes, each having five 0.10-m long, cobalt self-powered neutron detectors were mounted in the test train assembly as shown in Figures 4 and 6. The location of these devices relative to the test train is shown in Figure 3. These devices were employed to measure the relative axial neutron flux in the in-pile tube during the test. Each column of SPNDs was mounted so that its detectors were located with centers at 0.15, 0.31, 0.46, 0.62 and 0.78 m from the bottom of the active core.
- (7) A cobalt wire was mounted in the test train on the outside of each flow shroud to obtain the integrated axial neutron flux profile in the IPT (two flux wires were mounted on the flow shroud of Rod IE-016). The cobalt wire azimuthal locations

Fig. 5 Layout of the linear variable differential transformer, coolant flow orifice plate, turbine flowmeter, and coolant flow screen assembly typical of those used in Test IE-3.

Fig. 6 A typical self-powered neutron detector shown mounted on a support tube.

are shown in Figure 3. The cobalt wires were positioned to extend beyond the bottom and the top of the active core.

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

- (1) A 0 to 17-MPa strain post type pressure transducer was used to monitor fuel rod internal pressure on each fuel rod. The transducer was mounted on the instrumented end cap at the upper end of the fuel rod as shown in Figure 2.
- (2) Rod IE-017 was equipped with three spring-loaded thermocouples to measure the cladding surface temperature. They were located at 0.51, 0.61, and 0.71 m from the bottom of the fuel rod, and were all positioned at the zero degree azimuthal orientation. These thermocouples were tungsten-rhenium (W5%Re/W26%Re) with beryllium oxide insulation encased in a 1.17-mm diameter zircaloy sheath. The thermocouples were spring-loaded against the outer surface of the fuel cladding as shown in Figures 2 and 7 with an approximate four pound preload.

Rod IE-018 had four Chromel-Alumel (Type K) thermocouples brazed to the irradiated cladding using a special braze alloy consisting of Al 2.5%, Ti 46.75%, Zr 46.75%, and Be 4.0%. Figure 8 is a photograph of a typical thermocouple that has been remotely brazed to a fuel rod cladding using the same procedure as that used on this test. The thermocouple elevations above the bottom of fuel rod and azimuthal orientations were: 0.51 m (90°); 0.61 m (0° and 180°), and 0.71 m (270°).

Cladding surface temperature measurements were not made on Rods IE-015 and IE-016.

Fig. 7 A typical view of a portion of the test train as assembled for insertion into the Power Burst Facility in-pile tube.

EGG-A-1450

Fig. 8 Representative thermocouple that has been remotely brazed to a fuel rod cladding outer surface.

3. EXPERIMENT CONDUCT

The Irradiation Effects Test 3 conduct is shown in Figure 9. The test sequence for Test IE-3 consisted of three phases which were conducted over a 40 hour period, including about 30 hours of nuclear operations as shown in Figure 9. The three phases are discussed below.

- (1) The first test phase consisted of fuel rod preconditioning which included two power calibration cycles statistically designed to study the effects of inlet temperature, coolant flow rate, and reactor power and time upon the fuel rod behavior and coolant measurements. This period was followed by Cycle 3, a sequence of power oscillations and steady-state power levels to determine gap conductance values. An additional power cycle, Cycle 4, from 0 to 35 kW/m and returning to 0 kW/m at a rate of 0.2 kW/m per minute completed the preconditioning phase.
- (2) The second phase, referred to as Cycle 5, consisted of power ramp from zero to an average (of three rods)^[a] fuel rod peak power of 22 kW/m at a rate of 0.4 kW/m per minute and from 22 kW/m to 69 kW/m at a rate of 20 kW/m per minute followed by a high power steady-state period. The average fuel rod peak power was held at 69 kW/m^[b] for about one hour.
- (3) The third phase consisted of a flow reduction, to induce film boiling on the fuel rod cladding. During the flow reduction and the subsequent film boiling, the fuel rod peak power was held constant at 69 kW/m. After about one minute of film boiling on all four fuel rods, the reactor was shut down.

[a] The average peak power was determined by averaging Rods IE-015, IE-016, and IE-017. The differential temperature measurement used in determining the power of Rod IE-018 was less accurate than the measurements for the rest of the rods tested (see Appendix D).

[b] The peak-to-average ratio used was 1.32 based on SPND data discussed in Appendix D.

Fig. 9 Coolant flow rate, inlet temperature, fuel rod peak power, and reactor power during Test IE-3.

A detailed description of the test conduct is provided in this section. For convenience of discussion, the various power cycles are labeled Cycle 1 through 5 as designated on Figure 9.

3.1 Preconditioning Phase

As shown in Figure 9, Cycles 1 and 2 consisted of eight steps each at all possible combinations of two levels of flow rate ($505 \text{ cm}^3/\text{s}$ and $760 \text{ cm}^3/\text{s}$), inlet temperature (583 K and 604 K), and fuel rod peak power (28 kW/m and 35 kW/m)^[a]. The coolant pressure throughout the entire test sequence was nominally 14.8 MPa. To initiate Cycle 1, the fuel rod peak power was slowly increased at a rate of 0.2 kW/m per minute to a rod peak power of 28 kW/m, held constant at 28 kW/m for 20 minutes, and then increased to 35 kW/m at the same rate. Subsequent power changes during Cycle 1 were conducted at a rate of 0.7 kW/m per minute. After six hours of nuclear operations, the reactor was shut down due to an operational problem. At fuel rod peak powers below 20 kW/m, power changes were made at approximately 1.5 kW/m per minute. The power calibration sequence (Cycle 2) was started after about 90 minutes of reactor shutdown. The fuel rod power changes were made at a rate of 0.5 kW/m per minute during this cycle.

The gap conductance testing, designated as Cycle 3, consisted of power calibrations at nominal test peak rod power levels of 12 kW/m and 21 kW/m, followed by power oscillations at the same nominal power levels. The flow rate was $400 \text{ cm}^3/\text{s}$ (1970 kg/s-m^2) and the inlet temperature was 555 K. No attempt was made to retain the frequency content of the data during the oscillations in the plots in this report. These data are presented only for visual continuity.

[a] The combination of flow rate, inlet temperature, and rod linear heat rating was chosen using a 2^3 factorial design with one complete replicate to provide estimates of variances. The steps were randomly arranged in time with a constraint on the inlet temperature as shown in Figure 9. The design was additionally constrained so that the effect of time, electronic drift, etc., was confounded with the three factor interaction.

After a reactor shutdown of 6.4 hours, following the gap conductance testing, the remainder of the fuel rod preconditioning (Cycle 4) was conducted. Cycle 4 consisted of a power increase from 0 kW/m to a fuel rod peak power of 35 kW/m and a return to 0 kW/m. Power changes were made at a rate of 0.2 kW/m per minute.

3.2 Power Increase and Steady-State Operation

Following the fuel rod preconditioning phase, Cycle 5 was initiated with a power increase from 0 to 22 kW/m, at a rate of 0.4 kW/m per minute. The fuel rod peak power was held constant at this level for 30 minutes and then increased at a rate of 20 kW/m per minute to a predesignated power level. A thermal balance indicated that the average fuel rod peak power was about 64 kW/m, which was lower than desired. After about 15 minutes another power increase at a rate of 0.4 kW/m per minute was initiated to bring the fuel rod peak power to 71, 65, 71, and 62 kW/m for Rods IE-015, IE-016, IE-017 and IE-018, respectively. The fuel rod powers, coolant inlet temperature, and coolant flow rate were held essentially constant for 45 minutes. The high power period of steady-state operation was made primarily to test for short-term stress corrosion failure mechanisms due to the pellet-cladding mechanical interaction strains induced by the power ramps. In addition to fuel rod behavior data obtained during the one hour of high power operation, vigorous nucleate boiling caused degassing of the fuel rod cladding surface, thus reducing the possibility of premature occurrence of film boiling during the flow reduction phase.

3.3 Flow Reduction Phase

Following the high power steady-state period, while the average fuel rod peak power and coolant inlet temperature were held constant at 69 kW/m and 606 K, respectively, the coolant flow was reduced from $1000 \text{ cm}^3/\text{s}$ (4920 kg/s-m^2) in steps of about $50 \text{ cm}^3/\text{s}$ (250 kg/s-m^2). Following each flow reduction step, the coolant flow was held constant for about one minute prior to initiating another flow reduction step.

At an average coolant flow of $430 \text{ cm}^3/\text{s}$ (2120 kg/s-m^2) the internal pressure of Rod IE-015 and the cladding elongation on Rods IE-016, IE-017, and IE-018 increased (the LVDT on Rod IE-015 had failed), indicating film boiling had occurred. Once film boiling was detected, the average coolant flow was reduced an additional $100 \text{ cm}^3/\text{s}$ to $330 \text{ cm}^3/\text{s}$ (1620 kg/s-m^2). This reduction was followed by a final flow reduction to $280 \text{ cm}^3/\text{s}$ (1380 kg/s-m^2). The flow was held constant at this value for 50 seconds, after which the reactor was shut down.

4. EXPERIMENTAL RESULTS

Fuel rod experimental behavior data are presented in the form of a brief summary and figures showing the results of measurements made for each fuel rod over the entire test conduct. Each of these summary figures provides some perspective on the fuel rod behavior (fuel rod internal pressure, cladding elongation, and cladding surface temperature) responding to the test boundary conditions and driving functions (coolant inlet temperature, coolant flow, and reactor power). The quality of the cladding elongation, fuel rod internal pressure, and cladding surface temperature measurements for each fuel rod are briefly discussed.

Corrections that were applied to the data are discussed in Appendix C. Offsets were subtracted from LVDT and thermocouple measurements so that the average instrument signals at hot, zero power conditions were zero. Rod internal pressure data were also corrected for zero shifts due to decalibration of the pressure transducer throughout the test (a discussion of pressure transducer behavior is given in Appendix A). All pressure data shown in the text have been corrected for drift (a comparison of the uncorrected and corrected pressure transducer data is given in Appendix C).

4.1 Rod IE-015

The fuel rod internal pressure, coolant flow, inlet temperature, and fuel rod peak power for Rod IE-015 are shown in Figure 10. Since the LVDT failed at the start of the test no cladding elongation data were obtained.

Figure 11 shows the fuel rod internal pressure during the flow reduction. Approximately 45 seconds after reactor shutdown was initiated and film boiling had ceased on Rod IE-015, the pressure increased 5.6 MPa to the coolant system pressure of 14.8 MPa, indicating a loss of the fuel rod cladding integrity. Agreement between the pressure indicated by the fuel rod pressure transducer and the known coolant

Fig. 10 Rod IE-015 behavioral data and coolant conditions throughout Test IE-3.

Fig. 11 Rod IE-015 fuel rod internal pressure during the flow reduction phase of Test IE-3.

system pressure confirms that the pressure transducer drift correction was correctly applied and meaningful pressure data were acquired.

4.2 Rod IE-016

Figure 12 shows the fuel rod behavior data for Rod IE-016. The pressure transducer and the LVDT for this fuel rod appear to have functioned satisfactorily throughout the test. During the flow reduction, film boiling was indicated by an increase in both rod elongation and internal pressure.

Figures 13 and 14 show the cladding elongation, fuel rod internal pressure and coolant flow during the flow reduction. Film boiling is indicated by an increase in the cladding elongation, followed by an increase in fuel rod internal pressure. A decrease in the cladding

Fig. 12 Rod IE-016 behavioral data and coolant conditions throughout Test IE-3.

Fig. 13 Rod IE-016 cladding elongation and fuel rod internal pressure during the flow reduction phase of Test IE-3.

Fig. 14 Rod IE-016 coolant flow rate during the flow reduction phase of Test IE-3.

elongation occurs at the time of reactor shutdown, indicating initiation of the collapse of the vapor film on the fuel rod cladding surface.

4.3 Rod IE-017

In Figure 15 the fuel rod internal pressure, cladding elongation, coolant flow rate, coolant inlet temperature, and fuel rod peak power for Rod IE-017 are shown. The cladding surface temperature, as measured by the spring-loaded thermocouples at the 0.51 and 0.61-m elevations, is also shown. The thermocouple at the 0.61-m elevation failed during the power ramp. The output from the spring-loaded thermocouple at the 0.71-m elevation was inadvertently reversed during instrument range changes following Cycle 4 and is not shown.

As shown in Figures 16 and 17, film boiling is indicated on this fuel rod by increases in the cladding surface temperature at the 0.51-m elevation, cladding length, and fuel rod internal pressure. Comparison of the cladding surface temperature and the cladding elongation with the coolant flow rate indicates that film boiling did not extend down to the 0.51-m elevation until the next flow reduction that followed initiation of film boiling. The axial extent of film boiling probably increased upward on the fuel rod as well as downward during the flow reduction to $300 \text{ cm}^3/\text{s}$.

4.4 Rod IE-018

The behavior of Rod IE-018 during the test is shown in Figure 18. All four of the brazed thermocouples mounted on the cladding surface functioned throughout the test. During the flow reduction, the cladding elongation, fuel rod internal pressure, and all four cladding surface thermocouples indicated the occurrence of film boiling, although two thermocouples measured temperature increases of less than 20 K. These data are shown in Figures 19 and 20. The cladding elongation data for Rod IE-018 were very noisy and erratic at times and the output of the LVDT which appeared to have suffered a reduction in gain was not used to

Fig. 15 Rod IE-017 behavioral data and coolant conditions throughout Test IE-3.

Fig. 16 Rod IE-017 cladding surface temperature during the flow reduction phase of Test IE-3.

Fig. 17 Rod IE-017 cladding elongation and fuel rod internal pressure during the flow reduction phase of Test IE-3.

Fig. 18 Rod IE-018 behavioral data and coolant conditions throughout Test IE-3.

Fig. 19 Rod IE-018 cladding surface temperature during the flow reduction phase of Test IE-3.

Fig. 20 Rod IE-018 cladding elongation and fuel rod internal pressure during the flow reduction phase of Test IE-3.

quantify fuel rod response but was used to determine the timing of film boiling. The LVDT on Rod IE-018 did respond when film boiling occurred during the flow reduction, but the response shown in Figure 20 probably is an underestimation of the extent of cladding elongation during film boiling because this transducer had decalibrated.

Qualitative information on the propagation of the film boiling zone was obtained from cladding surface thermocouple and cladding elongation data. Thermocouple measurements at the 0.61-m elevation and LVDT measurements indicated the onset of film boiling during the flow reduction to $325 \text{ cm}^3/\text{s}$. During the subsequent flow reduction to $275 \text{ cm}^3/\text{s}$, the cladding elongation abruptly increased, indicating further growth of the film boiling zone. Approximately 10 seconds after the flow reduction to $275 \text{ cm}^3/\text{s}$, the cladding surface thermocouple at the 0.51-m elevation indicated film boiling.

5. PRELIMINARY POSTIRRADIATION EXAMINATION RESULTS

Preliminary results from the postirradiation examination are presented in this section. The test train was disassembled in the PBF canal and the rods were shipped to the hot cells in their flow shrouds. The rods were withdrawn, all intact, from their flow shrouds. The results of the visual and dimensional characterization examinations of the four rods are provided in the following subsections.

5.1 Visual Examination

5.1.1 Rod IE-015. The appearance of Rod IE-015 in the film boiling zone is shown in Figure 21. Posttest diameter measurements indicated that the film boiling zone extended between the 0.470 and 0.704-m rod elevations^[a]. The region between the 0.489 and 0.692-m rod elevations was characterized by oxide formations and oxide spalling. Fuel relocation and breakup apparently occurred and the cladding had collapsed into spaces between and around pellet fragments near the upper film boiling zone boundary, as shown in Figure 22. Although an internal pressure transducer indicated a loss of cladding integrity during the test, no visual evidence of a cladding perforation was noted in this or any other region of the rod. No evidence of cladding collapse into pellet interfaces was observed outside the region of fuel relocation. Most of the fuel relocation in this rod and Rod IE-016, discussed in the next section, probably occurred during the installation of the instrumented upper end cap on the fuel rods prior to testing in PBF (Section 2.3).

Several surface appearances in the film boiling zone were noted. As shown in Figure 23, where no apparent oxide spalling had occurred, the oxide had a "silvery" appearance. Where partial oxide spalling had occurred, the remaining oxide layer had a dark grey color. Bare zircaloy appears as a metallic grey.

[a] Distance was measured from the bottom of the fuel rod. The bottom of the fuel stack was 12.7 mm above the bottom of the fuel rod.

Fig. 21 Posttest photographs of the film boiling zones of Rods IE-015 and IE-016, Test IE-3.

Fig. 22 Cladding collapse between and over relocated fuel pellet fragments near the upper film boiling zone boundary of Rod IE-015, Test IE-3.

Fig. 23 Surface appearances in film boiling zone of Rod IE-015, Test IE-3.

Rod IE-015 had a cosine shaped bow in the 90-degree direction beginning near the 0.56-m elevation and extending to the top of the rod. The maximum deflection from the 0 to 180-degree plane was estimated to be 9 mm.

5.1.2 Rod IE-016. Figure 21 also shows the film boiling zone of Rod IE-016. Film boiling occurred between approximately the 0.470 and 0.712-m rod elevations. Figure 24 shows that at the 0.67-m elevation, near the top of the previously irradiated fuel column, the cladding collapsed into apparently wide gaps between fuel pellets and pellet fragments. As discussed in Section 2.3, fuel relocation and the resultant gaps between pellets and pellet chips probably occurred during pretest handling of the fuel rod in the hot cell. Cladding collapse also occurred into a 3-mm gap between relocated fuel at about the 0.61-m elevation. Other than these two regions where fuel relocation occurred, cladding collapse into pellet interfaces or chips was not observed. No evidence of cladding perforation was observed along the length of the rod.

Fig. 24 Cladding collapse into the region of relocated fuel near the upper film boiling zone boundary of Rod IE-016, Test IE-3.

Rod IE-016 bowed in the 90-degree direction between the 0.58 and 0.84-m elevations. An abrupt bend occurred at the 0.67-m elevation at which the cladding collapse shown in Figure 24 occurred. The maximum deflection from the 0 to 180-degree plane was estimated to be 6 mm.

5.1.3 Rod IE-017. Figure 25 shows the film boiling region of Rod IE-017. Based on posttest diameter measurements, the film boiling zone appears to extend between the 0.460 and 0.680-m rod elevations. Slight cladding collapse into pellet interfaces was noted at several locations at the 0-degree azimuthal rod orientation, but was not noted at any other location along the rod.

Rod IE 017 bowed in the 270-degree direction between the 0.53 and 0.64-m rod elevations, with an estimated maximum deflection of 3 mm from the 0 to 180-degree plane. The cladding above the 0.71-m elevation was also slightly bowed in the 270-degree direction.

5.1.4 Rod IE-018. Figure 25 also shows the film boiling region of Rod IE-018. The film boiling zone appears to extend between the 0.452 and 0.670-m elevations. The thermocouples brazed to the cladding surface appeared to be in excellent condition and remained firmly attached to the cladding surface.

Rod IE-018 had a cosine-shaped bow extending between the 0.51 and 0.71-m elevations in the 270-degree direction. The maximum deflection was 6 mm. A bend also occurred in the rod in the 180-degree direction at about the 0.81-m elevation. Wear marks were found on the cladding surface at the 0.62-m rod elevation indicating the rod had been in contact with the shroud (Figure 26). Difficulties in removing the rod from its flow shroud were encountered because of the degree of rod bowing.

5.2 Posttest Diameter Measurement

Diameter measurements were made on Rods IE-015, IE-016, and IE-018 at two perpendicular orientations with a micrometer at 50-mm increments

Fig. 25 Posttest photographs of the film boiling zones of Rods IE-017 and IE-018, Test IE-3.

Fig. 26 Wear marks on cladding outer surface of Rod IE-018, Test IE-3.

outside the film boiling zone and at 12-mm increments within the film boiling zone. The precision of the micrometer measurement is estimated to be ± 0.02 mm. A pulsed eddy current (PEC) scanning device,^[6] estimated to be accurate within ± 0.01 mm, was used to measure the posttest cladding diameter of Rod IE-017.

The micrometer measurements have a constant offset from the true diameter determined from pretest measurements. Since the cladding below the film boiling zone remained below approximately 620 K throughout the test, it was assumed that the cladding outside diameter in this region had not been affected by operation in PBF. From a comparison of the pretest and posttest diameter measurements, the offset was determined to be approximately -0.12 mm.

Posttest micrometer measurements and pretest PEC measurements of the cladding outside diameters along the lengths of Rods IE-015 and

IE-016 are shown in Figures 27 and 28. The offsets between the micrometer and PEC measurements have been removed. Pretest and posttest cladding outside diameters for Rod IE-017, both obtained by PEC scanning, are shown in Figure 29. Micrometer measurements of the cladding outside diameter of Rod IE-018 are shown in Figure 30. The PEC scan of the Rod IE-018 pretest cladding outside diameter has not been included in Figure 30 since the offset existing between the micrometer and PEC measurements could not be determined accurately. Figure B-1 of Appendix B presents PEC measurements of the Rod IE-018 outside diameter for comparison with Figure 30.

The measurements indicated significant swelling of the rod through the film boiling zones of Rods IE-015, IE-017, and IE-018 similar to the swelling observed on the rods from Test IE-1^[6], in which previously irradiated rods were also tested in film boiling. The swelling has been hypothesized to be induced by a combination of fuel swelling due to fission gas bubbles migrating and coalescing at grain boundaries and the volume change associated with fuel melting. Small amounts of swelling and collapse of the cladding occurred within the film boiling zone on Rod IE-016.

Fig. 27 Pretest and posttest diameter measurement of Rod IE-015 cladding, Test IE-3.

INEL-A-4762

Fig. 28 Pretest and posttest diameter measurement of Rod IE-016 cladding, Test IE-3.

Fig. 29 Pretest and posttest pulsed eddy current diameter measurement of Rod IE-0.7, Test IE-3.

INEL-A-4760

Fig. 30 Posttest diameter measurement of Rod IE-018, Test IE-3.

INEL-A-2551

6. COMPARISON OF CALCULATED AND MEASURED FUEL ROD BEHAVIOR

Measured fuel rod behavior variables are discussed according to type of measurement; that is cladding elongation, fuel rod internal pressure, and cladding surface temperature and presented as a function of fuel rod average power. Only measurements considered valid over selected time segments of the test are shown. To aid in the interpretation of the data, some of the data were compared with calculations from FRAP-T3, a computer code for predicting fuel rod behavior. A brief description of the FRAP-T3^[7] computer code^[a] and input used for the calculations is given in the following section.

6.1 Analytical Model

The FRAP-T3 computer code is a composite of various subcodes that are used to predict the thermal, mechanical, and chemical response of a fuel rod to input power and flow conditions. The code is designed to allow the user to specify a portion of the fuel behavior submodels, heat transfer correlations, axial and radial power distributions, and the physical characteristics of the fuel rod. A detailed description of the FRAP-T3 code including submodels and assumptions is found in Reference 7. A brief discussion of the fuel model and experiment conduct model is contained in the following subsections.

6.1.1 Fuel Rod Model. For this analysis, the fuel rods were modeled using 11 radial and 19 axial nodes. FRAP-T3 input included the nominal measured conditions of coolant flow, coolant inlet temperature, coolant pressure, fuel rod average power, and axial flux profile. The Westinghouse Electric Corporation W-3^[9] critical heat flux (CHF) correlation (with the cold wall factor) and the Groeneveld tubes and annulus (Version 5.9)^[10] film boiling heat transfer correlation were used. The free thermal expansion fuel deformation model and the Ross and Stoute^[11] model for gap conductance were specified.

[a] The analytical results were obtained using FRAP-T3 with load module T2P182 in conjunction with the material properties code MATPRO, MOD 009^[8].

It was necessary to model only Rods IE-016 and IE-017 using the FRAP-T3 computer code since they were representative of the other fuel rods tested. Input to model the fuel rods included the Saxton fuel rod design dimensions, burnup in MWd/tU (used only to determine the fuel melting point), coolant conditions, and fuel rod average power. The fuel cladding gap is the primary design variation in the four fuel rods. Rods IE-015 and IE-016 had a mean diametral gap of about 0.27 mm, and Rods IE-017 and IE-018 had a 0.22-mm mean diametral gap. However, the outside diameter of the cladding of Rods IE-017 and IE-018 was reduced by 0.05 mm during Saxton irradiation so a diametral gap of 0.17 mm was modeled in FRAP-T3 rather than the design gap of 0.22 mm. The cladding outside diameter of Rod IE-016 appeared to be unaltered after irradiation in the Saxton reactor so the nominal design gap of 0.27 mm was modeled by FRAP-T3.

Fuel rod average power was calculated from the experimental data using steady-state coolant flow and coolant temperature rise measurements. The time averaged axial flux profile was determined from a gamma scan of the cobalt wires that were positioned in the reactor test space throughout the test. Evaluation of the axially distributed self-powered neutron detector and flux wire data showed that the axial flux profile was not affected significantly by changes in the reactor power level. A change in peak to average power of less than 0.5 percent was observed for fuel rod peak powers between 10 to 69 kW/m. Consequently, the flux wire data, although it represents an integral of the power profile, is an accurate measurement of the instantaneous average fuel rod power distribution. A detailed explanation of the fuel rod power calculations and their uncertainty is provided in Appendix D.

6.1.2 Experiment Conduct Model. Power changes from Cycle 1 through Cycle 5 were slow enough that the fuel rods could be considered in thermal and mechanical quasi-equilibrium^[a], thus measured fuel rod

[a] Quasi-equilibrium is used since equilibrium would imply that fuel creep, restructuring, and fission gas release would be complete, which would not be the case for the fuel rods during the power ramp.

parameters are presented as a function of fuel rod average power. During the flow reduction and subsequent film boiling, fuel rod response was transient. Fuel rod behavior during this phase of the test is presented versus time. Therefore, both steady-state and transient calculations were made and compared with test data.

Steady-state FRAP-T3 analyses were performed by the following method. Calculations were performed using incremental steps in fuel rod average power of 5 kW/m, from zero to a maximum of 55 kW/m. At each calculational step, power was held constant, allowing the fuel rod to equilibrate. These equilibrium fuel rod variables (cladding surface temperatures, cladding elongation, etc.) at each 5 kW/m power step were used to generate the input data for the steady-state comparisons. FRAP-T3 calculations were performed for only one combination of coolant conditions. However, since the coolant flow, coolant inlet temperature, and fuel rod power were varied constantly, during Cycles 1 and 2 of the test, only steady-state experimental data recorded during the test with the same coolant conditions modeled by FRAP-T3 are compared with the FRAP-T3 calculations.

To calculate transient fuel rod behavior during the flow reduction phase, FRAP-T3 utilized the fuel rod average power, coolant pressure, coolant inlet temperature, and coolant flow history measured during the flow reduction. The FRAP-T3 calculation, using the W-3 CHF correlation, did not calculate the occurrence of film boiling at the same time in the flow reduction or over the same axial region on the fuel rod that actually occurred during the test^[a]. Therefore, to perform a post-CHF fuel behavior analysis, the FRAP-T3 code was modified to force the code to use post-CHF film boiling heat transfer coefficients at the actual

[a] The W-3 CHF correlation would not be expected to accurately predict the occurrence and extent of film boiling since the correlation was developed for a different thermal geometry than used in this experiment.

flow rate at which film boiling occurred during the experiment. A logic switch was used in FRAP-T3 to force various axial locations along the fuel rod into instantaneous film boiling. The transient propagation of the film boiling zone cannot be modeled by this method, but equilibrium conditions during film boiling should be more correctly calculated. The axial location of film boiling was determined from the visual post-irradiation examination of the fuel rods (Section 5). The film boiling zones were not modeled exactly because the rod node structure used in FRAP-T3 was selected to coincide with the cladding surface thermocouple locations.

6.2 Fuel Rod Behavior Data and Comparisons of FRAP-T3 Calculations with Selected Data

Calculations obtained using the FRAP-T3 fuel rod behavior code are compared with some of the data for Rods IE-016 and IE-017. Selected cladding elongation, fuel rod internal pressure, and cladding surface temperature data are also presented in this section. These data are shown to illustrate fuel rod behavior during normal, rapid power ramp, and film boiling conditions.

6.2.1 Cladding Elongation. Cladding elongation data for Rods IE-016 and IE-017 during part of Cycle 1, all of Cycles 4 and 5 and the flow reduction phase are presented in this subsection. Measured values and values calculated using the FRAP-T3 computer code are compared for Rods IE-016 and IE-017. The LVDT for Rod IE-015 failed and the cladding elongation data for Rod IE-018 appeared to be in error throughout the test. Hence, no data for Rod IE-015 are presented and only the data from the flow reduction part of the test are presented for Rod IE-018 to establish the time of film boiling. The cladding elongation data and calculations were converted from a millimeter displacement to a percent elongation by dividing the displacement by the length of the fuel stack (889 mm) and multiplying the result by 100.

An offset was subtracted from the cladding elongation data for all fuel rods to set the elongation to zero at the beginning of Cycle 1. An additional offset was subtracted from Cycle 4 to set the initial elongation for this cycle at zero. The difference between the offsets of

Cycle 1 and 4 were caused by a small movement of the support plate to which the LVDTs were attached. The support plate, located at the bottom of the test train, was not rigidly fixed to the main section of the test train. As the coolant flow rate changed from Cycles 1 to 4, there was sufficient dynamic head caused by the coolant bypass flow to slightly shift the position of the support plate resulting in an apparent cladding length change.

6.2.1.1 Rod IE-016. The cladding elongation data for Rod IE-016 during the first part of Cycle 1 and all of Cycles 4 and 5 are shown in Figure 31. The cladding elongation at zero power, although small, is not exactly zero because the offset caused by movement of the support plate could not be completely corrected. The calculated cladding elongation, which has a constant slope, is due to free thermal expansion of the cladding. Differences between the measured and calculated behavior are caused by pellet-cladding mechanical interaction.

Fig. 31 Cladding elongation versus fuel rod average power for Rod IE-016 during Cycles 1, 4, and 5 of Test IE-3.

During Cycle 1, PCI occurs at the beginning of the power cycle and the cladding elongation changed 0.006% per kW/m. During Cycles 4 and 5 the onset of PCI occurred at approximately 10 kW/m. This shift in onset of PCI is attributed to the irradiated fuel stack becoming redistributed during the power cycles, thereby increasing the effective gap between the fuel and the cladding. A more thorough description of the mechanism causing the fuel redistribution is found in Reference 6. The maximum slope of the cladding elongation during Cycle 5 was 0.009% per kW/m.

During Cycle 5, the high power ramp, a decrease in the rate of cladding length change occurred at about 35 kW/m. This amount of decrease was not observed for any of the fuel rods in previous IE tests, and apparently occurred for this fuel rod because of its design. (The initial preirradiated fuel stack length for this fuel rod was about 0.69 m, but became redistributed during handling and resulted in a preirradiated fuel stack length of 0.72 m. Several fresh fuel pellets were then added to bring the total fuel stack length up to about 0.89 m). Forces between the fuel stack and cladding may have caused this less densely packed fuel to relocate and may have resulted in the observed cladding elongation behavior. Further, the large effective gap in the region of relocated fuel may have resulted in comparatively high fuel temperatures which, in turn, could have produced plastic behavior in the fuel and contributed to the decrease in cladding elongation.

The cladding elongation and fuel rod average power versus time for Rod IE-016 during Cycle 5 is shown in Figure 32. The reductions in length that occurred during the steady-state periods were probably a combination of fuel creep, slippage between the fuel and cladding, and possibly some fuel redistribution. Any length decrease that was caused by fuel redistribution (that resulted because of the less densely packed fuel) would not be representative of normal fuel rod behavior. No means were available to determine the extent, if any, of the cladding length reduction that was a result of the redistributed fuel.

The cladding elongation during the flow reduction phase for Rod IE-016 was shown in Figure 13. Since the region of loosely packed fuel

Fig. 32 Cladding elongation and fuel rod average power versus time for Rod IE-016 during Cycle 5 of Test IE-3.

cannot be modeled in FRAP-T3, and the behavior of this fuel rod may significantly vary from the behavior of a fuel rod with a uniform fuel stack, no comparison of the data with FRAP-T3 calculations was made.

6.2.1.2 Rod IE-017. Figure 33 shows the cladding elongation data for Rod IE-017 during Cycles 1, 4 and 5. During Cycle 1, the onset of PCI occurred as the fuel rod power was increased from 0 kW/m and the cladding elongation changed 0.002% per kW/m. During Cycles 4 and 5, the beginning of hard^[a] PCI occurred at higher fuel rod powers. During Cycle 5 hard PCI was initiated at about 20 kW/m while the FRAP-T3

[a] "Hard" PCI is differentiated from "soft" PCI which is characterized by isolated points of fuel-cladding interaction due to pellet fragments or misaligned pellets, and does not result in lockup between the fuel and the cladding. During "hard" PCI the cladding elongation approximates that of the fuel stack.

Fig. 33 Cladding elongation versus fuel average power for Rod IE-017 during Cycles 1, 4, and 5 of Test IE-3.

calculations did not predict PCI initiation until a fuel rod average power of 40 kW/m was reached. The maximum slope of the cladding elongation increase was 0.013% per kW/m during Cycle 5. Fuel creep resulted in cladding length decreases of 0.045 and 0.040% at fuel rod average powers of 50 and 55 kW/m, respectively. Cladding elongation as a function of time is shown in Figure 34; the exponential decrease in cladding elongation during the constant power periods is indicative of fuel creep.

Differences in cladding elongation behavior existed between Rods IE-016 and IE-017. During Cycle 5, hard PCI occurred at approximately 10 kW/m in Rod IE-016. Rod IE-017, although it had a smaller diametral fuel-cladding gap, had hard PCI at 20 kW/m. The slope of cladding elongation versus power during Cycle 1 for Rod IE-016 was twice the slope of Rod IE-017. However, during Cycle 5, Rod IE-017 had a slope of cladding elongation versus power that was 40% higher than the slope of

Fig. 34 Cladding elongation and fuel rod average power versus time for Rod IE-017 during Cycle 5 of Test IE-3.

Rod IE-016. The behavior of relocated fuel may explain these measurements. A rod having relocated fuel may experience PCI at a low power level because the effective fuel-cladding gap would be small. At higher power levels, the cracked fuel may relocate upwards rather than press against the cladding. High fuel temperatures within the relocated fuel may cause plastic deformation of the fuel and contribute to slippage between the fuel and cladding. The net result would be a reduction in the rate of increase in cladding elongation with power, as seen in Rod IE-016.

Figure 35 shows the cladding elongation for Rod IE-017 during film boiling for both the experimental data and FRAP-T3 calculations. The region of the fuel rod that was in film boiling was modeled in the FRAP-T3 analysis using the logic switch discussed earlier. The rates of cladding length change are not comparable, but the total amounts of change are comparable. The measured and calculated increase in cladding

Fig. 35 Experimental and calculated (FRAP-T3) cladding elongation for Rod IE-017 during the flow reduction phase of Test IE-3.

elongation due to film boiling agreed within 15%. Based on the elongation data in Figure 35, the axial extent of the film boiling zone increased following the flow reduction step to $300 \text{ cm}^3/\text{s}$. If FRAP-T3 could have modeled the time dependent axial propagation of the film boiling zone, the agreement between the calculated and measured cladding elongation would have improved.

6.2.1.3 Rod IE-018. Data during the preconditioning cycles are not presented because the absolute magnitude of the output from the LVDT on this fuel rod was considered to be unreliable. However, the cladding elongation data during the flow reduction are presented in Figure 20 because the time in film boiling and relative axial extent of the occurrence of film boiling on the fuel rod are reliable.

6.2.2 Fuel Rod Internal Pressure. Fuel rod internal pressure data and transient fission gas release data, during film boiling are provided in this section.

6.2.2.1 Rod IE-015. Fuel rod internal pressure data during Cycles 4, 5, and the flow reduction are presented for Rod IE-015 in Figure 36. As shown in the figure, a pressure increase of approximately 2.3 MPa occurred during the two high power steady-state periods following the power ramp due to fission gas release. The hysteresis during Cycle 4 shown in this figure resulted from a small transducer drift that was not entirely removed from the data by the drift correction previously discussed.

Fig. 36 Fuel rod internal pressure versus fuel rod average power for Rod IE-015 during Cycles 4 and 5 of Test IE-3.

The fuel rod internal pressure during Cycle 5 and the flow reduction is shown in Figure 37 (a plot of the fuel rod internal pressure during the flow reduction on an expanded scale was shown in Figure 11). A fuel rod internal pressure increase of about 1.2 MPa was measured during film boiling on Rod IE-015. About 45 seconds after the reactor was shut down, the pressure transducer recorded cladding failure by

Fig. 37 Fuel rod internal pressure and fuel rod average power versus time for Rod IE-015 during Cycle 5 and flow reduction phase of Test IE-3.

measuring a rod internal pressure equal to the coolant pressure of 14.8 MPa. The failure location was not visually discernible during postirradiation examination.

6.2.2.2 Rod IE-016. The fuel rod internal pressure data for this fuel rod during Cycles 4 and 5 are shown in Figure 38. Internal pressure increase of 1.2 MPa due to fission gas release during the two steady-state periods at high power is clearly shown in this figure.

Void volume measurements indicated that approximately 3 cm^3 of additional void volume was present in the fuel stack due to the loosely packed fuel. The gases that occupied this additional void space in the fuel stack were heated to higher temperatures than were the gases in the diametral gap and plenum region. Since no provision was made to model this fuel stack configuration in the FRAP-T3 code, no comparison of FRAP-T3 calculations with experimental data was performed.

Fig. 38 Fuel rod internal pressure versus fuel rod average power for Rod IE-016 during Cycles 4 and 5 of Test IE-3.

Figure 39 shows the fuel rod internal pressure response from the start of Cycle 5 through the end of the test. Note the pressure increase of 1.1 MPa due to release of fission gas during the high power steady-state periods of operation. A fuel rod internal pressure increase of 2.0 MPa during film boiling can also be observed on this plot (an expanded plot of this increase was shown in Figure 13). A net increase in fuel rod internal pressure occurred between the beginning of Cycle 5 and after the reactor was shut down (the fuel rod power was 0 kW/m and the coolant inlet temperatures were nominally the same for both of these periods). This net increase in pressure, about 2.5 MPa, is attributed to fission gas release.

6.2.2.3 Rod IE-017. The fuel rod internal pressure during Cycles 4 and 5 is presented in Figure 40. Also shown in this figure is the FRAP-T3 calculated fuel rod internal pressure. The underestimation in pressure was consistent with comparisons made in previous IE Test

Fig. 39 Fuel rod internal pressure and fuel rod average power versus time for Rod IE-016 during Cycle 5 and the flow reduction phase of Test IE-3.

Fig. 40 Fuel rod internal pressure versus fuel rod average power for Rod IE-017 during Cycles 4 and 5 of Test IE-3.

Results Reports, however, the agreement of the pressure calculation of FRAP-T3 with test data is improved over earlier versions of the FRAP-T fuel rod behavior code. The improvement results from the assumption by the model that the gas temperature is equal to the fuel temperature at a point that is two-thirds of the pellet radius from the fuel centerline.

A pressure increase of 0.4 MPa was observed during the two high power steady-state periods of Cycle 5. Figure 41 shows the fuel rod internal pressure for this fuel rod from the beginning of Cycle 5 to the end of the test. A comparison between the calculated and measured fuel rod internal pressure response during film boiling is shown in Figure 42. An increase of about 0.6 MPa was measured during this period while calculations indicated a negligible fuel rod internal pressure increase. FRAP-T3 calculated a closed fuel-cladding gap in the film boiling region of Rod IE-017 prior to film boiling. Therefore, no gas was modeled in this region and no pressure increase during film boiling was calculated.

6.2.2.4 Rod IE-018. The fuel rod internal pressure data for Cycles 4 and 5 are shown in Figure 43. Some hysteresis remained in the corrected data for this pressure transducer. Also, the pressure data indicated a slight decrease in pressure during the high power periods of steady-state operation. On the basis of this and a previous history of pressure transducer data drift, it is felt that these data do not accurately represent the fuel rod pressure during the steady-state periods. The fuel rod internal pressure response during the flow reduction shown in Figure 20 occurred over a short duration and should be representative of the fuel rod pressure increase during film boiling. The fuel rod internal pressure increase that occurred while Rod IE-018 was in film boiling was 1.0 MPa.

6.2.3 Cladding Surface Temperature. Only Rods IE-017 and IE-018 were instrumented with cladding surface thermocouples. The experimental results obtained with these devices are discussed below.

Fig. 41 Fuel rod internal pressure and fuel rod average power versus time for Rod IE-017 during Cycle 5 and the flow reduction phase of Test IE-3.

Fig. 42 Experimental and calculated (FRAP-T3) fuel rod internal pressure for Rod IE-017 during the flow reduction phase of Test IE-3.

Fig. 43 Fuel rod internal pressure versus fuel rod average power for Rod IE-018 during Cycles 4 and 5 of Test IE-3.

6.2.3.1 Rod IE-017. The spring-loaded thermocouples on Rod IE-017 were installed on the fuel rod to detect the presence of local film boiling. Film boiling was dependably detected by these thermocouples, but the cladding surface temperature measured by these devices were typically poor for both steady-state and transient operation. This was due to systematic measurement errors associated with the contact resistance between the thermocouple and fuel rod cladding. Given the poor quality of the temperature measurements of the spring-loaded thermocouples, only their response during the flow reduction is presented to indicate the timing of events during that period. Test data from the spring-loaded thermocouples are not compared with the FRAP-T3 calculations. Figure 16 shows the response of the cladding surface thermocouple at the 0.51-m rod elevation on Rod IE-017 during the flow reduction. The response of the thermocouple located at the 0.71-m elevation is not shown because the polarity of this device was inadvertently reversed midway through the test during an instrument range change. The

cladding surface temperature measurement at the 0.61-m elevation is also not shown since the thermocouple failed during the power ramp phase of the test.

Peak cladding temperatures estimated from metallographic examination of the cladding in the film boiling zone of Rod IE-017, were approximately 1530 K which is in good agreement with the temperature of 1600 K calculated by FRAP-T3. Further results will be reported in the postirradiation examination^[3].

6.2.3.2 Rod IE-018. The cladding surface temperature measurements by the four brazed thermocouples on Rod IE-018 during the flow reduction are shown in Figure 19. The maximum temperature indicated by any of these devices, 880 K, was substantially lower than the cladding surface temperatures expected during film boiling. Based on previous tests, temperatures in excess of 1500 K were expected during film boiling at the coolant flow rates and fuel rod powers of this experiment.

6.2.4 Cladding Radial Deformation. The initial fuel-cladding gap is an important variable contributing to cladding radial deformation. FRAP-T3 calculations were performed only for Rods IE-016 and IE-017 because the initial gaps of the rods were similar to those of Rods IE-015 and IE-018, respectively. The cladding radial strains determined for the Test IE-3 fuel rods from posttest diameter measurements reported in Section 5.2 are compared with FRAP-T3 analyses in this section.

Figure 44 compares the permanent radial strains in Rods IE-015 and IE-016 with the strain calculated by FRAP-T3 for Rod IE-016. Diametral growth was measured in the film boiling zone of Rod IE-015 and was calculated by FRAP-T3. Both swelling and collapse occurred in the film boiling zone of Rod IE-016. The standard deviation of the strain shown in Figure 44 is estimated to be ± 0.026 mm or approximately $\pm 0.26\%$ strain for Rods IE-015 and IE-016.

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Fig. 44 Comparison of calculated and measured cladding permanent radial strains for Rod IE-015 and IE-016, Test IE-3.

The permanent cladding radial strain for Rod IE-017 determined from PEC measurements is shown in Figure 45 and compared with FRAP-T3 calculations. Although FRAP-T3 does not have a model for fuel swelling due to fission gas release, the code calculated a diametral plastic cladding strain similar to the measured value. The measured strain is estimated to be accurate within $\pm 0.12\%$.

Calculation of the cladding radial strain versus axial elevation on Rod IE-018 was not performed because the offset between pretest and posttest diameter measurements could not be determined. The peak cladding diameter increase is estimated to be 4.4%.

Fig. 45 Comparison of calculated and measured cladding permanent radial strains for Rod IE-017 Test IE-3.

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7. DISCUSSION AND CONCLUSIONS

In the following section the experimental results are discussed and compared with results from IE Scopint Test 2^[12], Test IE-1^[6], and Test IE-2^[13]. In Section 7.2 comparisons between the Test IE-3 data and FRAP-T3 calculations are discussed. Conclusions drawn from this experiment are contained in Section 7.3.

The Irradiation Effects Test 3, was conducted to determine the behavior of irradiated fuel rods during power ramp and film boiling conditions. The objectives of the test series include the acquisition of irradiated fuel rod behavior data for the evaluation and verification of advanced fuel rod behavior computer codes. Specific objectives of this test were (a) to determine the behavior of irradiated fuel rods during a fast power increase where the possibility of a pellet-cladding interaction failure is enhanced and (b) to determine the behavior of these fuel rods during a film boiling transient following this fast power ramp, and (c) to evaluate the effect of a small variation in initial (preirradiated) gap size on PCI.

Design diametral gap sizes for Rods IE-015 and IE-016 were 0.27 mm, whereas they were 0.22 mm for Rods IE-017 and IE-018. Irradiation of the rods in the Saxton reactor may have increased the diametral gap of Rod IE-015 to 0.30 mm and decreased the gap in Rods IE-017 and IE-018 to 0.17 mm, as discussed in Section 2.3. Data taken during the pre-conditioning cycles indicated that hard PCI was initiated on Rod IE-016 at a lower power (~ 10 kW/m) than on Rod IE-017 (~ 20 kW/m). This earlier lock up for Rod IE-016 may be attributed to the unusual fuel stack configuration. Normally since Rod IE-017 had a smaller initial gap than Rod IE-016, the onset of PCI would be expected to occur at a lower power for Rod IE-017. Therefore, no conclusions can be drawn on this aspect of the test.

During film boiling operation, increases in fuel rod internal pressure shown in Figure 46 ranged from 6 to 20%, and cladding elongation shown in Figure 47 ranged from 0.4 to 0.6%. Cladding surface

Fig. 46 Fuel rod internal pressure for all four rods during the flow reduction phase of Test IE-3.

Fig. 47 Cladding elongation for Rods IE-016, IE-017, and IE-018 during the flow reduction phase of Test IE-3.

temperature measurements reached a maximum of 880 K, which is well below the expected temperature of 1500 K. The low cladding surface temperature measured resulted from a systematic measurement error associated with the thermocouple installation. A more realistic estimate of the cladding surface temperature during film boiling will be obtained using cladding reaction layer thickness measurements during the postirradiation examinations.

Neutrographs of the fuel rods taken during preliminary postirradiation examination indicated that substantial molten fuel regions occurred within the film boiling zone of Rods IE-015 and IE-016. Lesser amounts of molten fuel were found in Rods IE-017 and IE-018. The visual inspection of the fuel rods indicated that oxide formations, oxide spalling, and local cladding collapse occurred in the film boiling region.

7.1 Comparison With Previous Experimental Data

In this section the experimental behavior of the irradiated fuel rods during the fast power ramp and during the film boiling event of Test IE-3 are compared with results of other tests in the Irradiation Effects Test Series. During Test IE-1, four irradiated fuel rods were tested in a similar experiment sequence, but the rate of power increase (3.2 kW/m per minute) during the high power ramp was about an order of magnitude less than the ramp rate (20 kW/m per minute) used in Test IE-3. The purpose for the increased power ramp rate for Test IE-3 was to determine if the ramp rate would affect the PCI that occurs during the ramp via weakening or even failure of the cladding from highly concentrated local stresses or enhanced stress-corrosion cracking. In addition to the cladding elongation data comparisons, fission gas release data from similarly tested irradiated fuel rods (IE Scoping Test 2 and Test IE-1) are compared with fission gas release data from this test.

7.1.1 Cladding Elongation. A summary of selected fuel rod design and test conduct parameters for Tests IE-1 and IE-3 are shown in Table II. Also listed in this table are the rate of power ramp and rate of cladding length increase for each test. This table provides a listing of a few similarities and differences in fuel rods and test conduct between the two tests. Cladding elongation versus fuel rod average power during the high power ramp of each test are shown in Figure 48. Data for two fuel

Fig. 48 Cladding elongation versus fuel rod average power for Rods IE-007 (Test IE-1), IE-009 (Test IE-1), IE-016 (Test IE-3), and IE-017 (Test IE-3) during the high power ramp of each test.

rods from each of Tests IE-1 and IE-3 are shown in this figure. Differences in the amount of cladding elongation per increase of fuel rod average power between Rods IE-007 and IE-009, as shown in Figure 48, have been attributed to differences in fabricated fuel density^[6]. Rods IE-007 and IE-017 have the same fuel density and only differ in the amount of burnup (6,830 and 14,780 MWd/tU, respectively), yet the amount of cladding elongation per increase in fuel rod power for Rod IE-017 is almost twice that of Rod IE-007. The cladding elongation of Rod IE-016 equals that of Rod IE-017, but the two cannot be directly compared due

TABLE II

SELECTED FUEL ROD DESIGN AND TEST CONDUCT PARAMETERS FOR TESTS IE-1 AND IE-3

<u>Test</u>	<u>Fuel Rod</u>	<u>Burnup (MWd/tU)</u>	<u>Fabricated Fuel Density (% TD)</u>	<u>Fabricated Diametral Gap (mm)</u>	<u>Power Ramp Rate kW/m per minute</u>	<u>Rate of Cladding Elongation % per kW/m</u>
IE-1	IE-007	6830	94	0.22	3.2	0.005
IE-1	IE-009	16640	92	0.22	3.2	0.003
IE-3	IE-016	8550	94	0.27	20	0.009
IE-3	IE-017	14780	94	0.22	20	0.013

to the presence of the relocated fuel at the 0.69-m elevation in Rod IE-016. The differences in cladding elongation between Rod IE-007 and IE-017 are due to the rate of power increase. During the fast power ramp (20 kW/m per minute) for Rod IE-017, the amount of slippage, fuel redistribution, and fuel creep were limited by the short duration of the power ramp. The flatter, almost convex shape of the cladding elongation data for Test IE-1 fuel rods (Rods IE-007 and IE-009) indicate that one or more of these relaxation type of phenomena occurred during the slower power ramp (3.2 kW/m per minute). The relocated fuel and enhanced plastic behavior of the fuel in Rod IE-016 caused by high fuel temperatures may have contributed to the relaxation phenomena mentioned and produced the elongation versus power curve shown for Rod IE-016 in Figure 48. The shape of the curve for Rod IE-016 is similar to the data for Rods IE-007 and IE-009, despite the greater power ramp rate. These data are in contrast to the distinctly concave shape of the cladding elongation data for Rod IE-017.

7.1.2 Fuel Rod Internal Pressure. Fuel rod internal pressure increases due to fission gas release during the high power, steady-state operation are discussed in this section. Pressure increases due to fission gas release and thermal effects during film boiling are also discussed.

The percent increase in fuel rod internal pressure due to fission gas released during the steady-state period high power operation for selected rods of the IE Test Series are shown in Table III. The duration of the steady-state period was approximately one hour in all of the tests. The amount of fission gas released during these periods of high power steady-state operation in PBF depended upon the fission gas inventory generated and retained in the fuel matrix during irradiation in the Saxton reactor and during the preconditioning phase in the PBF testing. The above fuel rods had differences in burnup, power history, and fuel temperature which could account for the range of fission gas release indicated by Table III.

TABLE III

MEASURED FUEL ROD INTERNAL PRESSURE INCREASES DUE TO
FISSION GAS RELEASE DURING HIGH POWER STEADY-STATE
OPERATION IN TESTS IE-ST-2, IE-1, and IE-3

<u>Test</u>	<u>Rod</u>	<u>Burnup (MWd/tU)</u>	<u>Fuel Rod Average Power (kW/m)</u>	<u>Fuel Rod Pressure Increase (%)</u>
IE-ST-2	IE-002	15800	40	5
IE-1	IE-007	6830	54	7
IE-1	IE-009	16640	51	12
IE-1	IE-010	15510	53	5
IE-3	IE-015	11060	55	24
IE-3	IE-016	8550	50	13
IE-3	IE-017	14780	55	5

Table IV lists the amounts of relative fuel rod internal pressure increases during film boiling for fuel rods in Tests IE-1 and IE-3. These increases, which range from 7 to 20%, were due to fission gas release from fuel in the film boiling zone, an increase in plenum temperature, and a change in void volume caused by fuel cladding differential thermal expansion, fuel swelling, and cladding collapse.

The relative amounts of fission gas released during the high power periods and during film boiling were calculated for Rods IE-007, IE-016, and IE-017 and the results are shown on Table V. This calculation was made by using the pressure data shown in Figures 37, 39, and 41, the perfect gas law, the posttest measured void volume, and the following assumptions: (a) a uniform radial and axial fission gas distribution in the rod; (b) 9500 MWd/tU per atom percent burnup; (c) that 30 fission gas atoms are produced per 100 fissions; and (d) that the total gas inventory generated in the rod was not released until high-power steady-state operation was reached in PBF. Actually, fission gas is released

TABLE IV

MEASURED FUEL ROD INTERNAL PRESSURE INCREASES DURING FILM BOILING OPERATION FOR
FUEL RODS IN TESTS IE-1 and IE-3

Test	Fuel Rod	Fuel Rod Average Power (kW/m)	Pre-Film Boiling Pressure (MPa)	Maximum Pressure During Film Boiling (MPa)	Pressure Increase During Film Boiling (MPa)	Increase (%)
IE-1	IE-007	53.0	5.9	6.5	0.6	10
IE-1	IE-009	50.3	5.0	5.9	0.9	18
IE-1	IE-010	51.7	6.8	7.4	0.6	11
IE-3	IE-016	49.2	10.7	12.8	2.1	20
IE-3	IE-017	54.5	8.2	8.8	0.6	7

TABLE V

FISSION GAS RELEASED DURING STEADY-STATE AND FILM BOILING OPERATION FOR FUEL RODS IN TESTS IE-1 and IE-3

Test	Fuel Rod	Calculated Fission Gas Inventory After Saxton Irradiation (moles)	Fuel Rod Peak Power (kW/m)	Fission Gas Released During Steady-State High Power Period		Fission Gas Released During Film Boiling Period	
				(moles)	(Percent of total available in com- plete fuel rod)	(moles)	(Percent of total available in film boiling zone)
IE-1	IE-007	4.04×10^{-3}	72	0.36×10^{-3}	9	1.06×10^{-3}	96
IE-3	IE-016	4.02×10^{-3}	65	1.23×10^{-3}	31	9.43×10^{-4}	68
IE-3	IE-017	8.95×10^{-3}	72	1.78×10^{-4}	2	1.42×10^{-4}	6

from the fuel matrix continuously so the calculated fission gas inventory in moles shown in Table V is a maximum value.

Differences in fission gas release between the fuel rods are attributed to differences in the amount of the original fission gas inventory remaining in each fuel rod prior to Test IE-3 and differences in the fuel temperatures of each rod during steady-state and film boiling operation. Preliminary postirradiation examination results indicate that much more fuel melting occurred in the film boiling region of Rods IE-007 and IE-016 than in Rod IE-017. It is generally accepted that 100 percent of the fission gases are released from regions of molten fuel which helps explain why substantially more fission gases were released from Rods IE-007 and IE-016 than from Rod IE-017 during film boiling operation. Although the amount of fission gas released from the 0.242-m long film boiling region in Rod IE-016 is significant, the total fuel rod internal pressure increase during film boiling due to both thermal effects and fission gas release was only 20 percent. For Rod IE-007 the total measured pressure increase during film boiling was 18 percent. Fission gas release during high power steady-state operation was 9, 31, and 2 percent for Rods IE-007, IE-016, and IE-017, respectively. These data indicate that the fuel rod pressure increases due to thermal effects and fission gas release were not large enough to cause large fuel rod internal pressure increases which would result in cladding ballooning or cladding failure. The pressure increase due to fission gas release during steady-state or film boiling operation is not likely to significantly affect rod behavior for rods with these levels of burnup and operated under similar conditions.

7.2 FRAP-T3 Results

Comparison between the experimental and calculated fuel rod internal pressures and cladding elongations were made in Section 6. A few comments regarding the FRAP-T3 comparisons are made in this section.

7.2.1 Cladding Elongation. Comparisons made between the cladding elongation data for Rod IE-017 and FRAP-T3 calculations indicate that the onset of hard PCI is not calculated well. The onset of hard PCI occurred at about 20 kW/m for Rod IE-017, while hard PCI was not calculated to occur until just over 40 kW/m. These results are consistent with comparisons made between test data and calculations in the IE-1^[6] and IE-2^[12] Test Results Reports.

Differences in the calculated and measured cladding elongation data result from simplifications made in the FRAP-T3 model. The PCI model in the FRAP-T3 code uses an annular gap, while a well defined annular gap does not exist in an irradiated fuel rod. Complex mechanisms of sintering, swelling, cracking, fuel creep and fuel pellet fragment redistribution contribute to an effective diametral gap different than that of a uniform annular model. The annular gap was also reduced due to cladding creepdown during irradiation in the Saxton reactor. The measured cladding outside diameter for the fuel rods prior to PBF testing is shown in Figure B-I in Appendix B. These data show evidence of cladding creepdown in Rods IE-017 and IE-018. In making the FRAP-T3 calculations, the minimum diameter indicated on Figure B-1 was used along with the nominal wall thickness to determine an input for a gap size.

Figure 49 shows a plot of cladding elongation versus fuel rod average power during the ramp for Rod IE-007 (Test IE-1) and Rod IE-017. Also shown in this figure are calculated cladding elongations using FRAP-T3 with input gap sizes ranging from 0.04 to 0.10 mm. These data indicate that the onset of hard PCI can be calculated correctly using a smaller diametral gap, but once the onset of PCI is calculated, the rate of cladding elongation is overestimated. The data shown in the figure indicate that the FRAP-T3 calculations provide a better estimate of the rate of cladding growth for the ramp rate of 20 kW/m per minute (Rod IE-017) than for the slower ramp rate of 3.2 kW/m per minute (Rod IE-007). Better agreement is expected because creep rate effects, which are not modeled by FRAP-T3, are less apparent at the higher ramp rate.

Fig. 49 Comparison of experimental and calculated (FRAP-T3) cladding elongation versus fuel rod average power for Rod IE-007 (Test IE-1) and Rod IE-017 (Test IE-3).

A plot similar to Figure 49 was presented in the IE-1 Test Results Report for Rod IE-007. There it was argued that the calculated cladding elongation was based on free thermal expansion of the fuel and calculations did not consider that the elastically extended cladding would deform the fuel. When the data from Rod IE-017 is compared to the FRAP-T3 calculations, this argument is still true but to a lesser extent. The amount of cladding elongation decrease during the high power steady-state period was greater for Rod IE-017 than for the Rods IE-007 and IE-009. This suggests that residual stresses built up during the rapid power ramp are causing creep and fuel relocation to occur to a greater extent during the steady-state period than occurred for Rods IE-007 and IE-009. The total amount of relocation during the steady-state period was greatest for Rod IE-017.

The total cladding elongation for Rod IE-017 was greater than for Rods IE-007 and IE-009 during the high power ramp. The increased amount

of strain caused by the higher power ramp rate was not sufficient to cause cladding failure. Fuel rod behavior during film boiling operation which followed the rapid power ramp was similar to the behavior of fuel rods subjected to slower power ramps.

7.2.2 Fuel Rod Internal Pressure. FRAP-T3 underpredicted the increase in steady-state internal pressure of Rod IE-017 as a function of reactor power by approximately 8% during the preconditioning phase. Some divergence between the measured and calculated values occurred due to fission gas release during the high power steady-state periods following the power ramp. Pressure increases which occurred at constant power were not calculated since FRAP-T3 does not contain a fission gas release model.

FRAP-T3 underpredicted transient fuel rod internal pressure increases during film boiling significantly. The measured pressure in Rod IE-017 increase was 0.6 MPa, while essentially no increase was calculated by FRAP-T3. During film boiling, the measured fuel rod internal pressure increase was due to both thermal effects (heated fuel and cladding) and fission gas release. A calculation indicates that about 4% of the 0.6 MPa increase during film boiling was due to fission gas release. It is apparent that the FRAP-T3 calculated fuel rod pressure increase is less than the measured pressure increase in the fuel rod due to thermal effects during film boiling. FRAP-T3 calculated gap closure prior to film boiling and consequently no gas existed within the film boiling zone. The fuel and cladding temperature increase during film boiling, therefore, has little affect on the gas temperature.

7.3 Conclusions

The following conclusions can be drawn from this experiment:

- (1) The rapid power ramp performed at a rate of 20 kW/m per minute did not result in fuel rod failure. The higher rate of power increase did cause an increased rate of cladding elongation and total strain when compared to fuel rods of Test IE-1,

performed at a rate of 3.2 kW/m, but there was no obvious effect of this fast ramp on fuel rod behavior during the subsequent steady high power operation or the film boiling event.

- (2) Cladding elongation, fuel rod internal pressure, and fission gas release data for Rods IE-015 and IE-016 are not typical of the other fuel rods in the IE Test Series. This is attributed to the extensively relocated fuel near the 0.69-m elevation of both rods.
- (3) Fission gas release occurred during steady-state operation at an average fuel rod peak power of 69 kW/m and during film boiling operation. The measured fuel rod internal pressure increase during film boiling due to a combination of fission gas release and thermal effects was small (less than 20 percent). This increase is generally insignificant with respect to fuel behavior unless the rod internal pressure prior to film boiling was near or above system pressure.
- (4) Rod IE-015 experienced a loss of cladding integrity approximately 45 seconds after reactor shutdown, similar to Rod IE-007 in Test IE-1. The failure location was not visually discernible during posttest examinations.
- (5) The test results indicate that the irradiated state of the fuel rods did not significantly affect rod behavior during normal, abnormal (power ramp at 20 kW/m per minute), and the accident (film boiling) conditions experienced during this test.

8. REFERENCES

1. United States Nuclear Regulatory Commission, Reactor Safety Research Program, *A Description of Current and Planned Reactor Safety Research Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research*, NUREG-75/058 (June 1975).
2. L. B. Thompson, O. Goetzmann, W. J. Quapp, and P. E. MacDonald, *Light Water Reactor Fuel Behavior Program Description: Irradiation Effects Experiment Requirements*, TFBP-TR-175, EG&G Idaho Inc. (February 1977).
3. S. A. Ploger et al., *Postirradiation Examination Results for the Irradiation Effects Test 3* (to be published as TREE-NUREG).
4. G. W. Gibson et al., *Characteristics of UO_2 -Zircaloy Fuel Rod Materials from the Saxton Reactor for Use in the Power Burst Facility*, ANCR-NUREG-1321 (September 1976).
5. M. E. Yancey et al., *Inspection System for Zircaloy Clad Fuel Rods*, ANCR-1259 (February 1976).
6. W. J. Quapp et al., *Irradiation Effects Test Series Test IE-1 Test Results Report*, TREE-NUREG-1046 (March 1977).
7. J. A. Dearien et al., *FRAP-T3: A Computer Code for the Transient Analysis of Oxidized Fuel Rods*, Report 1, TREE-NUREG-1173 (to be published).
8. P. E. MacDonald and L. B. Thompson (eds.), *MATPRO Version 09 - A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior*, TREE-NUREG-1005 (December 1976).
9. L. S. Tong, "Critical Heat Fluxes in Rod Bundles," *Two-Phase Flow and Heat Transfer in Rod Bundles Symposium*, ASME Winter Annual Meeting (November 1969) pp 31-46.

10. D. C. Groeneveld, *An Investigation of Heat Transfer in the Liquid Deficient Regime*, AECL-3281 (December 1968 revised August 1969).
11. A. M. Ross and R. L. Stoute, *Heat Transfer Coefficient Between UO_2 and Zircaloy-2*, AECL-1552 (June 1962).
12. W. J. Quapp et al., *Irradiation Effects Test Series Scoping Test 2 Test Results Report*, TREE-NUREG-1044 (September 1977).
13. C. M. Allison et al., *Irradiation Effects Test Series Test IE-2 Test Results Report*, TREE-NUREG-1074 (August 1977).

APPENDIX A

INSTRUMENTATION AND DATA ACQUISITION SYSTEM UNCERTAINTIES

The calibration equations that were used to convert the transducer output to engineering units during the data reduction process for Test IE-3 are presented in this appendix.

Data obtained during the conduct of the test are subject to three significant sources of error: (a) instrument calibration error, (b) data acquisition error, and (c) systematic measurement error. Estimates of both the instrument calibration error and data acquisition error are provided in this appendix. The systematic measurement error is a consequence of the transducer design and installation. To quantify the effect of the systematic measurement error it would be necessary, either by analysis or experiment, to simulate the in-use configuration of the transducer. Since this effort has not been made, an estimate of this error is not available. A qualitative assessment of the systematic measurement error indicates that this type of error is not significant (except in the case of the cladding surface temperature measurement, where large systematic errors are known to exist) and that the measurements provided by the instruments reflect values of the physical parameter measured, within the error estimates specified in this appendix. This statement is made assuming that the instruments do not decalibrate prior to or during the experiment, however, posttest calibrations were not performed on the test instrumentation to verify this assumption due to the difficulties involved in handling irradiated components. As indicated by the test data for the fuel rod internal pressure and cladding elongation, significant errors due to decalibration can occur.

Calibration equations for each instrument are presented in a tabular form in Table A-I. Included in the table is the 95% confidence interval estimate for each equation (the calibration error). The data system error and the quadratic sum of the data system and calibration instrument errors are also provided.

TABLE A-I

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTIES IN TEST IE-3 EXPERIMENTAL MEASUREMENTS

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
System Pressure	59-MPa Strain Post (SN 158)	[a]	$\text{MPa} = -12.031 + 4.4945 (\text{mV}) + 0.1438(\text{K})$	0	(MPa)	(MPa)	(MPa)	(MPa)
				25	0	0.22	0.58	0.62
				50	9.3	0.27	0.43	0.51
				75	18.6	0.35	0.33	0.48
				100	27.9	0.46	0.45	0.64
System Pressure	20-MPa Strain Post (SN 309)	[a]	$\text{MPa} = -3.254 + 0.59083(\text{mV})$	0	37.2 ^[b]	0.58	0.63	0.86
				25	0	0.11	0.07	0.13
				50	5.2	0.13	0.05	0.14
				75	10.3	0.17	0.04	0.17
				100	15.5	0.23	0.05	0.24
Coolant Inlet and Outlet Temperature	Chromel-Alumel Thermocouple (All Devices)	[a]	$\text{K} = 343.6 + 24.05 (\text{mV})$	0	20.7	0.29	0.07	0.30
				25	(K)	(K)	(K)	(K)
				50	578	0.3	1.1	1.1
				75	590	0.3	1.2	1.2
				100	602	0.4	1.2	1.3
				75	614	0.5	1.3	1.4
				100	626	0.7	1.3	1.5

[a] Calibrations are not unique to a particular fuel rod.

[b] Calibrated range of transducer output, operational range is 69 MPa.

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTIES IN TEST IE-3 EXPERIMENTAL MEASUREMENTS

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
Fuel Rod Pressure	17.2-MPa Strain Post (SN 440)	IE-015	$MPa = 7.791 + 3.370 (mV) - 0.0160 (K)$	0 25 50 75 100	(MPa) 0 4.3 8.6 12.9 17.2	(MPa) 0.09 0.11 0.14 0.14 0.24	(MPa) 0.03 0.02 0.02 0.02 0.03	(MPa) 0.10 0.11 0.14 0.14 0.24
Fuel Rod Pressure	17.2-MPa Strain Post (SN 441)	IE-016	$MPa = 0.495 + 3.160 (mV) + 0.00812 (K)$	0 25 50 75 100	0 4.3 8.6 12.9 17.2	0.09 0.11 0.14 0.19 0.24	0.03 0.02 0.02 0.02 0.03	0.10 0.11 0.14 0.19 0.24
Fuel Rod Pressure	17.2-MPa Strain Post (SN 436)	IE-017	$MPa = 7.462 + 3.4077 (mV) - 0.0169 (K)$	0 25 50 75 100	0 4.3 8.6 12.9 17.2	0.09 0.11 0.14 0.19 0.24	0.04 0.03 0.02 0.02 0.04	0.10 0.11 0.14 0.19 0.24
Fuel Rod Pressure	17.2-MPa Strain Post (SN 438)	IE-018	$MPa = -5.453 + 3.547 (mV) - 0.0162 (K)$	0 25 50 75 100	0 4.3 8.6 12.9 17.2	0.09 0.11 0.14 0.19 0.24	0.04 0.03 0.02 0.02 0.04	0.10 0.11 0.14 0.19 0.24

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTIES IN TEST IE-3 EXPERIMENTAL MEASUREMENTS

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
Cladding Elongation	LVDT (SN 427)	IE-015	$\text{mm} = 0.07725 + 8.8475 \text{ (mV)} + 1.5963 \text{ (mV}^2\text{)} [a]$	(mm) 0 25 50 75 100	(mm) 0 3.2 6.4 9.5 12.7	(mm) 0.07 0.08 0.11 0.14 0.18	(mm) 0.14 0.08 0.08 0.07 0.13	(mm) 0.16 0.11 0.14 0.16 0.22
Cladding Elongation	LVDT (SN 429)	IE-016	$\text{mm} = 0.03767 + 9.7842 \text{ (mV)} + 0.7076 \text{ (mV}^2\text{)} [a]$	0 25 50 75 100	0 3.2 6.4 9.5 12.7	0.07 0.08 0.11 0.14 0.18	0.06 0.04 0.04 0.03 0.06	0.09 0.09 0.12 0.18 0.19
Cladding Elongation	LVDT (SN 428)	IE-017	$\text{mm} = 0.03647 + 9.83606 \text{ (mV)} + 0.6272 \text{ (mV}^2\text{)} [a]$	0 25 50 75 100	0 3.2 6.4 9.5 12.7	0.07 0.08 0.11 0.14 0.18	0.07 0.04 0.04 0.04 0.07	0.10 0.09 0.12 0.15 0.19
Cladding Elongation	LVDT (SN 431)	IE-018	$\text{mm} = 0.05457 + 9.1959 \text{ (mV)} + 1.1767 \text{ (mV}^2\text{)} [a]$	0 25 50 75 100	0 3.2 6.4 9.5 12.7	0.07 0.08 0.11 0.14 0.18	0.10 0.06 0.06 0.06 0.10	0.12 0.10 0.12 0.15 0.21

[a] These equations are applicable to coolant inlet temperatures of 590 K.

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTIES IN TEST IE-3 EXPERIMENTAL MEASUREMENTS

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% Of Span)	Output	Data System Error	Calibration Error	Total Error
				(K)	(K)	(K)	(K)	(K)
Cladding Surface Temperature	Chromel-Alumel Thermocouple (Type K)	IE-018	$K = 355.4 + 2.4 \text{ (mV)} + 0.57 \text{ (}\mu\text{V}^2\text{)}$	0	544	6	4	7
				25	813	7	2	8
				50	1083	9	2	11
				75	1352	12	2	15
				100	1622	15	4	18
Cladding Surface Temperature	W5%Re/W26%Re Thermocouple	IE-017	$K = 346.17 + 55.92 \text{ (mV)} - 0.759896 \text{ (}\mu\text{V}^2\text{)}$	0	544	6	3	7
				25	813	7	2	9
				50	1083	9	2	12
				75	1352	12	2	16
				100	1622	15	2	21
Coolant Temperature Differential	Copper-Constantan Thermocouple (Type T)	[a]	$K = 0.0250 + 16.971 \text{ mV}^{[b]}$	0	0	0.12	0.01	0.12
				25	5.60	0.14	0.015	0.14
				50	11.20	0.19	0.034	0.19
				75	16.80	0.25	0.053	0.26
				100	22.40	0.31	0.076	0.32

[a] Calibration is not unique to a particular fuel rod.

[b] This equation is applicable to coolant inlet temperature of 606 K.

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTIES IN TEST IE-3 EXPERIMENTAL MEASUREMENTS

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output (cm ³ /s)	Data System Error (cm ³ /s)	Calibration Error (cm ³ /s)	Total Error (cm ³ /s)
Coolant Flow	Turbine Flowmeter (SN 104)	IE-015	$\text{cm}^3/\text{s} = -0.250 + 0.485 (\text{Hz})$	0	0	6.8	[a]	[a]
				25	315.8	8.1	0.75	8.1
				50	631.7	10.6	1.05	10.6
				75	947.5	14.1	1.80	14.2
				100	1263.3	17.7	[a]	[a]
Coolant Flow	Turbine Flowmeter (SN 101)	IE-016	$\text{cm}^3/\text{s} = 0.111 + 0.481 (\text{Hz})$	0	0	6.8	[a]	[a]
				25	315.5	8.1	0.10	8.1
				50	630.9	10.6	0.10	10.6
				75	946.4	14.1	0.15	14.1
				100	1261.8	17.7	[a]	[a]
Coolant Flow	Turbine Flowmeter (SN 102)	IE-017	$\text{cm}^3/\text{s} = 0.634 + 0.476 (\text{Hz})$	0	0	6.8	[a]	[a]
				25	317.8	8.1	0.70	8.1
				50	635.6	10.6	0.95	10.6
				75	953.5	14.1	1.70	14.2
				100	1271.3	17.7	[a]	[a]
Coolant Flow	Turbine Flowmeter (SN 103)	IE-018	$\text{cm}^3/\text{s} = 0.601 + 0.481 (\text{Hz})$	0	0	6.8	[a]	[a]
				25	315.5	8.1	0.55	8.1
				50	630.9	10.1	0.75	10.6
				75	946.4	14.1	1.85	14.2
				100	1261.8	17.7	[a]	[a]

[a] Output outside expected range of use so device not calibrated for these flow rates.

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

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

where

L_i = confidence interval estimate

Y_i = calculated value of the dependent variable from 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 error estimate applies.

The evaluation of L_i for several x_i values established the 95% confidence interval estimate.

For those transducers that had functional relationships that were non-linear or more 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 was 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) System Pressure. One 69-MPa and one 20-MPa strain post pressure transducer were used to monitor the coolant pressure.

(2) Coolant Inlet Temperature. Two Chromel-Alumel (Type K) thermocouples were used to measure the coolant inlet temperature. These devices were not calibrated. The 95% confidence interval estimate was 0.375% of the reading as stated by the manufacturer when using the standard calibration tables.

(3) Coolant Outlet Temperature. One Chromel-Alumel (Type K) thermocouple was provided to obtain the coolant outlet temperature of each fuel rod. Again, these devices were not calibrated and the error estimate for these devices was the same as the inlet temperature thermocouples, 0.375% of the reading as stated by the manufacturer.

(4) Coolant Temperature Rise. One copper-constantan differential temperature measurement thermocouple pair was positioned in each flow

shroud. Calibrations were obtained for the thermocouple pair from standard thermocouple tables. Inlet temperatures of 555, 583, and 606 K with a 20 K temperature rise were used in generating the thermocouple functions. The total estimated error is the square root of the sum of the squares of the calibration error, data system error, and 0.375% of the temperature differential. The calibration error was determined at a coolant inlet temperature of 606 K.

(5) Coolant Flow. Four flowmeters with graphite journal bearings were used to measure the coolant flow rate through each flow shroud. These flowmeters were calibrated by the manufacturer (Flow Technology Incorporated). Although calibrations were conducted with a slightly different inlet and outlet flow geometry than was used during the test, it was established that these deviations did not affect the measured coolant flow accuracy.

(6) Cladding Elongation. A linear variable differential transformer (LVDT) was used on each fuel rod to measure cladding elongation. The output of each LVDT is sensitive to transducer temperature, lead wire resistance, and the electronic settings on the signal conditioner used. The transducers that were utilized for this experiment were not calibrated at the temperature and with the same lead wire and electronic settings that were used during the test.

The procedure followed in determining a calibration function for the LVDTs is briefly described below. Initially the transducer outputs were monitored for various core displacements at several case temperatures (290, 370, 470, 580, 600, and 610 K). Then at a later date, the transducers were calibrated at room temperature with a different set of lead wires and electronic settings (phase and gain), that is, those used during the test. During this second calibration the phase and gain on the signal conditioner driving the LVDT were adjusted so that the room temperature calibration curve closely matched the previous calibration obtained at room temperature during the first calibration sequence. The LVDTs were then used in the test assuming that the calibration curves at

the elevated temperature found in the initial calibration would be applicable with the new lead wire resistance. This assumption does not result in an exact LVDT calibration, but it was estimated that this method resulted in a calibration function that was within two percent of the actual calibration.

Fuel Rod Instrumentation

(1) Fuel Rod Internal Pressure. Each fuel rod was fit with a 17-MPa strain post type pressure transducer to measure the fuel rod internal pressure. The error estimates provided in this appendix represent the uncertainty associated with a multivariate (transducer case temperature and diaphragm pressure) fit to the calibration data. The design of these devices is such that they are occasionally subject to zero offsets between repeated pressure increase and decrease cycles. Sometimes this drift is excessive. A review of calibration data for several of these pressure transducers has shown that the calibration of the transducer generally does not shift during a pressure increase cycle but instead, the shift usually occurs during a pressure decrease cycle. Consequently, the pressure differentials indicated during power increases are likely to be an accurate measure of the fuel rod internal pressure. The reason for the zero shift is not fully understood, but it is most likely due to the strain gauges not being integrally attached to the strain post in the pressure transducer.

(2) Cladding Surface Temperature. Tungsten-rhenium and Chromel-Alumel thermocouples were used in making the cladding surface temperature measurements. Again, standard calibration tables were used to convert these data. The total error is the square root of the sum of the calibration error, data system error and a percentage of the indicated reading. This percentage was 0.75% for the Type K thermocouples and 1% for the tungsten-rhenium thermocouples.

Cladding surface temperature data versus fuel rod local power for a 0.62-m elevation thermocouple (brazed thermocouple) on Rod IE-018

during Cycle 5 are shown in Figure A-1. The cladding surface temperature calculated by FRAP-T3 is also shown in this figure. The change in slope of the calculated cladding surface temperature is due to the transition from the forced convection regime to the nucleate boiling regime. As shown in the figure, the experimental data does not indicate a transition from forced convection to the nucleate boiling regime. This discrepancy between the calculated and measured temperatures is due to a systematic measurement error, which is discussed below.

The cladding surface temperature measurements on Rod IE-018 were made using a tantalum sheathed, Chromel-Alumel, grounded junction thermocouple that had a compacted magnesium oxide insulation. A thermocouple installed on the cladding surface is shown schematically in Figure A-2. The thermocouple junction was flattened to the approximate dimensions shown in the figure. For each thermocouple manufactured, the thermocouple wires are not always centrally placed in the sheath and the axis on which the thermocouple is flattened is not necessarily as shown in the figure. Therefore the thermocouple junction is positioned differently relative to the cladding surface for each thermocouple installation. Variation in the temperature of the thermocouple junction will occur for differences in the combined thickness of the braze filler, thermocouple sheath, tantalum insert and any contact resistance that might exist between the thermocouple sheath and the cladding. Therefore, a different systematic measurement error would be associated with each cladding surface thermocouple.

Shown in Figure A-3 is a plot of the steady-state cladding temperature at the inside surface midradius, and outside surface of the cladding calculated by FRAP-T3 for several fuel rod local powers. These calculations are for the cladding temperatures at the 0.61-m elevation during Cycle 5. At powers up to 18 kW/m, the cladding outside surface temperature changes about 4 K with each 6 kW/m increment in power. Heat is transferred from the cladding surface by forced convection. For a change in power from 24 to 60 kW/m, the cladding surface temperature changes only about 3 K. This occurs because the heat is transferred

Fig. A-1 Cladding surface temperature versus fuel rod local power on Rod IE-018 during Cycle 5, Test IE-3.

Fig. A-2 Cross section of a brazed thermocouple on the fuel rod cladding surface of Rod IE-018, Test IE-3.

Fig. A-3 Calculated cladding temperatures for various fuel rod local powers during Cycle 5, Test IE-3.

from the cladding surface more efficiently by nucleate boiling. As the heat flux (fuel rod power) increases from 24 to 60 kW/m, the cladding surface temperature remains relatively constant. Thus, the temperature gradient through the cladding must increase since both the thermal conductivity and dimensions of the cladding remain relatively constant during this period.

At fuel rod powers up to 18 kW/m, the data are offset from the calculated cladding surface temperature by a relatively constant value^[a] of about 2 K (Figure A-1). This constant offset results from the systematic measurement error discussed above. At fuel rod powers above 18 kW/m, the data diverges from the calculated values linearly, such that there is a 13 K difference in measured and calculated cladding surface temperature at 56 kW/m. The discrepancy between the measured and calculated value at fuel rod powers above 18 kW/m is also due to the systematic measurement error associated with the thermocouple installation.

The measurement error changes as a function of fuel rod power because the heat transfer, above fuel rod powers of 18 kW/m, is by nucleate boiling from the cladding surface. The cladding outer surface and thermocouple outer surface, where they contact the coolant, are maintained at a relatively constant temperature for fuel rod powers from 24 to 60 kW/m. For purposes of illustration, the thermocouple junction can be visualized as being located inside the cladding outer surface. Hence, the thermocouple will measure a temperature that is between the midradius and outer surface as shown in Figure A-3. As seen from this figure, a measurement made at an inner location in the cladding surface will diverge from the true cladding surface temperature similarly to the experimental data in Figure A-1 for fuel rod powers above 18 kW/m. It is realized that the temperature distribution in the thermocouple is not

[a] A constant offset of 6 K was added to these data so that the measured cladding surface temperature matched coolant temperature at 0 kW/m fuel rod power.

as simple as illustrated in Figure A-3 because of the thermal resistance and characteristic fin-like geometry of the thermocouple mounted on the cladding surface, but this diagram does show that the type of measurement errors indicated in Figure A-1 (13 K at 56 kW/m) result as discussed above.

Data Acquisition System

A summary of the data recording system errors associated with decalibration and drift of the electronics used to record and reduce the data are given for each transducer in Table A-I. A thorough discussion of the source of these errors is given in Appendix B of the IE Scoping Test 1 Report^[A-2].

The calculation procedure used in making these estimates of error is briefly outlined below. The data system amplification, gain, and offset are such that the input signal (transducer output) and the output signal (recorded data signal) are related by the equation:

$$V_o = (V_i - V_z) G \quad (A-3)$$

where

V_o = recorded data signal (volts)

V_i = transducer output signal (volts)

V_z = offset (volts)

G = gain.

The standard deviation of the recorded data signal, S_{V_o} , can then be determined from

$$S_{V_o} = \left[\left(\frac{V_o}{G} \right)^2 S_G^2 + G^2 S_{V_z}^2 + G^2 S_{V_i}^2 \right]^{1/2}. \quad (A-4)$$

where

S_G = standard deviation of the gain

S_{V_z} = standard deviation of the offset (volts)

S_{V_i} = standard deviation of the transducer output signal (volts).

For the calibration, S_{V_i} can be considered to be zero such that

$$S_{V_o} = \left[\left(\frac{V_o}{G} \right)^2 S_G^2 + G^2 S_{V_z}^2 \right]^{1/2}. \quad (A-5)$$

A test performed to evaluate the data system resulted in values of $\frac{S_G}{G}$ and GS_{V_z} to be 0.65% and 2.7 millivolts respectively.

For a total span ($Y_{\max} - Y_{\min}$) of 1000 millivolts, on the recording channel Equation A-5 can be simplified to

$$\frac{S_Y}{Y_{\max} - Y_{\min}} \% = \left[\left(\frac{Y - Y_{\min}}{Y_{\max} - Y_{\min}} \right)^2 \times (0.65\%)^2 + (0.27\%)^2 \right]^{1/2} \quad (A-6)$$

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 signal.

The total transducer measurement uncertainty, excluding the systematic measurement error, can be estimated by combining the uncertainty in the calibration and the uncertainty associated with the data acquisition system. The "total error" is given to be the square root of the sum of the squares of these two errors. Table A-I summarizes the data system error, calibration error, and the resultant total error for each of the transducers that functioned during the test.

REFERENCES

- A-1. M. G. Natrella, *Experimental Statistics, National Bureau of Standards Handbook 91*, U. S. Government Printing Office, 1963.
- A-2. W. J. Quapp et al., *Irradiation Effects Test Series Scoping Test 1 Test Results Report*, TREE-NUREG-1066 (September 1977).

APPENDIX B

FUEL ROD CHARACTERIZATION DATA

The pretest fuel rod characterization data for Test IE-3 is presented in this appendix. These data are reported to: (a) facilitate a comparison of fuel rod physical characteristics between IE experiments, (b) provide input for the analytical models used for calculations of fuel rod behavior during nuclear operations, and (c) provide input for posttest model verification activities. The fuel rod cladding measurements were obtained in U.S. Customary units and converted to SI units.

All four fuel rods were fabricated from irradiated Saxton fuel rods. The Saxton fuel rod numbers for Rods IE-015, IE-016, IE-017, and IE-018 were 721, 724, 888, and 890, respectively. A complete description of these fuel rods is given in Reference B-1. The upper end cap on each fuel rod was removed and replaced with one containing a pressure transducer. The fuel rods were then backfilled with a 77.7% helium and 22.3% argon gas mixture. Rods IE-015 and IE-016 had irradiated fuel stack lengths of about 0.69 m, whereas Rods IE-017 and IE-018 had irradiated fuel stack lengths of 0.89 m. To make all four fuel stack lengths equivalent, fresh fuel pellets were added to Rods IE-015 and IE-016. Although the irradiated fuel stacks for Rods IE-015 and IE-016 were 690-mm long, handling of the fuel rods caused the irradiated fuel stack lengths to increase about 5 mm. A gamma scan of these fuel rods indicated that the length increase was a result of the irradiated fuel pellets in the upper end of the fuel stack becoming less tightly packed and not a result of pellet-to-pellet gaps in the fuel stack. Eleven fresh fuel pellets, 15 mm in length, were added to each of the short fuel stacks to bring the total fuel stack length in Rods IE-015 and IE-016 to 889 mm.

Table B-I shows the preirradiated fuel pellet and cladding data (nominal design values) for these fuel rods. Table I, Section 2 provides a summary of additional fuel rod data peculiar to each fuel rod.

TABLE B-I

TEST IE-3 PRETEST FUEL ROD AND CLADDING DATA

Finished Cladding Data:[a]		
Mean Outside Diameter	-	9.93 mm
Mean Inside Diameter	-	8.75 mm
Overall Length	-	969.0 mm
Mechanical Properties of Fuel Rod Cladding:[b]		
Yield Tensile Strength		
At Room Temperature	-	572 MPa
At 658 K	-	389 MPa
Ultimate Tensile Strength		
At Room Temperature	-	773 MPa
At 658 K	-	490 MPa
Fuel Data:[a]		
Enrichment	-	12.5% ²³⁵ UO ₂
Mean Geometric Density	-	94 % TD
Burnup	IE-015	11,060 MWd/tU
	IE-016	8,550 MWd/tU
	IE-017	14,780 MWd/tU
	IE-018	15,860 MWd/tU
Mean Pellet Diameter[c]	IE-015	- 8.48 mm
	IE-016	- 8.48 mm
	IE-017	- 8.53 mm
	IE-018	- 8.53 mm
Mean Pellet Length[c]		- 15.24 mm
Stack Length		- 889.0 mm
Dish Dimensions		
Dish Chord	-	6.60 mm
Dish Depth	-	0.343 mm
Grain Size:[b]		
ASTM Micrograin Size	-	No. 13.2
Grain Diameter	-	3.7 x 10 ⁻³ mm
Plenum Length	-	50.8 mm

[a] All of these data are nominal dimensions for the preirradiated fuel rod except the overall length, which was measured.

[b] Data supplied by Westinghouse Electric Corporation.

[c] This only applies to the original fuel load, not to the unirradiated fuel pellets added to Rods IE-015 and IE-016.

Table B-II provides the characterization data for the fresh fuel pellets added to Rods IE-015 and IE-016.

Figure B-1 illustrates the cladding outside diameter dimensions following irradiation in the Saxton reactor and prior to testing in PBF obtained using the Pulsed Eddy Current Defect Inspection System equipped with LVDTs to measure rod diameter and rod bow^[B-2]. Cladding measurements were made at two azimuthal orientations, 90 degrees apart. These measurements are reported at 25.4-mm intervals over the length of the fuel rod. During irradiation in the Saxton reactor, Rods IE-015 and IE-016 had initial internal pressures of 13.2 and 9.0 MPa, respectively, whereas Rods IE-017 and IE-018 each had an initial pressure of 0.1 MPa. The effect of the high internal pressure on reducing cladding creep down is exhibited by Rods IE-015 and IE-016 in Figure B-1. The Saxton reactor operated with a coolant pressure of about 14 MPa. The peak fuel rod powers in the Saxton reactor were 33, 33, 49, and 49 kW/m for Rods IE-015, IE-016, IE-017 and IE-018, respectively.

TABLE B-II

TEST IE-3 FUEL PELLET CHARACTERIZATION DATA FOR FRESH FUEL PELLETS USED IN RODS IE-015 AND IE-016

Pellet No.	Diameter (cm)				Length (cm)			Weight (Grams)	Geometric Density (g/cm ³)	Immersion Density (g/cm ³)	End Face Per- pendicularity (cm)	
	Top	Center	Bottom	Average	0°	90°	Average				Top	Bottom
ROD IE-015												
1	0.8529	0.8532	0.8532	0.8531	1.5237	1.5286	1.5262	8.9142	10.3586	-	0.0041	0.0041
2	0.8532	0.8532	0.8532	0.8532	1.5192	1.5270	1.5231	8.9335	10.4000	-	-	-
3	0.8532	0.8532	0.8532	0.8532	1.5375	1.5428	1.5401	8.9853	10.3431	-	0.0071	0.0041
4	0.8532	0.8532	0.8532	0.8532	1.5250	1.5286	1.5268	8.9570	10.4018	-	-	-
5	0.8534	0.8534	0.8534	0.8534	1.5171	1.5212	1.5192	8.9142	10.3985	10.4316	0.0058	0.0015
6	0.8532	0.8532	0.8532	0.8532	1.5194	1.5230	1.5212	8.9399	10.4206	-	-	-
7	0.8529	0.8534	0.8534	0.8533	1.5227	1.5301	1.5364	8.9514	10.3959	-	-	-
8	0.8534	0.8534	0.8534	0.8534	1.5314	1.5362	1.5338	8.9846	10.3795	10.4380	-	-
9	0.8529	0.8529	0.8529	0.8529	1.5296	1.5367	1.5331	8.9974	10.4112	-	0.0036	0.0081
10	0.8529	0.8529	0.8529	0.8529	1.5423	1.5466	1.5444	9.0671	10.4140	-	-	-
11	0.8532	0.8532	0.8532	0.8532	1.5357	1.5323	1.5390	9.0134	10.3833	10.4144	0.0053	0.0018
ROD IE-016												
1	0.8529	0.8529	0.8529	0.8529	1.5270	1.5303	1.5287	8.9482	10.3848	-	0.0043	0.0028
2	0.8532	0.8532	0.8532	0.8532	1.5204	1.5253	1.5229	8.9368	10.4056	-	-	-
3	0.8534	0.8534	0.8534	0.8534	1.5390	1.5461	1.5425	9.0345	10.3770	-	0.0041	0.0066
4	0.8529	0.8529	0.8529	0.8529	1.5184	1.5217	1.5201	8.8771	10.3616	-	-	-
5	0.8529	0.8532	0.8532	0.8531	1.5268	1.5339	1.5303	8.9053	17.8109	10.3676	0.0036	0.0013
6	0.8532	0.8532	0.8532	0.8532	1.5270	1.5316	1.5293	8.9596	10.3873	-	-	-
7	0.8532	0.8532	0.8532	0.8532	1.5243	1.5288	1.5265	8.9340	10.3769	-	0.0015	0.0036
8	0.8532	0.8532	0.8532	0.8532	1.5380	1.5446	1.5413	8.9939	10.3453	-	-	-
9	0.8532	0.8532	0.8532	0.8532	1.5303	1.5352	1.5328	8.9928	10.4022	-	0.0018	0.0063
10	0.8529	0.8532	0.8532	0.8531	1.5390	1.5423	1.5406	9.0393	10.4039	10.4105	-	-
11	0.8532	0.8532	0.8532	0.8532	1.5387	1.5418	1.5403	9.0179	10.3798	-	0.0043	0.0033

Fig. B-1 Pretest mean cladding outside diameter measurement for all four rods in Test IE-3.

REFERENCES

- B-1. G. W. Gibson et al., *Characteristics of UO_2 Zircaloy Fuel Rod Materials from the Saxton Reactor for Use in the Power Burst Facility*, ANCR-NUREG-1321 (September 1976).
- B-2. M. E. Yancey et al., *Inspection System for Zircaloy Clad Fuel Rods*, ANCR-1259 (February 1976).

APPENDIX C

DATA REDUCTION AND EVALUATION

All data for Test IE-3 were recorded on the PBF Surveillance System^[a] in an analog format and then were digitized by the PBF PDP-15 data reduction facility. The resulting digital data were processed on an IBM/360, using the MAC/RAN Time Series Analysis Program^[C-2], and permanently stored on seven-track magnetic tapes in the MAC/RAN SIDU (Standard Input Data Unit) format.

During the processing on the IBM/360, several data reduction steps were completed as follows:

- (1) Data were converted from data system volts to engineering units (SI) using the calibration equations presented in Appendix A,
- (2) Any wild points were removed,
- (3) Zero power offsets due to data system drift were removed from coolant ΔT data channels,
- (4) SPND data were converted from detector nanoamps to neutron flux, using a comparison between gamma scan data from cobalt flux wires and integrals of SPND currents^[C-3],
- (5) Rod internal pressure data were corrected for large drifts in zero due to instrument decalibration.

[a] The Surveillance System is an FM multiplexed data recording system that records channels of analog data plus a standard time code (IRIG-A) on magnetic tape^[C-1].

Reduced Data Format

Three separate sets of data are stored on tape in the form of:

- (1) Data system volts
- (2) Engineering units (SI) obtained by applying calibration equations tabulated in Appendix A
- (3) A set reduced from (2) with calculated parameters added for additional calculations and plotting.

Each data channel of the first two sets contains approximately 28,000 data values with variable digitizing intervals ranging from 10 seconds down to 0.25^[a] seconds. Table C-I shows the digitizing intervals used.

TABLE C-I

DIGITIZING INTERVALS USED FOR TEST IE-3 DATA REDUCTION

<u>Time^[a] Period (sec)</u>	<u>Test Phase</u>	<u>Digitizing Interval (sec)</u>
0-138180	Preconditioning	10.0
139140-139980	Power Ramp	2.0
139980-143880	Steady-State Operation at 69 kW/m	10.0
143910-144780	Flow Reduction	0.25
144790-146460	After Shutdown	10.0

[a] The time corresponds to time on Figure 9 where time of zero designates the beginning of the first power increase.

[a] As the data were digitized, they were filtered with a low pass filter whose cutoff frequency corresponds to the Nyquist frequency ($1/2 \Delta T$).

The third data set was numerically filtered and reduced by a factor of 14 to give approximately 2000 points per channel. Several calculated parameters were added to the original data channels including:

- (1) Results from SPND data: (a) axial local power, (b) average neutron flux, (c) peak elevation, and (d) peak to average (see Appendix D).
- (2) Average rod power for each rod, calculated from thermal-hydraulic data (Appendix D).
- (3) Coolant mass flux for each rod, calculated using ASTEM^[C-4] water properties and inlet temperature.

In the SIDU format, each data channel has a unique number and name of eight characters. All of the data channels available are listed in Table C-II. The data channels are also shown in Figures C-1 through C-11.

Data Reduction

Corrections were applied to some of the channels to eliminate obvious errors in the data. Each of the corrections are discussed below.

The first correction was wild point removal. During the digitizing process, particularly at the beginning or end of an analog tape wild points were sometimes introduced into the data. Most of these points were removed using the MAC/RAN processor following conversion to engineering units.

The second step was elimination of zero power offsets in coolant ΔT data. These transducers had a very small output. Therefore, the signals were recorded through high gain amplifiers. For this reason,

TABLE C-II

DATA CHANNEL FORMAT FOR TEST IE-3

Comments		Transducer Name	Rod Number	Elevation (m)	Serial Number	Azimuthal (degrees)	Data System Channel Number	Reduced Tape No.	Channel ID	Units
Corrected for zero shift	--	Fuel Rod PxD	IE-015	--	440	--	2-3	4	RODPRS11	MPa
		Coolant ΔT	IE-015	--	452	--	5-3	15	DELTMP11	K
		Flowmeter	IE-015	--	10104	--	6-3	19	FLORAT11	cm ³ /s
Failed at beginning of test.	--	LVDT	IE-015	--	427	--	8-1	25	CLDDSP11	mm
		Outlet TC	IE-015	--	--	--	14-2	46	OUTEMP11	K
Corrected for zero shift	--	Fuel Rod PxD	IE-016	--	441	--	2-4	5	RODPRS21	MPa
		Coolant ΔT	IE-016	--	454	--	5-4	16	DELTMP21	K
		Flowmeter	IE-016	--	10101	--	6-4	20	FLORAT21	cm ³ /s
		LVDT	IE-016	--	429	--	8-2	26	CLDDSP21	mm
		Outlet TC	IE-016	--	--	--	9-3	31	OUTEMP21	K

TABLE C-II (Continued)

DATA CHANNEL FORMAT FOR TEST IE-3

<u>Comments</u>	<u>Transducer Name</u>	<u>Rod Number</u>	<u>Elevation (m)</u>	<u>Serial Number</u>	<u>Azimuthal (degrees)</u>	<u>Data System Channel Number</u>	<u>Reduced Tape No.</u>	<u>Channel ID</u>	<u>Units</u>
Corrected for zero shift	-- Fuel Rod Px̄	IE-017	--	436	--	3-1	6	RODPRS31	MPa
	Cladding TC (Spring)	IE-017	0.51	--	0	3-3	8	CLDTMP31	K
	Cladding TC (Spring)	IE-017	0.61	--	0	3-4	9	CLDTMP32	K
	Cladding TC (Spring)	IE-017	0.71	--	0	4-1	10	CLDTMP33	K
Corrected for zero shift	-- Coolant ΔT	IE-017	--	453	--	6-1	17	DELTMP31	K
	Flowmeter	IE-017	--	10102	--	7-1	21	FLORAT31	cm ³ /s
	LVDT	IE-017	--	428	--	8-3	27	CLDDSP31	mm
	Outlet TC	IE-017	--	--	--	14-3	47	OUTEMP31	K

TABLE C-II (Continued)

DATA CHANNEL FORMAT FOR TEST IE-3

Comments	Transducer Name	Rod Number	Elevation (m)	Serial Number	Azimuthal (degrees)	Data System Channel Number	Reduced Tape No.	Channel ID	Units
Corrected for zero shift	-- Fuel Rod PxD	IE-018	--	438	--	3-2	7	RODPRS41	MPa
	Cladding TC (Type K)	IE-018	0.61	--	0	4-3	11	CLDTMP41	K
	Cladding TC (Type K)	IE-018	0.51	--	90	4-4	12	CLDTMP42	K
	Cladding TC (Type K)	IE-018	0.61	--	180	5-1	13	CLDTMP43	K
	Cladding TC (Type K)	IE-018	0.71	--	270	5-2	14	CLDTMP44	K
Corrected for zero shift	-- Coolant ΔT	IE-018	--	455	--	6-2	18	DELTMP41	K
	Flowmeter	IE-018	--	10103	--	7-2	22	FLORAT41	cm ³ /s
	LDVT	IE-018	--	431	--	8-4	28	CLDDSP41	mm
	Outlet TC	IE-018	--	--	--	9-4	32	OUTEMP41	K

TABLE C-II (Continued)

DATA CHANNEL FORMAT FOR TEST IE-3

Comments	Transducer Name	Rod Number	Elevation (m)	Serial Number	Azimuthal (degrees)	Data System Channel Number	Reduced Tape No.	Channel ID	Units
	SPND 1	--	0.15	55209	45	10-1	33	NTFLUX01	n/cm ² ·s
	SPND 2	--	0.31	55206	45	10-2	34	NTFLUX02	n/cm ² ·s
	SPND 3	--	0.46	55208	45	10-3	35	NTFLUX03	n/cm ² ·s
	SPND 4	--	0.65	55204	45	10-4	36	NTFLUX04	n/cm ² ·s
	SPND 5	--	0.78	56082	45	11-1	37	NTFLUX05	n/cm ² ·s
	SPND 6	--	0.15	53964	225	11-2	38	NTFLUX06	n/cm ² ·s
	SPND 7	--	0.31	53979	225	11-3	39	NTFLUX07	n/cm ² ·s
	SPND 8	--	0.46	53968	225	11-4	40	NTFLUX08	n/cm ² ·s
	SPND 9	--	0.62	52771	225	13-1	41	NTFLUX09	n/cm ² ·s
	SPND 10	--	0.78	52769	225	13-2	42	NTFLUX10	n/cm ² ·s
	Loop Flowmeter	--	--	--	--	14-1	45	LOPFL001	cm ³ /s
Range change--	SLP-1	--	--	--	--	1-4	49	RECPOW01	MW
not removed	Fission Break Monitor	--	--	--	--	2-2	3	FISREL01	Log ₁₀ (R/hr)
	PPS #1	--	--	--	--	1-3	48	RECPOW02	MW
	PPS #2	--	--	--	--	2-1	2	RECPOW03	MW
Decalibrated--Coolant Pressure (10K)	--	--	--	--	--	7-3	23	SYSPRS01	MPa
Decalibrated--Coolant Pressure (10K)	--	--	--	--	--	7-4	24	SYSPRS02	MPa
	Inlet TC	--	--	--	--	9-1	29	INTEMP01	K
	Inlet TC	--	--	--	--	9-2	30	INTEMP02	K
	IPT Differential Pressure	--	--	--	--	13-1	43	IPT DP01	volts
	Loop Pressure	--	--	--	--	13-4	44	LOPPRS01	volts

TABLE C-II (Continued)
DATA CHANNEL FORMAT FOR IE-3

<u>Comments</u>	<u>Transducer Name</u>	<u>Rod Number</u>	<u>Elevation (m)</u>	<u>Serial Number</u>	<u>Azimuthal (degrees)</u>	<u>Data System Channel Number</u>	<u>Reduced Tape No.</u>	<u>Channel ID</u>	<u>Units</u>
(Calculated data, available on decimated data set only)									
0-1.47 x 10 ⁵	(Time)	--	--	--	--	12- thru 12-4	1	TIME	s
(Peak =	(Average Rcd Power)	IE-015	--	--	--	--	50	RODPWR01	kW/m
1.307 x	(Average Rcd Power)	IE-016	--	--	--	--	51	RODPWR02	kW/m
Average)	(Average Rod Power)	IE-017	--	--	--	--	52	RODPWR03	kW/m
	(Average Rcd Power)	IE-018	--	--	--	--	53	RODPWR04	kW/m
Average of --	(Average Rod Power)	--	--	--	--	--	54	RODPWRAV	kW/m
all four rods									
Average of --	(Average Rod Power)	--	--	--	--	--	60	RODPWR3V	kW/m
three Rods									
	(Standard deviation of rod power)	--	--	--	--	--	55	RODPWRSD	kW/m
	(Mass flux)	IE-015	--	--	--	--	56	MASFLX01	kg/s·m ²
	(Mass flux)	IE-016	--	--	--	--	57	MASFLX02	kg/s·m ²
	(Mass flux)	IE-017	--	--	--	--	58	MASFLX03	kg/s·m ²
	(Mass flux)	IE-018	--	--	--	--	59	MASFLX04	kg/s·m ²
	(Local Peaking Factor)	--	0.013	--	45	--	61	AXFLUX01	--
	(Local Peaking Factor)	--	0.159	--	45	--	62	AXFLUX02	--
	(Local Peaking Factor)	--	0.306	--	45	--	63	AXFLUX03	--
	(Local Peaking Factor)	--	0.452	--	45	--	64	AXFLUX04	--
	(Local Peaking Factor)	--	0.521	--	45	--	65	AXFLUX05	--
	(Local Peaking Factor)	--	0.599	--	45	--	66	AXFLUX06	--

TABLE C-II (Continued)

DATA CHANNEL FORMAT FOR IE-3

<u>Comments</u>	<u>Transducer Name</u>	<u>Rod Number</u>	<u>Elevation (m)</u>	<u>Serial Number</u>	<u>Azimuthal (degrees)</u>	<u>Data System Channel Number</u>	<u>Reduced Tape No.</u>	<u>Channel ID</u>	<u>Units</u>
(Local Peaking Factor)		--	0.662	--	45	--	67	AXFLUX07	--
(Local Peaking Factor)		--	0.724	--	45	--	68	AXFLUX08	--
(Local Peaking Factor)		--	0.745	--	45	--	69	AXFLUX09	--
(Local Peaking Factor)		--	0.892	--	45	--	70	AXFLUX10	--
(Average Neutron Flux)		--	--	--	45	--	71	FLUXAVER	n/cm ² ·s
(Elevation of Peak)		--	--	--	45	--	72	FLXMXELV	m
(Peak to Average)		--	--	--	45	--	73	FLUXMAX	--

Fig. C-1 Fuel rod internal pressure for all four rods in Test IE-3.

Fig. C-2 Cladding elongation for Rods IE-016, IE-017, and IE-018, Test IE-3.

Fig. C-3 Coolant flow rate for all four rods in Test IE-3.

Fig. C-4 Coolant temperature increase for all four rods and inlet and outlet temperature in Test IE-3.

Fig. C-5 Neutron flux at several axial locations for Test IE-3.

Fig. C-6 Reactor power and loop fission activity for Test IE-3.

Fig. C-7 Cladding surface temperature for Rods IE-017 and IE-018, Test IE-3.

Fig. C-8 Corrected and uncorrected fuel rod internal pressure for Rod IE-015, Test IE-3.

Fig. C-9 Corrected and uncorrected fuel rod internal pressure for Rod IE-016, Test IE-3.

Fig. C-10 Corrected and uncorrected fuel rod internal pressure for Rod IE-017, Test IE-3.

Fig. C-11 Corrected and uncorrected fuel rod internal pressure for Rod IE-018, Test IE-3.

these transducers were subject to small electronic zero drifts. This zero shift was corrected by adding an offset to the data so that the temperature rise was zero at zero reactor power. The constants that were applied are shown in Table C-III.

TABLE C-III
ZERO POWER OFFSET CORRECTIONS APPLIED TO
DIFFERENTIAL THERMOCOUPLE CHANNELS IN TEST IE-3

<u>Rod</u>	<u>Correction [a]</u>
IE-017	-0.08 K
IE-018	-0.5 K

[a] This correction was added to the data so that the average instrument signal at zero power was zero.

The SPND data were then converted from detector nanoamps to neutron flux. The conversion factors were determined from a comparison with three cobalt flux wires mounted in the IE-3 hardware. The fluence was determined from a gamma scan of the cobalt wires. Average neutron flux was calculated by dividing the fluence by the time at power. A conversion between individual SPND output and neutron flux was found by dividing the neutron flux at each SPND elevation by a corresponding average SPND current. The conversion factors are tabulated in Table C-IV.

Rod internal pressure data were also corrected for zero shifts due to decalibration of the pressure transducers. The correction was determined using the results of a multiple regression analysis of the data. The model chosen for the regression was determined as follows.

The pressure of the fill gas should be described fairly well using the perfect gas equation.

$$P = \rho RT \quad (C-1)$$

where

P = gas pressure (MPa)

ρ = gas density

R = gas constant

T = temperature (absolute scale).

The measured pressure (P_m) was assumed to be

$$P_m = P + bt + b_1 t^2 + C \quad (C-2a)$$

or

$$P_m = \rho RT + bt + b_1 t^2 + C \quad (C-2b)$$

TABLE C-IV

TEST IE-3 CONVERSION FACTORS RELATING SPND CURRENT AND NEUTRON FLUX

<u>Instrument</u>	<u>Conversion Factor^[a] 10^{20} n/cm²-s per ampere</u>
SPND 1	3.473
SPND 2	3.712
SPND 3	3.862
SPND 4	4.002
SPND 5	4.194
SPND 6	(Failed)
SPND 7	3.457
SPND 8	3.712
SPND 9	3.905
SPND 10	3.996

[a] The SPND data were multiplied by this constant to produce neutron flux.

where

- b and b_1 = quadratic time drift coefficients
- t = time from start of test (seconds)
- C = zero shift from the time of pressurization of the rod to the start of the test.

Since the average temperature of the gas cannot be measured, it has been assumed to be a quadratic function of average fuel rod power as

$$T = T_{in} + d\phi + d_1\phi^2 \quad (C-3)$$

where

- ϕ = average fuel rod power (kW/m)
- d and d_1 = quadratic power coefficients
- T_{in} = inlet temperature (K).

When equation (C-3) is substituted into (C-2b) and a general set of coefficients applied, measured pressure is

$$P_m = a_0 + a_1\phi + a_2\phi^2 + a_3T_{in} + a_4t + a_5t^2 \quad (C-4)$$

where

- $a_0 = C$
- $a_1 = \rho R d$
- $a_2 = \rho R d_1$
- $a_3 = \rho R$
- $a_4 = b$
- $a_5 = b_1$

The first coefficient, a_0 , was determined by using the cold (300 K) fill pressure adjusted to hot (606 K) conditions at the start of the test, i.e.

$$a_0 = P_m - P = P_m - P_{cold} \left(\frac{606}{300} \right) \quad (C-5)$$

TABLE C-V

TEST IE-3 MULTIPLE REGRESSION ANALYSIS RESULTS FOR FUEL ROD INTERNAL PRESSURE

Rod	Regression Equation Coefficients ^[a]					
	a_0	a_1	a_2	$a_3^{[b]}$	a_4	a_5
IE-015	-0.26	0.132	-0.961×10^{-3}	-0.146×10^{-1}	0.788×10^{-4}	-0.203×10^{-9}
IE-016	-0.18	0.140	-1.394×10^{-3}	0.480×10^{-2}	0.291×10^{-5}	-0.544×10^{-11}
IE-017	+0.46	0.088	-0.647×10^{-3}	0.160×10^{-2}	0.187×10^{-4}	-0.448×10^{-10}
IE-018	-0.02	0.101	-0.128×10^{-2}	0.137×10^{-1}	-0.369×10^{-4}	0.133×10^{-9}

[a] The regression analysis model was $P_m = a_1 \phi + a_2 \phi + a_3 T_{in} + a_4 t + a_5 t^2$ where a_0 was determined using cold fill pressure.

[b] The regression included very little data at different temperatures. Therefore the temperature coefficient, a_3 (Equation C-4), has little meaning.

A multiple regression analysis was then used to determine the remainder of the coefficients. The analysis was made using data from the first power ramp and the last two cycles of the preconditioning phase. This was done to avoid highly transient data such as the gap conductance segment and the power ramp. The results of this analysis are given in Table C-V.

The corrected pressure (P) was then calculated for the entire test using

$$P = P_m - a_0 - a_4 t - a_5 t^2. \quad (C-6)$$

The corrected pressure is presented in the text. A comparison between the corrected pressures and uncorrected pressures are shown in Figures C-8 through C-11.

Cladding elongation was also corrected to account for slight differences in mounting the LVDTs in the test hardware. When these devices are calibrated, displacement is defined as zero when the core is centered in the device. When the LVDTs are mounted in the hardware, the core is not necessarily centered. Therefore, to define cladding elongation in terms of the actual length of the cladding, the elongation must be set to zero at this point. This could have been handled in the equation to convert data system volts to engineering units by a suitable transformation. However, it was more convenient to apply the zero correction as a separate step. The elongation zero shifts applied to the data are given in Table C-VI.

TABLE C-VI
CORRECTIONS APPLIED TO CLADDING ELONGATION DATA IN TEST IE-3

<u>Rod</u>	<u>Correction^[a] (mm)</u>
IE-015	+ 0.79
IE-016	- 1.02
IE-017	- 1.03
IE-018	- 1.94

[a] This correction was added to the data so that the elongation was zero at cold (300 K) conditions.

REFERENCES

- C-1. W. J. Quapp et al., *Irradiation Effects Test Series Scoping Test 1 Test Results Report*, TREE-NUREG-1066 (September 1977).
- C-2. R. K. Otnes (ed.), *Reference Manual: MAC/RAN III-Time Series Data Analysis System*, Agababian Associates (1973).
- C-3. C. M. Allison, *Irradiation Effects Test Series SPND Calibration Test 3 Test Results Report*, TFBP-TR-142 (October 1976).
- C-4. 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

POWER CALIBRATION RESULTS

Fuel rod power was calculated using a combination of several techniques. For all of the test except for the flow reduction phase, fuel rod average power was calculated from a thermal balance. During the flow reduction phase, average power was determined from SPND 9 current using a linear regression equation relating the two variables. Local fuel rod powers were determined from both cobalt flux wires and SPNDs.

Fuel Rod Average Power

Each of the four fuel rods were instrumented for a thermal balance. Each rod was surrounded by an individual flow shroud which was instrumented with a differential thermocouple pair and a flowmeter. The test assembly was also instrumented for coolant inlet temperature and pressure. All of these instruments were available throughout the test except for the system pressure transducers. No useable pressure data were recorded. A constant pressure of 14.8 MPa was assumed based on PBF plant instrumentation.

Fuel rod average power was calculated from the thermal balance using a computer program which incorporates the equations presented in Reference D-1 and water properties determined from the ASTEM subroutines^[D-2]. The computer program used the decimated data (see Appendix C) for the preceding instruments and generated individual fuel rod average power, average (of four rods) fuel rod average power, and individual flow shroud coolant mass fluxes.

During the flow reduction phase of the test, saturated conditions existed at the outlets of the flow shrouds which prevented valid thermal balances. A set of linear regression equations relating SPND 9 current

TABLE D-I
 LINEAR REGRESSION EQUATIONS RELATING
 NEUTRON FLUX MEASURED BY SPND 9 AND FUEL ROD AVERAGE POWER IN TEST IE-3

<u>Rod Number</u>	<u>Test^[a] Phase</u>	<u>Regression^[b] Equation</u>	<u>95% Confidence Interval (kW/m)</u>
IE-015	Preconditioning Period	$y = 8.588 x - 0.18$	± 0.44
	Power Ramp, Steady-State Operation and Flow Reduction	$y = 8.406 x + 0.36$	± 0.77
IE-016	Preconditioning Period	$y = 8.197 x - 0.47$	± 0.57
	Power Ramp, Steady-State Operation and Flow Reduction	$y = 7.622 x + 0.79$	± 0.79
IE-017	Preconditioning Period	$y = 8.730 x - 0.02$	± 0.52
	Power Ramp, Steady-State Operation and Flow Reduction	$y = 8.367 x + 0.44$	± 0.81
IE-018	Preconditioning Period	$y = 6.938 x + 1.97$	± 1.21
	Power Ramp, Steady-State Operation and Flow Reduction	$y = 7.214 x + 0.58$	± 1.33
Average	Preconditioning Period	$y = 8.113 x + 0.33$	± 0.69
	Power Ramp, Steady-State Operation and Flow Reduction	$y = 7.902 x + 0.54$	± 0.93

[a] The gap conductance data was not included in the "Preconditioning Phase" regression.

[b] y = Average rod power (kW/m)

x = Neutron flux $n/cm^2 \cdot s$ measured by SPND 9 divided by 10^{14} .

and individual fuel rod average powers were generated using the data from the previous parts of the test. These equations, given in Table D-I, were used to calculate individual fuel rod average powers during the flow reduction.

Since individual fuel rod power data was available throughout the test, a comparison of the power in each rod was made. This was done to check for any unexpected variations in power due to instrument drift or asymmetry in the neutron flux due to control rod changes. There was an apparent problem with power in Rod IE-018. The coolant differential thermocouple pair on this rod showed a higher drift rate than did the other three pairs. The drift was evidently both a change in the transducer zero setting and sensitivity. However, the uncertainty in power for Rod IE-018 was effectively the same as the other rods because the magnitude of the drift was within the uncertainty limits for the thermocouple pair. After accounting for the drift in power on Rod IE-018, the ratios of individual rod power to average rod power (average of all four rods), shown in Table D-II, remained constant within the listed uncertainties.

TABLE D-II

RATIOS OF INDIVIDUAL FUEL ROD POWER TO FUEL ROD AVERAGE POWER IN TEST IE-3

<u>Power Ratios</u>	<u>Mean</u>	<u>95% Confidence Interval</u>
Rod IE-015/Average	1.027	± 0.035
Rod IE-016/Average	0.966	± 0.041
Rod IE-017/Average	1.056	± 0.019
Rod IE-018/Average	0.950	± 0.069

The uncertainties in fuel rod power for Rods IE-015, IE-016, and IE-017 were calculated using both instrument calibration errors and data acquisition errors. These results are shown in Table D-III.

TABLE D-III

UNCERTAINTIES IN FUEL POWER^[a] FOR FOR RODS IE-015, IE-016, AND IE-017

<u>Average Rod Power (kW/m)</u>	<u>95% Confidence Level (kW/m)</u>
0.0	<u>± 1.47</u>
29.2	<u>± 1.53</u>
60.7	<u>± 1.68</u>

[a] At system conditions of: pressure, 14.8 MPa; inlet temperature, 600 K; and flow rate, 1000 cm³/s. The tabulated results for rod power and total error are for the average coolant differential temperature and total error presented in Appendix A. The variation of the error between the rods was insignificant.

Local Power

A local power profile (power as a function of the elevation above the bottom of the rod) was obtained using the data from SPNDs and cobalt flux wires. The SPND results were used to determine the instantaneous peaking factors which were then multiplied by fuel rod average power to obtain an instantaneous local power. The instantaneous peaking factors were calculated by fitting a sine function to the output from each column of SPNDs^[a]. The sine function was then divided by the integral average of the fit over the length of the fuel to normalize the profile (see Reference D-1 for a discussion of the techniques).

Analysis of the two instantaneous profiles as a function of power indicated that any appreciable skewing in local power, due to change in control rod position, was limited to a fuel rod power less than 10 kW/m. Above that level, any change in local power was minor. In going from a fuel rod power of less than 30 kW/m up to 68 kW/m, the peak to average

[a] SPND 6 was failed prior to the beginning of the test so that its output was not included in the fit.

changed less than 0.5%. Even this change was limited to the lower half of the rod, so the normalized local power on the upper half of the rod remained fixed.

A comparison of the local power profiles from the two columns of SPNDs and the four^[a] cobalt wires extending the length of the fuel rods indicated that the azimuthal flux profile did not show any measureable asymmetry or tilting with change in power. Each cobalt wire mounted on the flow shroud of Rods IE-016, IE-017, and IE-018, at the same radial distance from the centerline of the test assembly as the centerline of the fuel rods, gave nearly identical average neutron fluences (the deviation was less than 1.3%, well within the uncertainties of the measured fluences).

Two cobalt wires were mounted on the flow shroud of Rod IE-016 at 90° and 180° orientations. The measured results from these wires indicate the slope in the radial neutron flux profile was 0.38% per millimeter at the radius of the fuel rods. This would correspond to a variation of 3.3% across the fuel. The local gradients within the fuel would be slightly different because of the flux depression within the fuel.

Normalized local power profiles versus axial elevation from both SPNDs and average of the four wires are shown in Figure D-1 and tabulated in Table D-IV. It can be seen that the uncertainties in local power for the 225-degree SPND column were significantly larger than the uncertainties of the 45-degree SPND column. This was due to the smaller number of SPNDs available and a greater uncertainty in the signals themselves.

[a] The one cobalt wire on the flow shroud of Rod IE-015 had become misaligned during handling prior to the test. The profile from the wire was not used.

Fig. D-1 Normalized local power profile for fuel rod average powers above 10 kW/m versus axial elevation of self-powered neutron detectors and an average of cobalt flux wires for Test IE-3.

TABLE D-IV

LOCAL POWER PROFILE WITH UNCERTAINTIES

Elevation	Instantaneous Local Power From SPNDs				Integrated Profile	
	Column at 45° ^[a]		Column at 225° ^[b]		From an Average of Flux Wires ^[c]	
	Normalized Local Power	95% Confidence Intervals	Normalized Local Power	95% Confidence Intervals	Normalized Local Power	95% Confidence Intervals
0.013	0.376	± 0.156	0.171	± 0.495	-----	-----
0.159	0.897	± 0.078	0.761	± 0.257	.8630	$\pm .0753$
0.306	1.238	± 0.065	1.300	± 0.114	1.2100	$\pm .0578$
0.452	1.319	± 0.064	1.420	± 0.173	1.3072	$\pm .0387$
0.521	1.298	± 0.075	1.344	± 0.149	1.2697	$\pm .0541$
0.599	1.217	± 0.081	1.205	± 0.124	1.1574	$\pm .0735$
0.662	1.097	± 0.075	1.075	± 0.114	1.0684	$\pm .0686$
0.724	0.932	± 0.055	0.941	± 0.113	0.9451	$\pm .1185$
0.745	0.867	± 0.056	0.893	± 0.113	0.9053	$\pm .1292$
0.892	0.388	± 0.090	0.511	± 0.390	0.5147	-----
<p>[a] Average Peak Elevation: 0.432 ± 0.050 Average Peak to Average: 1.323 ± 0.064</p> <p>[b] Average Peak Elevation: 0.411 ± 0.024 Average Peak to Average: 1.428 ± 0.175</p> <p>[c] Average of three wires Average Peak Elevation: 0.405 ± 0.0653 Average Peak to Average: 1.321 ± 0.0364</p>						

REFERENCES

- D-1. W. J. Quapp, et al., *Irradiation Effects Test Series Scoping Test 1 Test Results Report*, TREE-NUREG-1066 (September 1977).
- D-2. K. V. Moore, *ASTEM - A Collection of FORTRAN Subroutines to Evaluate the 1967 ASME Equations of State for Water/Steam and Derivations of these Equations*, ANCR-1026 (October 1971).

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