320
5-10-77

Doc # 1046

Doc # 1046
TRFEE-NUREG-1046
for U.S. Nuclear Regulatory Commission

IRRADIATION EFFECTS TEST SERIES
TEST IE-1
TEST RESULTS REPORT

W. J. QUAPP
C. M. ALLISON

L. C. FARRAR
A. S. MEHNER

March 1977

MASTER

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161
Price: Printed Copy \$6.75; Microfiche \$3.00

"The NRC will make available data tapes and operational computer codes on research programs dealing with postulated loss-of-coolant accidents in light water reactors. Persons requesting this information must reimburse the NRC contractors for their expenses in preparing copies of the data tapes and the operational computer codes. Requests should be submitted to the Research Applications Branch, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, D.C. 20555."

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the Energy Research and Development Administration, nor the Nuclear Regulatory Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

TREE-NUREG-1046
IRRADIATION EFFECTS TEST SERIES
TEST IE-1
TEST RESULTS REPORT

Approved:

W. J. Quapp, Manager
Experiments Specification and Analysis Branch

J. G. Crocker, Manager
Thermal Fuels Behavior Division

MASTER

TREE-NUREG-1046

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

IRRADIATION EFFECTS TEST SERIES
TEST IE-1
TEST RESULTS REPORT

By:

W. J. Quapp
L. C. Farrar
C. M. Allison
A. S. Mehner

EG&G IDAHO, INC.

Date Published - March 1977

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

ABSTRACT

This report describes the results of the first programmatic test in the Nuclear Regulatory Commission Irradiation Effects Test Series. This test (IE-1) used four, 0.97 m long PWR-type fuel rods fabricated from previously irradiated Saxton fuel. The objectives of this test were to evaluate the effect of fuel pellet density on pellet-cladding interaction during a power ramp and to evaluate the influence of the irradiated state of the fuel and cladding on rod behavior during film boiling operation.

The fuel rods, designated IE-007, IE-008, IE-009 and IE-010, were subjected to a preconditioning period, followed by a power increase to 70 kW/m peak at a coolant mass flux of $4800 \text{ kg/s}\cdot\text{m}^2$. After a flow reduction to $2520 \text{ kg/s}\cdot\text{m}^2$, film boiling occurred on Rod IE-007. Additional flow reductions to $1740 \text{ kg/s}\cdot\text{m}^2$ caused film boiling on the three remaining fuel rods. The maximum time in film boiling was 90 seconds for Rod IE-007. Rod IE-007 failed approximately three minutes after shutdown as a result of extensive cladding oxidation. Rod IE-008 was found to have cracks in the region below the film boiling zone during hot cell examination. Evaluation of the test data on Rod IE-008 and the posttest examination pointed to rod failure by hydride formation in the cladding.

Data are presented on the behavior of irradiated fuel rods during steady-state operation, a power ramp, and film boiling operation. The effects of as-fabricated gap size, as-fabricated fuel density, rod power, and power ramp rate on pellet-cladding interaction is discussed. Test data are compared with FRAP-T2 computer model predictions and comments on the consequences of sustained film boiling operation on irradiated fuel rod behavior are provided.

SUMMARY

This report describes the results of the first programatic test, IE-1, in the Irradiation Effects Test Series, which is part of the Nuclear Regulatory Commission's Fuel Behavior Program to define the behavior of fuel rods during transient operating conditions. This was the first Power Burst Facility (PBF) test consisting of four previously irradiated fuel rods.

The IE-1 test was conducted in June 1976 in the PBF reactor at the Idaho National Engineering Laboratory (INEL). Four Saxton^[a] fuel rods supplied by Westinghouse Electric Corporation were used. The fuel rods were designated as IE-007, IE-008, IE-009 and IE-010. The fuel rod upper end caps were removed at the INEL and replaced with endcaps containing a pressure transducer. The rods were repressurized with an argon and helium gas mixture that had a thermal conductivity similar to fission gases.

The test consisted of about thirty hours of preconditioning at rod peak powers not exceeding 33 kW/m, which included a period where the power was oscillated to obtain gap conductance data. The preconditioning period was followed by a power ramp to an average (of four rods) rod peak power of 70 kW/m at a ramp rate of 3.2 kW/m/minute to evaluate the potential for pellet-cladding interaction (PCI) induced cladding failure. One rod, IE-008, which developed a leak early in the preconditioning period, was postulated to have developed a small leak early in the power ramp due to pellet-cladding interaction and brittle fracture of the hydrided areas of the cladding. The other three rods survived the ramp with no apparent damage.

The rods were held at high power for about one hour, followed by a flow reduction to induce film boiling. Film boiling was first detected on Rod IE-007 at a flow rate of 514 cm³/s (2520 kg/s.m²). The

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

flow reduction was continued to $355 \text{ cm}^3/\text{s}$ ($1740 \text{ kg/s}\cdot\text{m}^2$) until all rods were determined to be in film boiling. Steady-state conditions were maintained for about one minute, after which the reactor was rapidly shut down.

Approximately three minutes after shutdown, the pressure transducer on Rod IE-007 indicated a failure. This first indication was followed by indications from the loop and building fission product activity monitors.

Visual examination of the rods following the test showed extensive levels of oxidation on Rod IE-007 and lesser amounts on the others. Rod IE-008 was found to have blister-like bulges on the lower region of the rod below the film boiling zone. Measurement of the cladding diameter in the film boiling zone showed that three of the rods (IE-007, IE-008, and IE-010) had increased in diameter through this region. Cladding collapse into pellet chips and pellet-to-pellet gaps was superimposed on the diametral expansion.

An analysis of the test data showed that fuel density variations affected the rate of fuel rod deformation during the power ramps. The fuel rod deformation rate of the 94% TD fuel was measured to be about twice that for the 92% TD fuel. It was also observed that the power at which hard PCI was initiated was not affected by initial gap sizes for these fuel rods. The two diametral gap sizes were 0.27 and 0.22 mm and the burnups ranged from 6,830 to 16,640 MWd/tU. Also, no evidence of substantial fission gas release was observed during the normal, off-normal, or accident (film boiling) conditions of the test.

Approximately one minute of film boiling, which caused severe cladding deterioration, and the resultant quenching did not cause cladding failure on the irradiated fuel rods. Rod IE-008 had an apparent cladding failure, allowing coolant to enter the fuel rod early in the experiment but did not show any behavior different than the unfailed fuel rods during film boiling.

The steady-state and transient data from this experiment were compared with the FRAP-T2 fuel rod behavior computer code calculations using the as-run test and fuel rod parameters.

Although some of the posttest characteristics of these irradiated rods have not been seen before, the overall rod behavior including the failure mode was not significantly different than that previously experienced on similarly operated unirradiated fuel rods.

CONTENTS

ABSTRACT	ii
SUMMARY	iii
1. INTRODUCTION	1
2. EXPERIMENT DESIGN.	3
2.1 Test Train.	3
2.2 Flow Shroud	3
2.3 Fuel Rods	8
2.4 Instrumentation	10
2.4.1 Test Train Instrumentation	10
2.4.2 Fuel Rod Instrumentation	12
3. EXPERIMENT CONDUCT	16
3.1 Preconditioning	17
3.2 Power Increase and Steady-State Operation	25
3.3 Flow Reduction.	26
4. PRELIMINARY POSTIRRADIATION EXAMINATION RESULTS.	33
4.1 Visual and Dimensional Characterization	33
4.1.1 Rod IE-007	33
4.1.2 Rod IE-008	38
4.1.3 Rod IE-009	30
4.1.4 Rod IE-010	40
4.2 Discussion.	40
5. EXPERIMENTAL RESULTS AND COMPARISON OF CALCULATED AND MEASURED FUEL ROD BEHAVIOR	42
5.1 Analytical Model.	43
5.1.1 Fuel Rod Model	43
5.1.2 Experiment Conduct Model	44
5.2 Cladding Elongation	44
5.2.1 Rod IE-007	44
5.2.2 Rod IE-008	50
5.2.3 Rod IE-009	50
5.2.4 Rod IE-010	57

5.3	Fuel Rod Internal Pressure.	59
5.3.1	Rod IE-007	64
5.3.2	Rod IE-009	69
5.3.3	Rod IE-010	70
5.4	Cladding Surface and Plenum Gas Temperature	75
5.5	Calculated Fuel Rod Behavior During Film Boiling.	80
5.5.1	Rod IE-007	80
5.5.2	Rod IE-009	80
6.	DISCUSSION AND CONCLUSIONS	83
6.1	Experimental Data	85
6.1.1	Cladding Elongation.	85
6.1.2	Fuel Rod Internal Pressure	90
6.1.3	Failure of Rod IE-008.	92
6.2	FRAP-T2 Results	96
6.2.1	Fuel Rod Internal Pressure	96
6.2.2	Cladding Elongation.	96
7.	REFERENCES	100
	APPENDIX A - INSTRUMENTATION AND DATA ACQUISITION SYSTEM ACCURACIES.	102
	APPENDIX B - FUEL ROD CHARACTERIZATION DATA	118
	APPENDIX C - TEST DATA.	121
	APPENDIX D - POWER CALIBRATION RESULTS.	132

FIGURES

1.	IE-1 four-rod experiment assembly and instrumentation shown installed in the PBF in-pile tube.	4
2.	Typical instrumented fuel rod for Test IE-1 shown installed in a flow shroud	5
3.	Top view of IE-1 test orientation test in the PBF in-pile tube.	6
4.	Schematic orientation of IE-1 self-powered neutron detectors (SPND)	7
5.	Expanded layout of the linear variable differential transformer, coolant flow orifice plate, turbine flow-meter, and coolant flow screen assembly.	11
6.	Self-powered neutron detectors as mounted on support bar . .	13
7.	View of a section of the test train, as assembled for insertion into the IPT	15
8.	Experiment conduct - inlet temperature, coolant flow rate, rod power, and reactor power for Test IE-1	18
9.	Rod IE-007 behavioral data during the IE-1 test up to the time immediately prior to film boiling	20
10.	Rod IE-008 behavioral data during the IE-1 test up to the time immediately prior to film boiling	21
11.	Rod IE-009 behavioral data during the IE-1 test up to the time immediately prior to film boiling	22
12.	Rod IE-010 behavioral data during the IE-1 test up to the time immediately prior to film boiling	23
13.	Rod IE-007 behavioral data during the IE-1 test from a time immediately prior to film boiling to shutdown	24
14.	Rod IE-008 behavioral data during the IE-1 test from a time immediately prior to film boiling to shutdown	29
15.	Rod IE-009 behavioral data during the IE-1 test from a time immediately prior to film boiling to shutdown	30
16.	Rod IE-010 behavioral data during the test from a time immediately prior to film boiling to shutdown	31
17.	Posttest view of Rods IE-007, IE-008, IE-009, and IE-010 . .	35
18.	Posttest diameter measurements on Test IE-1 irradiated rods.	36

19. View of Rod IE-007 fracture showing crazing of cladding ID .	37
20. View of Rod IE-008 below film boiling zone cladding perforations	39
21. View of Rod IE-009 showing cladding collapse into region of apparent missing fuel	41
22. Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 1. The nominal coolant flow rate and temperature are $500 \text{ cm}^3/\text{s}$ ($2554 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.	46
23. Cladding elongation for ROD IE-007 during the 75 minute steady power period of 18 kW/m for Cycle 1	46
24. Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 2. The nominal coolant flow rate and temperature are $750 \text{ cm}^3/\text{s}$ ($3682 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.	48
25. Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 4. The nominal coolant flow rate and temperature are $935 \text{ cm}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.	48
26. Cladding elongation fuel rod average power for Rod IE-007 during Cycle 5. The nominal coolant flow rate and temperature are $935 \text{ m}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.	49
27. Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 6. The nominal coolant flow rate and temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively	49
28. Cladding elongation for Rod IE-007 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively	51
29. Cladding elongation for Rod IE-007 during the flow reduction. The nominal fuel rod average power is 53.0 kW/m	51
30. Cladding elongation versus fuel rod average power for Rod IE-008 during Cycle 6. The nominal coolant flow rate and temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively	52
31. Cladding elongation for Rod IE-008 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively	52

32.	Cladding elongation for Rod IE-008 during the flow reduction. The nominal fuel rod average power is 49.4 kW/m.	53
33.	Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 1. The nominal coolant flow rate and temperature are 510 cm ³ /s (2503 kg/s·m ²) and 604 K, respectively	53
34.	Cladding elongation for Rod IE-009 during the 75 minute steady power period of 18 kW/m for Cycle 1	54
35.	Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 2. The nominal coolant flow rate and temperature are 770 cm ³ /s (3780 kg/s·m ²) and 604 K, respectively.	54
36.	Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 4. The nominal coolant flow rate and temperature are 950 cm ³ /s (4741 kg/s·m ²) and 601 K, respectively	55
37.	Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 5. The nominal coolant flow rate and temperature are 950 cm ³ /s (4741 kg/s·m ²) and 601 K, respectively	55
38.	Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 6. The nominal coolant flow rate and temperature are 990 cm ³ /s (4860 kg/s·m ²) and 604 K, respectively	56
39.	Cladding elongation for Rod IE-009 during Cycle 6. The nominal coolant flow rate and inlet temperature are 990 cm ³ /s (4860 kg/s·m ²) and 604 K, respectively	56
40.	Cladding elongation for Rod IE-009 during the flow reduction. The fuel rod average power is 50.3 kW/m.	58
41.	Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 1. The nominal coolant flow rate and temperature are 500 cm ³ /s (2554 kg/s·m ²) and 601 K, respectively	58
42.	Cladding elongation for Rod IE-010 during the 75 minute steady power period of 18 kW/m for Cycle 1	60
43.	Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 2. The nominal coolant flow rate and temperature are 750 cm ³ /s (3682 kg/s·m ²) and 604 K, respectively	60

44.	Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 4. The nominal coolant flow rate and temperature are 935 cm ³ /s (4666 kg/s·m ²) and 601 K, respectively.	61
45.	Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 5. The nominal coolant flow rate and temperature are 935 cm ³ /s (4666 kg/s·m ²) and 601 K, respectively	61
46.	Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 6. The nominal coolant flow rate and temperature are 975 cm ³ /s (4786 kg/s·m ²) and 604 K, respectively	62
47.	Cladding elongation for Rod IE-010 during Cycle 6. The nominal coolant flow rate and inlet temperature are 975 cm ³ /s (4786 kg/s·m ²) and 604 K, respectively	62
48.	Cladding elongation for Rod IE-010 during the flow reduction. The nominal fuel rod average power is 51.7 kW/m.	63
49.	Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 1. The nominal coolant flow rate and inlet temperature are 500 cm ³ /s (2554 kg/s·m ²) and 604 K, respectively	63
50.	Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 2. The nominal coolant flow rate and inlet temperature are 750 cm ³ /s (3682 kg/s·m ²) and 604 K, respectively	65
51.	Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 3. The nominal coolant flow rate and inlet temperature are 375 cm ³ /s (2166 kg/s·m ²) and 556 K, respectively	65
52.	Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 4. The nominal coolant flow rate and inlet temperature are 935 cm ³ /s (4666 kg/s·m ²) and 601 K, respectively	66
53.	Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 5. The nominal coolant flow rate and inlet temperature are 935 cm ³ /s (4666 kg/s·m ²) and 601 K, respectively	66
54.	Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 6. The nominal coolant flow rate and inlet temperature are 975 cm ³ /s (4786 kg/s·m ²) and 604 K, respectively	67

55. Fuel rod internal pressure on Rod IE-007 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K, respectively.	67
56. Fuel rod internal pressure for Rod IE-007 during flow reduction. The nominal fuel rod average power was 53.0 kW/m.	68
57. Fuel rod internal pressure versus fuel rod average power for Rod IE-009 during Cycle 6. The nominal coolant flow rate and inlet temperature are $990 \text{ cm}^3/\text{s}$ ($4860 \text{ kg/s}\cdot\text{m}^2$) and 604 K, respectively.	68
58. Fuel rod internal pressure on Rod IE-009 during Cycle 6. The nominal coolant flow rate and inlet temperature are $990 \text{ cm}^3/\text{s}$ ($4860 \text{ kg/s}\cdot\text{m}^2$) and 604 K, respectively	71
59. Fuel rod internal pressure for Rod IE-009 during flow reduction. The nominal fuel rod average power was 50.3 kW/m.	71
60. Fuel rod internal pressure versus fuel rod average power for Rod IE-010 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K, respectively	72
61. Fuel rod internal pressure on Rod IE-010 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K, respectively	72
62. Fuel rod internal pressure on Rod IE-010 during the flow reduction. The nominal fuel rod average power was 51.7 kW/m.	73
63. Cladding surface temperature at the 0.622-m elevation for a spring-loaded thermocouple on Rod IE-007 during the flow reduction. The other thermocouple located at this elevation failed.	73
64. Cladding surface temperature at the 0.622-m elevation for both of the spring-loaded thermocouples on Rod IE-008 during the flow reduction	76
65. Cladding surface temperature at the 0.521-m, 0.622-m and 0.724-m elevations on Rod IE-009 during the flow reduction. The plenum temperature is also shown for this fuel rod during this period.	76
66. Plenum temperature; inlet temperature and fuel rod average power for Rod IE-009 during Cycle 6.	77

67. Plenum, outlet and inlet temperature for Rod IE-009 during Cycle 6	77
68. Cladding surface temperature at the 0.622-m elevation for both of the spring-loaded thermocouples on Rod IE-010 during the flow reduction.	79
69. Cladding surface temperature on Rod IE-009 at the 0.521-m, 0.622-m and 0.724-m elevations during the flow reduction as calculated using FRAP-T2	79
70. Cladding elongation for Rods IE-007, IE-009, and IE-010 during Cycle 6	87
71. Composite response of cladding elongation during the flow reduction for all four fuel rods	87
72. Measured coolant flow rates through each flow shroud for Rods IE-007, IE-008, IE-009, and IE-010 during the flow reduction.	91
73. Fuel rod internal pressure and cladding elongation for Rod IE-010 during the flow reduction	91
74. Fuel rod pressure and cladding elongation for Rod IE-008 plotted with coolant flow, reactor power and fission activity monitor during preconditioning cycles when fuel cladding is postulated to have failed.	93
75. Parametric comparison between FRAP-T2 calculations and measured cladding elongation for Rod IE-007 during Cycle 6. Diametral gaps ranging from 0.04 to 0.010 mm were used in separate FRAP-T2 calculations to generate the analytical curves shown.	98

TABLES

I.	Parameters of Fuel Rods Used in Test IE-1.	9
II.	Cladding Surface Temperatures During Film Boiling.	28
III.	Film Boiling Zone on IE-1 Rods	34
IV.	Fuel Rod Internal Pressure Response During Cycle 6	74
V.	Fuel Rod Internal Pressure Response During Film Boiling	75
VI.	A Comparison of the Predicted and Measured Extent of Film Boiling on Rods IE-007 and IE-009	81
VII.	Summary of Selected Cladding Elongation Data	86
VIII.	Fuel Stack and Cladding Parameters Affecting PCI	88

IRRADIATION EFFECTS TEST SERIES
TEST IE-1
TEST RESULTS REPORT

1. INTRODUCTION

The Irradiation Effects (IE) Test Series is part of the Nuclear Regulatory Commission (NRC) Fuel Behavior Program^[1] to define behavior of irradiated fuel rods during transient reactor conditions. 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 off-normal or accident conditions. This test series is directed toward satisfying one of the major objectives of the Fuel Behavior Program - the development of verified analytical models for predicting the behavior of irradiated fuel rods. Detailed discussions of the irradiation effects and fuel behavior to be evaluated in this test series are described in the Irradiation Effects Experiment Requirements Document^[2]. This report describes the results of the IE-1 test.

The objectives of this test were (a) to evaluate the influence of fuel density on pellet-cladding interaction (PCI) during relatively slow ramps to about 33 kW/m and a fast power ramp to about 66 kW/m, and (b) to evaluate the influence of the irradiated state of the fuel and cladding on rod behavior during film boiling.

The four fuel rods were tested simultaneously with the same nominal power and flow history. A complete description of the fuel rods and test train design is given in Section 2.

The test consisted of a preconditioning period of about 30 hours followed by a power ramp, high power steady-state period, and a flow reduction transient to induce film boiling. During the early part of

the preconditioning period, gap conductance data were acquired using a technique which requires a sinusoidal oscillation of the reactor power level. Further discussion of the test conduct is provided in Section 3. The gap conductance test results will be presented in a separate report.

The results of the visual postirradiation examinations (PIE) are presented in Section 4. The balance of the PIE results will be reported separately after completion of the examinations. Section 5 contains the experimental results and a comparison of the calculated and measured fuel rod behavior. Conclusion and discussions of the test results and the comparisons made in Section 5 are provided in Section 6.

In Appendix A, an assessment of the instrumentation and data system accuracies are presented. The preirradiation fuel rod characterization data are contained in Appendix B. Appendix C contains additional test data not directly related to the fuel behavior data such as the coolant temperatures and pressures. The results of a detailed power calibration are presented in Appendix D.

2. EXPERIMENT DESIGN

Four fuel rods were radially positioned in the PBF reactor in-pile tube. A separate flow shroud was used to contain each fuel rod to hydraulically and thermally isolate the fuel rods from interacting with one another; essentially allowing four single fuel rod tests to be conducted simultaneously.

Each fuel rod and flow shroud assembly was instrumented to monitor fuel rod behavior during nuclear operations. The instruments associated with the test assembly and each fuel rod, and the principal components used in the experiment are described in this section.

2.1 Test Train

The test train hardware was designed to symmetrically position the four fuel rods in the PBF reactor test space and to allow for the removal of a fuel rod and shroud assembly, while the remainder of the test train is left intact. This provision allows for a quick change-out of fuel rods between tests. The test train hardware installed in the PBF in-pile tube is shown in Figure 1.

A fuel rod 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 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 Shroud

Each Zircaloy-4 flow shroud had a nominal inside diameter of 16.3 mm and a wall thickness of 3.15 mm. The pre-PBF test outside diameter of each fuel rod was nominally 9.89 mm. Resultant hydraulic and equivalent heated diameters were 6.39 mm and 16.9 mm respectively. The four flow shrouds were positioned in the in-pile tube as shown in Figures 1 and 3.

EGG-B-835

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

Fig. 2 Typical instrumented fuel rod for Test IE-1 shown installed in a flow shroud.

Fig. 3 Top view of IE-1 hardware orientation in the PBF in-pile tube. The 0 degree (0°) position for each flow shroud is toward the center of the test assembly.

Fig. 4 Schematic orientation of IE-1 self-powered neutron detectors (SPND).

2.3 Fuel Rods

The four fuel rods used for this test were of a similar design. The rods were clad with Zircaloy-4 tubing having an initial (pre-Saxton irradiation) outside diameter of 9.93 mm and a wall thickness of 0.564 mm. Based on pretest dimensional characterization^[3], the mean rod diameter in the central region of the rods was reduced (by creep) to about 9.89 mm as a result of service in the Saxton reactor. The fuel rods were approximately 0.97 m long (not including the instrumented end caps) and had a nominal active fuel stack length of 0.89 m. The fuel pellets (12.5 wt% ²³⁵U) were of a dished end design. Fuel pellet and fuel rod cladding characterization data for a typical irradiated fuel rod is given in Appendix B.

The fuel rods, designated Rod IE-007, IE-008, IE-009, and IE-010 were fabricated from Saxton (Westinghouse) Rods 119, 880, 835, and 836 respectively. Additional fuel rod information are summarized in Table I.

All of the fuel rods were fabricated from irradiated Saxton fuel rods with the upper end cap removed. A new upper end cap containing a pressure transducer replaced the original end cap. The fuel rods were backfilled with a gas mixture of 77.7% He and 22.3% Ar to about 2.6 MPa. The thermal conductivity of this mixture approximates the conductivity of the fission gases removed from the fuel rods. After backfilling, the rods were sealed and transported to the PBF canal to await testing. Complete details of the fabrication procedures are found in Reference 4.

By monitoring the fuel rod internal gas pressure transducers at various times prior to the test, it was determined that gas was leaking out of Rods IE-007, IE-008, and IE-009 via the 13 meter-long stainless steel cables attached to the plenum thermocouples. The cold rod pressure prior to heatup was 1.91 and 1.72 MPa for Rods IE-007 and IE-008. The pretest fuel rod internal pressure on Rod IE-009, uncertain because of a zero shift in the pressure transducer, was estimated to be 1.21 MPa. No pretest leak was discovered for Rod IE-010.

TABLE I

PARAMETERS OF FUEL RODS USED IN TEST IE-1

Parameter	Rod No.			
	IE-007	IE-008	IE-009	IE-010
Saxton Rod No.	119	880	835	836
Burnup MWd/tU	6,830	15,050	16,640	15,510
Fabricated Fuel Density (%TD)	94	94	92	92
Initial Fill Gas/Pressure (MPa)	He/3.45	He/1.48	He/2.52	He/2.52
Original Pellet Diameter (mm)	8.48	8.48	8.53	8.53
Fill Gas in 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 at Time of Assembly (MPa)	2.61	2.65	2.57	2.59
Pressure Prior to PBF Test (MPa)	1.91	1.72	1.21 [a]	2.49
Measured Void Volume (cm ³)	10.2	9.1	10.2	11.1

[a] This value was estimated since the pressure transducer had experienced a zero shift prior to the test.

2.4 Instrumentation

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

2.4.1 Test Train Instrumentation. The instrumentation associated with the test train is described below:

- (1) One 69 MPa and one 17 MPa strain post type pressure transducers were used to measure the system coolant pressure during the test. 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. The flowmeters were fabricated using a graphite journal bearing. These components are shown in Figures 2 and 5. Flow measurements are made by utilizing a magnetic pickup coil and a pulse rate indicator that monitors the turbine velocity.
- (3) A calibrated copper-constantan (Type T) differential thermocouple pair was provided to measure coolant temperature rise across each flow shroud. The coolant temperature increase, along with the measured coolant flow rate allows calculation of average fuel rod power for each fuel rod.
- (4) Two magnesium oxide insulated chromel-alumel (Type K) thermocouples were positioned near the inlet of the flow shroud assemblies to measure the coolant inlet temperature. The coolant outlet temperature of each

Fig. 5 Expanded layout of the linear variable differential transformer, coolant flow orifice plate, turbine flowmeter, and coolant flow screen assembly.

flow shroud was monitored with a similar chromel-alumel (Type K) thermocouple.

- (5) One linear variable differential transformer (LVDT) was positioned at the bottom of each fuel rod to measure changes in fuel rod cladding length (see Figures 2 and 5).
- (6) Eight, 4-inch long, cobalt self-powered neutron detectors (SPND) were mounted on support tubes in the test train assembly (see Figure 6). The location of these devices relative to the test train are shown in Figure 4. The SPNDs were employed to measure the relative axial neutron flux in the in-pile tube during the test. The data provided by these instruments, in conjunction with coolant differential temperature and flow rate measurements are used to monitor fuel rod power and neutron flux distribution. Five of the detectors were mounted on the test train with centers at 0.16, 0.31, 0.47, 0.63, and 0.78 m from the bottom of the active core. Three additional SPND's were mounted at the 0.63 m location to provide a relative measure of the radial neutron flux distribution across the test train.
- (7) A cobalt wire was mounted in the test train on the outside of one flow shroud to monitor the axial neutron flux profile in the in-pile tube. A posttest gamma scan of the wire yielded the integrated flux profile over the duration of the test. The cobalt wire was positioned to extend from below the bottom to above 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:

Fig. 6 Self-powered neutron detectors as mounted on support bar.

- (1) A 17 MPa strain post-type pressure transducer was used to monitor fuel rod internal pressure on each rod. The transducer was mounted on the instrumented end cap at the upper end of the fuel rod as shown in Figure 2.
- (2) A sheathed, magnesium oxide insulated, chromel-alumel (Type K) thermocouple was positioned in the plenum of all rods to monitor the plenum gas temperature. These devices were fabricated with grounded junctions and the ends of the sheath swaged to 0.46 mm. The positioning of the thermocouple is illustrated on Figure 2.
- (3) Each fuel rod was provided with two grounded junction, beryllium oxide insulated, tungsten-rhenium (W5%Re/W26%Re) alloy wire thermocouples to measure the cladding surface temperature. These zircaloy sheathed thermocouples were spring-loaded against the outer surface of the fuel cladding (see Figures 2 and 7) with an approximate three-pound preload. The thermocouples were located 0.62 m above the bottom of the fuel rods. On Rod IE-009 two additional thermocouple pairs were located at 0.52 and 0.72-m elevations.

Fig. 7 View of a section of the test train, as assembled for insertion into the IPT. The spring-loaded thermocouple mountings are shown at their respective positions on the outside of the flow shroud. The relative position of the self-powered neutron detectors are shown at their location in the test assembly.

3. EXPERIMENT CONDUCT

Irradiation Effects Test 1 (IE-1) consisted of the following three phases which spanned a total of 46 hours, including about 32 hours of nuclear operations:

- (1) A preconditioning phase which included two cycles designed to study the effects of inlet temperature, coolant flow rate, reactor power, and time upon fuel rod and test train measurements, a gap conductance piggy-back test, and two additional cycles of slowly varying power.
- (2) A power ramp which increased the rod peak power from 24 kW/m up to 70 kW/m.
- (3) A flow reduction to induce film boiling on the rods after approximately 45 minutes at 70 kW/m.

The power cycles were designated Cycle 1 through Cycle 6, as shown on Figure 8.

System pressure, based on PBF facility pressure measurement, was 14.82 ± 0.69 MPa. The two system pressure transducers mounted on the test train decalibrated prior to the test, therefore, the pressure data from these devices is not presented. The verification of the pressure and of the calculation of the uncertainty in pressure was made by analyzing the saturation temperature (measured by the outlet thermocouples during the flow reduction phase). The pressure was further verified at the end of the test, at which time Rod IE-007 failed and the internal pressure transducer went to 14.8 MPa.

Linear heat ratings, were determined using a thermal balance in conjunction with data from the axially distributed SPND's and a cobalt flux wire mounted in the test hardware. Average linear heat ratings

were calculated using a thermal balance^[a] while the SPND and flux wire data were used to calculate an axial power profile (see Appendix D and Reference 2 for a discussion of the techniques used). The data are summarized in this section and presented in detail in Section 5 with comparison with FRAP-T calculations.

3.1 Preconditioning

The first two preconditioning cycles consisted of 16 steps at all possible combinations of two levels of flow rate (505 and 750 cm³/s), inlet temperature (589 and 605 K), and average fuel rod peak linear heat rating (23 and 33 kW/m)^[b]. As seen in Figure 8, power was slowly increased at a rate of 0.35 kW/m per minute up to 23 kW/m at an inlet temperature of 605 K. Following this power step, the inlet temperature was decreased to 600 K (which lowers the outlet temperature well below saturation) verifying the rod power calculated at 605 K. When good agreement was obtained, the inlet temperature, flow rate, and power were increased. At 26.0 kW/m, a PBF transient rod power supply failure caused a scram of the reactor. After a shutdown of three hours, nuclear operation was continued. Power was then increased at 0.8 kW/m per minute up to 30 kW/m. The remainder of the cycles prior to gap conductance testing, was completed at a ramp rate of 0.8 kW/m per minute.

The gap conductance testing consisted of power calibrations at nominal test rod peak power levels of 10 and 17 kW/m, followed by power oscillations at nominal power levels of 10 and 17 kW/m. The

[a] Because of the large volume of data, it was not practical to calculate rod power using a thermal balance at each time step. Therefore, a computer program calculated rod power at several periods during the test. The results were then used in a regression analysis comparing SPND 4 and rod power. The resulting equation was used to plot rod power as a function of SPND 4.

[b] The combination of flow rate, inlet temperature, and rod linear heat rating was chosen using a 2³ 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. The design was additionally constrained so that the effect of time, electronic drift, etc., was confounded with the three factor interaction.

Fig. 8 Experiment conduct -- inlet temperature, coolant flow rate, rod power and reactor power for Test IE-1. Shutdowns shown were due to an automatic scram and range changes of the data acquisition system.

flow rate was $390 \text{ cm}^3/\text{s}$ and the inlet temperature was 557 K. No attempt was made to retain the frequency content of the data during the oscillations in the plots in this report. These data, which show strong aliasing effects, were presented only for visual continuity of the test conduct.

Following the gap conductance portion of the test, the reactor was shut down for instrument recalibration and range changes. The reactor was returned to power after five hours and the two additional cycles of preconditioning were completed. This period consisted of power steps from 0 to 33 kW/m at a nominal flow rate of $900 \text{ cm}^3/\text{s}$. The ramp rate was 0.31 kW/m per minute with a nominal 600 K inlet temperature.

During this phase, the fuel rod internal pressure transducers showed possible zero shifts (see Figures 9 through 13 for plots of key measurements for the test prior to the flow reduction). There was a slight (0.1 MPa) increase in indicated pressure at zero power for Rod IE-007, while there was a decrease for Rods IE-009 (1.5 MPa) and IE-010 (2.1 MPa). Possible fission gas release during these periods was not discernible to transducer drift. As seen in Figure 10, the indicated pressure for ROD IE-008 continued to increase throughout the test. Posttest analysis of the data and examination of the fuel rod determined that this pressure increase was due to a leakage of coolant water into the fuel rod.

All four LVDT's responded to changes in power with the LVDT on Rod IE-008 showing a shift in the zero, a reduction in sensitivity, with a few periods of erratic behavior. The other LVDT's showed no zero shift. The LVDT on Rod IE-010 showed a strong temperature sensitivity (0.03 mm/K). The temperature sensitivities of the other LVDTs were less than 0.002 mm/K.

All operating cladding surface thermocouples showed good agreement during this period. On Rod IE-007, one of the thermocouples failed. The two thermocouples on Rod IE-008 showed a possible asymmetry

Fig. 9 Rod IE-C07 behavioral data during the test up to the time immediately prior to film boiling.

Fig. 10 Rod IE-008 behavioral data during the test up to the time immediately prior to film boiling.

Fig. 11 Rod IE-005 behavioral data during the test up to the time immediately prior to film boiling.

Fig. 12 Rod IE-010 behavioral data during the test up to the time immediately prior to film boiling.

Fig. 13 Rod IE-007 behavioral data during the test from a time immediately prior to film boiling to shutdown.

in the cladding surface temperature since both read the same at zero power (see Figure 10). Both thermocouples on Rod IE-009 indicated the same temperature. On Rod IE-010, there was an apparent difference in the response of the two devices since there was a constant offset between the two throughout the test, including zero power levels.

The plenum thermocouples on Rods IE-008 and IE-009 showed good agreement up to the gap conductance cycle, at which time the Rod IE-008 thermocouple failed. The Rod IE-009 plenum thermocouple functioned throughout the test.

3.2 Power Increase and Steady-State Operation

The preconditioning phase ended with a steady-state fuel rod power of 24 kW/m. The test fuel rod power was then increased at a rate of 3.2 kW/m per minute up to an average fuel rod peak power of 69.7 kW/m^[a]. At high power steady-state conditions at the reactor peak power level, the fuel rod peak powers, determined using coolant flow and temperature differential, were 72.4, 67.0, 68.9, and 70.5 kW/m for Rods IE-007, IE-008, IE-009, and IE-010, respectively. Reactor power, coolant inlet temperature, and coolant flow were held constant for approximately 45 minutes following the power increase. Although reactor power was held constant, posttest analysis revealed that test fuel rod power drifted down about 2% during the 45 minute period.

Internal pressure transducers on Rods IE-007, IE-009, and IE-010 showed a significant increase in pressure as the power was increased (see Section 5, Table IV). The pressure transducer on Rod IE-008 did not respond to the power ramp. This was probably due to a loss of

[a] The reactor power necessary to produce 69.7 kW/m was predicted using a comparison between reactor power and rod power made during the last two cycles of preconditioning. Because of the nonlinearity between the two powers, the end point of the ramp was under-predicted. After a thermal balance was made at the end of the ramp, the reactor power was increased at 3.2 kW/m per minute to bring rod power up to the desired level. The fuel rods were operated at an average power of 62 kW/m prior to this last power increase for 12 minutes (see Figure 8).

cladding integrity in the regions of the rod which had hydrided such that the internal pressure was in equilibrium with the system pressure. The other three transducers indicated fission gas release during the high power steady-state operation, (Rod IE-009 had been showing a strong negative zero shift) with pressure increases of 0.3 MPa, 0.7 MPa, and 0.5 MPa for Rods IE-007, IE-009, and IE-010, respectively.

The LVDT's on all four rods indicated strong pellet to cladding mechanical interaction with a resulting increase in cladding elongation during the power increase. The cladding elongation decreased during the steady power conditions due to a relaxation of the stresses in the cladding as a result of fuel creep. Specifically, the LVDT's on Rods IE-007, IE-008, IE-009, and IE-010 showed increases of 0.14%, 0.12%, 0.10%, and 0.16% with relaxations of 0.013%, 0.013%, 0.021%, and 0.032%. The ratios of the relaxation due to fuel creep at steady power to the increase in elongation during the power increase were 9%, 11%, 21%, and 20% for Rods IE-007, IE-008, IE-009, and IE-010, respectively.

3.3 Flow Reduction

At the end of the high power steady-state period, the flow on all four rods was reduced to induce film boiling on the rods. The flow was reduced in steps of less than $60 \text{ cm}^3/\text{s}$ ($295 \text{ kg/s}\cdot\text{m}^2$) to a flow of $590 \text{ cm}^3/\text{s}$ ($2900 \text{ kg/s}\cdot\text{m}^2$), holding at each level for about 1 minute. The flow was then decreased in steps of $35 \text{ cm}^3/\text{s}$ ($171 \text{ kg/s}\cdot\text{m}^2$) until, at $514 \text{ cm}^3/\text{s}$ ($2520 \text{ kg/s}\cdot\text{m}^2$), the LVDT on Rod IE-007 indicated the onset of film boiling. At this point, the flow was further reduced to $415 \text{ cm}^3/\text{s}$ ($2030 \text{ kg/s}\cdot\text{m}^2$), held momentarily, and then further reduced to $355 \text{ cm}^3/\text{s}$ ($1740 \text{ kg/s}\cdot\text{m}^2$) to assure that film boiling was present on all four rods.

During the flow reduction, the LVDT's on the other three rods indicated that film boiling had occurred at flow rates of $345 \text{ cm}^3/\text{s}$ ($1690 \text{ kg/s}\cdot\text{m}^2$), $375 \text{ cm}^3/\text{s}$ ($1840 \text{ kg/s}\cdot\text{m}^2$), and $415 \text{ cm}^3/\text{s}$ ($2030 \text{ kg/s}\cdot\text{m}^2$) for Rods IE-008, IE-009, and IE-010 respectively. In Figures 13 through

16, the LVDT's on Rods IE-007, IE-008, IE-009, and IE-010 showed increases in cladding elongation from 0.15 to 0.39%^[a], 0.16 to 0.33%, 0.08 to 0.17%, and 0.11 to 0.44%, respectively. The reactor was shut-down after approximately one minute of film boiling Rod IE-007. After shutdown, cladding elongations decreased to 0.02, 0.22, 0.09 and 0.17%, respectively. Internal pressure for Rods IE-007, IE-009, and IE-010 increased from 6.0 to 6.5 MPa, 13 to 14 MPa^[b], and 6.8 up to 7.3 MPa, respectively, during film boiling. The plenum temperature thermocouple in Rod IE-009 increased from 620 to 663 K during the film boiling transient. The indicated cladding surface temperatures also increased as shown in Table II.

After 18 seconds in film boiling, the unstable vapor layer on Rod IE-009 (the rod with the second lowest power and highest flow) collapsed with a resultant decrease in elongation, plenum temperature, and cladding surface temperature (see Figures 15 and 71). After one minute of film boiling on Rod IE-007, the PBF reactor control rods were run in. As the power reduction was initiated, the cladding elongations decreased as the vapor layer partially collapsed. Seven to ten seconds after the decrease in elongation, at 25 to 40% power reduction, the cladding surface temperature on Rods IE-008 and IE-010 returned to pre-film boiling levels. After the rod power was decreased to 3 kW/m, the reactor was scrammed. Following shutdown, the pressure in Rod IE-007 returned to 3.9 MPa, indicating an increase in internal pressure of 0.6 MPa from the zero power level immediately prior to the last two cycles of preconditioning. Internal pressure for Rods IE-009

[a] The core of the LVDT on Rod IE-007 reached electronic saturation or a mechanical stop.

[b] The absolute value of these data are incorrect.

TABLE II
INDICATED CLADDING SURFACE TEMPERATURES DURING FILM BOILING

Location	Pre-Film Boiling Temperature (K)	Film Boiling Maximum Temperature [a] (K)
Rod IE-007 at 0.622 m and 0°	622	660
Rod IE-008 at 0.622 m and 180°	631	752
Rod IE-008 at 0.622 m and 0°	624	675
Rod IE-009 at 0.521 m and 180°	624	626
Rod IE-009 at 0.521 m and 0°	625	631
Rod IE-009 at 0.622 m and 180°	622	634
Rod IE-009 at 0.622 m and 0°	627	638
Rod IE-009 at 0.724 m and 180°	623	631
Rod IE-009 at 0.724 m and 0°	623	631
Rod IE-010 at 0.622 m and 180°	630	645
Rod IE-010 at 0.622 m and 0°	624	729

[a] Large systematic errors are present in these temperatures. The spring - loaded thermocouples serve effectively only as an indicator of local film boiling conditions.

Fig. 14 Rod IE-008 behavioral data during the test from a time immediately prior to film boiling to shutdown.

Fig. 15 Rod IE-009 behavioral data during the test from a time immediately prior to film boiling to shutdown.

Fig. 16 Rod IE-010 behavioral data during the test from a time immediately prior to film boiling to shutdown.

and IE-010 increased 1.0 and 1.2 MPa^[a]. About three minutes after test termination, the Rod IE-007 internal pressure sharply increased to coolant system pressure, indicating cladding failure. There was no positive indication of failure from the fission break monitor (see Appendix C, Figure C-7).

[a] The increase in internal pressure was corrected for the predicted shift in the zero during this time period. Based on data from the first cycle of the test, the predicted zero shifts were 0.10, -0.99, and -1.19 MPa for Rods IE-007, IE-009, and IE-010, respectively.

4. PRELIMINARY POSTIRRADIATION EXAMINATION RESULTS

The IE-1 test hardware was disassembled in the PBF canal and the four rods were shipped to the hot cells in their flow shrouds. Of the detailed postirradiation examinations scheduled, the visual and preliminary dimensional characterizations are described in this section. Complete postirradiation examination results will be published in a separate report.

4.1 Visual and Dimensional Characterization

Overall views of the rods are shown in Figure 17. When removed from the flow shroud, Rod IE-007 was fractured into two pieces. The other three rods were intact. The film boiling zones were clearly defined by the axial extent of surface oxidation and diameter changes (see Table III). Observations on each rod are discussed in more detail in the following sections. Diametral measurements of each rod are presented in Figure 18.

4.1.1 Rod IE-007. This rod (IE-007) was in two pieces with a circumferential cladding fracture occurring at the 0.514-m rod elevation. Cladding cracks up to 30 mm in length ran axially from the main fracture. The main cladding fracture and axial cladding cracks exhibited no evidence of plastic deformation; indicating the cladding was in a highly embrittled state when the failure occurred.

At the main fracture, where the fuel apparently had washed out following failure, the ID surface of the cladding was crazed with a mud-crack type pattern (see Figure 19). However, where fuel had fallen out during the posttest examination, the surface was relatively smooth with no obvious surface crazings or mud-crack patterns. Since operational data indicated the rod failed shortly after termination of the test when cladding surface temperatures were relatively low (near the ambient coolant temperature of about 600 K), a low temperature reaction apparently took place; probably a reaction between water and the Zr-UO_2 reaction products forming the surface patterns.

TABLE III

FILM BOILING ZONE ON IE-1 RODS

Rod No.	Film Boiling Zone Boundaries (meters from bottom of rod)		Length of Film Boiling Zone (m)	Observation ^[a]
	Lower	Upper		
IE-007	0.445	0.685	0.240	Rod failure in FBZ, severe cladding oxidation, longest FBZ, diameter increase in FBZ.
IE-008	0.540	0.686	0.146	Rod intact, cladding cracks below FBZ, minor cladding oxidation, short FBZ, diameter increase in FBZ
IE-009	0.540	0.680	0.140	Rod intact, shortest FBZ, least cladding oxidation, cladding collapse in FBZ
IE-010	0.508	0.686	0.178	Rod intact, relatively severe oxidation, 2nd longest FBZ, diameter increase in FBZ

[a] FBZ - film boiling zone

Fig. 17 Posttest view of Rods IE-007, IE-008, IE-009, and IE-010.

Fig. 18 Posttest diameter measurements on Test IE-1 irradiated rods.

Fig. 19 View of Rod IE-007 fracture showing crazing of cladding ID.

The film boiling zone extended from the 0.445-m rod elevation to about the 0.685-m rod elevation. In the film boiling zone, the rod surface was heavily oxidized comparable to the appearance of Rod IE-001, from IE Scoping Test 1, which also failed following termination of the test^[5]. Through the film boiling zone, Rod IE-007 increased in diameter up to a maximum of about 0.4 mm. The diameter increases were not associated with the axial cracks. This diameter increase was contrary to experience on unirradiated rods where the cladding collapsed onto the fuel, resulting in diameter decreases in the film boiling zone.

4.1.2 Rod IE-008. This rod (IE-008) was intact; however a series of cladding perforations in the form of axial cracks in the cladding were found below the film boiling zone between the 0.25 to 0.38-m rod elevations (see Figure 20). Rod bulging appeared to be associated with some of these defect regions and in one instance fuel was protruding from the rod through the cladding cracks. The cladding, as evidenced from the rod bulging, did not exhibit a total absence of ductile behavior, although little plastic deformation was found along the edges of the cracks. This rod failed by cracks formed in regions of the cladding determined to be hydrided. The hydrogen source was coolant water leaking into the rod as discussed in Section 3.

The film boiling zone extended from the 0.540 to 0.686-m rod elevations. Cladding surface oxidation appeared to be considerably less than that observed on Rod IE-007. In the film boiling zone, this rod also increased in diameter by as much as 0.3 mm.

4.1.3 Rod IE-009. This rod (IE-009) was intact with no cladding perforation observed. The film boiling zone extended from about 0.540 to 0.680-m rod elevation. Cladding surface oxidation was considerably less severe than seen on the other rods. The rod was bowed or "kinked" in the film boiling zone.

The film boiling zone was characterized by uniform cladding collapse onto the fuel column and into pellet interfaces, pellet chips, and

Major Cladding Perforation at 0.360 m Rod Elevation on Rod IE-008. The Rod Had Bulged and Fuel Appeared to be Protruding Through Breach in Cladding.

Longitudinal Cladding Splits (arrows) Just Below Major Cladding Perforation

Fig. 20 View of Rod IE-008 below film boiling zone showing cladding perforations.

regions where the fuel had broken up and relocated. The cladding apparently survived the rather severe collapse deformations associated with the collapse into the regions where fuel was apparently missing (see Figure 21). Fuel redistribution during prior irradiation or handling prior to this test is the probable cause of spaces in the fuel stack.

4.1.4 Rod IE-010. This rod (IE-010) was intact with no cladding failures. The film boiling zone was characterized by rather severe cladding surface oxidation; visually estimated to be less severe than that observed on Rod IE-007 and more severe than on Rods IE-008 and IE-009. The film boiling zone extended from the 0.508 to 0.686-m rod elevation. This rod also increased in diameter by about 0.4-mm through the film boiling zone.

4.2 Discussion

The axial extent of the film boiling zone correlated well with the visual assessment of the severity of cladding oxidation, i.e., the longer the film boiling zone the greater the surface oxidation in the zone. Rod IE-007, with the longest and most severe cladding oxidation, failed in the film boiling zone. The cladding, highly embrittled due to the external Zr-water and internal Zr-UO₂ reactions^[a], apparently was not capable of withstanding even minor strains during cooldown due to fuel-cladding interaction, pressure differential, or thermal gradients. Rod IE-008 failed below the film boiling zone during the test due to hydride defects.

As previously observed on Test IE-ST-1 and IE-ST-2, the cladding on Rod IE-009 collapsed through the film boiling zone. The cladding surface temperature or the time at temperature was apparently much less for Rod IE-009 than for the other rods as indicated by the relatively minor surface oxidation. The cladding was capable of withstanding very severe collapse deformation strains.

[a] Preliminary metallography indicated that near the location of rod failure the internal and external cladding reaction zones penetrated nearly through the cladding wall. Cladding temperature was estimated to be 1780 K at this location.

During film boiling, the internal pressure for all the rods was much less than the external coolant pressure. However, three rods, Rods IE-007, IE-008, and IE-0010 increased in diameter in the film boiling zone. The cause of the diameter increase was postulated to be a combination of (a) transient fuel swelling associated with gases trapped in the fuel and (b) fuel volume increase due to melting^[a]. It was postulated that at the onset of film boiling and prior to any significant oxidation or cladding embrittlement, the cladding collapsed onto the fuel column and then fuel swelling occurred, causing a net increase in fuel rod diameter. The fuel rod swelling mechanism will be carefully evaluated by cladding and fuel metallography.

Cladding Collapse

Fig. 21 View of Rod IE-009 showing cladding collapse into region of apparent missing fuel.

[a] Preliminary results from neutrography and metallographic examination indicate extensive fuel melting occurred in the film boiling zones of all four rods.

5. EXPERIMENTAL RESULTS AND COMPARISON OF CALCULATED AND MEASURED FUEL ROD BEHAVIOR

In this section, fuel rod behavior data is presented from selected portions of the test. Cladding displacement, fuel rod internal pressure, plenum temperature, and cladding surface temperature data are given. Power changes made during the preconditioning cycles were slow enough that the fuel rods were in quasi- thermal and mechanical equilibrium^[a], thus allowing the fuel rod parameters to be plotted versus fuel rod average or local power. During the flow reduction and film boiling portion of the test, the response of the fuel rods to film boiling was transient, therefore, these measured values are presented versus time.

For Rods IE-007 and IE-009 the measured data are compared with analytical results obtained using the FRAP-T2^[b] computer code calculations. Comparisons are made for data obtained during both the preconditioning cycles and flow reduction phase of the test. Separate calculations were not performed for Rods IE-008 and IE-010 because of their similarity to rods IE-007 and IE-009, respectively.

During Cycle 2 of the test, the coolant flow, temperature, and fuel rod power were varied constantly, making complete comparisons of steady-state experimental data and calculations during these cycles impractical, therefore, only a portion of the possible comparisons were made.

[a] The term quasi-equilibrium is used since equilibrium would imply that fuel creep restructuring and fission gas release would be complete, which would not (in the strictest sense) be the case for these fuel rods during the power cycles.

[b] The analytical results were obtained using FRAP-T2, MOD 002, with load module TOL888, in conjunction with the material properties code MATPRO MOD 006.

5.1 Analytical Model

The FRAP-T2 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 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-T2 code, including submodels and assumptions, is found in Reference 6.

5.1.1 Fuel Rod Model. For this analysis, the fuel rods were modeled using 11 radial and 15 axial nodes. FRAP-T2 input included the nominal measured conditions of coolant flow, coolant inlet temperature, coolant pressure, average fuel rod power, and axial flux profile. The Westinghouse W-3^[7] CHF correlation using the cold wall factor and the Groeneveld tubes and annulus^[8] film boiling heat transfer correlation were used. The free thermal expansion fuel deformation model and the MacDonald-Broughton^[9] model for gap conductance were specified.

Fuel rod average power was calculated from the experimental data using steady-state coolant flow and coolant temperature rise measurements. The axial flux profile was determined from a gamma scan of a cobalt wire that was positioned in the reactor test space throughout the test. Evaluation of the axially distributed self-powered neutron detector data and the flux wire data showed that the axial flux profile was not affected significantly by changes in the reactor power level (less than 5% change in peak to average power was observed for fuel rod powers from 10 to 60 kW/m.) Consequently, the flux wire data, although it represents an integral of the power profile, is the best available measure of the average test power distribution. A detailed explanation of the fuel rod power calculations and their uncertainty is provided in Appendix D.

Input to model the irradiated fuel rods included the Saxton fuel rod nominal design dimensions, burnup in MWd/t and pre-PBF testing

cladding outside diametral measurements. These data are summarized in Section 2, Table I and in Appendix B.

5.1.2 Experiment Conduct Model. The steady-state FRAP-T2 analyses for comparison with the experimental data were made by the following method. Calculations were performed using incremental steps in the average fuel rod power of 6 kW/m, from 0 to a maximum of 48 kW/m. At each step, this power was held constant, allowing the fuel rod to equilibrate. These equilibrium fuel rod parameters (cladding surface temperatures, cladding elongation, etc.) at each 6 kW/m power step were used to generate the data for the steady-state comparisons.

To model the transient fuel rod behavior during the flow reduction, the average fuel rod power, appropriate coolant pressure and coolant inlet temperature, and the as-run coolant flow history during the flow reduction steps were used. The flow history for Rods IE-007 and IE-010 was presented in Section 3.

5.2 Cladding Elongation

Cladding elongation data for Rods IE-007, IE-008, IE-009 and IE-010 during Cycle 1, part of Cycle 2, all of Cycles 4 through 6, and during the flow reduction are presented. Comparisons between the measured values and values calculated using the FRAP-T2 computer code are made for Rods IE-007 and IE-009. The cladding elongation data are presented as a percent of length change from zero power. The reference length used was 879 mm.

5.2.1 Rod IE-007. A plot of cladding elongation versus fuel rod average power for the power increase of Cycle 1 is shown in Figure 22. The measured and calculated values diverge above 2 kW/m. The calculated cladding elongation was due to free thermal expansion of the cladding, caused by the increasing cladding temperatures with increasing fuel rod power. The experimental rate of cladding elongation indicated that pellet-to-cladding interaction (PCI) began at about 2 kW/m, thus the elongation of the cladding was dominated by the fuel stack

growth. The onset of PCI at very low fuel rod power is typical^[5,10] for the first power cycle of a fuel rod that has a diametral fuel stack to cladding gap similar to fuel rods used in this test (0.22 to 0.27 mm).

A uniform annular gap between the fuel column and inner cladding diameter does not exist. In irradiated rods, off center, skewed, and chipped pellets initially cause an effectively smaller than theoretical diametral gap. This smaller effective gap causes PCI during the initial power cycles at very low fuel rod powers. These low power interactions are reduced as the fuel rod power is repeatedly cycled. The PCI and thermal stresses on the fuel during the power cycles tends to increase the effective diametral gap between the pellets and cladding by fragmenting and repositioning of the fuel. Once the rate of PCI as a function of fuel rod power reaches a maximum on a given power ramp, it is often termed "hard" PCI. The power at which hard PCI occurs depends primarily upon the diametral gap size and the previous operating history.

Although the irradiated fuel rods used in this test had been subjected to many previous power cycles in the Saxton Reactor, the fragmented fuel pellets in the fuel rods were disturbed during handling prior to this test, causing a redistribution of the cracked fuel pellets. Consequently, as the fuel rod power was increased in Cycle 1, the frictional forces between the fuel stack and cladding were sufficient to cause PCI at very low fuel rod average powers.

The cladding elongation data in Figure 22 indicates that the rod decreased in length while the fuel rod power was held constant at 18 kW/m over a 75 minute period. A plot of the cladding response during this period of time is shown in Figure 23. The reduction in cladding length probably resulted from redistribution of pellet fragments, fuel creep, and slippage between the fuel stack and elastically strained cladding. Once the power increase was again started, the rate of cladding growth was unchanged from what it was prior to the steady power period. The power reduction portion of Cycle 1 is not plotted, since the reactor scrammed just after the maximum power shown in Figure 22 was obtained.

Fig. 22 Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 1. The nominal coolant flow rate and temperature are $500 \text{ cm}^3/\text{s}$ ($2554 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 23 Cladding elongation for Rod IE-007 during the 75 minute steady power period of 18 kW/m for Cycle 1.

During Cycle 2, several combinations of fuel rod power, coolant inlet temperature, and coolant flow were used. Only the data during the first part of the cycle, where coolant inlet temperature and flow were held constant, is presented. Figure 24 shows the cladding elongation versus fuel rod average power for a power increase from 0 to 24 kW/m. The power was held steady for about 23 minutes at 24 kW/m, then reduced to 21 kW/m and held constant for 23 minutes, and then increased back to 24 kW/m. Since the fragmented fuel stack was partially redistributed during Cycle 1, hard PCI did not occur during Cycle 2 until a fuel rod average power of 12 kW/m was reached.

An offset was subtracted from the cladding elongation data for each fuel rod to set the displacement of the cladding to a zero value for zero fuel rod average power at the beginning of Cycle 1. The cladding elongation data at zero fuel rod power for the second and subsequent cycles (Figures 24 and 35) are shown to be negative. This observation is attributed to either transducer or data system drift and/or a combination of coolant temperature effect on the cladding and the sensitivity of the LVDT. The LVDT temperature sensitivity on Rod IE-010 was up to 0.004%/K but was negligible on Rods IE-007 and IE-009. For the length of the rod to shorten as much as shown in Figure 24 or 35, the average cladding temperature would have to decrease more than 10 K. A calibration was not obtained for a transducer temperature of 555 K, therefore, the cladding elongation data during Cycle 3, where the inlet temperature was 555 K, is not presented.

Fuel rod elongation data for Cycles 4, 5 and 6 are shown on Figures 25, 26, and 27. The onset of hard PCI on these cycles continued to shift to a higher fuel rod average power than the previous cycles. On the last cycle, Cycle 6, PCI began at about 19 kW/m.

During Cycle 6, the fuel rod average power was raised to 49 kW/m at a rate of 3.2 kW/m per minute. The reactor control rods were then leveled, which caused an approximate 2 kW/m reduction in fuel rod average power over a 17 minute period. The power was then increased to

Fig. 24 Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 2. The nominal coolant flow rate and temperature are $750 \text{ cm}^3/\text{s}$ ($3682 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 25 Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 4. The nominal coolant flow rate and temperature are $935 \text{ cm}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 26 Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 5. The nominal coolant flow rate and temperature are $935 \text{ cm}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 27 Cladding elongation versus fuel rod average power for Rod IE-007 during Cycle 6. The nominal coolant flow rate and temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

54 kW/m, where the control rods were again leveled. Another reduction of 2 kW/m in the fuel rod average power occurred. The power was held at about 52 kW/m for 38 minutes during this period. During both of these steady power periods, a cladding length decrease, due to fuel creep, occurred. The cladding response plotted against time during Cycle 6 is shown on Figure 28. The exponential shape of the cladding length change was indicative of fuel creep.

During the flow reduction, film boiling first occurred on Rod IE-007. The response of the LVDT during film boiling is shown on Figure 29. The flat top on the elongation data trace resulted from the fuel rod reaching a mechanical limit or the LVDT saturating electronically, hence, the data shown in this plot does not represent the full extent of cladding elongation during film boiling.

5.2.2 Rod IE-008: Cladding elongation data for Rod IE-008 appeared to shift about 0.10 percent at the beginning of Cycle 1 and was very noisy throughout that cycle. During the subsequent cycles, the transducer signal was less noisy, but drifted and indicated about one-half the relative increase in cladding elongation as did Rod IE-007, which was of a similar design. Figures 30 and 31 show the response of Rod IE-008 during Cycle 6. The quantitative response of this measurement is not usable, but the relative shape of the cladding elongation decay during the steady power periods is of interest. Figure 32 shows the cladding elongation response during film boiling on this fuel rod. Again, the data shows only the relative cladding response due to uncertainty in the LVDT sensitivity.

5.2.3 Rod IE-009. Cladding elongation data and FRAP-T2 calculations for Rod IE-009 are shown on Figures 33 to 39. PCI for this fuel rod started at about zero fuel rod average power during Cycle 1, as it did for Rod IE-007. The rate of cladding elongation though, was less for this fuel rod than for Rod IE-007. For example, the rate of cladding elongation as a function of fuel rod average power for Rod IE-007

Fig. 28 Cladding elongation for Rod IE-007 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 29 Cladding elongation for Rod IE-007 during the flow reduction. The nominal fuel rod average power is 53.0 kW/m .

Fig. 30 Cladding elongation versus fuel rod average power for Rod IE-008 during Cycle 6. The nominal coolant flow rate and temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg}/\text{s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 31 Cladding elongation for Rod IE-008 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg}/\text{s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 32 Cladding elongation for Rod IE-008 during the flow reduction. The nominal fuel rod average power is 49.4 kW/m.

Fig. 33 Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 1. The nominal coolant flow rate and temperature are 510 cm^3/s ($2503 \text{ kg/s}\cdot\text{m}^2$) and 604 K, respectively.

Fig. 34 Cladding elongation for Rod IE-009 during the 75 minute steady power period of 18 kW/m for Cycle 1.

Fig. 35 Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 2. The nominal coolant flow rate and temperature are $770 \text{ cm}^3/\text{s}$ ($3780 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 36 Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 4. The nominal coolant flow rate and temperature are $950 \text{ cm}^3/\text{s}$ ($4741 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 37 Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 5. The nominal coolant flow rate and temperature are $950 \text{ cm}^3/\text{s}$ ($4741 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 38 Cladding elongation versus fuel rod average power for Rod IE-009 during Cycle 6. The nominal coolant flow rate and temperature

Fig. 39 Cladding elongation for Rod IE-009 during Cycle 6. The nominal coolant flow rate and inlet temperature are $990 \text{ cm}^3/\text{s}$ ($4860 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

was about 0.0035% per kW/m, while for Rod IE-009 it was 0.0015% per kW/m. A discussion of these differences in cladding elongation rate is presented in Section 6.

Figure 34 shows the time response of the cladding length decrease due to slippage and fuel pellet redistribution during the steady power period at 18 kW/m. In Figures 35 to 38 it is apparent that the onset of hard PCI occurred at a progressively higher fuel rod average power as the fuel was power cycled. The negative zero power intercept was explained in Section 5.2.1. Cladding elongation during Cycle 6 is depicted in Figure 38 and 39. Fuel creep was again evident as an exponential reduction in cladding length during the steady power periods (see Figure 39). FRAP-T2 calculations, shown on Figure 38, indicate that PCI was predicted to occur at 36 kW/m. Once the PCI was calculated to occur though, the rate of cladding length increase was overpredicted relative to the data.

The cladding elongation data during film boiling for Rod IE-009 is provided on Figure 40. This fuel rod showed the smallest elongation increase of all four fuel rods during the film boiling period. The length change for this fuel rod was a 0.05% elongation. This figure also shows that film boiling did not occur on Rod IE-009 until the final flow reduction was made.

5.2.4 Rod IE-010. PCI on Rod IE-010 during Cycle 1 occurred from 0 to 22 kW/m fuel rod average power (see Figure 41). The cladding elongation rate during this cycle was 0.0020% per kW/m, which is similar to the 0.0015% per kW/m observed for Rod IE-009 (Rods IE-009 and IE-010 were of a similar design and burnup). During the 75 minute steady power period of Cycle 1, the cladding elongation increased about 0.004%, (see Figure 42). There is no apparent mechanism for cladding growth during a steady power period. Shortly after the power increase during Cycle 1 was reinitiated, the cladding length showed a sudden decrease. This behavior was unusual, but could be a result of stresses

Fig. 40 Cladding elongation for Rod IE-009 during the flow reduction. The fuel rod average power is 50.3 kW/m.

Fig. 41 Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 1. The nominal coolant flow rate and temperature are 500 cm³/s (2554 kg/s·m²) and 601 K, respectively.

built up during the power ramp causing a sudden readjustment of the cladding length.

Cycles 2 through 6 are shown on Figures 43 through 47. Cladding elongation behavior for this fuel rod was similar to that observed for Rod IE-009, i.e., the transition from soft to hard PCI was smoother than observed for Rod IE-007. Recall that the fabricated diametral gap for Rod IE-007 was larger than for Rods IE-009 or IE-010. This larger gap may explain why the transition to hard PCI was smoother for Rods IE-009 and IE-010 than for Rod IE-007.

Redistribution of the fuel stack during the power cycles also occurred for this fuel rod (IE-010). This was apparent by hard PCI occurring at higher fuel rod average powers during subsequent cycles.

Figures 46 and 47 show the cladding elongation response during Cycle 6. No change in cladding length occurred at the 18 kW/m power level for Rods IE-008 and IE-009. A cladding length change did not occur at this power level since the fuel rod had been cycled through several power cycles and redistribution of the fuel stack had occurred. The combination of residual stresses in the extended cladding and fuel temperatures was low enough that fuel creep did not occur at this low power level. The average fuel rod power was then ramped at a rate of 3.2 kW/m per minute to 48 kW/m and then to 53 kW/m, and held constant for 17 and 38 minutes, respectively, at each power level. During the two high, steady power periods, a cladding length reduction occurred, resulting from fuel creep.

The cladding response during film boiling for Rod IE-010 is shown in Figure 48. The maximum indicated cladding length change during film boiling was a 0.32 percent elongation.

5.3 Fuel Rod Internal Pressure

Several plots of fuel rod internal pressure versus fuel rod average power and time are presented in this section. Both test data and

Fig. 42 Cladding elongation for Rod IE-010 during the 75 minute steady power period of 18 kW/m for Cycle 1.

Fig. 43 Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 2. The nominal coolant flow rate and temperature are 750 cm³/s (3682 kg/s·m²) and 604 K, respectively.

Fig. 44 Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 4. The nominal coolant flow rate and temperature are $935 \text{ cm}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 45 Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 5. The nominal coolant flow rate and temperature are $935 \text{ cm}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 46 Cladding elongation versus fuel rod average power for Rod IE-010 during Cycle 6. The nominal coolant flow rate and temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 47 Cladding elongation for Rod IE-010 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 48 Cladding elongation for Rod IE-010 during the flow reduction. The nominal fuel rod average power is 51.7 kW/m.

Fig. 49 Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 1. The nominal coolant flow rate and inlet temperature are 500 cm³/s (2554 kg/s·m²) and 604 K, respectively.

FRAP-T2 values are presented for Rod IE-007. Only a limited amount of test data for Rods IE-009 and IE-010 are given. Fuel rod internal pressure data versus fuel rod average power for Rod IE-008 is not presented since this fuel rod appeared to have developed a leak during Cycle 2.

5.3.1 Rod IE-007. The fuel rod internal pressure for Rod IE-007 plotted as a function of fuel rod average power was very repeatable throughout the test. Plots for the six power cycles are shown in Figures 49 to 56. Fuel rod internal pressure data during Cycle 3 had a noticeably lower intercept than the other five cycles because of the reduced inlet temperature which affected cladding temperature in the plenum region and therefore the internal gas pressure.

FRAP-T2 calculations, presented on these plots, show that the rate of pressure increase was consistently underpredicted. This underprediction was probably caused by a too low calculated internal gap gas temperature. Since the gap gas in the fuel rod was distributed throughout cracks and voids in the fuel column, the gas had a higher temperature than it would if it were completely located in an annular gap between the fuel stack outer diameter and the cladding inner diameter. The calculated model used by the FRAP-T2 code assumed that the gap gas temperature was the average temperature of the cladding inside surface and fuel stack outer surface temperatures. Also, a small volume of gas in the fuel cracks and in the fuel pellet dish volumes is modeled, but the temperature of this gas is assumed to be the same as the fuel pellet surface temperature. Consequently, the gas temperature, hence the gas pressure, was underpredicted.

The repeatability of the fuel rod internal pressure data for Cycles 1 through 5 indicated that the transducer did not drift and the fuel rod did not leak during the test.

Figures 54 and 55 show the fuel rod internal pressure during Cycle 6. Evidence of fission gas release during the high power steady

Fig. 50 Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 2. The nominal coolant flow rate and inlet temperature are 750 cm³/s (3682 kg/s·m²) and 604 K, respectively.

Fig. 51 Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 3. The nominal coolant flow rate and inlet temperature are 375 cm³/s (2166 kg/s·m²) and 556 K, respectively.

Fig. 52 Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 4. The nominal coolant flow rate and inlet temperature are $935 \text{ cm}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 53 Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 5. The nominal coolant flow rate and inlet temperature are $935 \text{ cm}^3/\text{s}$ ($4666 \text{ kg/s}\cdot\text{m}^2$) and 601 K , respectively.

Fig. 54 Fuel rod internal pressure versus fuel rod average power for Rod IE-007 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 55 Fuel rod internal pressure on Rod IE-007 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 56 Fuel rod internal pressure for Rod IE-007 during flow reduction. The nominal fuel rod average power was 53.0 kW/m.

Fig. 57 Fuel rod internal pressure versus fuel rod average power for Rod IE-009 during Cycle 6. The nominal coolant flow rate and inlet temperature are 990 cm³/s (4860 kg/s·m²) and 604 K, respectively.

state periods of this cycle is shown on these figures. The fuel rod power was held constant during three different periods of Cycle 6. Once at 18 kW/m, again at about 49 kW/m, and finally at 54 kW/m. No measurable fission gas release occurred during the hour of steady power at 18 kW/m, while a 0.17 MPa pressure increase occurred during a 17 minute steady power period at 49 kW/m. During the 38 minute steady power period of 54 kW/m, the fuel rod pressure again increased by about 0.20 MPa to a pressure of 5.85 MPa. These pressure increases are attributed to a release of fission gases that were trapped in the fuel matrix during the initial irradiation of the fuel rod prior to this test.

The fuel rod internal pressure response during the film boiling is shown in Figure 56. The moderate increase of fuel rod internal pressure, 0.6 MPa, to a maximum of 6.5 MPa was attributed primarily to heating of the gases in the fuel rod, a void volume reduction due to cladding collapse, and fuel swelling (see Section 4). No evidence of substantial transient fission gas release appeared during this off-normal operating condition even though a relatively large fraction of the fuel in the film boiling region was molten. This statement is made on the basis of similar magnitude pressure increases observed in unirradiated fuel rods undergoing similar transients.

Also shown on these figures is the FRAP-T2 calculation for the flow reduction period. The underprediction of the fuel rod internal pressure increase was attributed to an incorrect estimate of the temperature of the gases in the fuel rod.

5.3.2 Rod IE-009. Only data during Cycle 6 and during the flow reduction are provided for Rod IE-009. An evaluation of the data plots (see Figure 11) during earlier cycles indicated that either the pressure transducer drifted or the fuel rod leaked pressure through an extension cable connected to the plenum temperature thermocouple. It is also possible that both of these events occurred. Helium leakage was measured through the plenum thermocouple cables prior to the test for this fuel rod and is the probable explanation for the pressure decrease.

Figure 57 shows a plot of fuel rod internal pressure versus fuel rod average power. The fuel rod power was held constant during this cycle at powers of approximately 17, 47, and 51 kW/m. The fuel rod internal pressure response plotted against time, Figure 58, shows these steady power periods. During the first steady power period at 17 kW/m, the fuel rod pressure first indicated 1.85 MPa and then decayed to about 1.80 MPa over a 60 minute period. During the power increase to 47 kW/m, the fuel rod internal pressure increased to 3.95 MPa, where the fuel rod power was held nearly constant for about 17 minutes. An increase of 0.15 MPa was observed for this steady power period. Finally, the fuel rod average power was increased to 51 kW/m and held constant for about 38 minutes while the fuel rod internal pressure increased from 4.4 to 4.8 MPa. The pressure increases during the constant power periods were a result of fission gas release. These increases slightly underestimated the amount of fission gas released because of the slow leak through the plenum thermocouple cable, as indicated during the 60 minute steady power period at 17 kW/m. The measured pressure for this fuel rod increased from 5.0 MPa to a peak of 5.9 MPa during film boiling (see Figure 59).

5.3.3 Rod IE-010. The fuel rod internal pressure response for Rod IE-010 is similar to Rod IE-009, i.e., Rod IE-010 also showed evidence of a slow leak or transducer drift. Again, only the response of the fuel rod internal pressure during Cycle 6 and the flow reduction are presented. Figure 60 shows a reduction of fuel rod pressure of 0.05 MPa at a constant fuel rod average power of about 18 kW/m. This decrease in pressure at the low steady power gives some estimate of the leak rate or transducer drift. Fission gas release during the two steady power periods at about 48 kW/m and 53 kW/m are shown in Figures 60 and 61. Figure 62 shows the fuel rod internal pressure response during the flow reduction and subsequent film boiling.

Tables IV and V summarize the measured pressure changes during the three quasi-steady power periods of Cycle 6 and the flow reduction for all three fuel rods.

Fig. 58 Fuel rod internal pressure on Rod IE-009 during Cycle 6. The nominal coolant flow rate and inlet temperature are $990 \text{ cm}^3/\text{s}$ ($4860 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 59 Fuel rod internal pressure for Rod IE-009 during flow reduction. The nominal fuel rod average power was 50.3 kW/m .

Fig. 60 Fuel rod internal pressure versus fuel rod average power for Rod IE-010 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 61 Fuel rod internal pressure on Rod IE-010 during Cycle 6. The nominal coolant flow rate and inlet temperature are $975 \text{ cm}^3/\text{s}$ ($4786 \text{ kg/s}\cdot\text{m}^2$) and 604 K , respectively.

Fig. 62 Fuel rod internal pressure on Rod IE-010 during the flow reduction. The nominal fuel rod average power was 51.7 kW/m.

Fig. 63 Cladding surface temperature at the 0.622-m elevation for a spring-loaded thermocouple on Rod IE-007 during the flow reduction. The other thermocouple located at this elevation failed.

TABLE IV

FUEL ROD INTERNAL PRESSURE RESPONSE DURING CYCLE 6

Parameter	Rod No.								
	IE-007			IE-009			IE-010		
Fuel Rod Average Power (kW/m)	18	49	54	17	47	51	18	48	53
Initial Pressure (MPa)	4.25	5.34	5.65	1.9	3.95	4.4	4.55	6.25	6.50
Final Pressure (MPa)	4.25	5.51	5.85	1.85	4.1	4.8	4.50	6.30	6.75
Pressure Change (MPa)	0	+0.17	+0.20	-0.05	+0.15	+0.40	-0.05	+0.05	+0.25
(%)	0	+3	+4	-3	+4	+8	-1	+1	+4
Time For Change (Minutes)	60	17	38	60	17	38	60	17	38

TABLE V
FUEL ROD INTERNAL PRESSURE RESPONSE DURING FILM BOILING

Parameter	Rod No.		
	IE-007	IE-009	IE-010
Fuel Rod Average Power (kW/m)	53.0	50.3	51.7
Initial Pressure (MPa)	5.9	5.0	6.8
Maximum Pressure (MPa)	6.5	5.9	7.4
Pressure Increase (MPa)	0.6	0.9	0.6
(%)	10	18	11

5.4 Cladding Surface and Plenum Gas Temperature

All four fuel rods were fit with diametrically opposed spring-loaded thermocouples, configured as discussed in Section 2, to monitor the cladding surface temperature and to detect the presence of local film boiling. Film boiling was dependably detected by the thermocouples, however, the temperature measurements were typically poor for both steady-state and transient operations 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. Data from the spring-loaded thermocouple is not compared with the FRAP-T2 calculations.

Figures 63 through 66 show the cladding surface thermocouple response during the flow reduction for Rods IE-007, IE-008, IE-009, and IE-010, respectively. The maximum cladding surface temperature measured by cladding thermocouples, approximately 750 K, was well below the cladding temperature associated with the oxidation reported in Section 4.

Fig. 64 Cladding surface temperature at the 0.622-m elevation for both of the spring-loaded thermocouples on Rod IE-008 during the flow reduction.

Fig. 65 Cladding surface temperature at the 0.521-m, 0.622-m and 0.724-m elevations on Rod IE-009 during the flow reduction. The plenum temperature is also shown for this fuel rod during this period.

Fig. 66 Plenum temperature, inlet temperature and fuel rod average power for Rod IE-009 during Cycle 6.

Fig. 67 Plenum, outlet, and inlet temperature for Rod IE-009 during Cycle 6.

The plenum thermocouples on Rods IE-007 and IE-010 failed prior to the test. Rod IE-009 had a plenum thermocouple that worked throughout the test. The plenum thermocouple on Rod IE-008 failed a few hours after the fuel rod pressure transducer on that fuel rod indicated a leak. The plenum temperature measurement was not compared to a plenum gas temperature using the FRAP-T2 computer code because of calculational instabilities in the plenum temperature model that occurred when attempting to make this calculation during film boiling. The plenum thermocouple response in Rod IE-009 during film boiling is shown in Figure 65.

Figure 67 shows a plot of the plenum temperature measurement, coolant inlet temperature and the fuel rod average power for Rod IE-009 during Cycle 6. The coolant outlet temperature, coolant inlet temperature and plenum temperature measurement are shown on Figure 68 for Rod IE-009 during Cycle 6. These two plots show the dependence of plenum temperature on the coolant conditions.

No tests or error analyses were performed to provide an estimate of a systematic measurement error associated with the plenum thermocouple. The installation of this device was such that it was expected to exchange radiation with the cladding inner surface and with the top of the fuel stack. Additionally, heat conduction along the sheath of this shielded thermocouple would affect any temperature measurement. Contribution of these effects to the systematic error has not been determined.

FRAP-T2 calculations of cladding surface temperatures on Rod IE-009 at the 0.541-m, 0.622-m and 0.724-m elevations during film boiling are shown on Figure 69. The FRAP-T2 calculations indicated the highest cladding surface temperature (1470 K) would occur on the fuel rod at the 0.470-m elevation. This elevation was below the lower limit of film boiling (0.540-m) as reported in Section 4.

Fig. 68 Cladding surface temperature at the 0.622-m elevation for both of the spring-loaded thermocouples on Rod IE-010 during the flow reduction.

Fig. 69 Cladding surface temperature on Rod IE-009 at the 0.521-m, 0.622-m and 0.724-m elevations during the flow reduction as calculated using FRAP-T2.

5.5 Calculated Fuel Rod Behavior During Film Boiling

Some additional parameters that were calculated using FRAP-T2 for Rods IE-007 and IE-009 are included in the following subsections.

5.5.1 Rod IE-007. FRAP-T2 calculations indicated film boiling would first occur at the 0.622-m elevation on Rod IE-007 at a flow rate of $590 \text{ cm}^3/\text{s}$ ($2895 \text{ kg/s}\cdot\text{m}^2$). With additional reductions in flow, similar to the experimental flow history, film boiling was predicted to occur over a region from the 0.343-m elevation to the top of the fuel stack. Cladding collapse was calculated from the 0.343 to 0.622-m elevation. The minimum coolant flow used in the calculations was $355 \text{ cm}^3/\text{s}$ ($1743 \text{ kg/s}\cdot\text{m}^2$). The maximum cladding surface temperature was calculated to be 1490 K at the 0.419-m elevation. Fuel melting was predicted to occur from the 0.343-m elevation to the 0.572-m elevation, and the total molten fuel fraction was 0.065. Metal-water reaction was calculated to occur over the region on the surface of the cladding extending from the 0.343 to 0.673-m elevation, with a maximum oxide depth of 0.026 mm. Fuel rod failure was not predicted to occur.

5.5.2 Rod IE-009. Calculations made using FRAP-T2, for Rod IE-009 indicated a less severe fuel rod response during the flow reduction than was predicted for Rod IE-007. Rod IE-009, though, had a fuel rod power about 2.7 kW/m less than Rod IE-007. Film boiling was calculated to initiate at the 0.673-m elevation, at a flow rate of $569 \text{ cm}^3/\text{s}$ ($2792 \text{ kg/s}\cdot\text{m}^2$). A calculated flow reduction to $535 \text{ cm}^3/\text{s}$ ($2626 \text{ kg/s}\cdot\text{m}^2$), then to $430 \text{ cm}^3/\text{s}$ ($2111 \text{ kg/s}\cdot\text{m}^2$) and finally to $375 \text{ cm}^3/\text{s}$ ($1841 \text{ kg/s}\cdot\text{m}^2$) indicated film boiling would occur over a region from the 0.419-m elevation to the top of the fuel stack. Molten fuel was calculated to exist from the 0.419 to 0.521-m elevation. The total calculated molten fuel fraction was 0.021. The maximum calculated cladding surface temperature was 1460 K at the 0.470-m elevation. Metal-water reaction was predicted to occur on the cladding surface from the 0.419 to 0.622-m elevation, with a maximum oxide depth of 0.015 mm.

A posttest visual examination of the fuel rods indicated that the film boiling zone on Rod IE-007 extended from the 0.445 to 0.685-m elevation. On Rod IE-009, the posttest examination revealed evidence of film boiling from the 0.540 to 0.680-m elevation. A comparison of the observed film boiling regions with the predicted zones is shown in Table VI. Also shown on this table are the measured and calculated cladding elongations.

TABLE VI
A COMPARISON OF THE PREDICTED AND MEASURED
EXTENT OF FILM BOILING ON RODS IE-007 AND IE-009

Rod No.	Predicted Film Boiling Region [a]			Measured Film Boiling Region [a]		
	Beginning	End	Length	Beginning	End	Length
IE-007	0.343	0.889	0.546	0.445	0.685	0.307
IE-009	0.419	0.889	0.470	0.540	0.680	0.331
	Calculated Cladding Elongation %			Measured Cladding Elongation %		
IE-007	0.80			0.25		
IE-009	0.65			0.10		

[a] Indicated dimensions are elevations (in meters) above bottom of fuel rod.

The discrepancies in calculated and measured cladding elongations shown on Table VI are largely a result of the axial film boiling region not being predicted well. This lack of agreement between the experiment and the FRAP-T2 calculations is expected since the critical heat flux correlation used to determine the onset of film boiling was developed from experimental data obtained using a different flow geometry than was used in this test. Therefore, an extensive evaluation of the PCI

model in FRAP-T2 cannot be made using a comparison of the actual and predicted cladding elongations. But it should be noted that the differences in predicted cladding elongation compared to measured values are significantly greater than the differences in the predicted and measured film boiling zones. This suggests that the PCI model in FRAP-T2 still accounts for a portion of the overpredicted cladding elongation.

6. DISCUSSION AND CONCLUSIONS

Test objectives included (a) an evaluation of the effect of fuel density on PCI during power ramps and (b) an evaluation of the effect of irradiation on fuel rod performance during off-normal, or film boiling, conditions. Three of the four irradiated fuel rods were successfully monitored for cladding length changes throughout the test. The cladding elongation data indicated that higher density fuel causes increased fuel deformation rate during power ramps. This increase in elongation rate is attributed to the more dense, harder fuel not yielding to stresses as readily as does the less dense ceramic. This effect of density had been previously demonstrated^[10,11] in Halden testing on unirradiated fuel rods at beginning of life. This is the first time that fuel rods having a substantial prior irradiation history were used to evaluate this effect. These data indicate, as do data on fresh fuel rods, that hard pellet-cladding-interaction (length change per kW/m) can be reduced even at moderate burnup levels, by the use of low initial fabrication density fuel pellets.

The 94% TD fuel rods, IE-007 and IE-008, had an initial pellet-cladding diametral gap of 0.27-mm while the 92% TD fuel rods, IE-009 and IE-010, had an initial pellet-cladding diametral gap of 0.22-mm. The differences in diametral gaps, resulting from the variation in pellet size, did not significantly affect the elongation rate of the irradiated fuel rods. The power at which hard PCI repeatably occurs does not seem to be affected by the differences in initial diametral gap that these fuel rods had.

During the flow reduction, film boiling was initiated on all four of the irradiated fuel rods. One of the fuel rods, IE-007, indicated failure due to a severely oxidized embrittled cladding shortly after the film boiling had been terminated. Rod IE-008 failed during the preconditioning cycles of the test. This fuel rod failure was most likely due to PCI strain of the hydrided zircaloy cladding at the beginning of the power ramp initiating Cycle 6. The locations of the hydride

blisters were found below the film boiling zone in the region of expected maximum PCI. The source of the hydride is postulated to be a result of water leakage into the fuel rod beginning about ten hours after the start of the test (see the internal pressure transducer response in Figure 10). The cause of this leakage will be investigated during postirradiation examinations.

Film boiling, which causes severe cladding deterioration, and the resultant quenching did not cause cladding failure on Rods IE-008, IE-009 or IE-010 even though one rod probably contained substantial water vapor. These results indicate that irradiated rods can survive an extended period of film boiling (about 1.5 minutes for Rod IE-010) and have enough ductility to maintain cladding integrity through a severe quenching. Postirradiation visual examination of the fuel rods showed evidence of a wide range of peak cladding temperatures among the four rods. Diametral measurements showed cladding swelling in the film boiling zone for three of the four rods. The rod behavior (cladding temperatures and the extent of oxidation) appears very sensitive to small differences in power and flow conditions.

A detailed evaluation of the cladding condition (metallography) will be made available in a separate report after the postirradiation examinations of the fuel rods have been completed.

With the exception of the underlying cause of the leak which produced the hydrided region on Rod IE-008, the PCI that occurred (even during the power ramp at a rate of 3.2 kW/m per minute) did not cause the irradiated cladding to fail.

In addition to the above information, fuel rod internal pressure data during steady-state, power ramp, and film boiling operations was accumulated. These data and the cladding elongation information will be used to help evaluate fuel rod codes such as FRAP-T.

6.1 Experimental Data

A discussion of the experimental data is made to (a) address the effect of fuel density on PCI, (b) summarize fission gas release data, and (c) review the failure of Rod IE-008.

6.1.1 Cladding Elongation.

Cladding elongation data presented during Cycle 1, for the four fuel rods, demonstrated that the rate of cladding deformation was strongly affected by small variations in fuel density. The PCI that occurred at the beginning of Cycle 1 was attributed to the small effective diametral gap which resulted from a fragmented fuel stack.

During Cycle 2, hard PCI first occurred at 10 to 12 kW/m for Rods IE-007, IE-008 and IE-010 (the output of the LVDT on Rod IE-008 was considered unreliable throughout the test). The shift in power relative to Cycle 1 for the onset of hard PCI was caused by the fragmented fuel stack being redistributed by the elastically stressed cladding. Relocation of the fuel stack resulted in a larger effective diametral gap. Subsequent power cycles caused the point of initial hard PCI to saturate at about 16 kW/m in Cycles 5 and 6 for all three fuel rods (IE-007, IE-009, and IE-010). This suggests that the power at the onset of hard PCI was not affected by the small variations in the initial diametral gaps and fuel densities.

Transition from soft to hard PCI was less abrupt for Rods IE-009 and IE-010 than for Rod IE-007. This effect was probably influenced more by fuel density than diametral gap.

From Figure 70 and Table VII it is evident that the difference in cladding elongation rates between the two densities of fuels was less apparent during Cycle 6 than it was during Cycle 1, but still existed.

TABLE VII

SUMMARY OF SELECTED CLADDING ELONGATION DATA

Parameter	Rod No.		
	IE-007	IE-009	IE-010
Fabricated Fuel Density (%TD)	94	92	92
Initial Pellet Diameter (mm)	8.48	8.53	8.53
Measured Pretest Mean Cladding Outside Diameter (mm)[a]	9.89	9.99	9.90
Rate of Cladding Elongation (percent per kW/m) During:			
Cycle 1	0.0035	0.0015	0.0020
Cycle 6	0.0053	0.0031	0.0043
Amount of Cladding Length Reduction (percent) During the High Power Steady-State Periods of Cycle 6:			
Period 1	0.012	0.020	0.028
Period 2	0.015	0.025	0.029

[a] These are measured mean values obtained from the irradiated fuel rods prior to the PBF test.

Table VIII lists fuel stack and cladding parameters affecting PCI.

Rods IE-007 and IE-009, were of a high enough burnup that differences in burnup (6,830 MWd/tU for Rod IE-007 and 15,050 MWd/tU for Rod IE-009) probably did not affect PCI. Also, it has been demonstrated elsewhere that the differences in original diametral gap, as these two fuel rods had, did not affect the rate of cladding elongation during hard interaction^[11]. Consequently, the differences in cladding elongation rate (% per kW/m) were probably due to the differences in initial fuel density. The relative difference in rate of cladding elongation between the two fuel rods (about a factor of 2, for a 2% difference

Fig. 70 Cladding elongation for Rods IE-007, IE-009, and IE-010 during Cycle 6.

Fig. 71 Composite response of cladding elongation during the flow reduction for all four fuel rods.

TABLE VIII

FUEL STACK AND CLADDING PARAMETERS AFFECTING PCI

Parameter	Rod No.			
	IE-007	IE-008	IE-009	IE-010
Fabricated Fuel Density (% TD)	94	94	92	92
Smear Density (% TD)	88	88	87	87
Original Pellet Diameter (mm)	8.48	8.48	8.53	8.53
Original Cladding Outside Diameter (mm)	9.93	9.93	9.93	9.93
Original Diametral Gap (mm)	0.27	0.27	0.22	0.22
Irradiated Cladding Outside Diameter (mm)	9.89	9.86	9.88	9.90
Burnup (MWd/tU)	6,830	15,050	16,640	15,510

in density) is consistent with elongation rate changes in fuel rods of similar densities at the Halden reactor^[11].

Experiments made to delineate factors affecting cladding elongation^[11] conducted at the Halden Reactor indicated that cladding elongation rates for high density fuel are larger than those for low density fuel. The effect of initial gap size was to cause an increase in elongation rate for fuel rods with smaller gaps, although, the report^[11] concluded the influence of gap size on irradiated fuel rods was relatively minor. Therefore, these experimental results are consistent with those obtained in the Halden work.

The differences in relative cladding elongation between the higher and lower density fuel rods are summarized in Table VII. The amount of cladding relaxation due to fuel creep is less in the high density fuel, which is consistent with known behavior on the creep rates of ceramics^[11] (the creep rate of ceramic materials increases with a decreasing density).

Shown on Figure 71 is the cladding elongation data for all four fuel rods during the flow reduction. Because of the large temperature excursions that occurred on the cladding surface at the onset of film boiling, an immediate cladding length change was caused by thermal expansion of the cladding. The cladding growth was also supplemented by PCI, caused by an increase in fuel stack growth resulting from thermal expansion of the fuel. Therefore, the cladding elongation data is a good indicator of when film boiling occurred and the relative axial extent of the film boiling on a fuel rod. Likewise, a reduction in cladding length indicates a reduction in the extent, or collapse of film boiling on a fuel rod.

Film boiling was initiated at a different time in the flow reduction on each fuel rod because: (a) each fuel rod was operating at a different average power, and (b) the relative coolant flows through each shroud were different. Fuel rod average powers during the flow reduction for Rods IE-007, IE-008, IE-009, and IE-010 were 53.0, 49.4, 50.3, and 51.7 kW/m, respectively. Figure 72 shows the coolant flow rate through each shroud during the flow reduction. DNB occurred first on Rod IE-007, the highest-power fuel rod. Rod IE-010, a fuel rod with a similar flow rate and slightly lower fuel rod average power (51.7 kW/m), was forced into film boiling during the first flow reduction step made after film boiling was indicated on Rod IE-007. At the end of this first flow reduction step, film boiling occurred on Rod IE-008. The average fuel rod power for Rod IE-008 was the lowest at 49.4 kW/m, but Rod IE-008 also had the lowest relative coolant flow of the four fuel rods. Finally, an additional flow reduction step initiated film boiling on Rod IE-009.

Figure 72 shows that Rod IE-009 had the highest relative flow rate of all four fuel rods. Rod IE-009, at 50.3 kW/m, was one of the lower power fuel rods. This effect, combined with the high flow rate, explains why film boiling occurred last on this fuel rod. Another variation that Rod IE-009 had from the other three fuel rods was that it

was fit with three pair of spring-loaded cladding surface thermocouples, where the other fuel rods only had one pair. These devices were not expected to significantly affect the rod behavior, but they probably caused a slight perturbation to the cladding surface heat flux and provided some coolant mixing.

6.1.2 Fuel Rod Internal Pressure. Table V in Section 5 summarized fuel rod internal pressure data during Cycle 6. These data indicated that the maximum pressure increase due to fission gas release occurred on Rod IE-009 during the 38 minute steady power period at 51 kW/m fuel rod average power. The fuel rod pressure increased about 9%, from 4.4 to 4.8 MPa during this period. Rods IE-007 and IE-010 increased in pressure by about 3 and 4% respectively during this same period. The absolute pressure indicated on Rod IE-009 could be in error because of a zero shift in the calibration which occurred between the time of calibration and when it was installed in the reactor. The calibration used for the data in this report represents a best estimate of the transducer behavior.

The fuel rod pressure response during film boiling on each fuel rod is tabulated in Table V. These data indicated that there was no significant release of fission gas during film boiling on the fuel rods. The fuel rod internal pressure and cladding displacement data (see Figure 73) for Rod IE-010 during the flow reduction offers some insight to events occurring during film boiling. The steeply increasing cladding displacement data indicated significant cladding temperature rise and axial propagation of the film boiling region on the fuel rod. Additional cladding surface temperature increases resulted until the rates of heat generation and removal equilibrated. The increase in fuel rod internal pressure for Rod IE-010 corresponded with the onset of film boiling on the fuel rod, as indicated by the LVDT. Once the film boiling region was established (as indicated by a decreasing rate of cladding elongation increase), the fuel rod internal pressure remained relatively constant until the LVDT indicated film boiling had ceased. Since no additional pressure increases were indicated after the onset

Fig. 72 Measured coolant flow rates through each flow shroud for Rods IE-007, IE-008, IE-009, and IE-010 during the flow reduction.

Fig. 73 Fuel rod internal pressure and cladding elongation for Rod IE-010 during the flow reduction.

of film boiling on Rod IE-010, it is implied that substantial fission gas release did not occur during the approximate 45 seconds of film boiling. The fuel rod internal pressure data for Rods IE-007 and IE-009 show similar results (see Figures 58 and 61). The pressure increases shown in Table V, from 10 to 18%, were greater than the 8 percent pressure increase measured on the unirradiated fuel rod during film boiling during IE Scoping Test 1^[5]. These two experiments are not completely comparable, due to minor variations in test conditions, but they do indicate that any fission gas released from an irradiated fuel rod during film boiling is not significant enough to affect cladding deformation because of high fuel rod internal pressure. These conclusions of course, are based on limited data. Additional testing of other irradiated fuel rods from different reactors is needed before the above conclusions can be generally validated.

6.1.3 Failure of Rod IE-008. As noted in Section 3, Rod IE-008 began leaking early in the test. The history plot (Figure 10, Section 3) of the fuel rod internal pressure transducer for Rod IE-008 indicated a pressure increase after about 10 hours of testing. It is postulated that the seal on a tube used to backfill the fuel rod with the simulated fission gas mixture failed, providing a small leak path for the coolant water (system pressure 14.8 MPa) to enter the fuel rod at this point in the test. The fuel rod pressure continued to increase, presumably as more water leaked into the fuel rod and was vaporized by the hot fuel. The leak would have to of been a small one, since during Cycle 4 the internal fuel rod pressure increased to a maximum value of 16.3 MPa, which was about 1.5 MPa above the coolant pressure. The fuel rod average power was approximately 33 kW/m at this point. During Cycle 5 at the same fuel rod average power, the fuel rod pressure was 15.8 MPa, which indicates that the vaporized water either reacted with the cladding or the fuel rod gases were leaking out of the fuel rod.

Figure 74 shows an expanded history of cladding elongation, fuel rod internal pressure, loop fission product monitor, loop flow, and reactor power from the end of Cycle 5 to the end of the test. These

Fig. 74 Fuel rod pressure and cladding elongation for Rod IE-008 plotted with coolant flow, reactor power and fission activity monitor during preconditioning cycles when fuel cladding is postulated to have failed.

data traces are provided to illustrate the suggested mechanisms leading to the cladding failure on Rod IE-008. The fission product monitor data provides a measure of the relative level of gamma activity in the experiment loop flow. This data trace has a logarithmic scale. The loop flow data indicates a relative amount of coolant flowing in the experiment loop. This data is shown to assist in interpreting the fission product monitor data trace. When the loop flow increased so did the fission monitor data trace, due to the decreased transit time of the coolant and the associated reduced decay time for the ^{16}N activity in the coolant. Thus, if the reactor power and loop flow had remained constant and the fission product monitor increased, that would be a positive indication that additional fission products had been deposited in the experiment loop. Unfortunately, both reactor power and loop flow were changing at the same time as the fission product monitor began to increase.

At a time of 154000 seconds, the fuel rod average power started to increase, initiating Cycle 6. The fuel rod internal pressure also began to increase at this point. Shortly after the power increase started, at a time of 154500 seconds, the fuel rod internal pressure showed a decrease from 14.1 MPa to about 13.8 MPa, which was the pressure of the coolant within the accuracy of the pressure transducer. Coincident with the pressure decrease was a small increase in the fission monitor level. Since the loop coolant flow remained constant, the fission product indication increase was either associated with the power increase, or attributed to an increase in fission products in the loop, or both. As the fuel rod average power increased to 18 kW/m, the fuel rod internal pressure remained essentially constant at coolant system pressure. During the 60 minutes at this power, both the fuel rod internal pressure and fission product monitor remained at constant values. At the end of this constant power period, the fuel rod average power was increased at a rate of 3.2 kW/m per minute from 18 to 51 kW/m. During this increase the loop fission product monitor indicated a major increase, while the loop coolant flow remained constant. The fuel rod internal pressure data was noisy, but indicated a very small increase

and then returned to a pressure equivalent to the coolant pressure. It is suggested that the strong PCI occurring during this ramp caused a break in a previously formed hydrided area of the cladding (see Section 4).

During the flow reduction, the fuel rod internal pressure reading was very noisy. A small pressure increase of about 0.15 MPa occurred at the time film boiling was indicated by the LVDT. This spike was followed by a slight reduction in rod pressure. At a time corresponding to a collapse of the film boiling on Rod IE-008, the pressure transducer indicated a sharp reduction to about 13.0 MPa. The fuel rod internal pressure then returned to system pressure. The sharp reduction in pressure at the termination of film boiling on Rod IE-008 was probably caused by the steam in the fuel rod being rapidly condensed as the fuel rod was quenched by the coolant water (Note rod power was approaching zero at these conditions also thus removing the energy source inside the rod).

Posttest examination of the cladding on Rod IE-008 indicated a cladding rupture at the 0.25 and 0.38-m elevation (about 0.16-m below the film boiling zone). These cladding defects were a result of hydriding, as discussed in Section 4. It is postulated that water leaking into the gas fill tube early in the test caused the hydriding. Since zircaloy hydrides are known to be very brittle, a cladding failure occurred due to PCI during the rapid power ramp of Cycle 6, as indicated by the fission product detector.

Rod IE-008 indicated a cladding failure prior to film boiling occurring on the fuel rod. Thus it is assumed that water and water vapor would be inside of the cladding during the flow reduction and subsequent film boiling. This off-normal condition, which raised the fuel temperatures above melting, did not result in a massive or destructive failure due to a vapor explosion, fuel cladding or fuel water interactions.

Since a small percent of fuel rods in commercial reactors typically have failed cladding, it can be inferred from this experiment that

the behavior of a failed fuel rod in the event of film boiling probably would not result in any phenomena different than that known to occur on unfailed rods. In fact, at the end of the actual test, Rod IE-008 was not suspected to have failed. The internal pressure increase was improperly attributed to instrument malfunctioning. Only the posttest examination, coupled with a very detailed review of the pressure data, led to the conclusions presented above.

6.2 FRAP-T2 Results

Comparisons between the experimental and calculated fuel rod internal pressures and cladding elongations were made in Section 5. A few comments regarding the FRAP-T2 models used in calculating these parameters are given in this section. Some results on calculated fuel and cladding temperatures are also given.

6.2.1 Fuel Rod Internal Pressure. Plots depicting FRAP-T2 calculations and measured values for fuel rod internal pressure as a function of fuel rod average power show that the FRAP-T2 model consistently undercalculated the rate of fuel rod internal pressure increase (up to 20% at 50 kW/m fuel rod average power as shown in Figure 56). The measurements indicate that an increase in fuel rod internal pressure resulting from fission gas release is small (1 to 8%).

6.2.2 Cladding Elongation. Cladding elongation comparisons made for Rods IE-007 and IE-008 indicated that the onset of hard PCI was not correctly predicted using the FRAP-T2 models. The fuel rods modeled in these comparisons were irradiated (IE-007 with 6,830 MWd/tU and IE-009 with 16,640 MWd/tU). The fuel in an irradiated fuel rod would be fragmented, and the diametral gap would not be a well defined annulus. Also, the cladding would have a reduced diameter resulting from creep that occurred during irradiation (that is, if the pressure differential across the cladding wall was sufficient, as was the case for these fuel rods). The cladding diameter reduction for these fuel rods was obtained from pretest measurement of the cladding outer diameter. From these measurements and a nominal cladding wall thickness, a mean

cladding inside diameter was estimated for each fuel rod. The data are presented in Appendix B.

Complex mechanisms of densification, pore migration, sintering, swelling, cracking, creep, and fuel pellet fragment redistribution contribute to fuel pellet effective diameter change during irradiation. No attempt was made to estimate the effect of burnup on the fuel stack diameter in these fuel rods, even though it was known that diametral changes did occur due to both densification, swelling and cracking. Code input for the fuel stack was composed of the nominal, as-built dimensions for each fuel rod analyzed.

It is apparent that a significant weakness in the analysis was the estimate of the cladding diametral gap. This poor estimate was, in part, the reason PCI was not calculated to occur for Rod IE-007 in the data comparisons in Section 5. This estimate of the diametral gap could also be partly responsible for a lack of comparison between the predicted and measured pressures.

Figure 75 shows a comparison between FRAP-T2 calculations and the measured cladding elongation for Rod IE-007 during Cycle 6. Several diametral gaps, ranging from 0.04 to 0.10 mm, were used in separate FRAP-T2 calculations to generate the analytical values shown on this figure. This compares to a diametral gap size of 0.23 mm used for the calculations in Section 5 for Rod IE-007.

This parametric study indicated that to predict the onset of hard PCI, a very small diametral gap must be used in the model. But, once PCI was predicted to occur, the FRAP-T2 analysis overestimated the rate of cladding deformation. The interaction model used in the code did not consider that the forces exerted by the elastically extended cladding deform the fuel, as indicated by a decrease in slope of the cladding elongation data at a fuel rod average power of about 30 kW/m on Figure 75. Also, once a diametral gap was calculated to have been closed by thermal expansion at a node, the cladding length change of the cladding node segment was calculated to be the same as the fuel stack

Fig. 75 Parametric comparison between FRAP-T2 calculations and measured cladding elongation for Rod IE-007 during Cycle 6. Diametral gaps ranging from 0.04 to 0.10 mm were used in separate FRAP-T2 calculations to generate the analytical curves shown.

length change. No allowance was provided for slip between the cladding and fuel stack.

Another restriction imposed on the calculational model was the input of a constant diametral gap over the entire length of the fuel rod. Cladding diameter on an irradiated fuel rod is smaller toward the center, higher power section of the fuel rod. The smaller diametral gap, caused by these necked down regions would allow PCI to be calculated at lower powers and at a different rate than a simplified mean diameter required by the FRAP-T2 model. This model simplification helps explain why the FRAP-T2 analysis overpredicted the extent of PCI once the initial pellet to cladding contact had occurred.

7. 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, W. J. Quapp, P. E. MacDonald, and O. Goetzmann, Light Water Reactor Fuel Behavior Program Description: Irradiation Effects Experiment Requirements, TFBP-TR-175, EG&G Idaho Inc. (February 1977).
3. G. W. Gibson et al, Characteristics of UO₂-Zircaloy Fuel Rod Materials From the Saxton Reactor for Use in Power Burst Facility ANCR-NUREG-1321 (September 1976).
4. P. M. Mandiloff, W. C. Francis, V. Strickholm, "Assembly of Fuel Rods for Power Burst Facility Testing", 24th Conference on Remote Systems Technology, Winter Meeting - American Nuclear Society, Nov 14-19, 1975.
5. W. J. Quapp, C. M. Allison, and L. C. Farrar, Irradiation Effects Test Series, Scoping Test 1, Test Results Report, TREE-NUREG-1066 (To be Published).
6. J. A. Dearien, L. J. Siefken, et. al., FRAP-T2: A Computer Code for the Transient Analysis of Oxide Fuel Rods, TREE-NUREG-1040 (March, 1977).
7. L. S. Tong, "Critical Heat Fluxes in Rod Bundles", pp 31-46 of Two-Phase Flow and Heat Transfer in Rod Bundles Symposium, ASME Winter Annual Meeting (November 1969).
8. D. C. Groeneveld, An Investigation of Heat Transfer in the Liquid Deficient Regime, AECL-3281 (Rev.) (December 1968 revised August 1969).

9. P. E. MacDonald, et al., MATPRO - A handbook of Material Properties For Use in the Analysis of Light Water Reactor Fuel Behavior, ANCR-1263 (February 1976).
10. W. J. Quapp and P. E. MacDonald, Pellet-Cladding Interaction of Mixed Oxide Fuel During Startup, Paper E-6, Presented at the Enlarged Halden Program Group Meeting (March 1973).
11. I. Ruyter, et al., Factors Affecting Cladding Elongation and Relaxation of Strain, Presented at Enlarged Halden Programme Group Meeting, Sanderstolen, Norway (8th - 12th March 1976).

APPENDIX A

INSTRUMENTATION AND DATA ACQUISITION SYSTEM ACCURACIES

The calibration equations that were used to convert the transducer output to engineering units during the data reduction process for the IE-1 Test 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 system acquisition error, and (c) measurement error. Estimates of both the instrument calibration error and data system 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 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 the measurement error is not available. A qualitative assessment of the systematic measurement error allows the conclusions that the systematic measurement errors are not significant except in the case of the spring-loaded thermocouples and that the measurements provided by the other instruments should reflect values of the physical parameter measured, within the error estimates specified in this appendix. This assumption is also questionable with respect to the plenum temperature measurement. Also, this statement is made assuming that the instruments do not decalibrate prior to or during the experiment. As indicated by the test data for fuel rod internal pressure, significant errors due to decalibration can occur. Posttest calibrations were not performed on the test instrumentation to verify this assumption.

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 (calibration error) for each equation. The data system error and linear sum of the data system error and instrument calibration error is also provided.

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

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

where L_i = is the confidence interval estimate

and 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)

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

n = number of variates

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 is assumed to be specified by the accuracy limits as established by a manufacturer or an ANSI standard. The type of error estimate used for each transducer is specified below:

Test Train Instrumentation

(1) System Pressure. One 69 MPa and one 17 MPa strain post pressure transducer were used to monitor the coolant pressure. The 17 MPa pressure transducer (SN 155) failed prior to the test, therefore only the calibration for SN 156 is presented. As discussed in Section 3, this transducer drifted excessively.

(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 taken to be 0.375% of 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 for the inlet temperature thermocouples.

(4) Coolant Temperature Rise. One copper-constantan differential temperature measurement thermocouple pair was positioned in each flow shroud. Each thermocouple pair was calibrated for inlet temperatures of 583 and 605 K for a 20 K temperature rise at each inlet temperature. All of the differential thermocouples correlated well.

(5) Coolant Flow. Four flowmeters using 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). These calibrations were conducted with a slightly different inlet and outlet flow geometry than was used during the test. It was assumed 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 the cladding elongation. The calibration equations for these devices are shown on Table A-I.

The output of each LVDT is sensitive to the transducer temperature, the 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 electronics 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 output was 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 electronics settings (phase and gain) i.e., than 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 LVDT was 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 did not result in an exact LVDT calibration, but it was estimated that this method resulted in a calibration function that was within 2% of the actual calibration.

Additionally, the LVDTs on Rods IE-008 and IE-010 underwent a reduction in output for a given displacement at case temperatures between 580 and 610 K. This change in output was associated with a change in the ferro-magnetic properties of the transducer shield material in this temperature range. Tests with similarly constructed LVDTs have provided an estimate of this variation. Thus, a correction factor was applied to the calibrations for the LVDTs on Rods IE-008 and IE-010.

Due to the above factors, a precise estimate of the 95% confidence interval is not possible, but it is estimated that the calibrations used described the cladding elongation changes within 5%. The calibration procedure outlined above is not considered optimum, but practical limitations prevented correcting the deficiency in time for this test.

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 and that any zero offset usually occurs during the pressure decrease portion of a cycle. Consequently, the pressure differentials indicated during power increases are likely to be an accurate measure of the fuel rod internal pressure change.

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. Design changes are currently underway to resolve this problem.

(2) Plenum Temperature. A chromel-alumel (Type K) thermocouple was installed in the plenum of Rods IE-008 and IE-009 to monitor plenum gas temperature variations. These thermocouples were stainless sheathed and had an ungrounded junction, the end of which was swaged to 0.46 mm. Standard calibration tables were used for the conversion of thermocouple output to temperature.

(3) Cladding Surface Temperature. Uncalibrated tungsten-rhenium thermocouples were used for the spring-loaded thermocouples in making the cladding surface temperature measurements. Again, standard calibration tables were used to convert these data.

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 are 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:

$$V_o = (V_I - V_Z) G \quad (A-5)$$

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, SV_o , can then be determined to be

$$SV_o = \left[V_o^2 \frac{SG^2}{G^2} + G^2 SV_Z^2 + G^2 SV_I^2 \right]^{1/2} \quad (A-6)$$

For the calibration, SV_I can be considered to be zero such that

$$SV_o = \left[V_o^2 \frac{SG^2}{G^2} + G^2 SV_Z^2 \right]^{1/2} \quad (A-7)$$

A test performed to evaluate the data system resulted in values of $\frac{SG}{G}$ and $G SV_Z$ to be 0.0065 and 2.7 millivolts respectively.

For a total span ($Y_{max} - Y_{min}$) on the recording channel of 1000 millivolts, Equation A-7 can be simplified to

$$\frac{SY}{Y_{max} - Y_{min}} \% = \left[\left(\frac{Y - Y_{min}}{Y_{max} - Y_{min}} \right)^2 \times (0.65\%)^2 + (0.27\%)^2 \right]^{1/2} \quad (A-8)$$

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.

TABLE A-I

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTY IN EXPERIMENTAL MEASUREMENTS.

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
System Pressure	69 MPa Strain Pos. (SN: 156)		$MPa = 122.4 - 0.02465(T) + 10.233(mV)$	0 25 50 75 100	(MPa) 0 10.3 21.7 31.0 41.4	(MPa) 0.22 0.27 0.35 0.46 0.58	(MPa) 0.76 0.71 0.70 0.71 0.76	(MPa) 0.79 0.76 0.78 0.85 0.96
Coolant Inlet and Outlet Temperature	Chromel-Arume1 Thermocouple (All Devices)	[a]	$K = 343.6 + 24.05 (mV)$	0 25 50 75 100	(K) 578 590 602 614 626	(K) 0.3 0.3 0.4 0.5 0.7	(K) 1.1 1.2 1.2 1.3 1.3	(K) 1.1 1.2 1.3 1.4 1.5

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

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTY IN EXPERIMENTAL MEASUREMENTS.

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
					(cm ³ /s)	(cm ³ /s)	(cm ³ /s)	(cm ³ /s)
Coolant Flow	Turbine Flowmeter (SN 329)	IE-007	$\text{cm}^3/\text{s} = -0.5718$ $+0.4213(\text{mV})$	0	0	6.8	1.9	7.1
				25	315.8	8.1	1.8	8.3
				50	631.7	10.6	1.7	10.7
				75	947.5	14.1	1.8	14.2
				100	1263.3	17.7	1.9	17.8
Coolant Flow	Turbine Flowmeter (SN 325)	IE-008	$\text{cm}^3 = -0.4107$ $+0.4203 (\text{mV})$	0	0	6.8	1.4	6.9
				25	315.5	8.1	1.3	8.2
				50	630.9	10.6	1.3	10.7
				75	946.4	14.1	1.3	14.2
				100	1261.8	17.7	1.4	17.8
Coolant Flow	Turbine Flowmeter (SN 328)	IE-009	$\text{cm}^3 = 1.087$ $+ 0.4234 (\text{mV})$	0	0	6.8	0.8	6.9
				25	317.8	8.1	0.8	8.1
				50	635.6	10.6	0.7	10.6
				75	953.5	14.1	0.8	14.1
				100	1271.3	17.7	0.8	17.7

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTY IN EXPERIMENTAL MEASUREMENTS.

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
Coolant Flow	Turbine Flowmeter (SN 327)	IE-010	$\text{cm}^3 = -0.5684$ $+ 0.4214 \text{ (mV)}$	0	$\frac{\text{cm}^3}{\text{s}}$ 0	$\frac{\text{cm}^3}{\text{s}}$ 6.8	$\frac{\text{cm}^3}{\text{s}}$ 3.6	$\frac{\text{cm}^3}{\text{s}}$ 7.7
				25	35.5	8.1	3.3	8.8
				50	630.9	10.6	3.2	11.1
				75	946.4	14.1	3.3	14.5
				100	1261.8	17.7	3.6	18.1
Coolant Temperature Differential	Copper Constantan Thermocouple (Type T) (SN DTC-7)	IE-007	$K = 16.10 \text{ (mV)}^{[b]}$	0	$\frac{\text{K}}{}$ 0	$\frac{\text{K}}{}$ 0.12	$\frac{\text{K}}{}$ 0.12	$\frac{\text{K}}{}$ 0.17
				25	5.60	0.14	0.11	0.18
				50	11.20	0.19	0.11	0.22
				75	16.80	0.25	0.11	0.27
				100	22.40	0.31	0.12	0.34
	(SN DTC-10)	IE-008	$K = 16.12 \text{ (mV)}^{[b]}$	0	0	0.12	0.08	0.14
				25	5.60	0.14	0.08	0.16
				50	11.20	0.19	0.07	0.20
				75	16.80	0.25	0.08	0.26
				100	22.40	0.31	0.08	0.32

[b] These equations are applicable to coolant inlet temperature of 605 K.

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTY IN EXPERIMENTAL MEASUREMENTS.

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
Coolant Temperature Differential	(SN DTC-8)	1E-009	$K = 16.08 \text{ (mV)}^{[b]}$		(K)	(K)	(K)	(K)
				0	0	0.12	0.07	0.14
				25	5.60	0.14	0.07	0.16
				50	11.20	0.19	0.07	0.20
				75	16.80	0.25	0.07	0.26
				100	22.40	0.31	0.08	0.32
	(SN DTC-9)	1E-010	$K = 16.08 \text{ (mV)}^{[b]}$					
				0	0	0.12	0.16	0.20
				25	5.60	0.14	0.15	0.21
				50	11.20	0.19	0.15	0.24
				75	16.80	0.25	0.16	0.30
				100	22.40	0.31	0.17	0.36

[b] These equations are applicable to coolant inlet temperatures of 605 K.

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTY IN 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 256)	IE-007	MPa = - 1.244 - .0000651 (T) + 3.291 (mV)	0	(MPa) 0	(MPa) 0.09	(MPa) 0.06	(MPa) 0.11
				25	4.3	0.11	0.04	0.12
				50	3.6	0.14	0.04	0.15
				75	12.9	0.14	0.04	0.20
				100	17.2	0.24	0.06	0.25
	(SN 259)	IE-008	MPa = - 1.200 + 0.0005245 (T) - 3.850 (mV)	0	0	0.09	0.08	0.12
				25	4.3	0.11	0.06	0.13
				50	3.6	0.14	0.05	0.15
				75	12.9	0.19	0.06	0.20
				100	17.2	0.24	0.09	0.26
	(SN 258)	IE-009	MPa = -7.788 + - 0.0005126 (T) + 3.401 (mV)	0	0	0.09	0.06	0.11
				25	4.3	0.11	0.04	0.12
				50	3.6	0.14	0.04	0.15
				75	12.9	0.19	0.04	0.20
				100	17.2	0.24	0.06	0.25
	(SN 235)	IE-010	MPa = - 2.150 + 0.001567 (T) + 3.356 (mV)	0	0	0.09	0.06	0.11
				25	4.3	0.11	0.05	0.12
				50	8.6	0.14	0.04	0.15
				75	12.9	0.19	0.05	0.20
				100	17.2	0.24	0.06	0.25

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTY IN 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-039)	IE-007	$\text{mm} = 0.105 \times 10^{-1} \text{ mV}_2$ $- 0.717 \times 10^{-6} \text{ mV}_3$ $+ 0.751 \times 10^{-9} \text{ mV}^3 \text{ [d]}$	0	0	[c]	[c]	[c]
				25	2.63	[c]	[c]	[c]
				50	5.25	[c]	[c]	[c]
				75	7.875	[c]	[c]	[c]
				100	10.50	[c]	[c]	[c]
	LVDT (SN-212)	IE-008	$\text{mm} = 0.138 \times 10^{-1} \text{ mV}_2$ $- 0.416 \times 10^{-5} \text{ mV}_3$ $+ 0.404 \times 10^{-8} \text{ mV}^3 \text{ [d]}$	0	0	[c]	[c]	[c]
				25	3.42	[c]	[c]	[c]
				50	6.84	[c]	[c]	[c]
				75	10.26	[c]	[c]	[c]
				100	13.68	[c]	[c]	[c]
	LVDT (SN 433)	IE-009	$\text{mm} = 0.120 \times 10^{-1} \text{ mV}_2$ $- 0.244 \times 10^{-5} \text{ mV}_3$ $+ 0.134 \times 10^{-8} \text{ mV}^3 \text{ [d]}$	0	0	[c]	[c]	[c]
				25	2.73	[c]	[c]	[c]
				50	5.45	[c]	[c]	[c]
				75	8.18	[c]	[c]	[c]
				100	10.90	[c]	[c]	[c]
	LVDT (SN-210)	IE-010	$\text{mm} = 0.145 \times 10^{-1} \text{ mV}_2$ $- 0.594 \times 10^{-5} \text{ mV}_3$ $+ 0.527 \times 10^{-8} \text{ mV}^3 \text{ [d]}$	0	0	[c]	[c]	[c]
				25	3.46	[c]	[c]	[c]
				50	6.92	[c]	[c]	[c]
				75	10.37	[c]	[c]	[c]
				100	13.83	[c]	[c]	[c]

[c] The calibration error is not defined for this instrument. See explanation in this appendix.

[d] These equations are applicable to coolant inlet temperatures of 600 K.

TABLE A-I (Continued)

SUMMARY OF CALIBRATION EQUATIONS AND UNCERTAINTY IN EXPERIMENTAL MEASUREMENTS.

Measurement	Transducer (serial number)	Fuel Rod Number	Calibration Equation	Output (% of Span)	Output	Data System Error	Calibration Error	Total Error
Plenum Temperature	Chromel-Alumel Thermocouple (Type K)	[a]	$K = 343.6$ $+ 24.05 \text{ (mV)}$	0	540	1	2	2
				25	600	2	2	3
				50	660	2	3	3
				75	720	3	3	4
				100	780	3	4	5
Cladding Surface Temperature	W5%Re/W26%Re Thermocouple	[a]	$K = 351.2$ $+ 58.50 \text{ (mV)}$ $- 0.6598 \text{ (mV)}^2$ $+ 0.02044 \text{ (mV)}^3$	0	540	10	2	10
				25	990	11	4	12
				50	1430	15	5	16
				75	1870	20	7	21
				100	2320	25	9	26

REFERENCES

- A-1. M. G. Natrella, Experimental Statistics, National Bureau of Standards Handbook 91, August, 1963.
- A-2. W. J. Quapp, C. M. Allison, and L. C. Farrar, Irradiation Effects Test Series, Scoping Test 1, Test Results Report, TREE-NUREG-1066 (to be published).

APPENDIX B

FUEL ROD CHARACTERIZATION DATA

The pretest fuel rod characterization data for the IE-1 test 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 prediction of fuel rod behavior during nuclear operations and (c) provide input for post-test 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 only modifications made to each of the fuel rods was to remove the original upper end cap and replace it with an instrumented one. The fuel rods were then backfilled with a 77.7% He - 22.3% Ar gas mixture. Table B-1 shows the preirradiated (nominal design values) fuel pellet and cladding data for these fuel rods. Table I, Section 2, provides a summary of additional fuel rod data particular to each fuel rod.

Figure B-1 provides a plot of the pretest cladding outside diameter dimensions obtained using the Pulsed Eddy Current Defect Inspection System equipped with LVDTs setup to measure rod diameter and rod bow. Cladding measurements were made at two azimuthal orientations, 90° apart. These measurements are reported at discrete 25.4 mm intervals over the length of the fuel rod. The two diametral measurements at each interval were averaged to provide data for the plot (Figure B-1).

TABLE B-I

PRETEST FUEL ROD AND CLADDING DATA

FINISHED CLADDING DATA: [a]

MEAN O/D	-	9.931 mm
MEAN I/D	-	8.750 mm
OVERALL LENGTH	-	969.4 mm

MECHANICAL PROPERTIES OF FUEL ROD CLADDING: [b]

YIELD TENSILE STRENGTH

AT ROOM TEMP	-	$5.72 \times 10^8 \text{ N/m}^2$
AT 658 K	-	$3.89 \times 10^8 \text{ N/m}^2$

ULTIMATE TENSILE STRENGTH

AT ROOM TEMP	-	$7.73 \times 10^8 \text{ N/m}^2$
AT 658 K	-	$4.90 \times 10^8 \text{ N/m}^2$

FUEL DATA: [a]

MEAN GEOMETRIC DENSITY	IE-007	-	94 % TD
	IE-008	-	94 % TD
	IE-009	-	92 % TD
	IE-010	-	92 % TD

MEAN DIAMETER	IE-007	-	8.48 mm
	IE-008	-	8.48 mm
	IE-009	-	8.53 mm
	IE-010	-	8.53 mm

MEAN PELLET LENGTH	-	15.24 mm
--------------------	---	----------

STACK LENGTH	-	879. mm
--------------	---	---------

DISH DIMENSIONS

DISH CHORD	-	6.60 mm
DISH DEPTH	-	0.343 mm

GRAIN SIZE: [b]

ASTM MICROGRAIN SIZE	-	No. 13.2
GRAIN DIAMETER	-	$3.7 \times 10^{-3} \text{ mm}$

PLENUM LENGTH	-	50.8 mm
---------------	---	---------

VOID VOLUME	IE-007	-	10.2 cm ³
	IE-008	-	9.1 cm ³
	IE-009	-	10.2 cm ³
	IE-010	-	11.1 cm ³

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

[b] Data supplied by Westinghouse Electric Corporation.

Fig. B-1 Pre-PBF test mean cladding outside diameter for Rods IE-007, IE-008, IE-009, and IE-010 as determined by a Pulsed Eddy Current Defect Inspection System with LVDT's set up to measure the fuel rod diameter.

APPENDIX C

TEST DATA

All of the test data from Irradiation Effects Test 1 were digitized, processed on an IBM 360 using the MAC/RAN Time Series Analyses Program^[1], and permanently stored on a nine track magnetic tape in a MAC/RAN SIDU (Standard Input Data Unit) format. Each data channel (the data from each measurement) contains approximately 20,000 data values with a variable digitizing interval from 10 seconds down to 0.25 seconds. All of the response plots (data versus time plots) were made using data which had been decimated by a factor of 10.

The data was converted from electrical units (volts) to SI engineering units using instrument calibration equations and then, any wild points were removed. In addition, the coolant ΔT had small zero power offsets removed using an additive constant (see Table C-1) such that the data would indicate zero power at zero reactor power. This was done to remove the errors caused by a zero drift in the electronics as discussed in Appendix A.

All of the data channels are presented in this appendix (see Figures C-1 through C-9). Certain key measurements presented in the text are repeated in this section, comparing the same measurements on different rods.

TABLE C-I

ADDITIVE CORRECTION FACTORS APPLIED TO COOLANT AT SIGNAL

Rod Number	Additive Correction Factor		
	Power Calibration	Gap Conductance	Remainder of Test
IE-007	0.0	0.0	0.0
IE-008	-0.130	-0.129	-0.130
IE-009	0.0	0.0	0.0
IE-010	+0.336	+0.336	+0.336

Fig. C-1 - Fuel rod internal pressures for Test IE-1.

Fig. C-2 - Cladding elongation for Test IE-1.

Fig. C-3 - Coolant flow rates for Test IE-1.

Fig. C-4 - Coolant outlet temperatures and plenum temperatures for Test IE-1.

Fig. C-5 - Coolant inlet temperature and coolant temperature increases for Test IE-1.

Fig. C-6 - Neutron flux at several axial elevations for Test IE-1.

Fig. C-7 - Reactor power and fission loop activity for Test IE-1.

Fig. C-8 - Cladding surface temperatures for Rods IE-007, IE-008, and IE-010 for Test IE-1.

Fig. C-9'- Cladding surface temperatures for Rod IE-0C9 for Test IE-1.

APPENDIX D

POWER CALIBRATION RESULTS

Average test rod power, expressed in terms of linear heat rating, was calculated using a thermal heat balance. The thermal balance was calculated with a computer program based on the equations presented in Reference D-1. Water properties were calculated by the ASTEM code^[D-2]. Because of the large amount of data, the thermal balance was calculated for the first cycle of the test (see Figure 8), the first part of the gap conductance testing-Cycle 3, and Cycle 5 and the first part of Cycle 6. A linear regression was then made between the thermal hydraulically determined average rod power for each rod and SPND 4 for those parts of the test. The regression relationships to SPND 4 were used as follows: (a) for detailed plots, rod power was represented by SPND 4 using the appropriate equation for each section of the test, and (b) the response curves showing the entire test used SPND 4 with a combined equation to represent rod power. The thermal hydraulically determined average rod power was used for any calculations such as the determination of variances for the factorial design experiment run during Cycles 1 and 2.

Local power (the power profile along the rod) was calculated using the normalized fluence profile measured by the cobalt wire inserted in the test train. The flux wire determined local power is an integrated profile covering the length of the test at all power levels. This local power was verified against the instantaneous profiles calculated from the SPND signals. These instantaneous profiles indicated a slight shift in the flux peak versus reactor power. However, this is limited to reactor powers producing a rod power of 10 kW/m or less. No shift was apparent beyond that power up to the maximum power used in the test (70 kW/m).

Thermal Balance Results

All of the instruments necessary for a thermal balance for each rod were available throughout the test. Therefore, a comparison between the average power in each rod was possible to check for any unexpected variation in rod power due to instrument drift or asymmetry in the neutron flux field from PBF reactor control rod changes; none were found. The slight variations in power between rods due to rod design or assembly remained consistent during the test. The ratios of individual rod power to average rod power, shown in Table D-I, remained constant within the listed uncertainties.

TABLE D-I
RATIOS OF ROD POWER TO AVERAGE ROD POWER

<u>Power Ratios</u>	<u>Mean</u>	<u>95% Confidence Interval</u>
Rod IE-007/Average	1.038	0.014
Rod IE-008/Average	0.971	0.013
Rod IE-009/Average	0.989	0.013
Rod IE-010/Average	1.002	0.021

The linear regression between SPND 4 and rod average power (average power for each rod) indicated a high degree of fit and was relatively insensitive to change in moderator density due to inlet temperature change. The resulting equations are shown in Table D-II.

The uncertainties in rod power for Rod IE-007 through IE-010 were calculated using both instrument calibration errors and data acquisition errors. These results are shown in Table D-III.

TABLE D-II
LINEAR REGRESSION EQUATIONS RELATING NEUTRON FLUX
MEASURED BY SPND 4 AND ROD POWER

Rod Number	Test ^[a] Phase	Regression Equation	Uncertainty In Power ^[b] (kW/m)
IE-007	Cycles 1 and 2	$RP = 16.932 \times 10^{-13} SPND4 - 0.20$	0.893
	Cycle 3	$RP = 17.866 \times 10^{-13} SPND4 - 0.71$	10.349
	Cycles 4, 5, and 6	$RP = 15.365 \times 10^{-13} SPND4 + 1.40$	1.444
	Combined	$RP = 15.468 \times 10^{-13} SPND4 + 1.35$	1.877
IE-008	Cycles 1 and 2	$RP = 15.854 \times 10^{-13} SPND4 - 0.25$	4.737
	Cycle 3	$RP = 15.532 \times 10^{-13} SPND4 - 0.31$	5.333
	Cycles 4, 5, and 6	$RP = 14.263 \times 10^{-13} SPND4 + 1.58$	1.044
	Combined	$RP = 14.418 \times 10^{-13} SPND4 + 1.32$	3.728
IE-009	Cycle 1 and 2	$RP = 15.700 \times 10^{-13} SPND4 - 0.22$	0.684
	Cycle 3	$RP = 18.157 \times 10^{-13} SPND4 - 1.27$	21.228
	Cycles 4, 5, and 6	$RP = 14.529 \times 10^{-13} SPND4 + 1.54$	1.038
	Combined	$RP = 14.742 \times 10^{-13} SPND4 + 0.94$	1.298
IE-010	Cycles 1 and 2	$RP = 15.638 \times 10^{-13} SPND4 - 0.09$	0.804
	Cycle 3	$RP = 16.510 \times 10^{-13} SPND4 - 0.75$	11.838
	Cycles 4, 5, and 6	$RP = 15.179 \times 10^{-13} SPND4 + 0.50$	1.079
	Combined	$RP = 15.238 \times 10^{-13} SPND4 + 0.37$	1.025
Average	Cycles 1 and 2	$RP = 16.031 \times 10^{-13} SPND4 - 0.19$	1.298
	Cycle 3	$RP = 17.016 \times 10^{-13} SPND4 - 0.76$	11.965
	Cycles 4, 5, and 6	$RP = 14.834 \times 10^{-13} SPND4 + 1.26$	1.045
	Combined	$RP = 14.966 \times 10^{-13} SPND4 + 1.00$	1.571

[a] The much larger uncertainty in Cycle 3 (the gap conductance test) is due to the aliasing problem. Not only is the frequency content altered during this time due to a large digitizing interval but more importantly for a regression analysis, phase shift has taken place.

[b] 95% confidence intervals.

TABLE D-III
UNCERTAINTIES IN ROD POWER^[a]

<u>Average Rod Power (kW/m)</u>	<u>Total Error^[b] (kW/m)</u>
0.0	1.58
28.9	1.65
57.7	1.79

[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.

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

Calculation of Local Power

Local power was obtained using the data from SPNDs and the cobalt wire. The normalized axial flux profile obtained from the SPNDs gave the instantaneous profile which can show any skewing as the power is changed. This profile was calculated by fitting a sine function to the SPND currents from SPNDs at distributed elevations. Only the upper SPNDs (SPND 3 to SPND 8) were available for the entire test^[a], so the

[a] SPND 2 was available the last half of the test. However, there appears to be a decalibration of the device since use of it's signal in the fit causes unreasonable results.

instantaneous power profile was much better defined for the upper half of the rod^[a].

Any appreciable skewing in the power profile due to changes in control rod position was limited to a rod power less than 10 kW/m. At the high power hold period (~70 kW/m), the peak-to-average as indicated by the fit to the SPND signals was slightly reduced from 1.32 to 1.28. This was due mainly to the increase in inlet temperature from 600 to 605 K during the power increase (see Figure 8). The effect of this change was limited to the lower half of the rod so the local power on the upper half of the rod remained fixed. As seen in Figure D-1, there was some effect upon the local power due to voiding in the shroud during the flow reduction. The peak-to-average was reduced and the peak shifted upward on the rod. The average neutron flux also appeared to be decreasing as the flow was reduced. The effect of voiding can be seen clearly on Figures D-2 and D-3. These plots show peak-to-average and average neutron flux (the flux profile integral averaged over the length of the fuel stack). Peak-to-average, a very weak function of rod power with a regression equation of

$$\text{Peak-to-Average} = 0.00131 \times \text{Average Rod Power} + 1.33$$

showed a shift as the flow was reduced. The average flux, which is linearly related to power with a regression equation of

$$\text{Average flux} \times 10^{-13} = 0.0864 \text{ Average Rod Power} - 0.00103,$$

[a] The SINE function fit, which represents the local power profile over the full length of the rod, had a limited number of sample points spanning the upper half of the rod only. Therefore, the bottom half of the profile is an extrapolation of these few points. Small indicated local shifts in the peak, the peak-to-average (since the peak occurs in the lower half of the rod) might have small systematic biases because of the extrapolation to the bottom half of the rod. This effect is expected to be small based upon comparison with the flux wire results and results from previous tests [D-2].

Fig. D-1 Histories for peak-to-average, peak power elevation, average neutron flux (integrated over length of fuel stack), and average rod power from just prior to and including the flow reduction for Test 1E-1.

Fig. D-2 Peak power elevation versus rod average power for Cycle 5, power increase and flow reduction for Test IE-1.

Fig. D-3 Peak to average versus rod average power for Cycle 5, power increase and flow reduction for Test IE-1.

also showed a shift. Both of these shifts have only negligible effects on average and local rod power. Since average rod power for the flow reduction is determined by using the linear fit between SPND 4 and rod power, average rod power shows a slight decrease (<1%) during the flow reduction. This is because the drop in average flux is due to drops in thermal flux seen by both SPND 4 and SPND 5. Physics calculations have indicated that the effect of the voiding could be lessened for the fuel rod itself. Although the moderator density in the shroud is reduced, with a resulting drop in thermal neutrons around the fuel, the neutron absorption is also reduced allowing more of the neutrons to reach the fuel. The shift in peak elevation is not taken into account. Since the local power profile is relatively flat, the change in local power due to a shift in peak elevation is well within the uncertainty limits calculated for the profile. Normalized flux profiles (normalized local power profiles) are shown in Figure D-4 for the high power hold period and the last two cycles of preconditioning at inlet temperatures of 600 and 605 K, respectively. The 95% confidence intervals are also shown. These results are also tabulated in Table D-IV.

The normalized local power profile, obtained from a gamma scan of the flux wire at 0.3-inch intervals, gives a profile integrated over time and all power levels. As seen in Figure D-4, this profile is indicative of the power distribution experienced during most of the test. It had a peak-to-average of 1.33 with a standard deviation of 0.053 and a peak elevation of 0.378 with a standard deviation of 0.038. FRAP-T calculations (see Section 5) were made using this profile because of the close agreement with the instantaneous profiles and the much larger number of points defining the curve (150 as opposed to 3 for the SPNDs).

TABLE D-IV

LOCAL POWER PROFILE WITH UNCERTAINTIES BASED ON SPND OUTPUT

Elevation	Cycles 4 and 5 ^[a]		Cycle 6 ^[b]	
	Normalized Local Power	95% Confidence Intervals	Normalized Local Power	95% Confidence Intervals
0.013	0.509	± 0.036	0.494	± 0.014
0.159	1.154	0.060	1.115	0.026
0.306	1.312	0.024	1.264	0.014
0.452	1.186	0.034	1.166	0.009
0.521	1.109	0.045	1.113	0.016
0.599	1.023	0.045	1.056	0.020
0.662	0.941	0.039	0.992	0.020
0.724	0.833	0.031	0.894	0.018
0.745	0.788	0.028	0.849	0.017
0.892	0.328	0.011	0.361	0.007

[a] Average Peak Elevation: 0.301 ± 0.024
 Average Peak-to-Average: 1.313 ± 0.022

[b] Average Peak Elevation: 0.308 ± 0.011
 Average Peak-to-Average: 1.266 ± 0.012

Fig. D-4 Local power profile for Cycles 4, 5, and 6 (based on SPND signals) and entire test (based on flux wire data).

REFERENCES

- D-1 W. J. Quapp, C. M. Allison, and L. C. Farrar, Irradiation Effects Test Series, Scoping Test 1, Test Results Report, TREE-NUREG-1066 (to be published).
- D-2 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).

DISTRIBUTION RECORD FOR TREE-NUREG-1046

Internal Distribution

- 1 - Chicago Patent Group - ERDA
9800 South Cass Avenue
Argonne, Illinois
- 2-4 - A. T. Morphew
Classification and Technical Information Officer
Idaho Operations Office - ERDA
Idaho Falls, ID 83401
- 5 - R. J. Beers, ID
- 6 - P. E. Litteneker, ID
- 7 - R. E. Tiller, ID
- 8 - R. E. Wood, ID
- 9 - H. P. Pearson, Supervisor
Technical Information
- 10-19 - INEL Technical Library
- 20-39 - Authors
- 40-184 - Special Internal

External Distribution

- 185-186 - Saul Levine, Director
Office of Nuclear Regulatory Research, NRC
Washington, D.C. 20555
- 187-507 - Distribution under NRC-3, Water Reactor Safety Research
Fuel Behavior