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TREE-NUREG-1177

for U.S. Nuclear Regulatory Commission

POSTIRRADIATION EXAMINATION RESULTS FOR THE IRRADIATION EFFECTS TEST SERIES IE-ST-2 ROD IE-002

BEVERLY A. MURDOCK

MASTER

December 1977

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

DEPARTMENT OF ENERGY

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

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TREE-NUREG-1177

POSTIRRADIATION EXAMINATION RESULTS FOR THE IRRADIATION EFFECTS TEST IE-ST-2 ROD IE-002

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TREE-NUREG-1177

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

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FOR THE IRRADIATION EFFECTS TEST
IE-ST-2 ROD IE-002

By

Beverly A. Murdock

EG&G IDAHO INC.

December 1977

PREPARED FOR THE
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ABSTRACT

A postirradiation examination was conducted on a zircaloy-clad, UO_2-fueled, pressurized water reactor (PWR) type rod which had been tested in the Power Burst Facility as part of the Irradiation Effects Test Series of the Thermal Fuels Behavior Program. The fuel rod, previously irradiated to a burnup of 15 800 MWd/t was subjected to a power ramp from 28 to 55 kW/m peak power at an average ramp rate of 4 kW/m/min. Posttest fuel restructuring and relocation, fission product redistribution, and fuel rod cladding deformation were evaluated and analyzed.

SUMMARY

A PWR-type, zircaloy-clad, UO_2-fueled rod was examined after testing in the Thermal Fuels Behavior Program Irradiation Effects Test Series Test IE-ST-2. The test was performed in a pressurized water reactor environment in the in-pile test loop of the Power Burst Facility (PBF). The posttest examination was conducted to determine the condition of a previously irradiated (15 800 MWd/t) fuel rod which had been exposed to a power ramp from 28 to 55 kW/m rod peak power at an average ramp rate of 4 kW/m/min.

Extensive fuel cracking and possible central void formation was observed in a region just above the rod axial peak power location. The fission product cesium was determined to have migrated away from this region and toward the bottom of the fuel rod. Both of these observations suggest fuel temperatures were high and fuel melting may have occurred in this region. This region, as well as another region about one-third of the distance from the bottom of the rod, showed substantial cladding deformation. The maximum ovality (ratio of major to minor diameter) observed was 1.029 and the maximum decrease in wall thickness was 17% of the original wall thickness.

The rod did not fail as a result of the power ramp even though relatively large cladding deformations occurred, probably as a result of pellet/cladding interaction. The cause of the apparent high fuel temperatures and unusually large cladding deformations cannot be defined at this time.

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POSTIRRADIATION EXAMINATION RESULTS FOR THE IRRADIATION EFFECTS TEST IE-ST-2 ROD IE-002

I. INTRODUCTION

Fuel rod behavior studies are being conducted in the Thermal Fuels Behavior Program (TFBP) as part of the Nuclear Regulatory Commission's Water Reactor Safety Research Fuel Behavior Program[1]. The TFBP is conducted by EG&G Idaho, Inc., under contract to the Department of Energy. The work is performed in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory (INEL). Experimental data are being obtained from in-pile tests and postirradia-tion examinations for use in the verification of analytical models being developed to predict the behavior of light water nuclear reactor fuel rods under normal and postulated accident conditions.

The Irradiation Effects Test IE-ST-2 was conducted to provide data on the behavior of irradiated fuel rods subjected to a power ramp and to film boiling operation. The Test IE-ST-2 results have been compared to model calculations and are reported in Reference 2. The postirradiation examination results from the two rods which operated in film boiling are reported in Reference 3.

The purpose of this report is to document the results of the non-destructive postirradiation examinations performed on Rod IE-002, one of four fuel rods used in Test IE-ST-2. This rod was not examined along with the other rods[3] since it had not experienced film boiling - a major objective for the four rods in Test IE-ST-2. However, this provided an opportunity to determine the condition of a previously irradiated fuel rod[a] which had been exposed to a power ramp without subsequent film boiling operation. The nondestructive examination was performed to evaluate the general condition of the rod and to obtain pellet-cladding interaction strain and fission product redistribution data which can be used in model verification studies.

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[a] Irradiated in the Saxton reactor, a small, prototype pressurized water reactor, designed by Westinghouse Electric Corporation for the USAEC.

A brief description of the Rod IE-002 fuel rod design, previous irradiation history, adaptation for PBF testing, and test history in PBF is provided in Section II of this report. The results of the examination are presented in Section III and are discussed in Section IV. The appendices discuss the measurement techniques used in the examination.

II. EXPERIMENT DESIGN AND CONDUCT

Fuel Rod IE-002 was a previously irradiated fuel rod subsequently operated in the PBF Test IE-ST-2. Section 1 describes the fuel rod dimensions and fabrication procedures as well as the previous irradiation history. Section 2 presents a summary of Test IE-ST-2. These topics are discussed in greater detail in Reference 2.

1. FUEL ROD DESIGN AND IRRADIATION HISTORY

Fuel Rod IE-002 was remotely fabricated from Saxton fuel Rod 829, which had a previous average burnup of 15 800 MWd/t and a peak cladding fluence of approximately 1.1 x 10^{21} n/cm^2 (> 1 MeV). Reported time-averaged peak power for this rod in the Saxton reactor was 39.5 kW/m. The reported axial power peak from the Saxton irradiation was located approximately 0.3 m from the bottom of the rod. The top end cap was replaced with an instrumented end cap and the fuel rod was backfilled with a 77.7% helium and 22.3% argon gas mixture to a pressure of 2.67 MPa. The gas mixture used simulates the thermal conductivity of fill gas and fission gases removed from similar Saxton fuel rods[4]. However, in the several weeks between the time the rod was backfilled and the time it was loaded into the reactor test space, some of the backfill gas had leaked out through the MgO insulation in the 12 m long extension cable connecting the plenum thermocouples to the instrumentation outside of the reactor. At the beginning of the test the rod internal pressure was 2.0 MPa.

The rod was clad with zircaloy-4 tubing having a nominal outside diameter of 9.93 mm and a nominal wall thickness of 0.592 mm. The fuel rod was approximately 0.97 m long (not including the instrumented end cap) and had a nominal active fuel stack length of 0.88 m. The fuel pellets (9.5 wt% ^{235}U) were of a dished-end design. Additional information on the characteristics of Saxton fuel rods is presented in Reference 4.

A 17-MPa strain-post pressure transducer, mounted on the instrumented end cap at the upper end of the rod, was used to monitor fuel rod internal pressure. A sheathed, magnesium-oxide-insulated, Chromel-Alumel (Type K) thermocouple was positioned in the plenum to monitor the plenum gas temperature.

The fuel rod was provided with two, grounded junction, beryllium-oxide-insulated, tungsten-rhenium (W5%Re/W26%Re) alloy wire, zircaloy-sheathed thermocouples to measure the cladding surface temperature. These thermocouples were spring-loaded against the outer surface of the fuel cladding with an approximately three-pound preload. The thermocouples were located 0.62 m above the bottom of the rod.

2. TEST CONDUCT

Rod IE-002 was one of four rods tested in the Irradiation Effects Test IE-ST-2. The test was conducted in a simulated pressurized water reactor (PWR) environment in the PBF in-pile test loop. The test sequence and rod power during the test are shown in Figure 1. A discussion of the methods used to measure and calculate rod powers and uncertainties in these measurements is provided in Reference 2. The rod was operated at peak power levels up to 28 kW/m for a period of about 39 hours. This preconditioning period included nine hours during which a gap conductance test was performed. Following the preconditioning period, the power in Rod IE-002 was ramped at 4 kW/m/min from 28 to 55 kW/m peak power; about 16 kW/m higher than the reported peak power (39.5 kW/m) during Saxton operation[4].

After one hour at this high power level, the coolant flow was reduced to about 55% of the original flow for 30 seconds and then to 50% for an additional 60 seconds to force the rods into film boiling operation. The test was then terminated and rod power quickly reduced to zero. While film boiling did occur on two of the four fuel rods in this test, fuel rod instrumentation indicated film boiling did not occur on Rod IE-002 due to the lower power of this rod relative to the others.

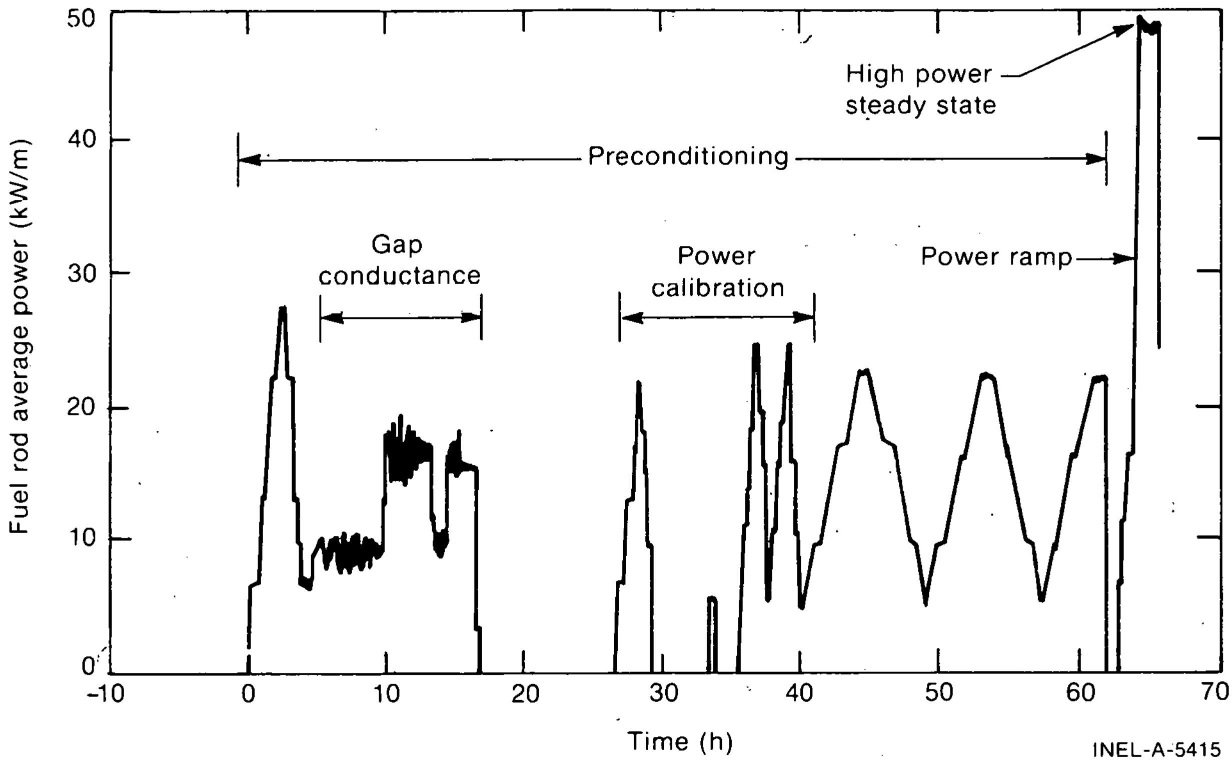

Fig. 1 Fuel rod power for Rod IE-002 during irradiation in Test IE-ST-2.

Figure 2 shows the time-averaged axial power distribution for the previous irradiation of the fuel rod in the Saxton reactor and the normalized axial power profile of the rod from the irradiation in Test IE-ST-2. The power peak from the Saxton irradiation is located approximately 0.3 m from the bottom of the rod and the power peak from Test IE-ST-2 is located at approximately 0.5 m.

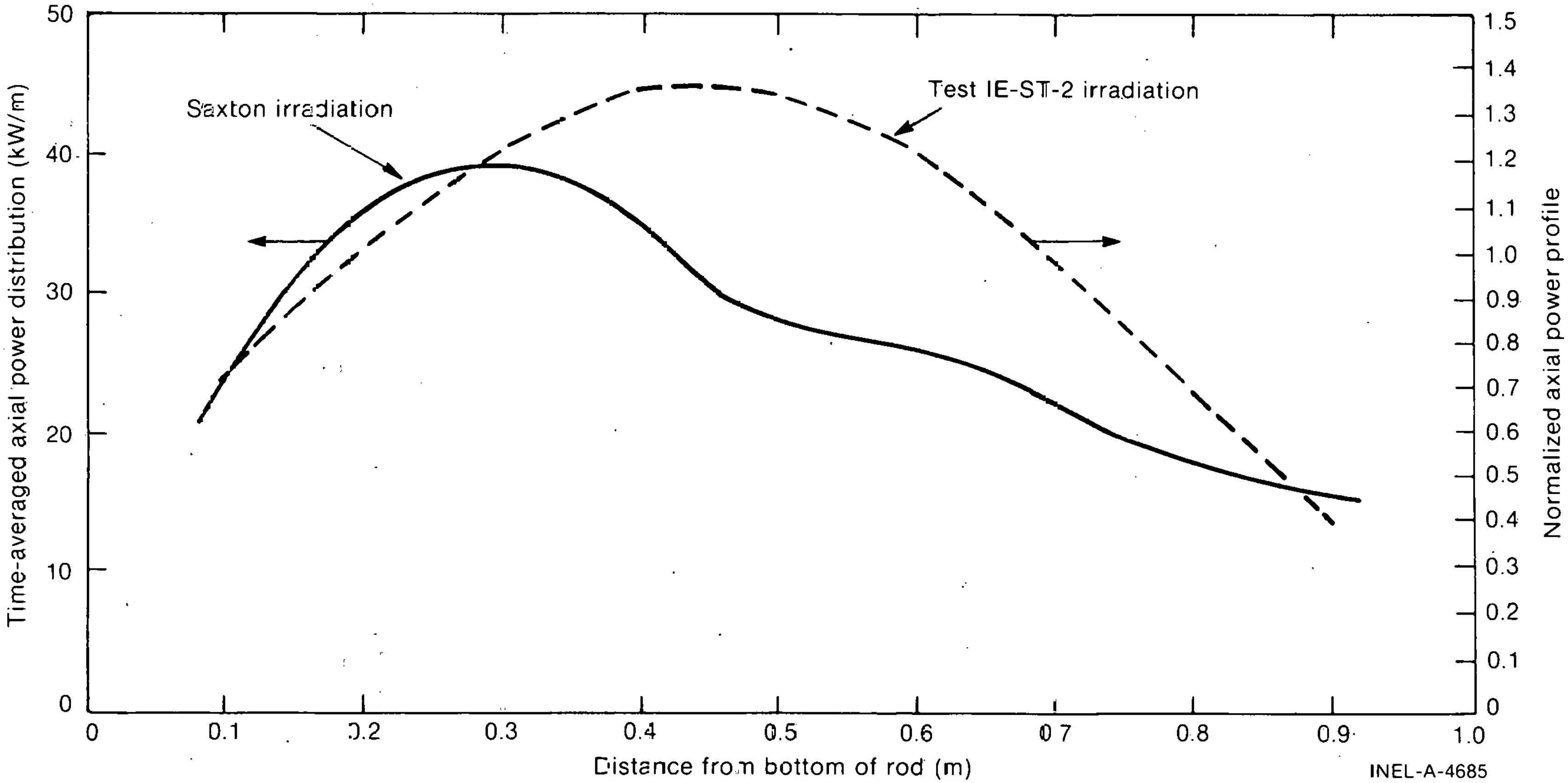

Fig. 2 Axial power profile of Rod IE-002 from Saxton irradiation and Test IE-ST-2 irradiation.

III. EXAMINATION RESULTS

Rod IE-002 was nondestructively examined to determine the overall rod condition, any restructuring or relocation of the fuel, cladding deformation, and fission product redistribution.

1. POSTTEST FUEL ROD CONDITION

Rod IE-002 was examined to establish the overall posttest condition of the rod. The entire rod was covered with a thin, dark oxide. A wear mark at the 0.46-m elevation resulted from contact with one of the fuel rod centering screws.

Figures 3 and 4 show the rod at 0 and 180 degrees, respectively. The peak power location (0.492 m) and the region which contained cracked fuel (0.55 to 0.62 m) (Section III, 2) are indicated in the figures. Some ^{137}Cs migrated away from this region (Section III, 4) and wall thinning also occurred in this region (Section III, 3.2). No discoloration or other obvious change was seen in the cladding outer surface at this location, which confirmed the instrumentation indications that the rod did not operate in film boiling as a result of the flow reduction.

2. FUEL RESTRUCTURING AND RELOCATION

Rod IE-002 was posttest neutrographed to evaluate the restructuring and relocation of the fuel as a result of the test operation. Fuel Rod 829, from which Rod IE-002 was fabricated, had been neutrographed prior to test rod assembly. Figure 5 shows the pre- and posttest neutrographs. The fuel rod was neutrographed at the General Electric Company (GE) Vallecitos Nuclear Center near Pleasanton, California during the pretest nondestructive examination[4]. Following Test IE-ST-2 the rod was neutrographed at the Argonne National Laboratory Transient Reactor Test Facility (TREAT) at the INEL.

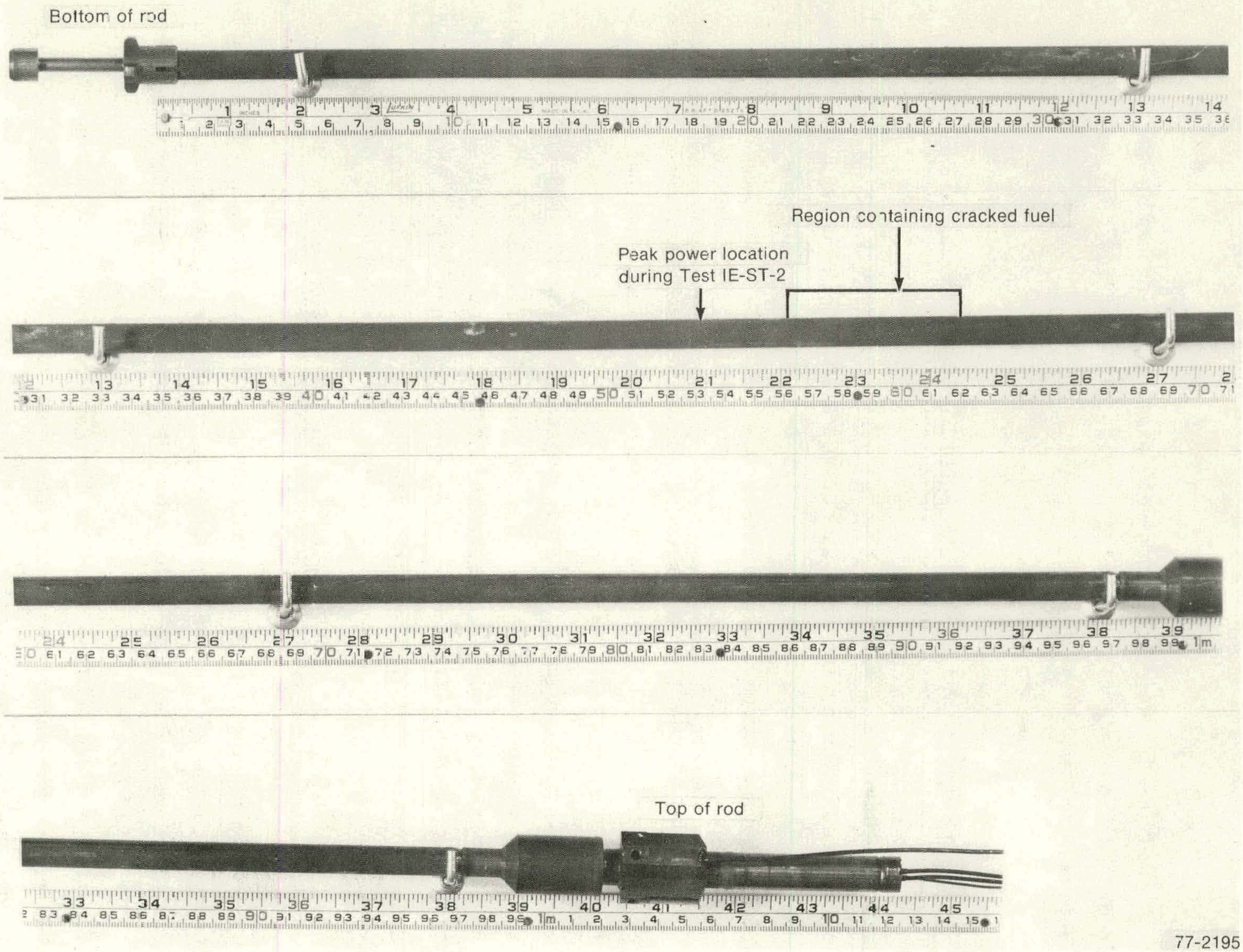

Fig. 3 Rod IE-002 at zero degrees.

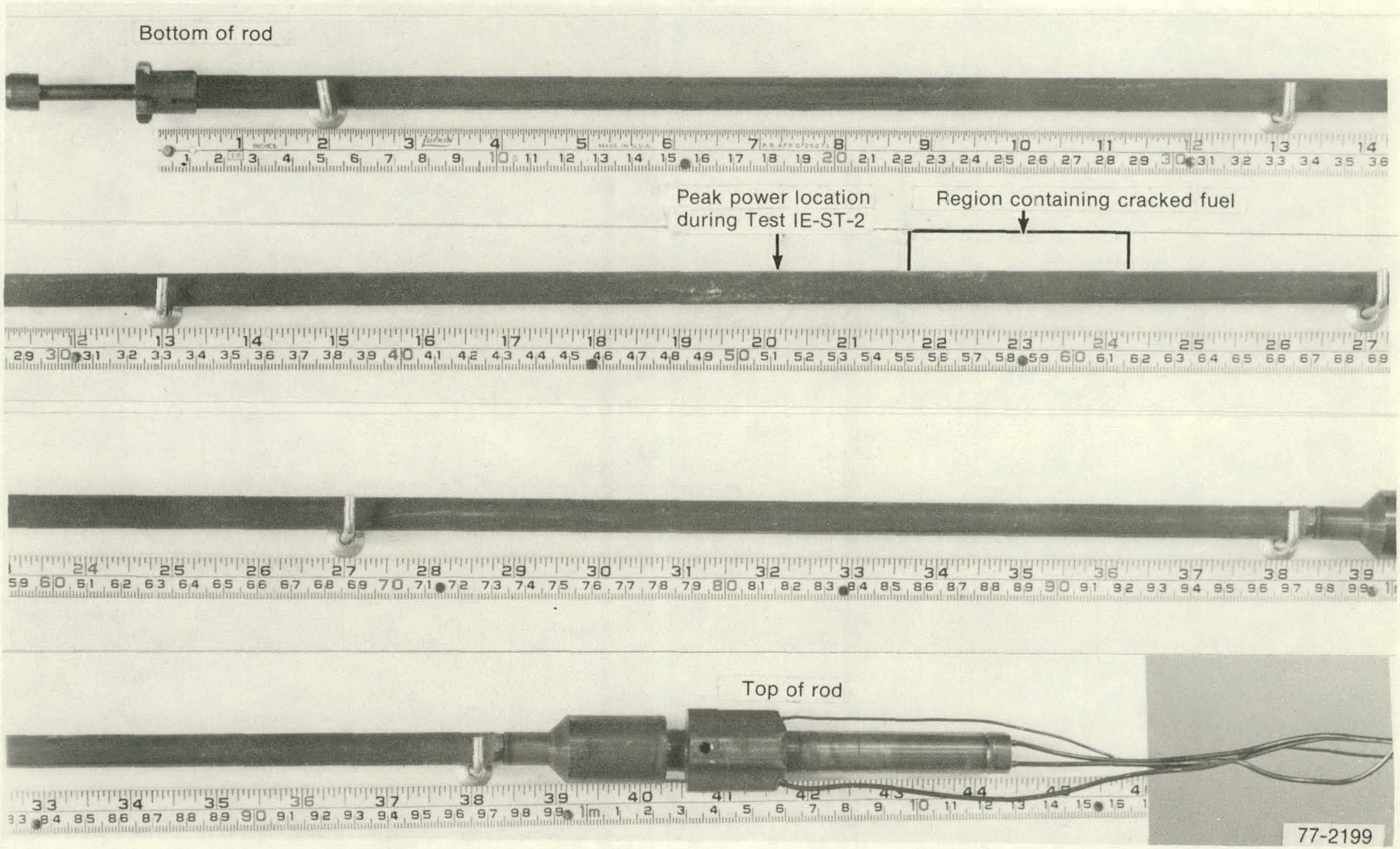

Fig. 4 Rod IE-002 at 180 degrees.

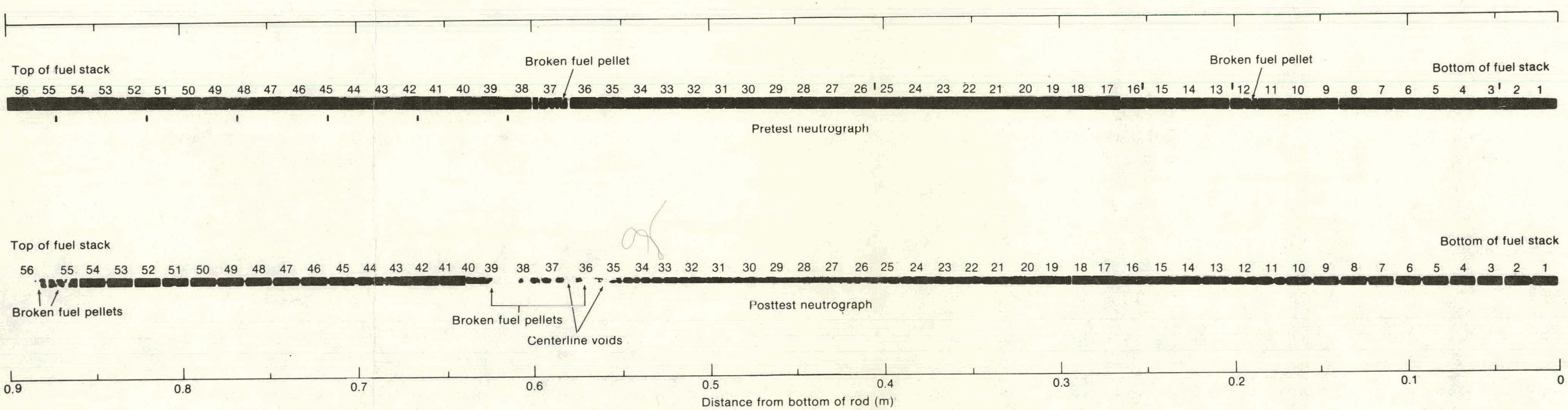

Fig. 5 Pre- and posttest neutron radiographs of Rod IE-002.

The GE neutrograph shows greater definition of individual pellet-to-pellet gaps and cracks, whereas these features appear exaggerated in the TREAT neutrograph. The pretest neutrograph shows fuel cracking resulting from operation in the Saxton reactor. Pellet number 37, located about 0.8 m from the bottom of the fuel rod,[a] was severely cracked. In the same general region (Pellets 36 through 39), the posttest neutrograph shows four pellets that were severely fractured. The degree of cracking appears somewhat exaggerated by the TREAT neutrograph. Whether this fuel breakup is a result of handling while the rod was being modified for use in the test or whether it is a result of the test is not clear. Two pellets at the top of the fuel stack were also broken. This breakage was probably due to handling when the end cap was installed. Pellet 12 at 0.20 m was cracked as shown in the pretest neutrograph; however, no additional cracking of this pellet occurred as a result of testing.

Some indication of centerline void formation which may be associated with fuel melting was noted in the region from 0.55 to 0.62 m. Destructive examination would be required to verify this assumption; however, the migration of ^{137}Cs, as reported in Section III, 4, is an indication of elevated fuel temperatures which, again, may be associated with fuel melting.

3. CLADDING DEFORMATION

The amount of cladding deformation was determined by a pulsed eddy current scan[5]. Cladding wall thickness and outer diameter were measured. A discussion of the results of these measurements is provided in the following sections. Appendix A presents a description of the scanning equipment and a discussion of the accuracy and repeatability of the results.

---

[a] The bottom of the fuel stack was 0.013 m from the bottom of the rod.

3.1 Diameter

The cladding outer diameter was generally unchanged from the pretest condition, except for two axial locations: 0.345 and 0.580 m. Figure 6 shows the cladding pre- and posttest outer diameter at these locations. The pretest diameter measurements showed some ovality, with the maximum diameter at 60 degrees and the minimum diameter at 150 degrees. (Ovality is defined as the ratio of maximum diameter to the diameter at 90 degrees from the location of the maximum diameter.) The maximum pretest ovality was 1.007, occurring at the 0.58-m elevation. Posttest results indicated the circumferential location of the maximum diameter had shifted to 30 degrees, with the minimum diameter at 120 degrees. The maximum posttest ovality was 1.029, also located at 0.58 m.

3.2 Cladding Wall Thickness Variations

Cladding wall thickness was uniform over the length of the fuel rod after irradiation in the Saxton reactor. However, after irradiation in Test IE-ST-2, decreases in wall thickness were measured at two regions: 0.22- to 0.36-m elevation, the region of maximum power in the Saxton reactor; and 0.42- to 0.62-m elevation, the region of maximum power in PBF operation. Figure 7 shows the wall thickness as a function of orientation for two elevations in the lower region and three elevations in the upper region. The maximum decrease in wall thickness, 0.102 mm (17%), occurs at 0.330 m. The average wall thickness in the lower region was 0.546 mm, a reduction of 0.046 mm (8%). The average wall thickness in the upper region was 0.575 mm, a reduction of 0.017 mm (3%). No evidence of cladding ridging was observed at pellet interfaces.

The minimum wall thickness and maximum ovality were determined along the length of the fuel rod. These data, plotted in Figure 8, show two regions of cladding deformation. However, from the plot it can be seen that the regions of diameter change are offset axially from the regions of wall thinning by approximately 0.05 m, upward.

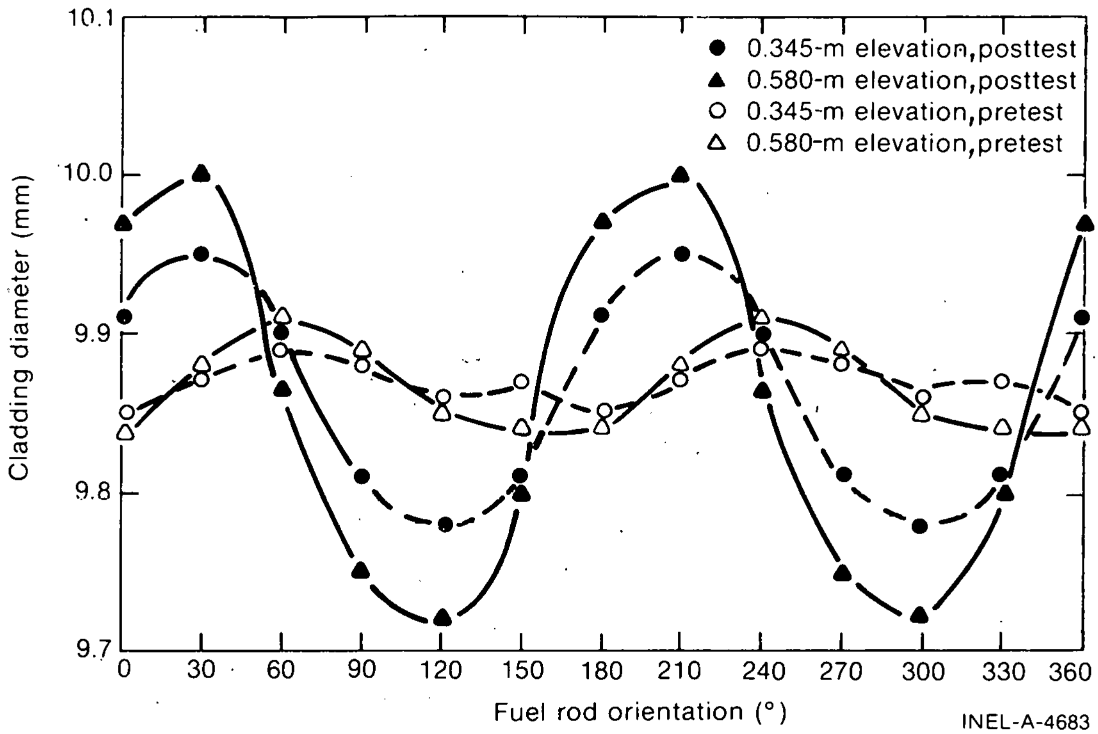

Fig. 6 Pre- and posttest cladding diameters of Rod IE-002.

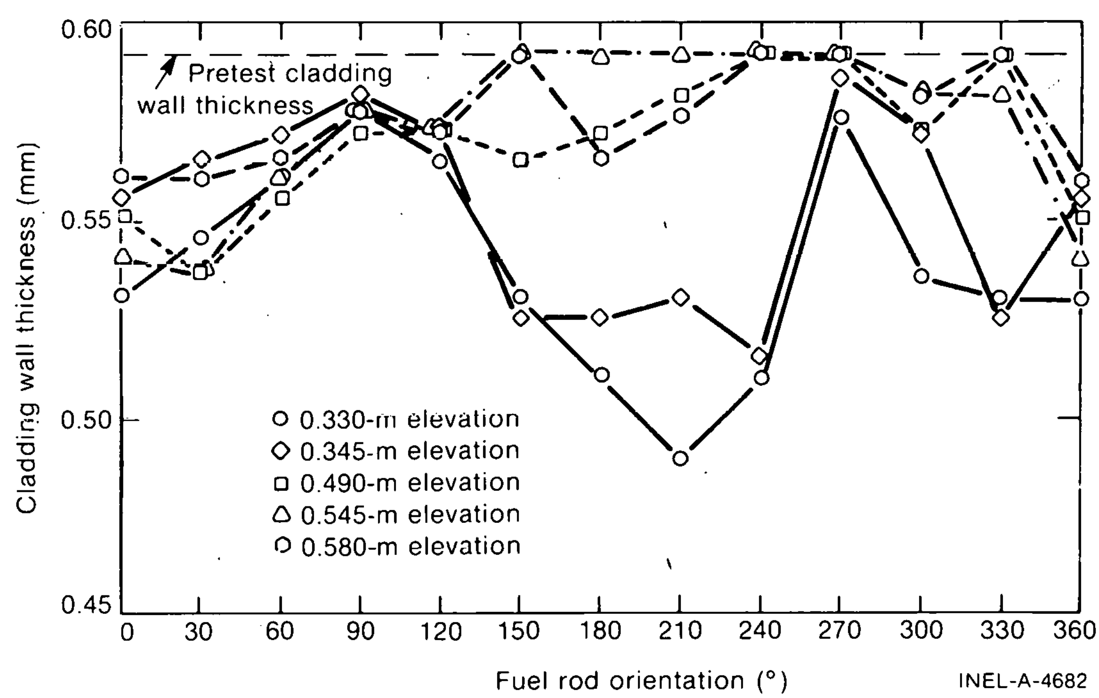

Fig. 7 Posttest cladding wall thickness at several axial locations of Rod IE-002.

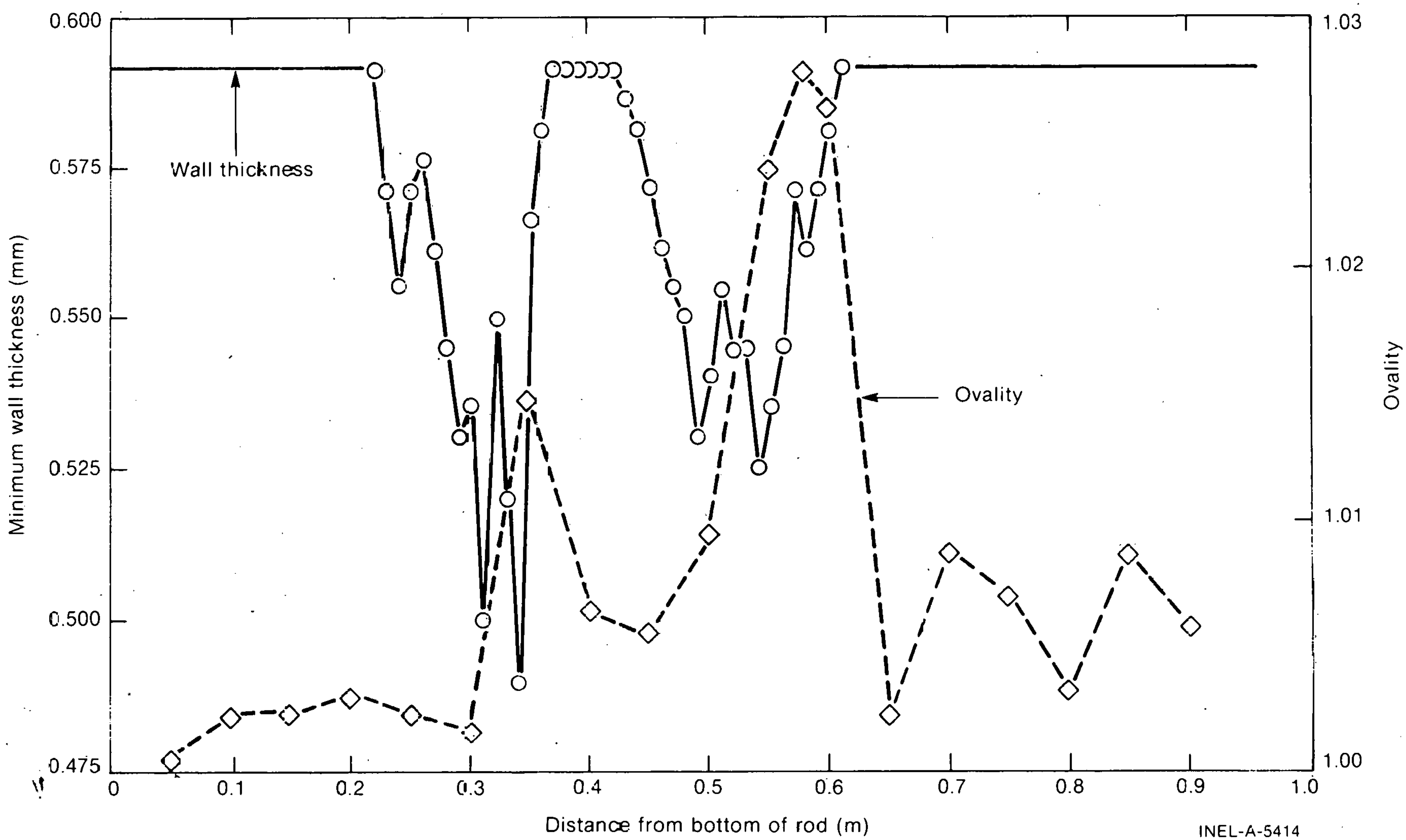

Fig. 8 Minimum wall thickness and maximum ovality along the length of Rod IE-002.

4. FISSION PRODUCT REDISTRIBUTION

Fuel Rod IE-002 was gamma scanned pre- and posttest. These data are presented in Appendix B. A decrease in gross gamma activity in both scans correlates well with the area of broken fuel shown in Figure 5.

The fuel rod was also spectral gamma scanned for the following: 497 keV ^{103}Ru, 622 keV ^{106}Rh, 662 keV ^{137}Cs, 724 keV ^{95}Zr, 756 keV ^{95}Zr, 765 keV ^{95}Nb, and 2186 keV ^{144}Ce. Results are presented in Table 1. The 765 keV ^{95}Nb, and 662 keV ^{137}Cs gamma intensities are plotted as a function of axial location in Figure 9. Both the ^{95}Nb and ^{137}Cs intensities decrease in the region containing broken fuel (0.55 to 0.62 m). The ^{137}Cs intensity increases toward the bottom of the rod. Figure 10 shows the ratio of 662 keV ^{137}Cs intensity to the 765 keV ^{95}Nb intensity as a function of axial location. This plot shows the migration of ^{137}Cs from the 0.55- to 0.61-m region, above the power peak location, toward the bottom 0.2 m of the fuel rod.

TABLE I

MEASURED GAMMA RAY INTENSITIES FROM FUEL ROD IE-002

Location[a] (distance from bottom of rod m)	Gamma Ray Intensity (cps) 497 keV ^{103}Ru	522 keV ^{106}Rh	662 keV ^{137}Cs	724 keV ^{95}Zr	756 keV ^{95}Zr	765 keV ^{95}Nb	2186 keV ^{144}Ce
0.008[b]	---	---	5.5 $\pm$ 0.2	---	---	0.12 $\pm$ 0.01	---
0.018	---	0.85 $\pm$ 0.07	164.0 $\pm$ 5	1.40 $\pm$ 0.05	1.59 $\pm$ 0.07	5.8 $\pm$ 0.2	0.19 $\pm$ 0.01
0.098	0.46 $\pm$ 0.07	1.08 $\pm$ 0.07	146.0 $\pm$ 5	1.81 $\pm$ 0.07	2.11 $\pm$ 0.09	7.6 $\pm$ 0.3	0.17 $\pm$ 0.02
0.168	0.64 $\pm$ 0.08	1.66 $\pm$ 0.06	175.0 $\pm$ 5	2.47 $\pm$ 0.07	2.71 $\pm$ 0.06	9.5 $\pm$ 0.3	---
0.248	0.55 $\pm$ 0.07	1.94 $\pm$ 0.10	163.0 $\pm$ 5	2.96 $\pm$ 0.10	3.02 $\pm$ 0.10	11.1 $\pm$ 0.3	0.33 $\pm$ 0.02
0.328	0.92 $\pm$ 0.11	1.98 $\pm$ 0.07	145.0 $\pm$ 4	3.12 $\pm$ 0.09	3.49 $\pm$ 0.08	12.3 $\pm$ 0.3	0.35 $\pm$ 0.02
0.398	0.73 $\pm$ 0.08	2.02 $\pm$ 0.07	140.0 $\pm$ 4	3.39 $\pm$ 0.13	3.61 $\pm$ 0.09	12.8 $\pm$ 0.4	0.31 $\pm$ 0.03
0.478	1.10 $\pm$ 0.10	2.04 $\pm$ 0.07	147.0 $\pm$ 5	3.19 $\pm$ 0.12	3.68 $\pm$ 0.14	12.9 $\pm$ 0.4	0.35 $\pm$ 0.02
0.548	0.55 $\pm$ 0.11	1.68 $\pm$ 0.08	93.0 $\pm$ 3	2.75 $\pm$ 0.08	3.09 $\pm$ 0.06	11.1 $\pm$ 0.3	0.27 $\pm$ 0.02
0.558	0.28 $\pm$ 0.04	0.65 $\pm$ 0.03	12.5 $\pm$ 0.4	1.03 $\pm$ 0.06	1.24 $\pm$ 0.05	4.3 $\pm$ 0.2	0.09 $\pm$ 0.01
0.578	0.67 $\pm$ 0.09	1.37 $\pm$ 0.07	65.0 $\pm$ 2	2.35 $\pm$ 0.10	2.65 $\pm$ 0.08	9.5 $\pm$ 0.3	0.21 $\pm$ 0.02
0.608	0.49 $\pm$ 0.07	0.85 $\pm$ 0.05	9.7 $\pm$ 0.2	1.79 $\pm$ 0.06	1.99 $\pm$ 0.07	7.1 $\pm$ 0.3	0.15 $\pm$ 0.02
0.628	0.70 $\pm$ 0.10	1.38 $\pm$ 0.07	100.0 $\pm$ 3	2.31 $\pm$ 0.07	2.79 $\pm$ 0.09	9.7 $\pm$ 0.4	0.23 $\pm$ 0.02
0.708	0.52 $\pm$ 0.08	1.59 $\pm$ 0.08	125.0 $\pm$ 4	2.30 $\pm$ 0.11	2.50 $\pm$ 0.10	9.7 $\pm$ 0.3	0.26 $\pm$ 0.02
0.728	0.33 $\pm$ 0.05	1.37 $\pm$ 0.06	107.0 $\pm$ 3	1.77 $\pm$ 0.07	2.13 $\pm$ 0.07	7.3 $\pm$ 0.2	0.23 $\pm$ 0.02
0.858	0.39 $\pm$ 0.05	0.91 $\pm$ 0.06	89.0 $\pm$ 3	1.41 $\pm$ 0.04	1.61 $\pm$ 0.05	5.9 $\pm$ 0.2	0.19 $\pm$ 0.02
0.894	0.50 $\pm$ 0.06	0.62 $\pm$ 0.04	56.0 $\pm$ 2	1.06 $\pm$ 0.04	1.35 $\pm$ 0.05	4.7 $\pm$ 0.2	0.12 $\pm$ 0.01
0.906[b]	---	---	4.0 $\pm$ 0.1	0.04 $\pm$ 0.01	0.05 $\pm$ 0.01	0.05 $\pm$ 0.01	0.01 $\pm$ 0.001

[a] Top of fuel stack is at 0.898 m; bottom of fuel stack is at 0.013 m.
[b] Background measurement - fuel not viewed by scanner

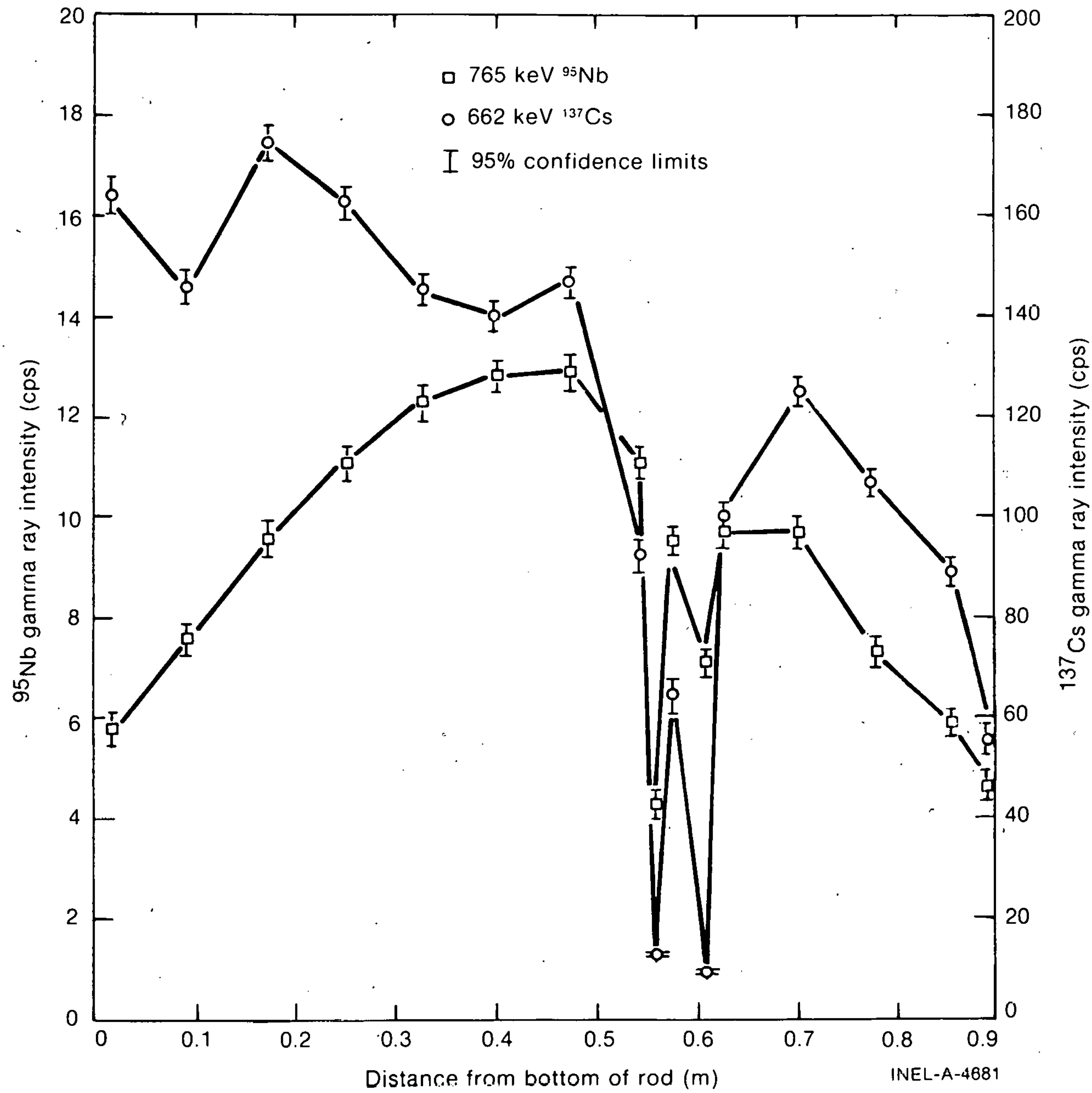

Fig. 9 Gamma ray intensities from Rod IE-002.

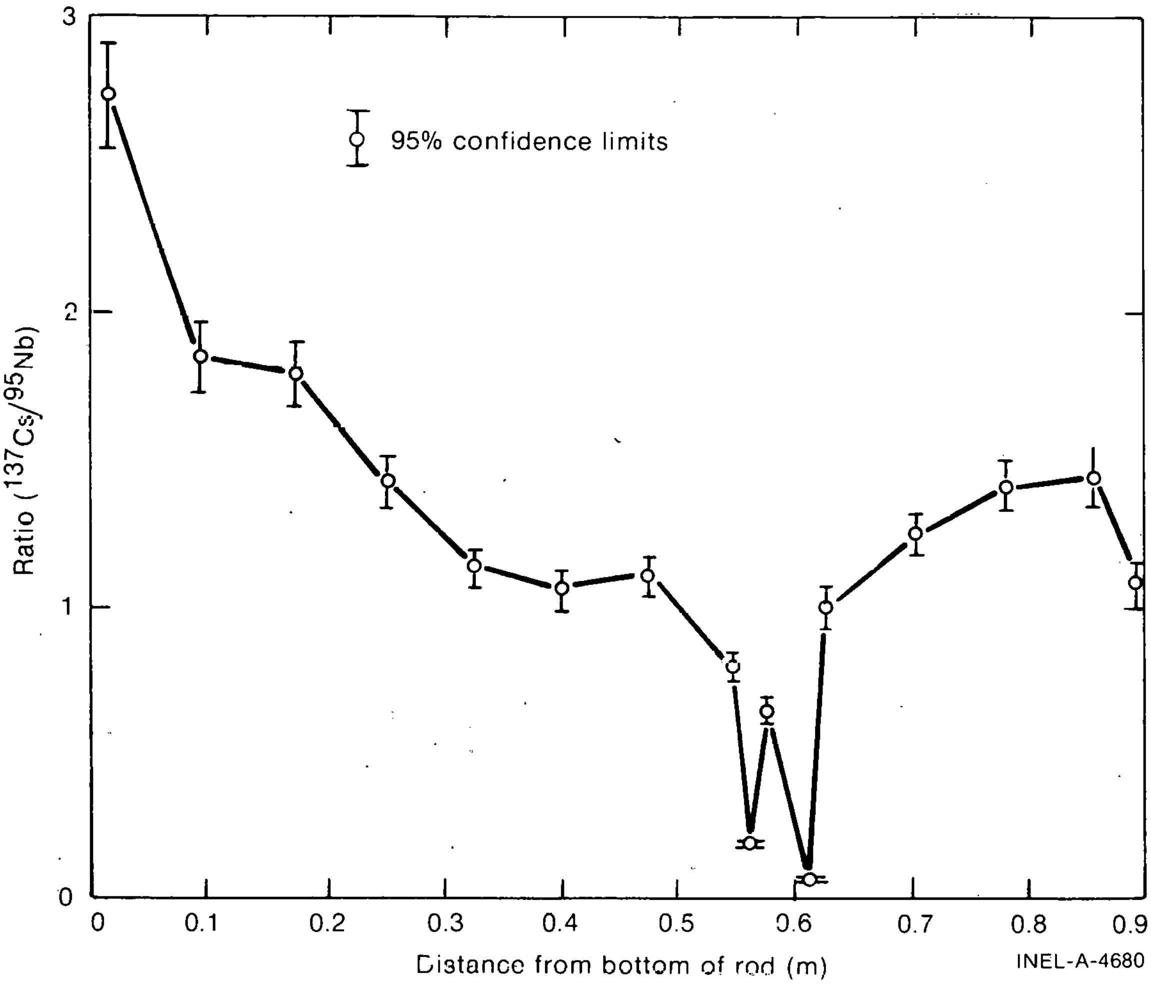

Fig. 10 Ratio of 662 keV ^{137}Cs to 765 keV ^{95}Nb gamma ray intensities.

IV. DISCUSSION

Rod IE-002 was nondestructively examined to determine the effects of the power ramp on the condition of the rod. The rod survived the ramp and high-power hold period without cladding failure. A uniform black oxide covered the entire surface of the rod. No cladding defects were visually detected. However, unusual cladding deformation, fuel relocation and restructuring evidence, and fission product redistribution were found in the postirradiation examination in two axial locations along the rod corresponding to the reported[4] high-power region (0.30 m) during Saxton operation and just above the high-power region (0.49 m) during the test in PBF. The reported[4] power skewing toward the bottom of the rod during Saxton operation was not detected in the gross gamma scan of the rod.

A spectral gamma scan detected gross migration of fission product ^{137}Cs from the 0.55- to 0.62-m region toward the bottom of the rod. In this same region, indications of void formation at the center of the pellets were observed in the posttest neutrograph. These findings indicate that fuel temperatures were very high in this region and possibly some fuel melting had occurred. This same region is the location expected to operate in film boiling during the flow reduction phase of the test. However, on-line instrumentation, the visual appearance of the rod, and the absence of cladding collapse all indicate that the rod did not operate in film boiling. The cause of the anomalously high fuel temperature cannot be explained at this time.

In this same region, relatively large cladding deformations occurred with maximum cladding wall thickness locally decreasing by about 9% at the 0.58-m rod elevation as compared with wall thickness measured after operation in the Saxton Reactor. Cladding deformations also occurred from the 0.22- to 0.36-m rod elevation, the region of maximum rod power during Saxton operation. A maximum wall thickness decrease of 17% occurred at the 0.33-m rod elevation. Prior to PBF operation wall thickness was relatively uniform, with no regions showing evidence of cladding wall thinning.

Rod IE-002 survived a high power ramp without failure. Fuel restructuring and fission product migration evidence suggest anomalously high fuel temperatures occurred in a region just above the peak power region of the rod. In this same region and in the region of maximum power during Saxton operation, large and unusual cladding deformations occurred, apparently a result of pellet/cladding interaction strains induced by the power ramp. The irradiation-damaged cladding ($>1 \times 10^{21}$ nvt) was capable of withstanding these large deformations without failure. The cause of the anomalous cladding deformations, fuel restructuring, and fission product migration cannot be explained at this time. These occurrences have not been observed to date on other Saxton-irradiated rods similarly tested in the Irradiation Effects Test Series.

V. REFERENCES

1. United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Water Reactor Safety Research - Status Summary Report, NUREG 75/046 - 5 (May 15, 1975).

2. W. J. Quapp et al, Irradiation Effects Test Series Scoping Test 2, Test Results Report, TREE-NUREG-1044 (September 1977).

3. A. S. Mehner, Postirradiation Examination Results for the Irradiation Effects Scoping Test 2, TREE-NUREG-1022 (January 1977).

4. G. W. Gibson et al, Characteristics of UO_2-Zircaloy Fuel Rod Materials From Saxton Reactor for Use in Power Burst Facility, ANCR-NUREG-1321 (September 1976).

5. W. C. Francis et al, Nondestructive Examination of Irradiated Fuel Rods by Pulsed Eddy Current Techniques, ANCR-1282 (February 1976).

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APPENDIX A

PULSED EDDY CURRENT SCANNING

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APPENDIX A

PULSED EDDY CURRENT SCANNING

Fuel Rod IE-002 from Test IE-ST-2 was scanned by pulsed eddy current (PEC) methods[A-1] both before and after testing. The PEC scanner provided outer diameter and wall thickness information. Scans were taken at 30-degree intervals around the fuel rod.

Many of the orientations on fuel Rod IE-002 were scanned twice. Figure A-1 shows an example of two scans at one orientation for both the outer diameter channel and the wall thickness channel. The repeatability of these measurements was excellent. The accuracy of the diameter and wall thickness measurements was estimated to be within ± 0.008 mm. The precision of these measurements was 0.003 mm.

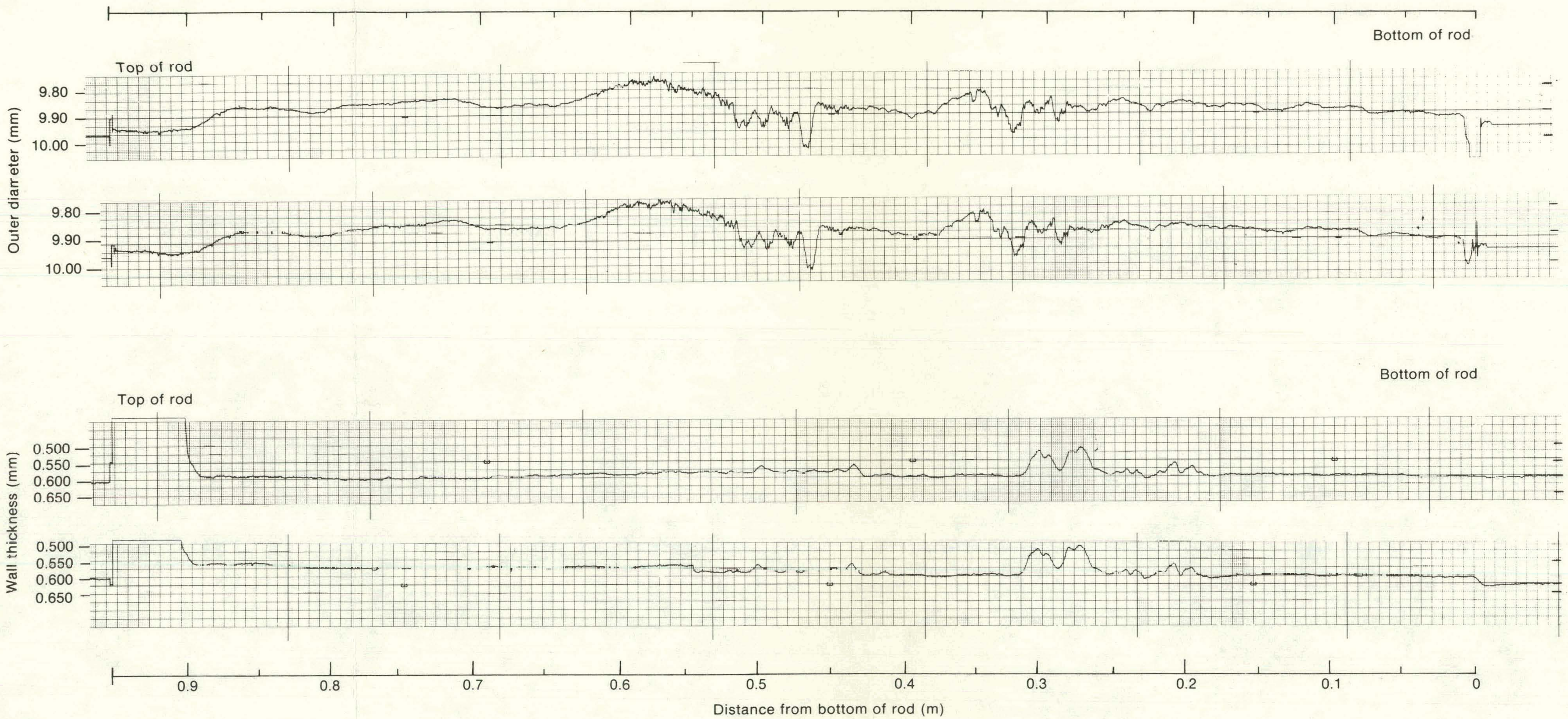

Fig. A-1 Pulsed eddy current scans of Rod IE-002.

REFERENCE

A-1. W. C. Francis et al, <u>Nondestructive Examination of Irradiated Fuel Rods by Pulsed Eddy Current Techniques</u>, ANCR-1282 (February 1976).

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APPENDIX B

GAMMA SCANS

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APPENDIX B

GAMMA SCANS

Fuel Rod IE-002was gamma scanned both before and after Test IE-ST-2 to document the relative lievel of gamm ray activity as a function of distance along the fuel stack and to determine any changes that occured as a result of the PBF testing. The data were taken with a NaI scintillation detector coupled to a multicannel pulse height analyzer and strip chart recorder.

A Slo-Syn stepping motor provides the rod scanning drive and positioning mechanism which allows setting the test interval to $\pm$0.03 mm. The fuel rod was situated in front of the Nai collimator slit (0.051 x 1.270 mm) and the output of the NaI detector was received by the strip chart recorder as the fuel rod was driven past the collimator slit at a slow, constant rate. The strip chart recording speed was matched to the positioning rate (8.46 mm/s) of the fuel rod past the collimator. A standard ^{137}Cs source was monitored before and following the gamma scan to determine whether any electronic drift in the initial and final data recording occurred during the scan.

The top and bottom of the fuel stack were identified by step-scanning past the detector-collimator. A rapid change in the gamma ray intensity occurred at the end of the fuel stack.

Figure B-1 shows the pre- and posttest gross gamma scans. A different scale was used for the two scans, as shown in the figure by the ^{137}Cs shource run before and after each scan. The pretest scan shows a decrease in activity at the area of cracked fuel located 0.546 to 0.577 m from the bottom of the fuel stack. The decrease in posttest activity from 0.55 to 0.62 m indicates more extensive fuel cracking as a result of the test. This activity decrease was also found to be a result of ^{137}Cs migration away from this area.

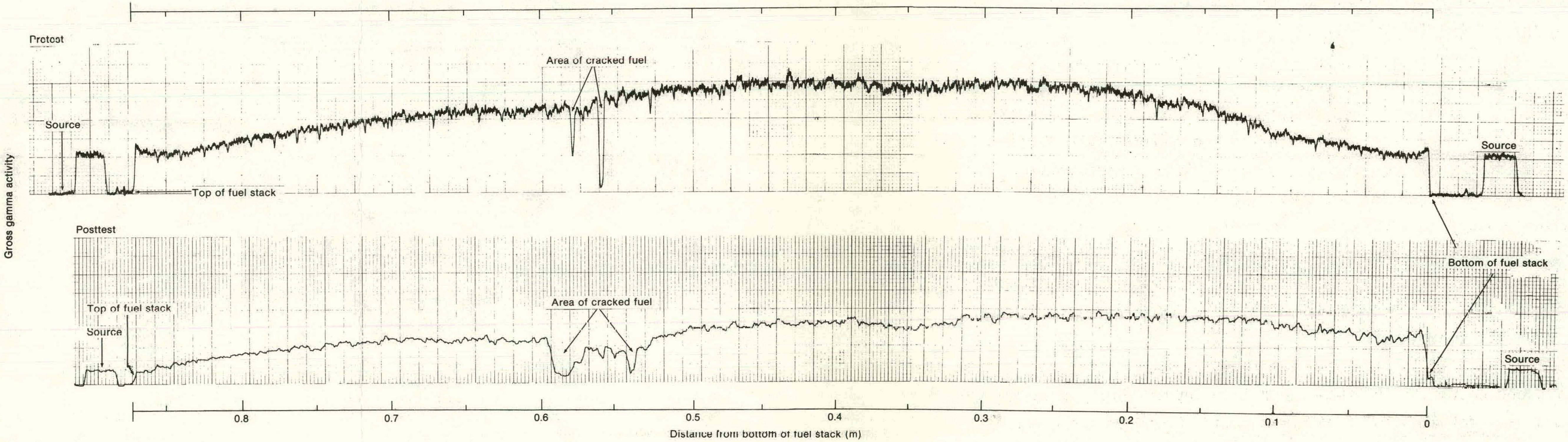

Fig. B-1 Pre- and posttest gross gamma scans of Rod IE-002.

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