NUREG/CR-2567
PNL-4240

Final Data Report for the Instrumented Fuel Assembly (IFA)-432

Prepared by E. R. Bradley, M. E. Cunningham, D. D. Lanning

Pacific Northwest Laboratory
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Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

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NUREG/CR-2567
PNL-4240
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Manuscript Completed: May 1982
Date Published: June 1982

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Prepared for
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Washington, D.C. 20555
NRC FIN B2043

ACKNOWLEDGMENTS

The authors wish to thank the Fuel Behavior Branch, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (NRC), for their continued support and encouragement of the experimental program. We also thank the Halden Boiling Water Reactor staff and A. D. Appelhans, the NRC representative at Halden, for their efforts in recording and transmitting the experimental data. Special thanks are given to W. D. Bennett, who produced the many data plots, and to S. K. Edler for editing this report.

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ABSTRACT

This report presents the in-reactor data collected during the irradiation of the six-rod instrumented fuel assembly (IFA)-432 in the Halden Boiling Water Reactor (HBWR) from June 1980 through June 1981. This Pacific Northwest Laboratory (PNL)-designed assembly was one of a series of U.S. Nuclear Regulatory Commission (NRC)-sponsored tests to obtain data for the development and verification of steady-state fuel performance computer codes. IFA-432 operated from December 1975 until June 1981, when it was removed from the reactor. Two of the rods were removed for examination, and the assembly was reinserted in December 1981 to obtain additional data.

Fuel centerline temperatures, cladding elongations, internal fuel rod pressures, and local powers at thermocouple positions were monitored during the irradiation of IFA-432; and the resulting data are presented in this report. The local powers were derived from neutron detector readings while the other variables were measured directly. Detailed analysis of the data is not made, but topical reports discussing certain aspects of the data are referenced. Descriptions of the assembly, instrumentation and calibration, and data processing methods are presented.

SUMMARY

Pacific Northwest Laboratory (PNL) designed four instrumented fuel assemblies (IFA-431, -432, -513, and -527) for a series of U.S. Nuclear Regulatory Commission (NRC)-sponsored tests in the Halden Boiling Water Reactor (HBWR), Halden, Norway. IFA-432 was loaded in the reactor and began operating in December 1975, and peak burnups in excess of 2990 GJ/kgU (34.7 GWd/MTU) were reached as of June 1981. Valuable information on fuel performance has been obtained from neutron detectors, fuel thermocouples, cladding elongation monitors, and pressure transducers. Much of this instrumentation survived to burnups typical of power reactor fuel (25 to 35 GWd/MTU). IFA-432 was removed from the reactor in June 1981, and the decision was made to remove two rods for postirradiation examination (PIE) and to reinsert the assembly in the reactor. Peak burnups in excess of 3900 GJ/kgU (45 GWd/MTU) are projected for three of the four rods that are still in the reactor.

This report presents in-reactor data collected from IFA-432 for the period from June 1980 through June 1981. Data collected prior to June 1980 were presented in previous reports (Hann et al. 1978b; Bradley et al. 1981). Fuel temperatures, power levels, and elongation data are presented in the form of plots of the variables versus time; internal pressure data and calculated burnups are tabulated. Some PIE results from rod 8 are also presented. Descriptions of the test rationale, assembly and rod designs, test facility, instrument array and calibration, and data processing methods are included. Topical reports discussing specific aspects of the data analysis are referenced.

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INTRODUCTION

The thermal stored energy in a fuel rod is the driving function for the severest postulated nuclear energy-related accident--the loss-of-coolant accident (LOCA). Because of this, the final acceptance criteria for emergency core cooling systems require calculation of the stored energy and gap conductance of a fuel rod, both for normal operation and for the duration of the LOCA. Although these calculations are used in the regulation of commercial nuclear power plants, uncertainties in them have caused temporary derating of some power plants and delays in the startup of other plants. Many of these uncertainties can be attributed to the lack of well-characterized data for fuel irradiated throughout the normal operating power range of commercial nuclear power plants.

To focus on these uncertainties, four instrumented fuel assemblies (IFAs) were designed by Pacific Northwest Laboratory (PNL)[a] and were irradiated in the Halden Boiling Water Reactor (HBWR), Halden, Norway. The first two tests in this U.S. Nuclear Regulatory Commission (NRC)-sponsored test series--IFA-431 and IFA-432--were identical six-rod assemblies with the same variations in gap size and fuel type that were operated at different power levels. The third assembly in the series--IFA-513--contained six identical rods with varied fill gas compositions and pressures. The fourth assembly--IFA-527--used xenon fill gas to study the effects of fuel pellet cracking and relocation.

The subject of this report is IFA-432 (the second assembly), which had a design power of 49 kW/m (15 kW/ft) and reached its goal burnup of 1720 GJ/kgU (20 GWd/MTU) in late 1978. Since most of the instruments in IFA-432 were still functioning properly at that time, it was left in the HBWR core to obtain data at higher burnups. IFA-432 has provided a vast amount of well-characterized experimental data under conditions that realistically simulate light-water reactor (LWR) conditions. The data have been used extensively for analyzing fission gas release (Bradley et al. 1979a; Bradley et al. 1979b) and thermal and mechanical fuel rod performance[b] and for evaluating error propagation in stored energy calculations (Cunningham et al. 1978; Cunningham et al. 1980). As a result of the data analysis, improved models for computer code calculations of fuel rod performance in LWRs are being developed.

The experimental data collected for IFA-432 from startup (December 1975) through May 1980 were presented in previous reports (Hann et al. 1978b; Bradley et al. 1981). This report presents the experimental data collected from June 1980 through June 1981 and represents the final formal data report for IFA-432. Additional data collected from the four IFA-432 fuel rods that are still in the reactor will be presented periodically in quarterly reports.

(a) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute.
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INTRODUCTION

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TEST DESCRIPTION

A collection of mathematical models (i.e., a computer code) is used to simulate the wide range of conditions postulated during an evaluation of reactor fuel safety. Any computer code that is forced to rely on a collection of empirical and semiempirical models for much of the analysis is limited and should be primarily used for interpolation. Some extrapolation can be accomplished with models based on first principles; however, well-characterized data are needed in either case to test code predictions. When this program began in July 1974, very little data were available describing the effects of burnup on LWR fuel and no data were available describing the effects of fuel densification on fuel temperatures. Accordingly, a test matrix was developed (see Table 1), and two IFAs were designed to provide the data.[a]

CROSS-CORRELATION EFFORTS

Much thought went into the design of this test to:

- insure a means for cross-correlating the data
- provide as many independent checks of data validity as possible
- insure against instrument failure
- insure at least internal consistency on a relative basis
- provide some reference points to commercial plant designs and other fuel research programs.

One of the basic premises of the test design was to provide a systematic approach that would allow adequate interpolation and extrapolation with computer codes. The first step in this approach was the decision to begin with two identical assemblies since this would enhance the ability to interpolate over a range of powers and replicate initial conditions. (For example, all the data from the first power ramp of IFA-431 were duplicated with IFA-432.) Uncertainties associated with assembly and rod power distributions would also be reduced with identically designed assemblies.

The power profile in the HBWR (Figure 1) was also considered during the design. The tops of the rods were placed at the peak, which forced the bottoms of the rods to operate at 70 to 80% of peak rod power. To take advantage of the power distribution, thermocouples (TCs) were placed in the top and bottom of each rod. No tests had ever been run at Halden with TCs penetrating both end caps; however, Halden developed a workable design. TCs in both ends allowed modelers to check the ability of various codes to extrapolate over a

(a) IFA-432 and IFA-431 are identically designed assemblies; IFA-431 was irradiated from June 1975 to February 1976 (Hann et al. 1978a; Nealley et al. 1979).

TABLE 1. Design Parameters and Instrumentation for Instrumented Fuel Assembly (IFA)-432[a]

Rod No.	Pellet Diameter, mm	Pellet Diameter, in.	Cold Diametral Gap[b], mm	Cold Diametral Gap[b], in.	Fill Gas	Fuel Density, % TD	Fuel Type[c]	Instrumentation: Temperature, Upper	Instrumentation: Temperature, Lower	Instrumentation: Pressure	Instrumentation: Cladding Length
1	10.681	0.4205	0.229	0.009	He	95	Stable	TC[d]	TC	PT[e]	EM[f]
2	10.528	0.4145	0.381	0.015	He	95	Stable	UT[g]	TC	--	EM
3	10.833	0.4265	0.076	0.003	He	95	Stable	TC	TC	--	EM
4	10.681	0.4205	0.229	0.009	Xe	95	Stable	TC	TC	--	EM
5	10.681	0.4205	0.229	0.009	He	92	Stable	TC	TC	PT	EM
6	10.681	0.4205	0.229	0.009	He	92	Unstable	TC	TC	PT	EM
7	10.528	0.4145	0.381	0.015	He	95	Stable	--	--	--	--
8	10.681	0.4205	0.229	0.009	He	95	Stable	--	--	--	--
9	10.732	0.4225	0.179	0.007	He	95	Stable	--	--	--	--

(a) IFA-432 peak power = 49 kW/cm (15 kW/ft).
(b) Cladding OD = 12.789 mm (0.5035 in.), and cladding ID = 10.909 mm (0.4295 in.). Diametral gap is cladding ID minus pellet diameter.
(c) With respect to in-reactor densification.
(d) TC = thermocouple.
(e) PT = pressure transducer.
(f) EM = elongation monitor.
(g) UT = ultrasonic thermometer.

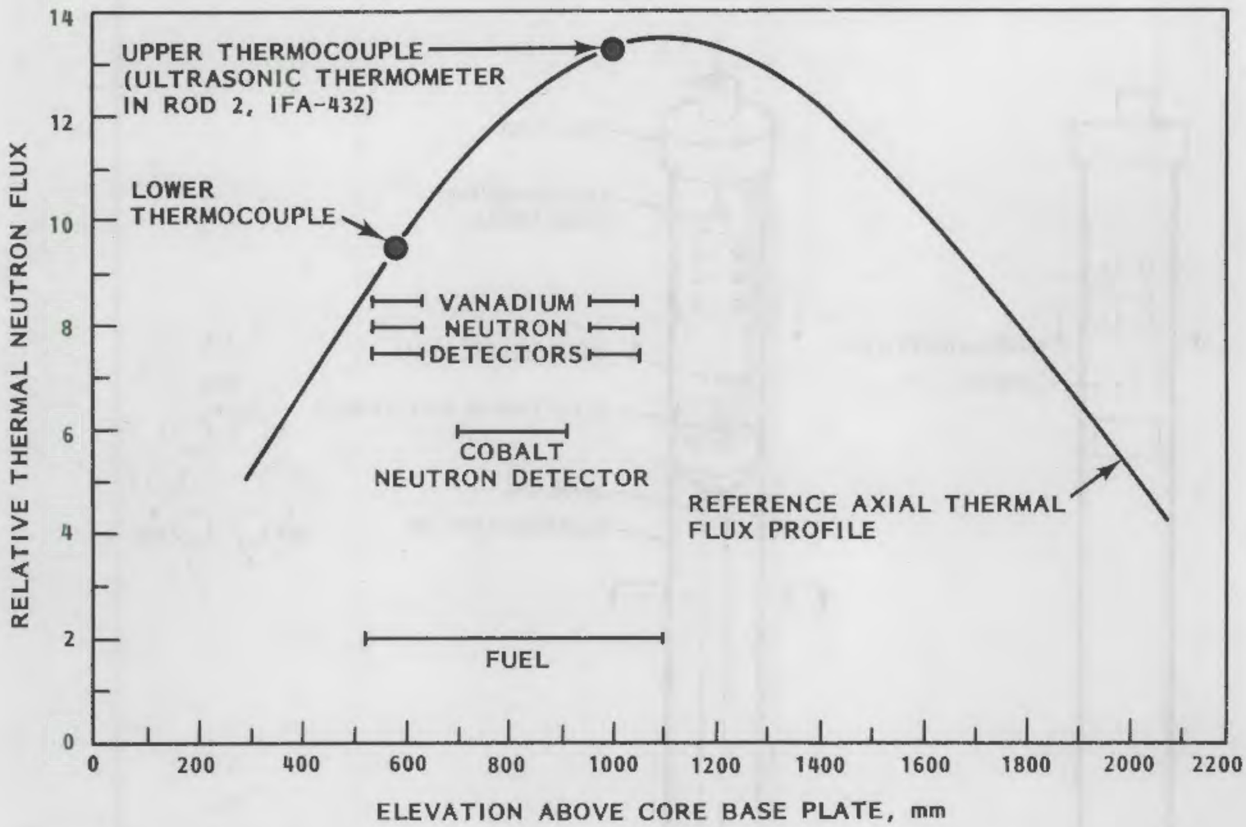

FIGURE 1. Arrangement of Temperature Sensors, Neutron Detectors, and Fuel Relative to Reference Axial Thermal Flux Profile

short power range within the same rod. If a code cannot perform these calculations adequately, calculations of the temperature distribution over an ~4-m fuel length would also be suspect.

Reference points to commercial plants and other fuel research programs were also developed by selecting a BWR-6 fuel geometry, procuring commercial-quality tubing, and selecting appropriate assembly powers. Some of the cladding procured for this program was shipped to Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho, for use in their Halden tests. The PNL and INEL programs both used the same starting powder for fuel manufacture. Some of the fuel structures were similar to those investigated in the Edison Electric Institute/Electric Power Research Institute (EEI/EPRI) UO_2 fuel densification study (Brite et al. 1975) to provide a reference point to a much larger structural characterization program.

The correct assessment of rod powers and the distribution of power within the rods are of utmost importance to assure the best possible thermal data. Therefore, seven neutron sensors were placed in each assembly (see Figure 2):

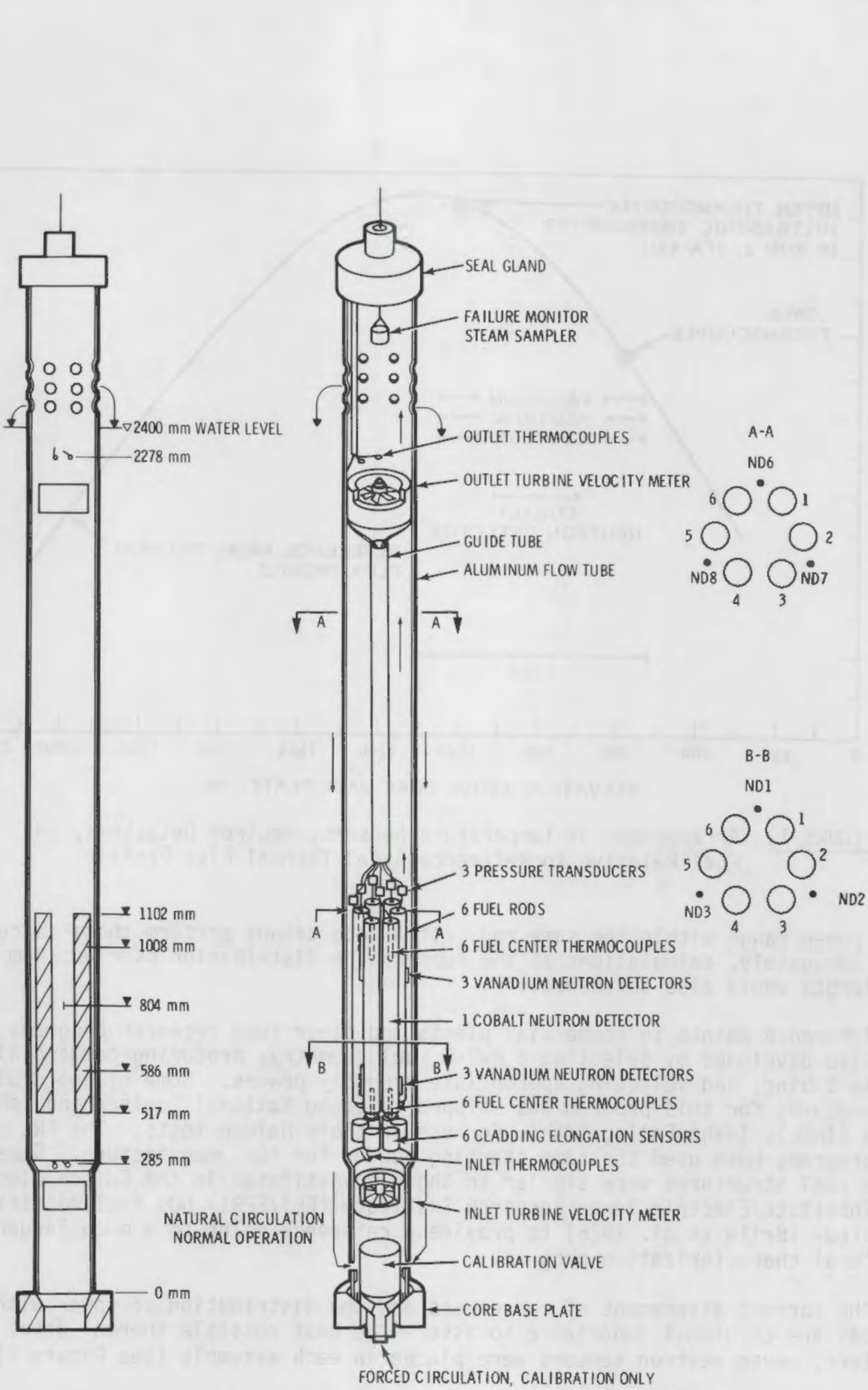

FIGURE 2. Schematic of Instrumented Fuel Assembly (IFA)-432

one cobalt detector in the center, three vanadium detectors at the top plane of the TCs, and three vanadium detectors at the bottom plane of the TCs. An extensive calibration of the vanadium sensors was conducted during the initial startup of any assembly. In addition, rod 3 (0.076-mm diametral gap) was included as an internal standard. The small gap is closed at power; thus, the temperature gradient across the gap is minimized. Since the coolant temperature and fuel centerline temperatures are known, an independent check of rod power at both the top and bottom planes in the assembly can be obtained. Rod powers and fuel temperatures in both assemblies have been compared to assure consistent data. A cladding elongation monitor was mounted for each rod; rods could be changed without disrupting the elongation sensors. Rods 1, 5, and 6 also had null-balance fission gas pressure transducers.

The amount of cross-correlation that is possible is illustrated in Table 2. In addition to the rod-to-rod comparisons, top-to-bottom comparisons can be made in each rod; and separate effects as a function of burnup and power can be evaluated.

TABLE 2. Cross-Correlation Matrix

	Rod 1(a) (9-He-95-S)	Rod 2 (15-He-95-S)	Rod 3 (2-He-95-S)	Rod 4 (9-Xe-95-S)	Rod 5 (9-He-92-S)	Rod 6 (9-He-92-U)
Gap Size	X	X	X			
Fuel Relocation	X	X	X	X	X	X
Fuel Eccentricity				X		
Fuel Stability					X	X
Gas Composition	X			X		
Fuel Density	X				X	
Rod Powers			X			
Rod Pressures	X				X	X
Dynamic Temperature	X	X	X	X	X	X

(a) (9-He-95-S) indicates that the rod has a 9-mil diametral gap and is backfilled with helium, that the fuel has a density of 95% theoretical, and that the fuel is stable with respect to densification.

TEST FACILITY

The HBWR uses natural circulation of heavy water for cooling. Reactor operating data are shown in Table 3. The schematic of the HBWR core loading in November 1975 (see Figure 3) indicates the locations of IFA-431 and IFA-432.

TABLE 3. Operating Data for the Halden Boiling Water Reactor

Power level	12 MW
Reactor pressure	3.4 MPa (500 psi)
Heavy-water saturation temperature	513K (464°F)
Plenum inlet temperature	510K (459°F)
Thermal flux	$\sim 2 \times 10^{16}$ n/m^2-s/(W/g)
Fast flux (>1 MeV)	$\sim 5 \times 10^{15}$ n/m^2-s/(W/g)
Average fuel power density	14.8 W/g

FUEL AND CLADDING PRECHARACTERIZATION

Extensive precharacterization of the fuel and cladding was essential to assure quality data and to reduce calculational uncertainties. Since the precharacterization is presented elsewhere (Hann et al. 1977), only the main objectives will be presented in this report.

Establishing the initial dimensions and void volumes within the pins was also an essential part of assessing all thermal calculations and fission gas releases. Consequently, the lengths and diameters of each pellet and the cladding for each rod were measured. Each pellet was identified with a unique number to trace pellet types and position within the rod. With this information the axial distribution of gap volume and the plenum volume were obtained with considerable accuracy. Pellet and cladding roundness profiles were also obtained to illustrate the departure from ideal coaxial cylinders used in most computer code models.

Geometric densities were determined for all pellets, and immersion densities were determined for a significant fraction of the pellets. A correlation was developed relating immersion density to geometric density. These data were used in two ways: to correct rod powers for differences in mass distribution and to verify NRC resintering models used to characterize the propensity of the fuel to densify. Resintering tests conducted on each fuel type are discussed in Hann et al. (1977).

The EEI/EPRI UO_2 densification program demonstrated the importance of pore-size distribution measurements in characterizing the stability of various fuel types. Therefore, the pore-size distributions of the three fuel types used in these experiments were measured prior to irradiation to assure that the desired

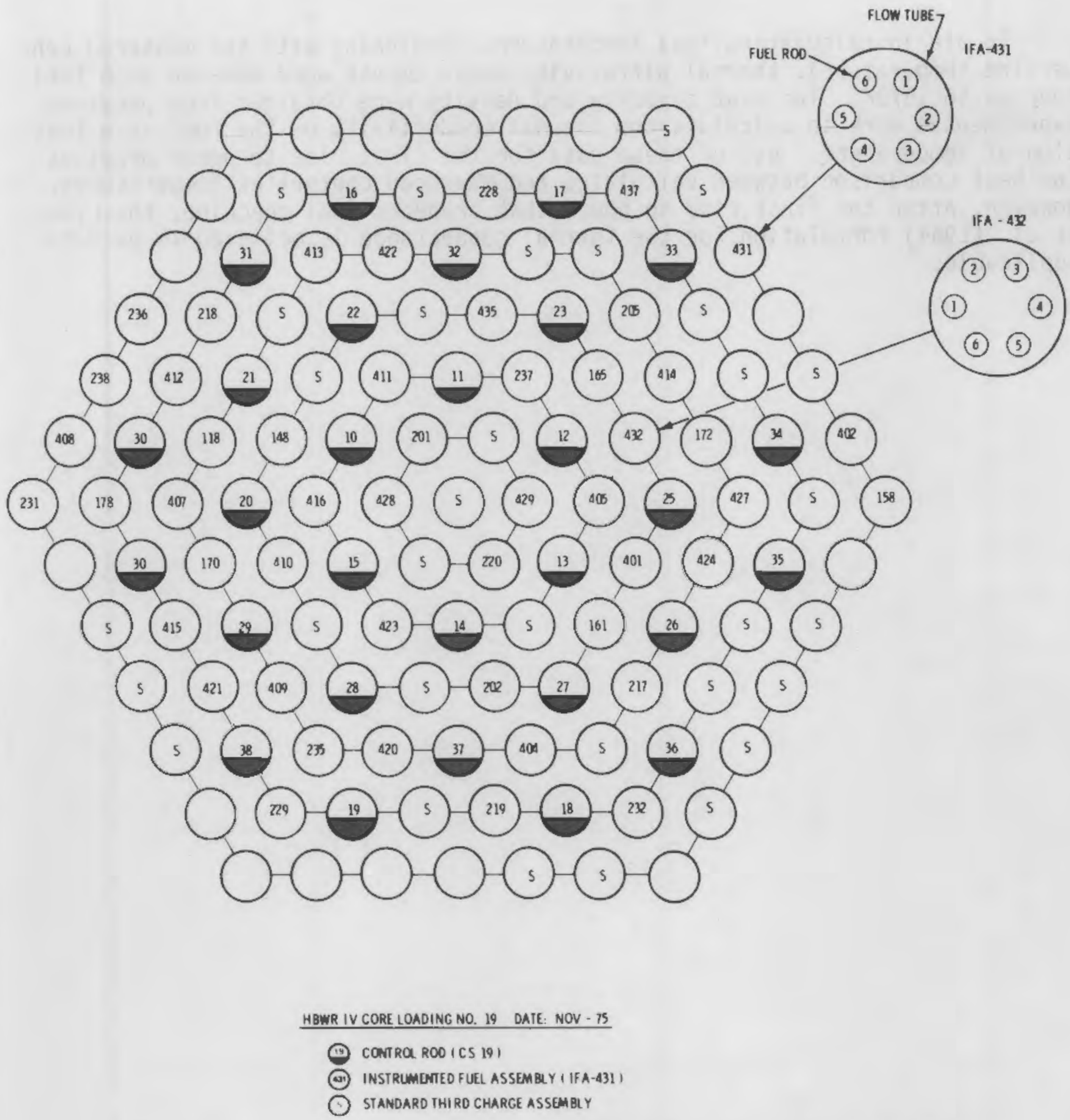

FIGURE 3. Location of IFA-431 and IFA-432 in the Halden Boiling Water Reactor Core

response to irradiation would be achieved. Both fuel densities and pore-size distribution will be measured during the postirradiation examination (PIE) of rods 1, 5, and 6 at Harwell, U.K. Archive pellets from each fuel type were retained to reduce variances associated with potential differences in examination techniques used in the pre- and post-test measurements.

To aid in calculating fuel temperatures (beginning with the measured centerline temperature), thermal diffusivity measurements were made on each fuel type up to 1873K. The heat capacity and density were obtained from previous experimental work to calculate the thermal conductivity of the fuel as a function of temperature. Use of these data for the first rise to power provides the best comparison between calculated and measured centerline temperatures. However, after the first rise to power that produces fuel cracking, the Lyons et al. (1964) formulation for the thermal conductance is believed to be more applicable.

IRRADIATION SUMMARY

IFA-432 was initially charged into the reactor in December 1975, and a total of eight fuel rods have been irradiated in the six fuel rod positions as illustrated in Figure 4. The assembly reached its goal burnup of 1720 GJ/kgU (20 GWd/MTU) in late 1978; but since most of the instruments were still operating properly, it remained in the reactor to obtain data at higher burnups.

Rod 4 was designed to study the effects of fill gas composition and fuel eccentricity on thermal performance; it was replaced by rod 8 after the first reactor operating cycle. Rod 8 was removed from the assembly in January 1980 to undergo mechanical compliance testing at the Harwell, U.K., hot cell facility; and rod 9 was inserted in its place. Rod 7 was used as an unirradiated standard for the mechanical compliance tests.

In June 1981, the assembly was removed from the reactor after the original fuel rods (1, 2, 3, 5, and 6) had attained average burnups of ~2600 GJ/kgU (30 GWd/MTU). At this time, the decision was made to remove rods 1 and 6 from the assembly for destructive PIE and to continue irradiating the remaining four fuel rods. It is projected that these fuel rods will remain in the reactor until the original fuel rods attain average burnup levels in excess of 3500 GJ/kgU (40 GWd/MTU).

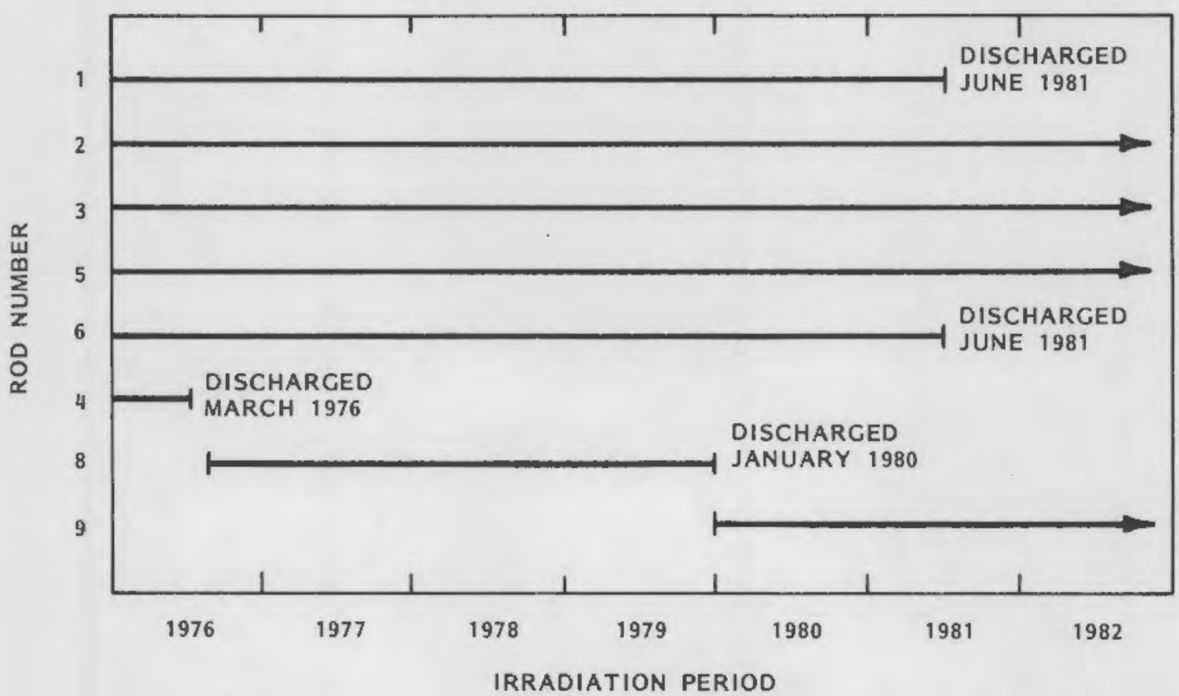

FIGURE 4. Irradiation History for IFA-432 Fuel Rods

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FIGURE 4. Irradiation History for IFA-432 Fuel Rods

INSTRUMENT PERFORMANCE

IFA-432 is a highly instrumented fuel assembly that originally contained the following instruments:

- 7 neutron detectors for assessing rod power levels
- 11 TCs and 1 ultrasonic thermometer for measuring fuel centerline temperatures (one at both ends of each fuel rod)
- 6 cladding elongation monitors
- 3 pressure transducers for measuring internal fuel rod gas pressure.

All of these instruments provided reliable experimental data at startup except for one vanadium neutron detector and the ultrasonic thermometer, which was located at the upper end of rod 2. Although some instrument failures occurred during the irradiation, the majority of the instrumentation was still operating when rods 1 and 6 were removed from the assembly in June 1981 after more than 5 yr of irradiation. Reliable data are currently being obtained from six neutron detectors, three lower TCs, one cladding elongation monitor, and one pressure transducer. The upper TC from rod 3 continues to provide signals, but the data appear to be unreliable.

The burnup levels to which reliable experimental data are or will be available from IFA-432 are summarized in Table 4. None of the neutron detectors have failed since startup; thus, power data are available for all fuel rods at two axial locations throughout the irradiation. It is estimated that the power data are accurate to ±10% at a 3σ confidence level.

TABLE 4. Summary of Burnup Levels[(a)] To Which Reliable Experimental Data Are or Will Be Available for IFA-432

Rod No.	Power Data Upper	Power Data Lower	Temperature Data Upper	Temperature Data Lower	Cladding Elongation Data	Internal Pressure Data	Postirradiation Examination Data
1	2972	2218	840	2218	204[(b)]	2595[(b)]	2595[(b)]
2	(c)	(c)	0	(c)	(c)	---	(d)
3	(c)	(c)	1933	(c)	1633	---	(d)
4	247	183	72	54	215	---	---
5	(c)	(c)	322	(c)	220	(c)	(d)
6	2997	2265	689	1600	2115	2630	2630
8	2134	1629	---	---	1882	---	1882
9	(c)	(c)	---	---	---	---	(d)

(a) Values in GJ/kgU; to convert to GWd/MTU multiply by 0.0116.
(b) Rod average burnup for elongation, pressure, and PIE data.
(c) Instrumentation is still operating.
(d) Irradiations are not completed.

All 11 TCs provided fuel centerline temperature data. The two TCs in rod 4 (xenon-filled) failed after about two weeks of operation because of the high operating temperatures (~2300K) for this rod. Higher temperatures at the upper position also shortened the useful lifetime of the upper TCs in the remaining fuel rods. For rods 1, 5, and 6, the upper TC failures were associated with temperature increases caused by fission gas release (Bradley et al. 1979a). Data from the upper TC of rod 3 appeared to be questionable after 1933 GJ/kgU (22.4 GWd/MTU).

With the exception of rod 6, TCs at the lower position have operated throughout the irradiation; and temperature data from this location are currently being obtained from rods 2, 3, and 5. The relative temperatures among the fuel rods were fairly constant after about 1300 GJ/kgU (15 GWd/MTU); thus, the fuel temperatures at the lower position of rod 6 can be estimated from the measured temperatures of the other rods.

Neutron irradiation-induced TC decalibration should be considered when using the temperature data because the measured temperatures will be less than the true temperature. The rate of decalibration is currently estimated to be 1.75% per 10^{24} n/m^2 thermal neutron fluence at the TC tip (Crouthamel and Freshley 1980). This decalibration rate suggests that the measured temperatures at 2200-GJ/kgU (25-GWd/MTU) local burnup are about 10% below the true temperature. However, analysis of transient temperature data taken during reactor scrams in August 1979 and January 1980 indicates up to 20% TC decalibration for local burnup levels below 1800 GJ/kgU (21 GWd/MTU). The decalibration rate given by Crouthamel and Freshley (1980) represents a reasonable estimate of the existing data, but additional experimental data are needed to clearly define neutron irradiation-induced TC decalibration.

All six cladding elongation monitors operated during the initial irradiation period, and three monitors were still operating in June 1981. Elongation data for rods 1 and 5 are available for burnups to about 200 GJ/kgU (2.3 GWd/MTU), and elongation data for rod 3 are available for burnups to 1663 GJ/kgU (19 GWd/MTU). Elongation data for rods 2, 6, and 9 are available through June 1981; data for rods 4 and 8 are available for the periods they were in the reactor. When rods 1 and 6 were removed from the assembly (June 1981), rod 9 was repositioned in the assembly; and elongation data are no longer being obtained from this rod. The only elongation data presently being obtained are from rod 2.

Internal pressure data are available from the three rods with pressure transducers (1, 5, and 6) throughout most of the irradiation period. The pressure transducers in rods 1 and 5 operated continuously during the irradiation period; however, pressure data are not available from rod 6 for burnup levels from 1700 to 2462 GJ/kgU (20 to 28 GWd/MTU). After this period, the pressure transducer in rod 6 appeared to be operating correctly once again; it is not presently known what caused the change in operating status. The pressure data from rod 1 showed increased scatter with possibly a small pressure decrease after 1700 GJ/kgU (20 GWd/MTU). PIE of this rod will verify the accuracy of these data.

PIE data for IFA-432 fuel rods will be available in the near future. The examination of rod 8 after nearly 1900-GJ/kgU (22-GWd/MTU) burnup was recently completed. Mechanical compliance between the fuel and cladding was emphasized in these examinations, and the majority of the data will be reported separately. Data regarding fission gas release and ceramography are presented in the data section of this report. Rods 1 and 6 were removed from the assembly in June 1981 after attaining rod average burnup levels near 2600 GJ/kgU (30 GWd/MTU). These rods are scheduled for destructive PIE at Harwell; details are given in the next section of this report.

IFA-432 has provided an extensive data base for assessing the in-pile performance of UO_2 fuels operating under conditions typical of commercial LWRs. The data obtained through May 1980 can be found in two previous data reports (Hann et al. 1978b; Bradley et al. 1981), and the data from June 1980 to June 1981 are presented in a subsequent section of this report. All of the IFA-432 data are stored in a computer file and are available for future analysis.

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FUTURE PLANS

IFA-432 reached its goal burnup of 1720 GJ/kgU (20 GWd/MTU) in late 1978 but has remained in the reactor to obtain well-characterized data for assessing the effects of higher burnup on fuel rod performance. The future plans for the four IFA-432 fuel rods that are currently being irradiated and the scheduled PIE for rods 1 and 6 are discussed in this section.

RODS 2, 3, 5, AND 9

After rods 1 and 6 were removed from the assembly, the four remaining fuel rods were reinserted in the reactor in December 1981 to obtain additional burnup. Currently, in-pile data are being obtained from the lower TCs of rods 2, 3, and 5; the cladding elongation monitor for rod 2; the pressure transducer on rod 5, and the six operating neutron detectors. The experimental data obtained from these instruments will be presented in future quarterly reports.

The assembly is expected to remain in the reactor until the rod average burnups for rods 2, 3, and 5 exceed 3500 GJ/kgU (40 GWd/MTU); and burnups approaching 4300 GJ/kgU (50 GWd/MTU) may be possible. An exact date for assembly discharge has not been established, but it will depend on future funding and the status of the fuel rods and instrumentation.

The primary reason for extending the irradiation of IFA-432 was to obtain fission gas release data at higher burnups. Consequently, fuel rod puncture and fission gas analyses represent the minimum PIE expected for rods 2, 3, and 5. Rod 2 was fabricated with a large fuel-to-cladding gap (380 µm) and has operated at higher fuel temperatures than the other rods in IFA-432. Rod 3 was fabricated with a small original gap (75 µm) and has operated at temperatures 400 to 600K below those in rod 2. Comparing the gas release data from these two rods with data obtained from rods 1 and 8, which operated at intermediate temperatures to lower burnups, will assist in separating the effects of temperature and burnup on fission gas release in UO_2 fuels. These fuel rods all contained the same type of fuel and were irradiated under identical conditions, which makes these comparisons especially beneficial.

The effects of initial fuel density and microstructure on fission gas release at high burnups can be assessed by comparing the gas release data from rod 5 to the data obtained from rods 1, 2, 3, and 8. Rod 5 contained fuel with lower density and smaller grain size than the other fuel rods, which could potentially affect fission gas release. However, the internal pressure data from rods 1 and 5 suggest little difference in gas release for these two fuel types at 2600 GJ/kgU (30 GWd/MTU). Fuel rod puncture data will confirm the accuracy of the gas release estimates derived from the internal pressure data.

In addition to providing fission gas release data, detailed examinations of the fuel and cladding after more than 4300-GJ/kgU (40-GWd/MTU) burnup would provide an excellent opportunity to assess the effects of high burnup on UO_2 fuel rods. These rods are especially suited for determining the effects of high burnup because:

- Extensive characterization of the fuel and cladding was made prior to irradiation.
- Detailed irradiation histories of the fuel rods are available.
- PIE data will be available from rods 1, 6, and 8 at lower burnup levels.

At present, definite plans for PIE of rods 2, 3, and 5 have not been made. The extent of the examinations will depend on future funding, which has not been identified.

RODS 1 AND 6

Rods 1 and 6 of IFA-432 were removed from the assembly in June 1981 and are scheduled to undergo PIE at Harwell from April to September 1982. The examinations to be performed on these rods were selected to obtain the maximum amount of information for the available funding. Optional examinations have also been identified in the event that additional funding becomes available.

The examinations for rods 1 and 6 are divided into three categories: base program, special examinations, and optional examinations (see Table 5). The base program will establish the condition of the fuel and cladding, fission gas release, fuel microstructure, and fission product deposits on the inside of the cladding. Under special examinations, the radial distribution of fission products (Xe, Zr, and Cs) in the fuel will be measured by electron probe microanalysis (EPMA) for comparison with computer code calculations. Each fuel rod will be internally pressurized with helium to 3.4 MPa to verify the integrity of the TC connections.

The optional examinations will not be made unless additional funds are allocated. These examinations would generally support and quantify the data obtained under the existing program. For instance, examination of the krypton-85 concentrations in microcores taken from the fuel pellet at various radial locations would verify and quantify the EPMA scans. The relative concentration of fission gas at grain boundaries versus grain interiors would also be determined by comparing the amount of krypton-85 released when the microcores are crushed with the amount released during fusion of the crushed powders. Neutron-induced TC decalibration is an important consideration in evaluating in-pile temperature data. The TC examination to characterize and quantify transmutation products would help quantify neutron-induced decalibration of tungsten-rhenium TCs.

TABLE 5. Postirradiation Examinations for Rods 1 and 6 of IFA-432

Base Program	
Visual	Rods 1 and 6
Gamma scan	Rods 1 and 6
Profilometry	Rod 1
Puncture/fission gas analysis	Rods 1 and 6
Ceramography	Rods 1 and 6 (2 locations)
Bulk density	Rods 1 and 6
Cladding ID examination	Rod 1
Fuel burnup	Rod 6
Special Examinations	
Radial distribution of fission products (electron probe microanalysis)	Rods 1 and 6 (20 points across diameter at one location)
Fuel rod pressurization	Rods 1 and 6
Optional Examinations	
Radial distribution of ^{85}Kr (in microcores)	
Additional ceramography and metallography	
Thermocouple examination to characterize and quantify transmutation products	

DATA PRESENTATION

In-reactor data collected from IFA-432 by the Halden IBM/1800 on-line computer data acquisition system for the period from June 1980 through June 1981 are presented in this section. Linear heat generation rates (LHGRs), fuel temperatures, and cladding elongation data are plotted as a function of time. In each plot, the relative position of the rod number that appears in the upper left-hand corner corresponds to the relative position of the curve in each figure.

Internal pressure data were taken manually and are presented in tabular form along with moderator temperatures and reactor and assembly power levels. All of the pressure data taken from June 1980 to June 1981 are presented. The calculated burnup levels at upper and lower TC locations are also given on a monthly basis. Finally, fuel rod puncture data and ceramography results from the PIE of rod 8 are presented.

POWER HISTORIES

Power histories for the upper and lower TC locations for all six rods are presented in Figures 5 through 24. These values were deduced from the vanadium self-powered neutron detector (SPND) readings after applying correction factors to account for local mass distribution, radial flux tilt, and axial flux shape (see Appendix A). A correction of -0.66% per GWd/MTU was made for the burnup-dependent depletion of uranium-235 (taken from depletion calculations performed at Halden). The assembly power calibration is discussed in Appendix B.

The neutron detector readings during transient periods have not been corrected for the response lag of the detector caused by incomplete saturation of the vanadium emitter. This lag amounts to about 5 min during a power ramp or one-third of the normal data collection frequency.

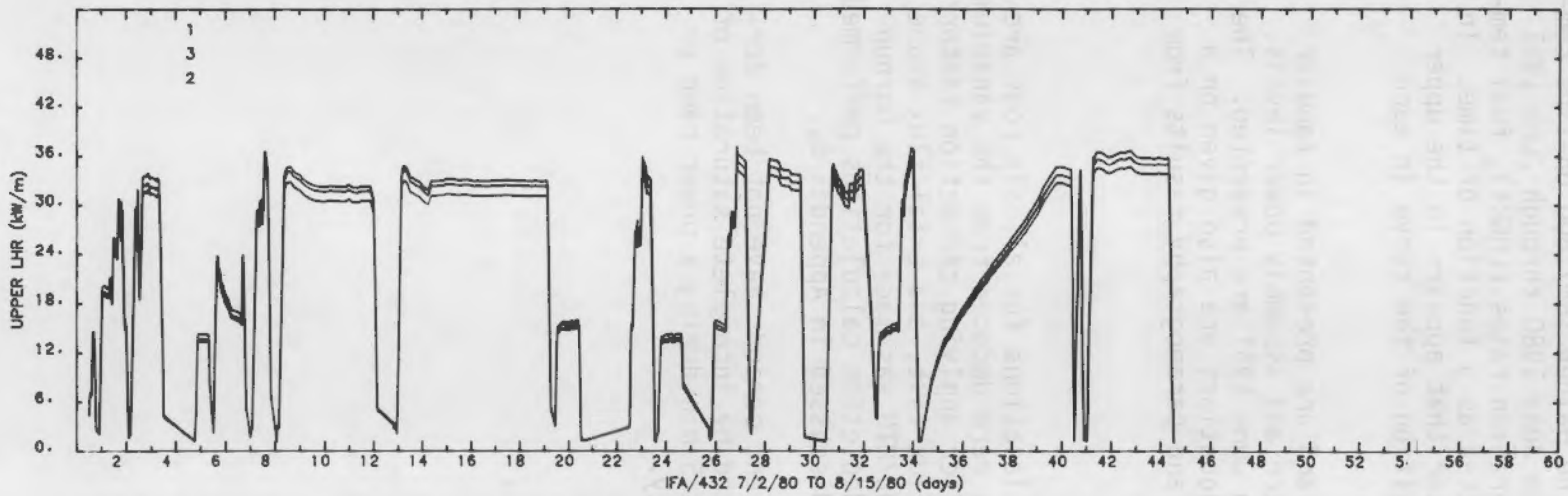

FIGURE 5. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from July 2, 1980, to August 15, 1980

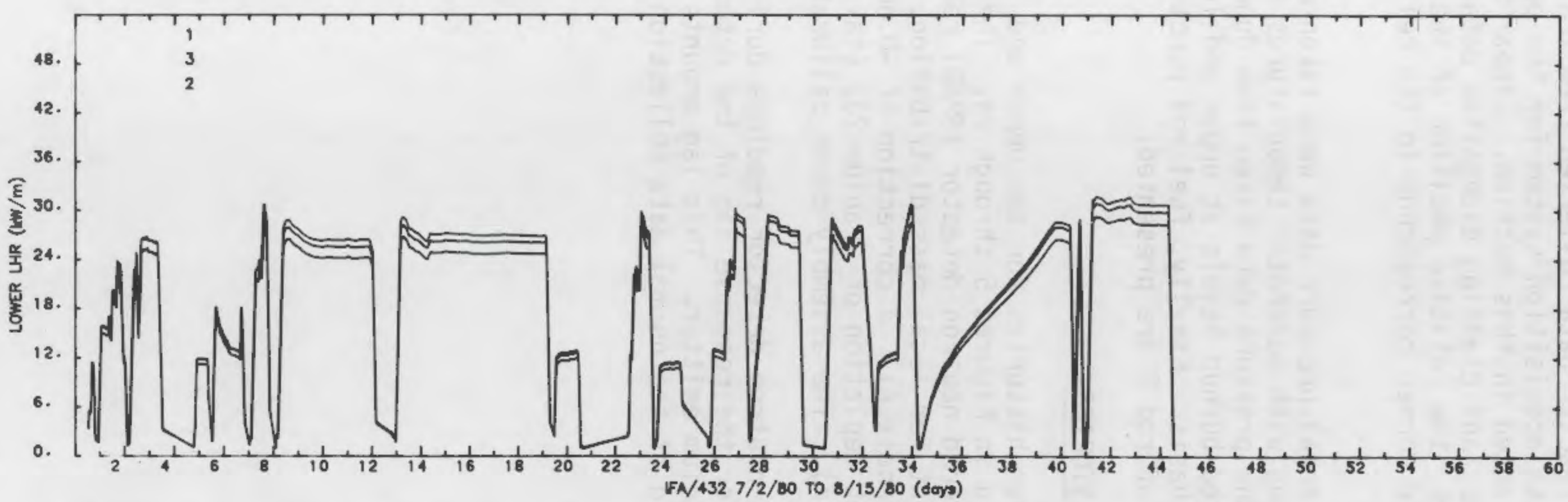

FIGURE 6. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from July 2, 1980, to August 15, 1980

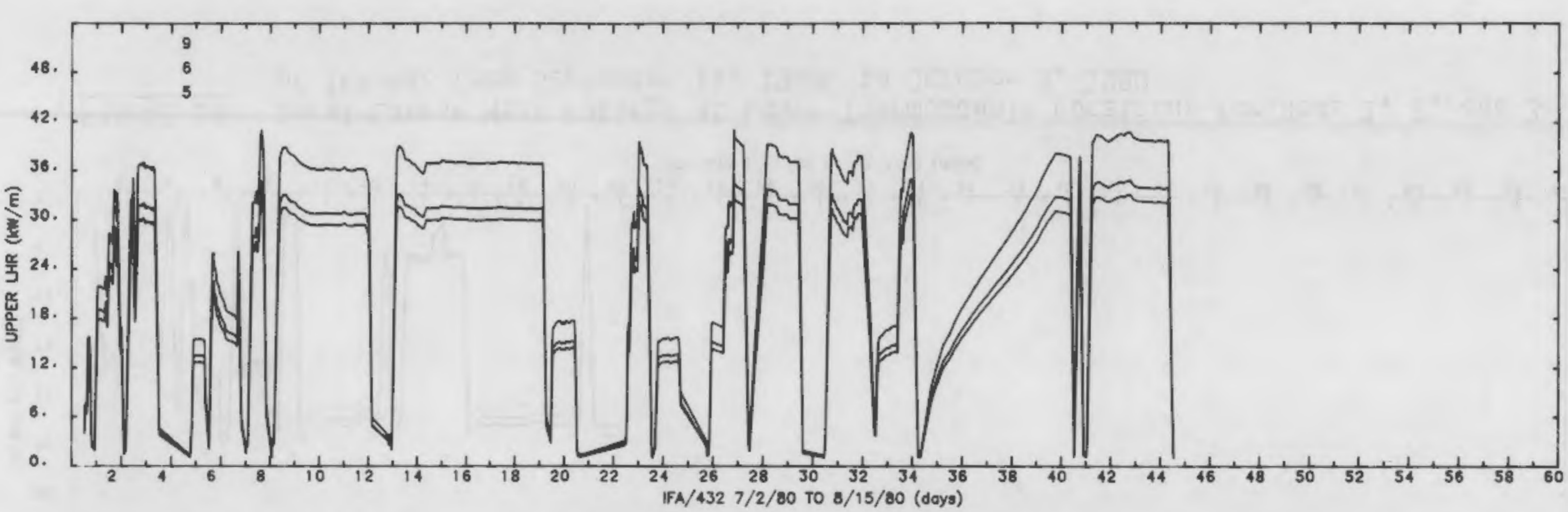

FIGURE 7. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from July 2, 1980, to August 15, 1980

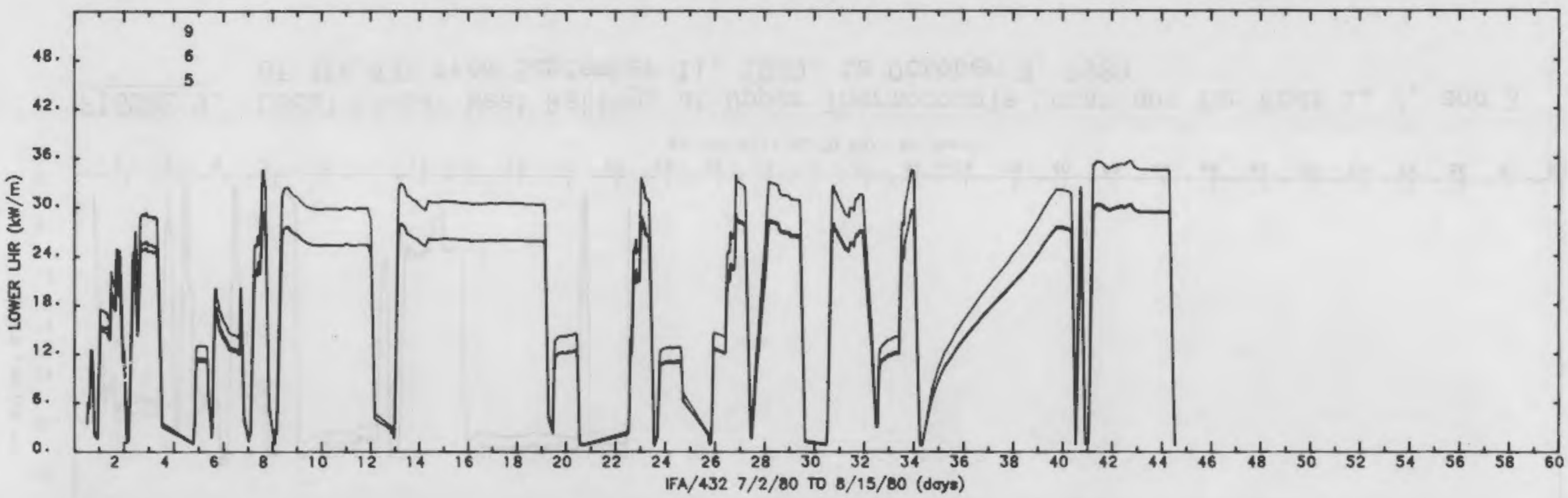

FIGURE 8. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from July 2, 1980, to August 15, 1980

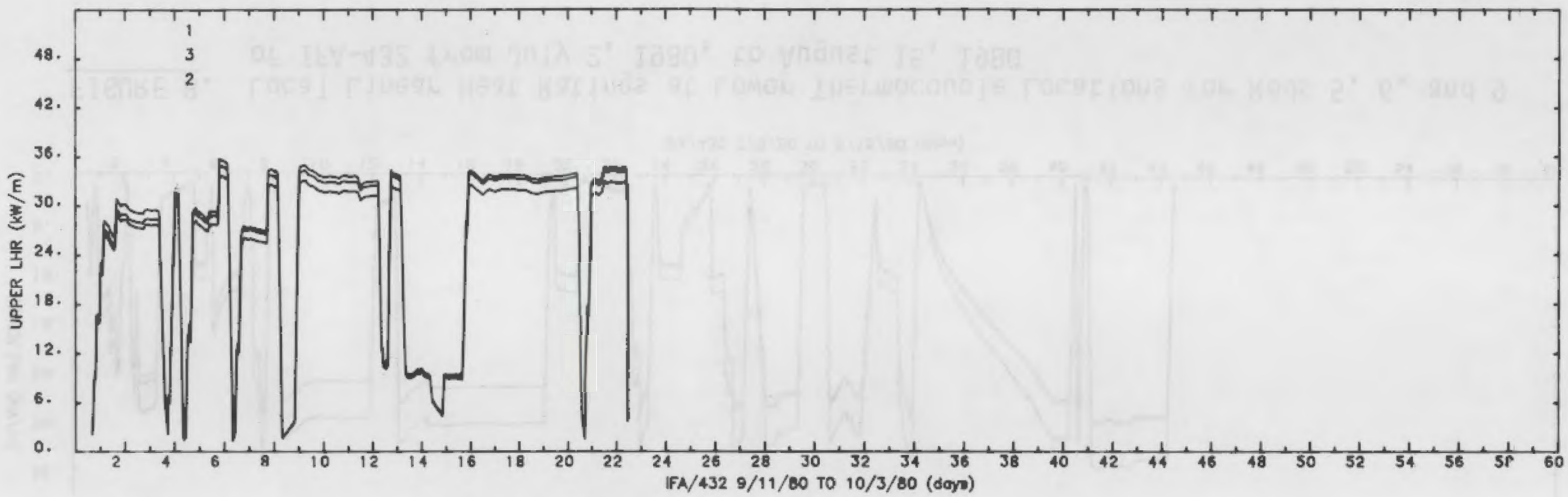

FIGURE 9. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from September 11, 1980, to October 3, 1980

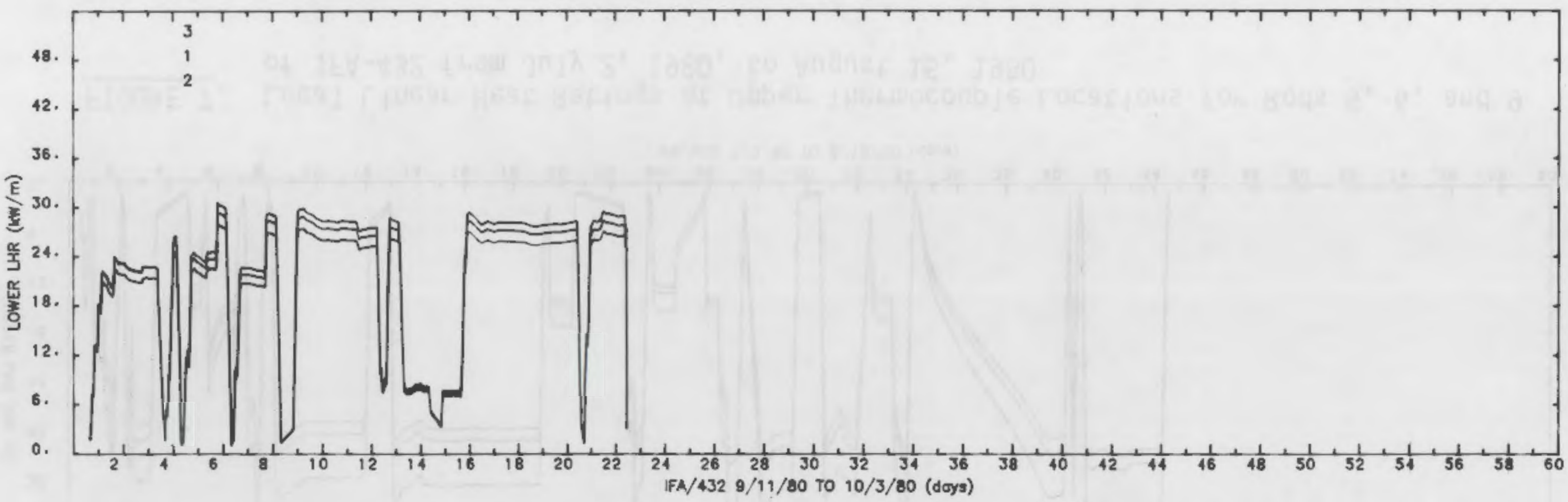

FIGURE 10. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from September 11, 1980, to October 3, 1980

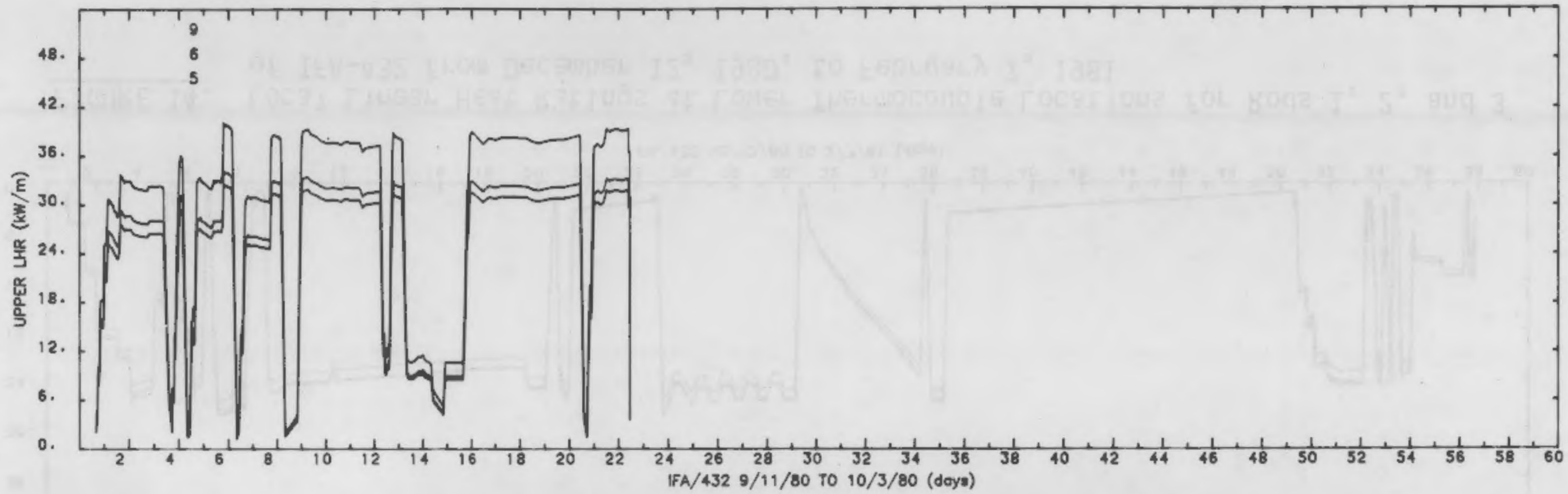

FIGURE 11. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from September 11, 1980, to October 3, 1980

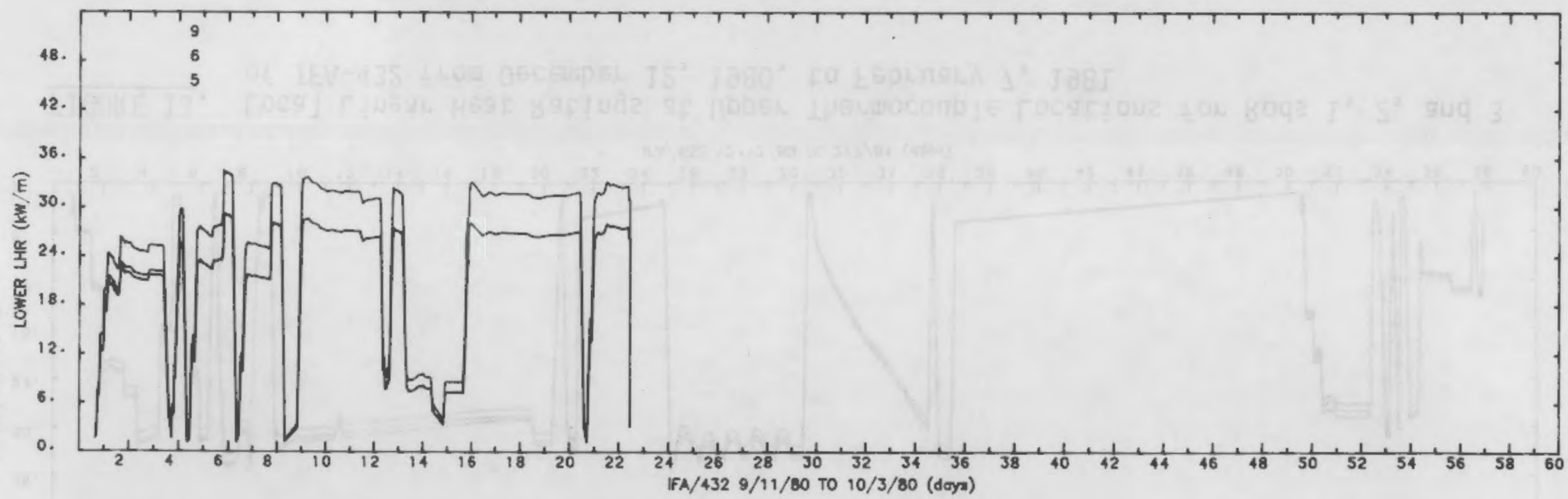

FIGURE 12. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from September 11, 1980, to October 3, 1980

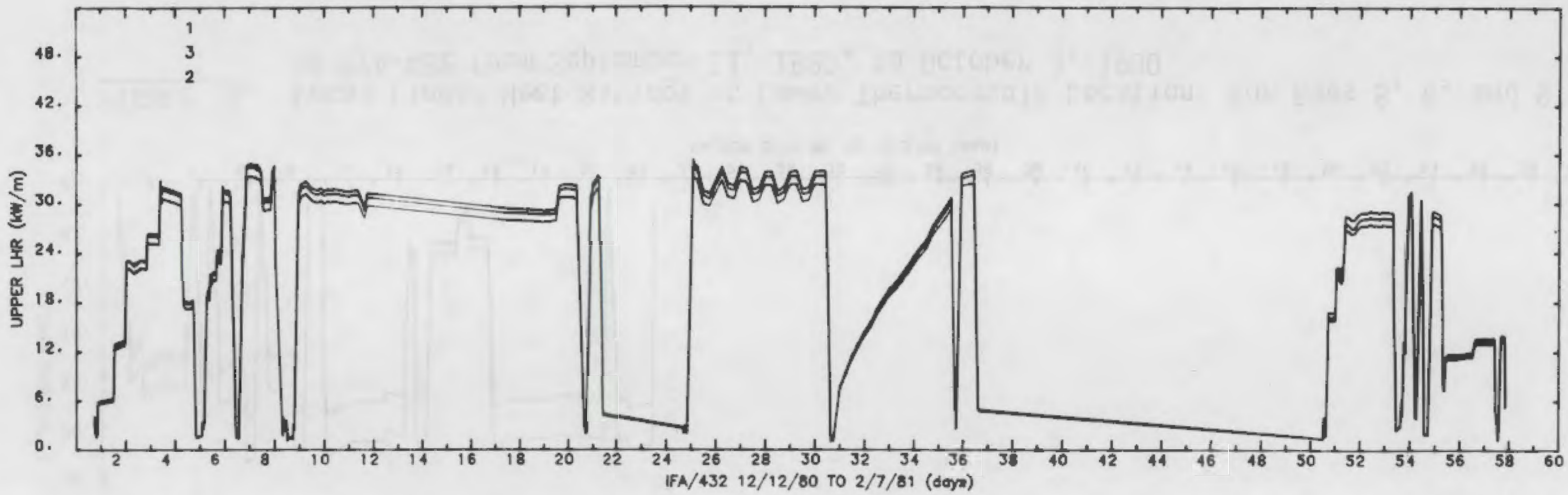

FIGURE 13. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from December 12, 1980, to February 7, 1981

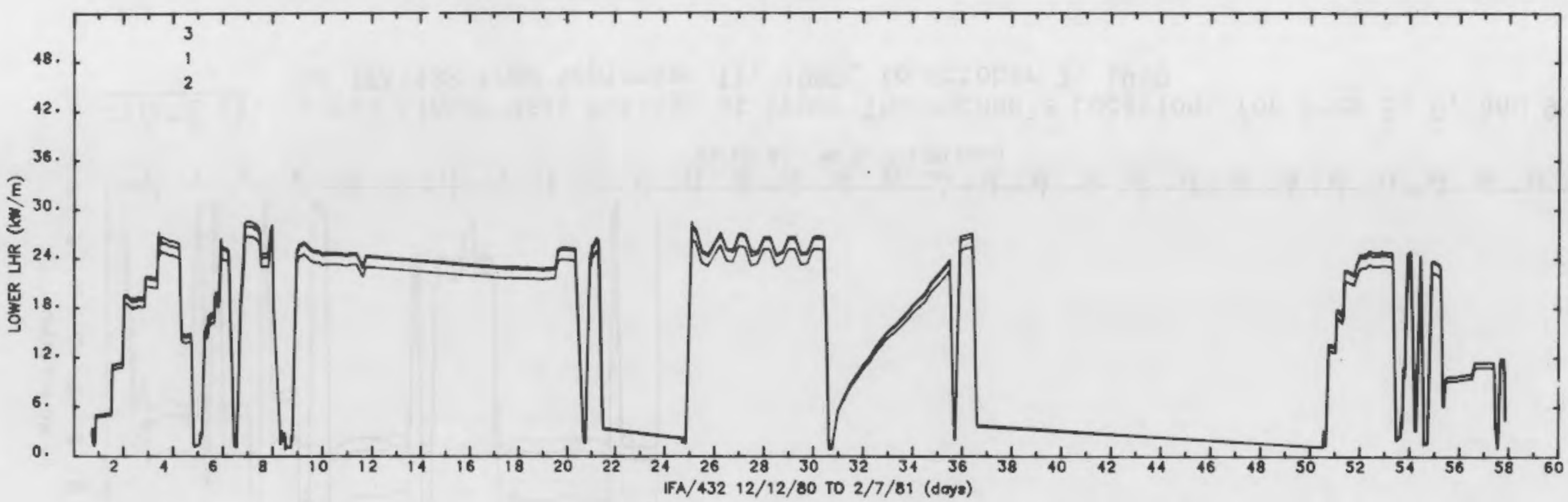

FIGURE 14. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from December 12, 1980, to February 7, 1981

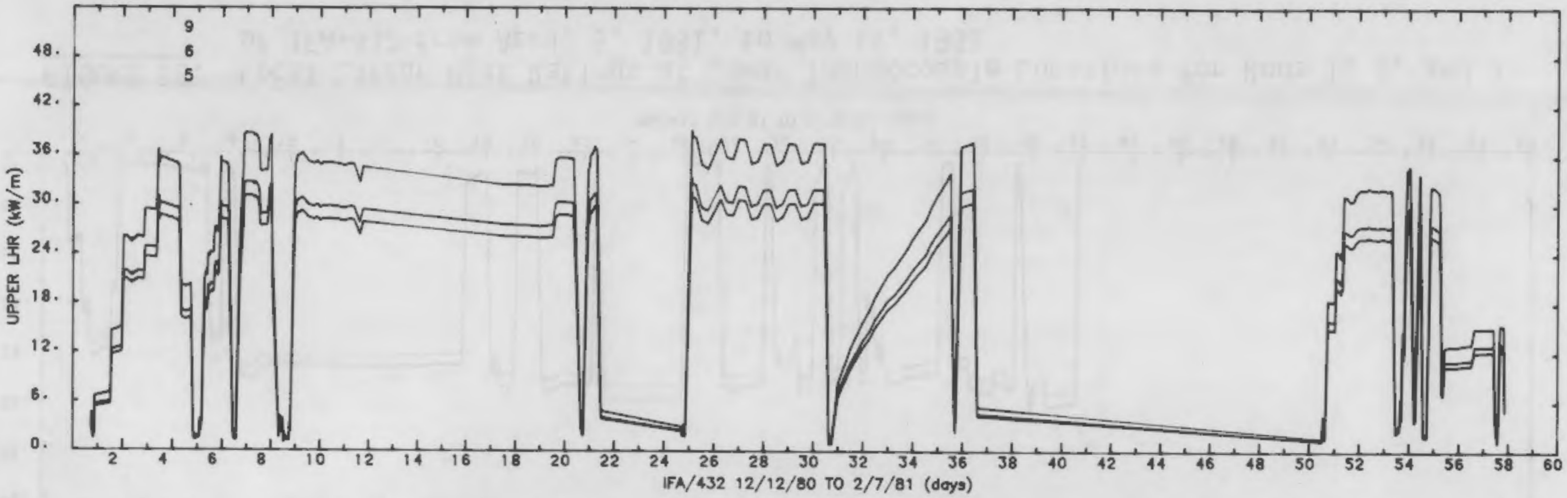

FIGURE 15. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from December 12, 1980, to February 7, 1981

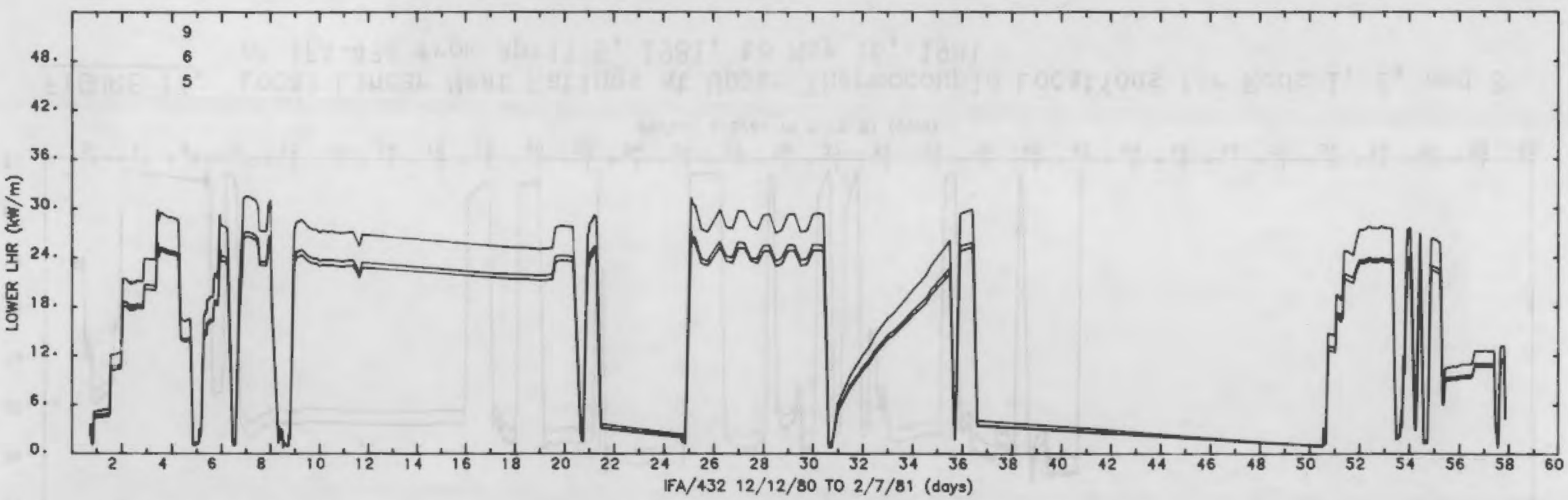

FIGURE 16. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from December 12, 1980, to February 7, 1981

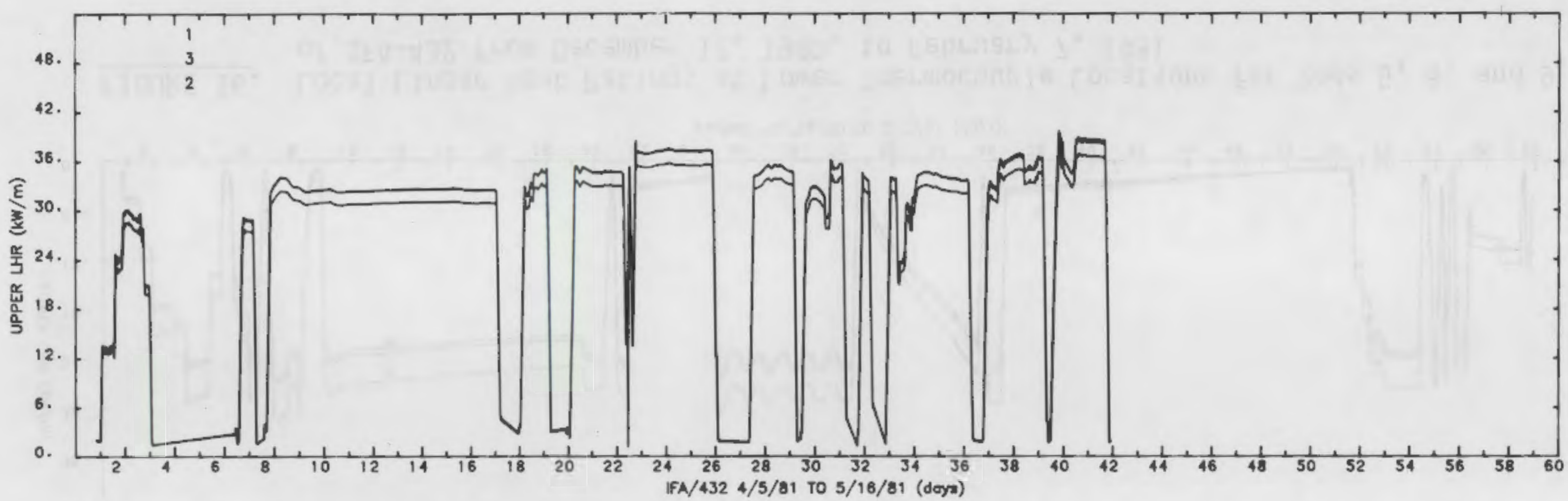

FIGURE 17. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from April 5, 1981, to May 16, 1981

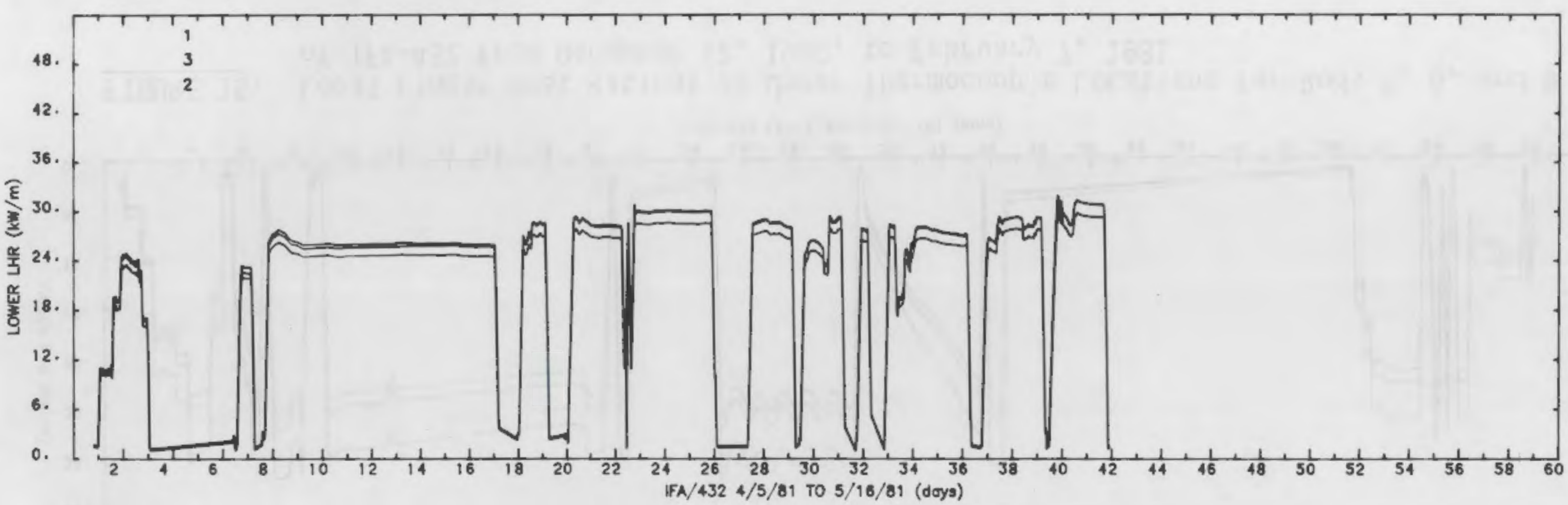

FIGURE 18. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from April 5, 1981, to May 16, 1981

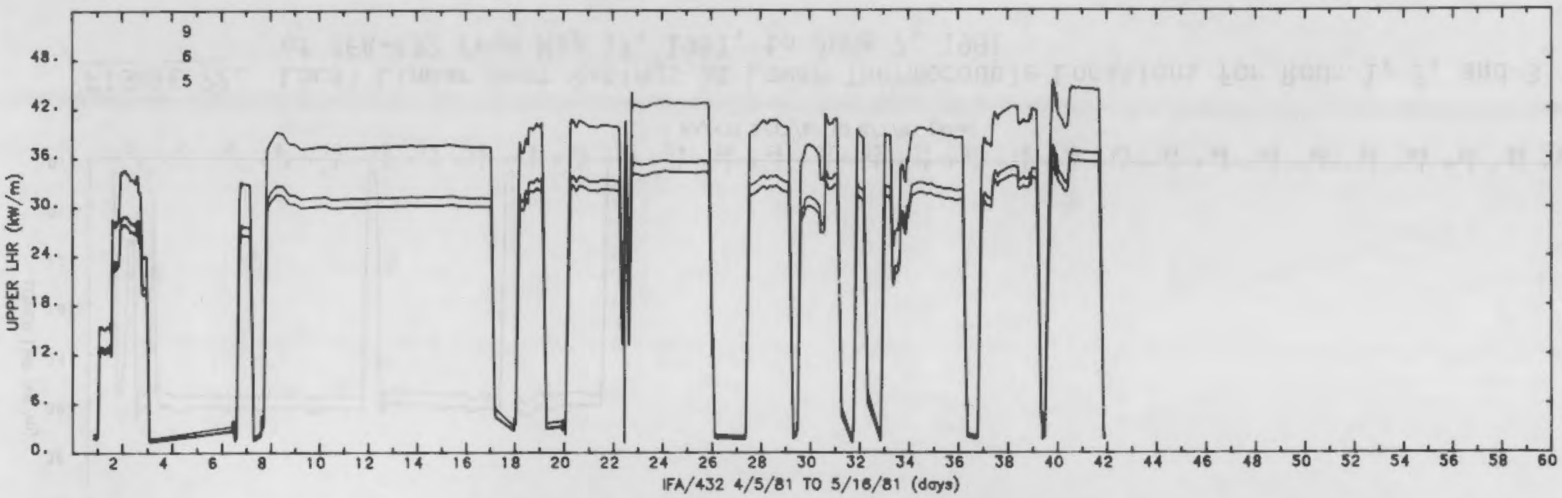

FIGURE 19. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from April 5, 1981, to May 16, 1981

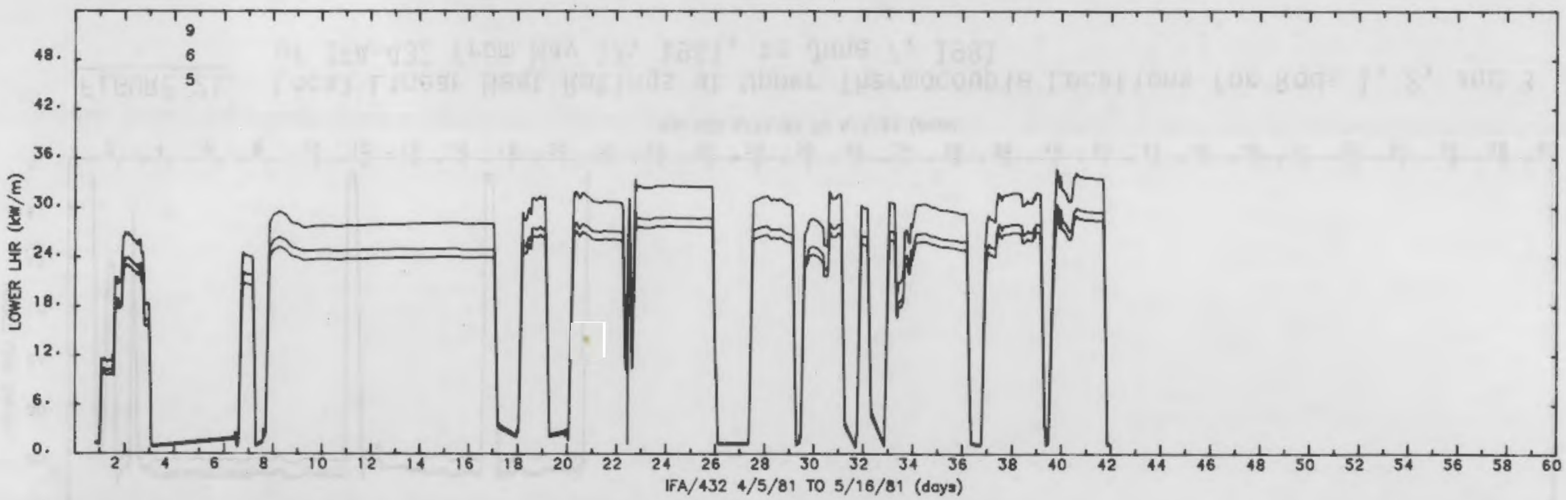

FIGURE 20. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from April 5, 1981, to May 16, 1981

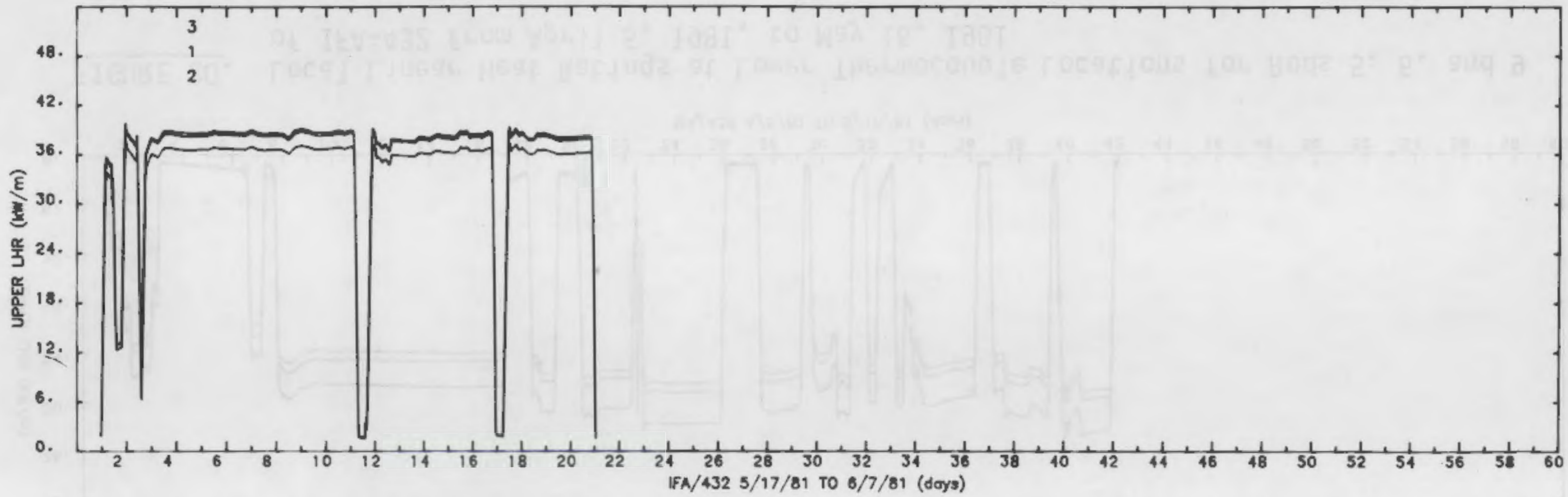

FIGURE 21. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from May 17, 1981, to June 7, 1981

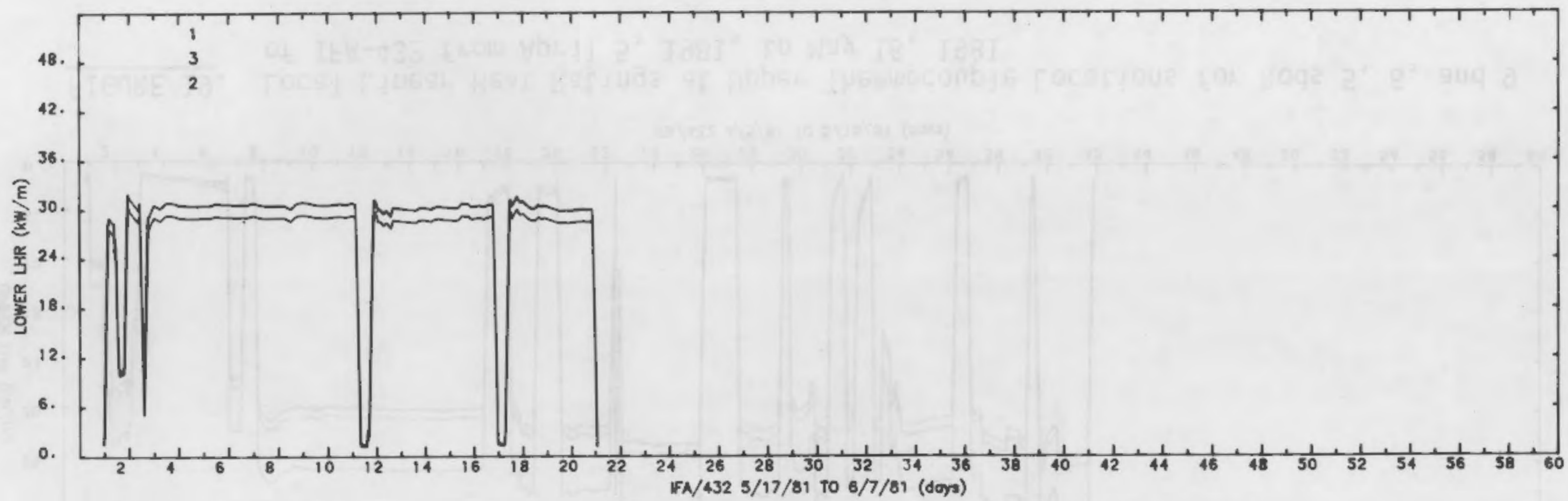

FIGURE 22. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from May 17, 1981, to June 7, 1981

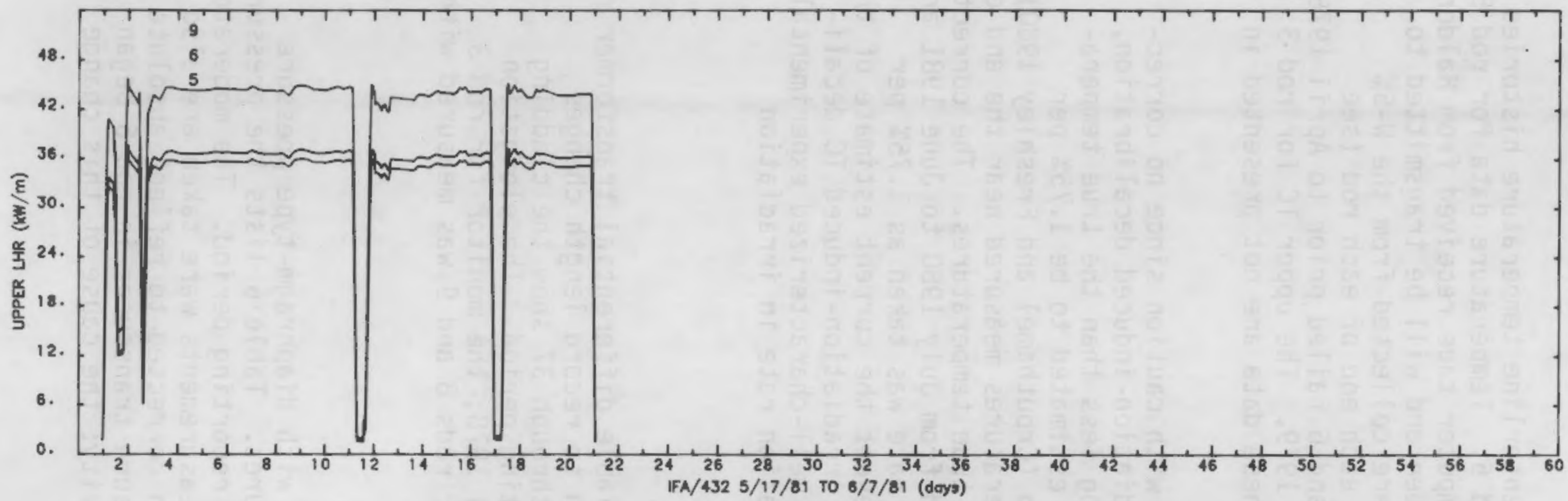

FIGURE 23. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from May 17, 1981, to June 7, 1981

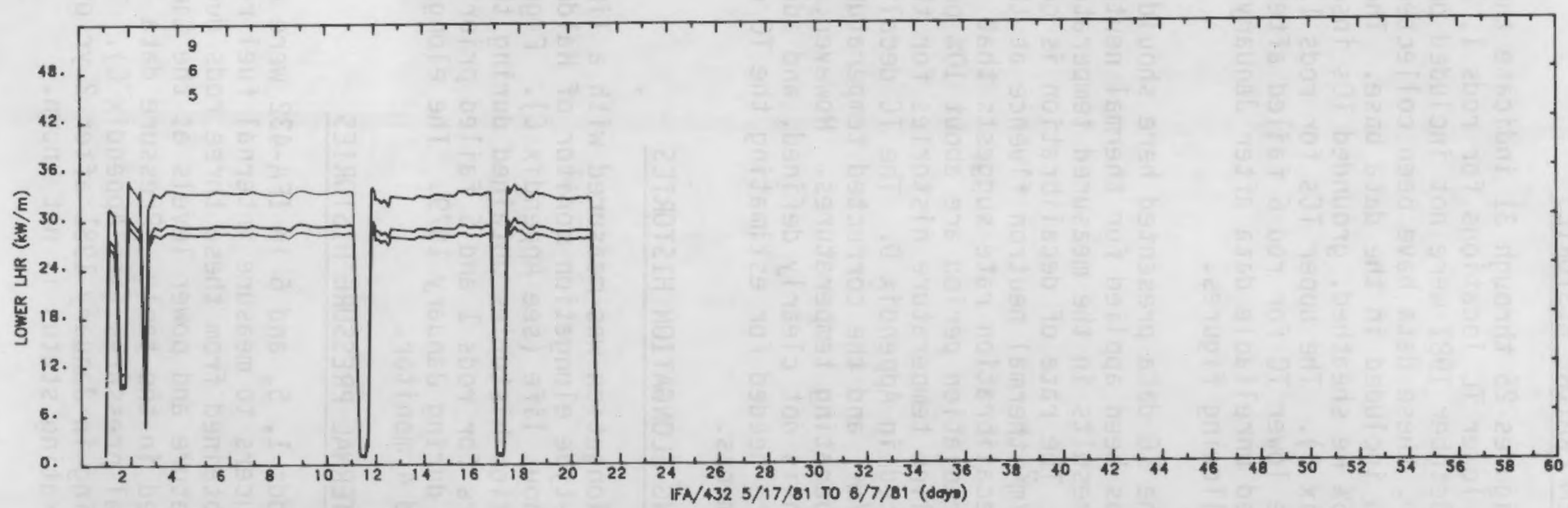

FIGURE 24. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from May 17, 1981, to June 7, 1981

FUEL TEMPERATURE HISTORIES

Figures 25 through 31 indicate the fuel centerline temperature histories at the lower TC locations for rods 1, 2, 3, and 5. Temperature data for rod 5 after October 1981 were not included on the computer tape received from Halden; however, these data have been collected by Halden and will be transmitted to PNL and included in the data base. The data were collected from the W-5% Re/W-26% Re sheathed, grounded TCs inserted in each end of each rod (see Appendix C). The upper TCs for rods 1, 2, 5, and 6 failed prior to April 1978; and the lower TC for rod 6 failed after August 1979. The upper TC for rod 3 provided unreliable data after January 1979; these data are not presented in the following figures.

The TC data presented here should be used with caution since no correction has been applied for thermal neutron irradiation-induced decalibration, which results in the measured temperatures being less than the true temperatures. The rate of decalibration is currently estimated to be 1.75% per 10^{24} n/m^2 thermal neutron fluence at the TC tip (Crouthamel and Freshley 1980). This decalibration rate suggests that the temperatures measured near the end of the irradiation period are about 10% below the true temperatures. The corrected centerline temperature histories for the period from July 1980 to June 1981 are presented in Appendix D. The TC decalibration rate was taken as 1.75% per 10^{24} n/m^2, and the corrected temperatures represent the current estimate of the true operating temperatures. However, neutron irradiation-induced TC decalibration is not clearly defined, and additional well-characterized experimental data are needed for estimating the TC decalibration rate in irradiation experiments.

CLADDING ELONGATION HISTORIES

Elongation was measured with a linear variable differential transformer (LVDT)-type elongation monitor of Halden design to record length changes throughout life (see Appendix C). Figures 32 through 37 show the cladding elongation histories obtained during the reporting period. The elongation monitors for rods 1 and 5 failed prior to April 1978; the monitor for rod 3 failed during January 1979. The elongation for rods 8 and 9 was measured with the rod 4 monitor.

ROD INTERNAL PRESSURE HISTORIES

Rods 1, 5, and 6 in IFA-432 were equipped with diaphragm-type pressure transducers to measure internal fuel rod pressures. Table 6 lists the pressure data obtained from these three rods during the reporting period. The moderator temperature and power levels at the time the measurements were taken are also included in the table. Pressure data have been corrected to reflect absolute internal pressures (see Appendix C). The pressure transducer in rod 6 began operating in January 1981 after 2 yr of inactivity; the cause of this change in operating status is not known.

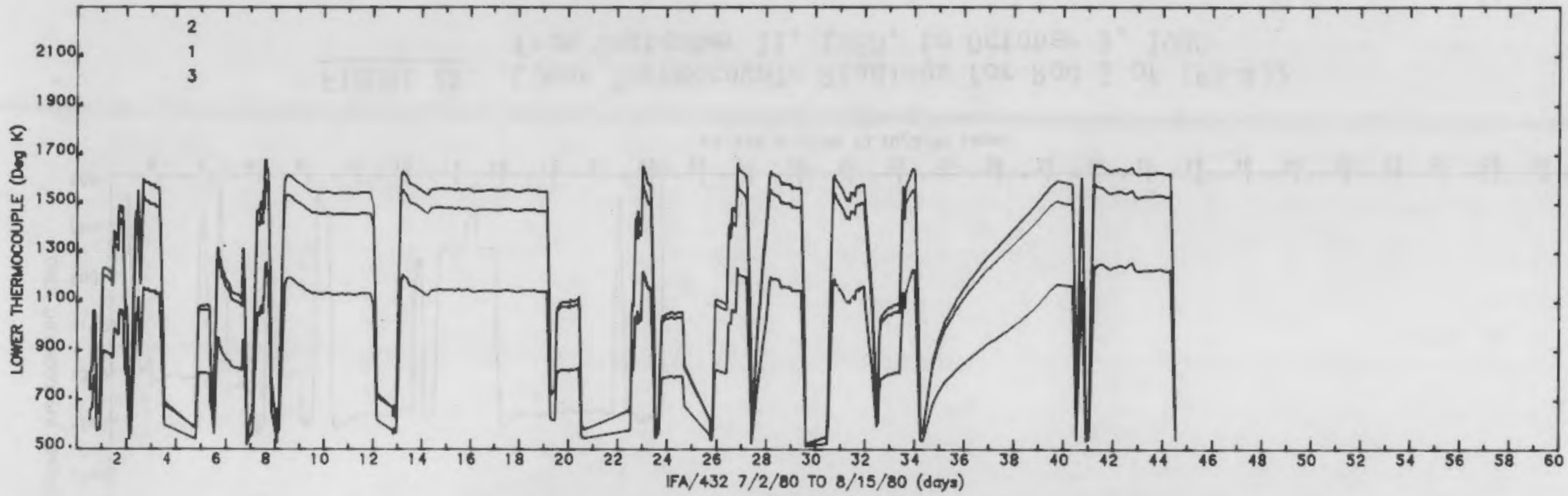

FIGURE 25. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from July 2, 1980, to August 15, 1980

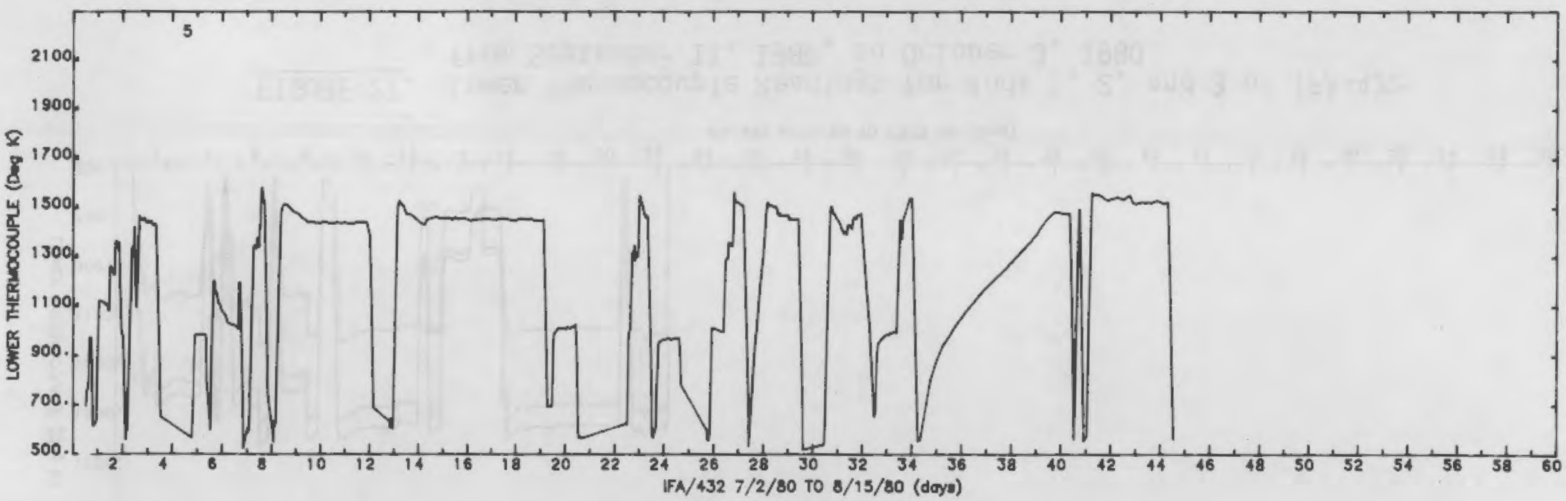

FIGURE 26. Lower Thermocouple Readings for Rod 5 of IFA-432 from July 2, 1980, to August 15, 1980

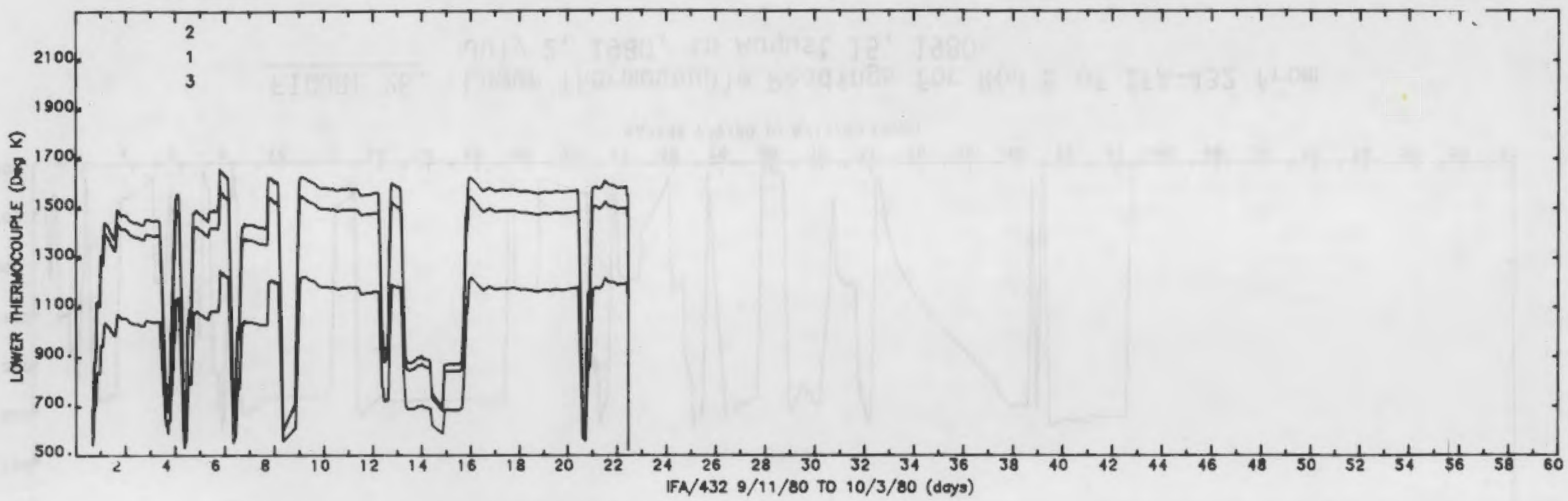

FIGURE 27. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from September 11, 1980, to October 3, 1980

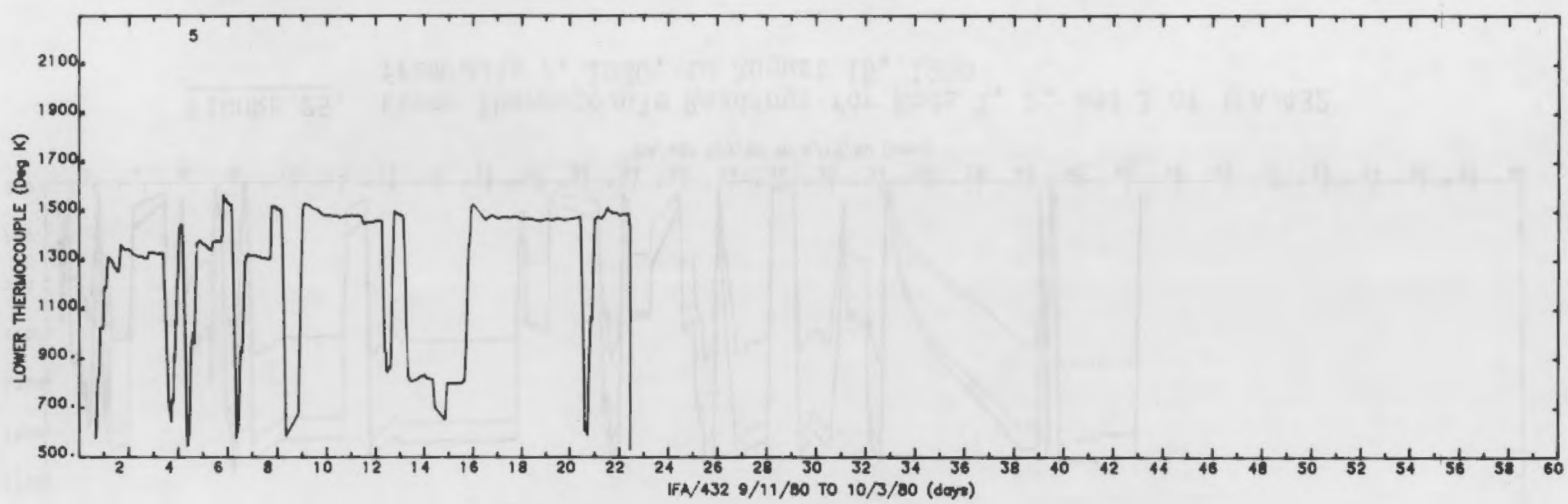

FIGURE 28. Lower Thermocouple Readings for Rod 5 of IFA-432 from September 11, 1980, to October 3, 1980

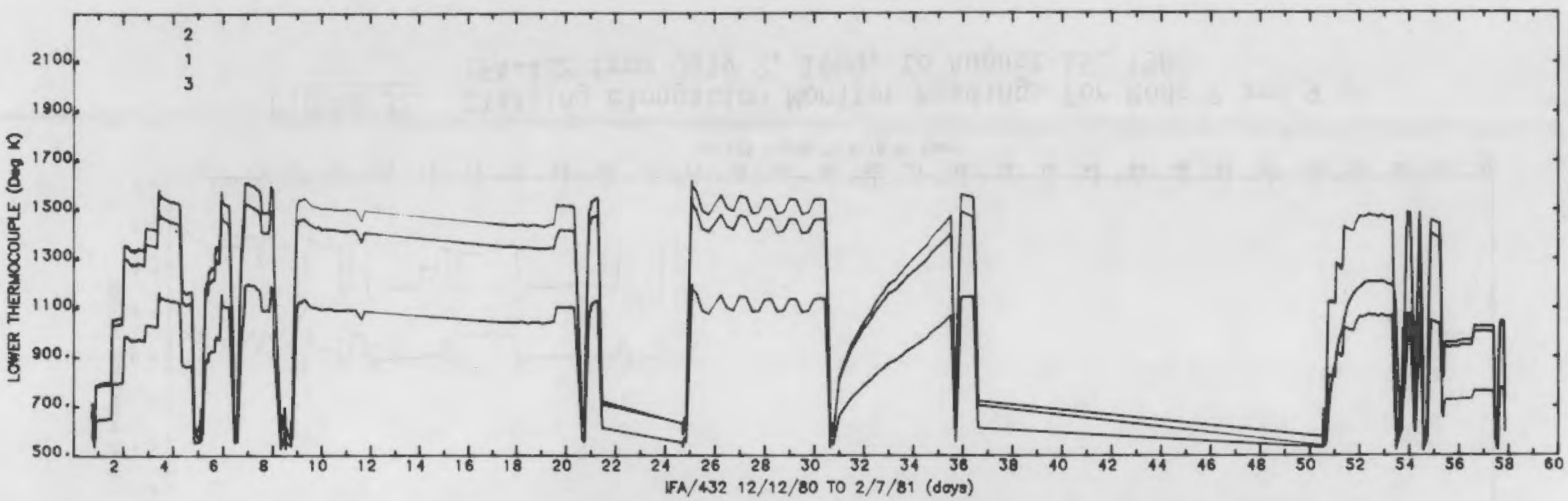

FIGURE 29. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from December 12, 1980, to February 7, 1981

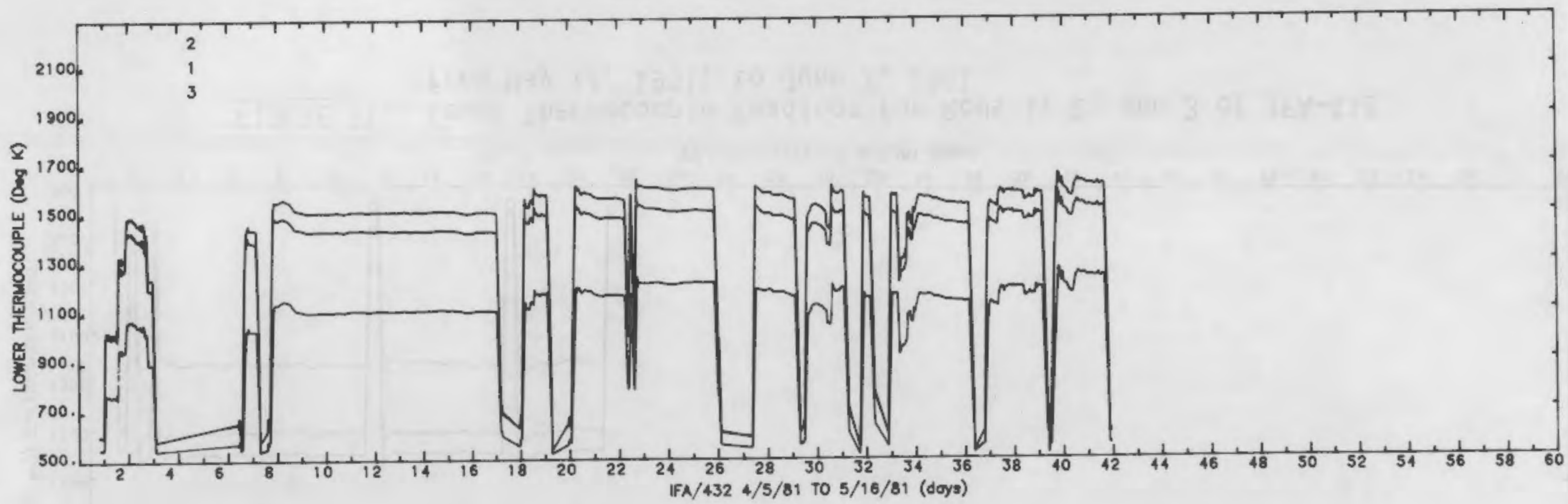

FIGURE 30. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from April 5, 1981, to May 16, 1981

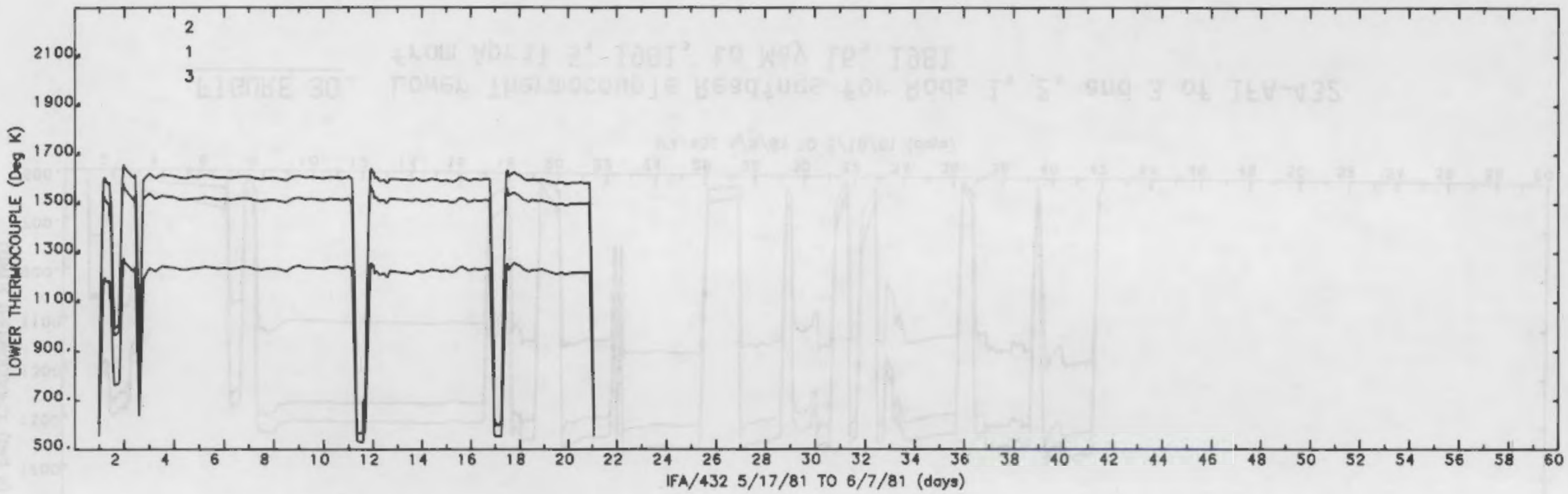

FIGURE 31. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from May 17, 1981, to June 7, 1981

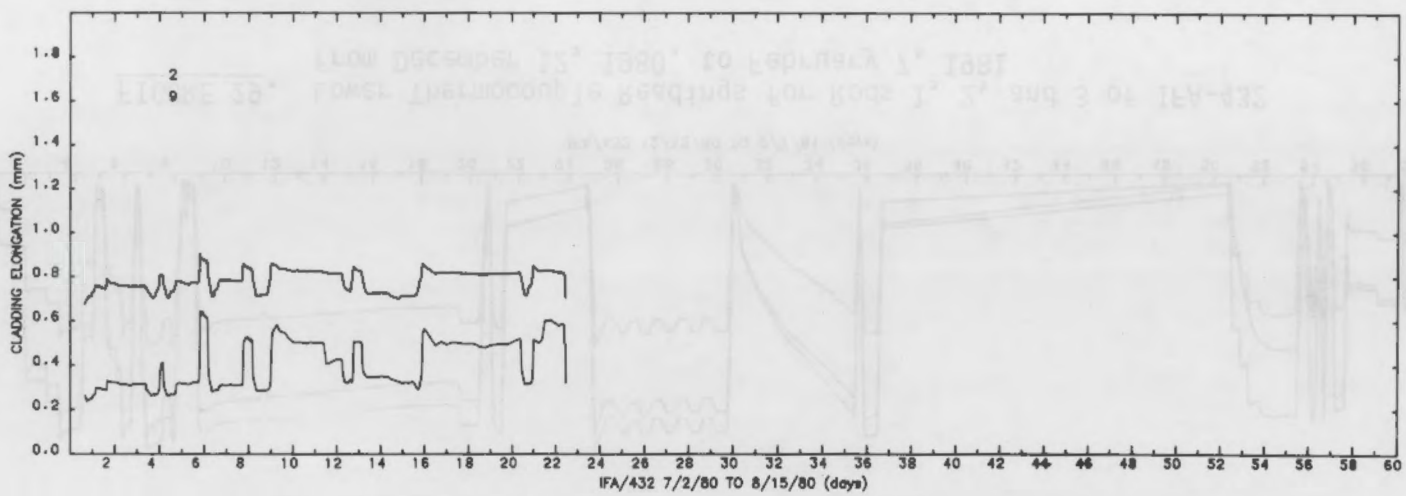

FIGURE 32. Cladding Elongation Monitor Readings for Rods 2 and 9 of IFA-432 from July 2, 1980, to August 15, 1980

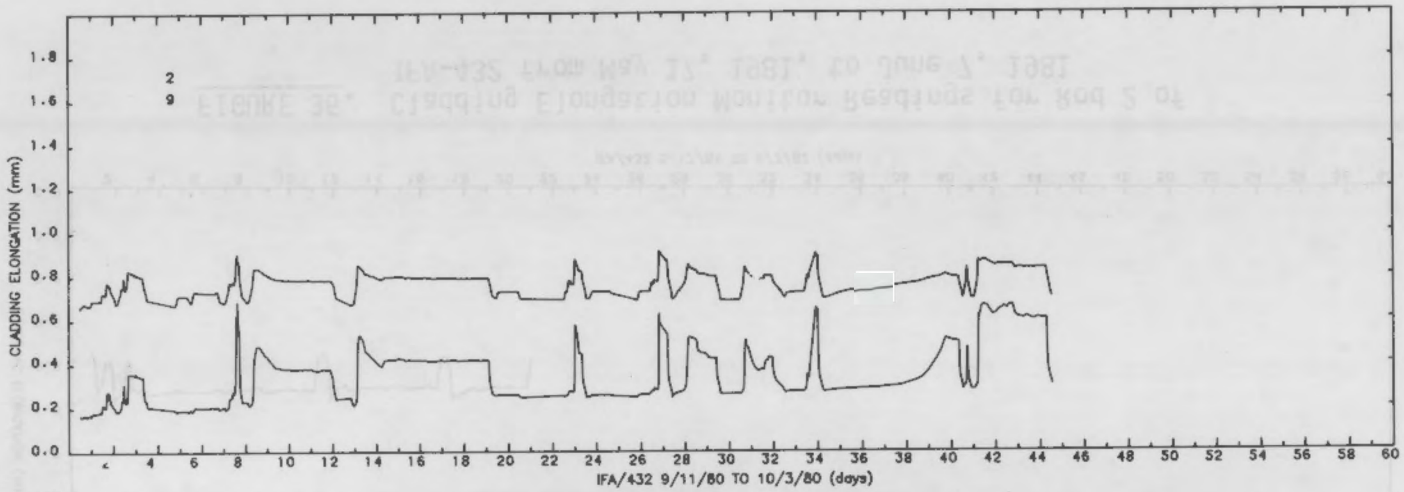

FIGURE 33. Cladding Elongation Monitor Readings for Rods 2 and 9 of IFA-432 from September 11, 1980, to October 3, 1980

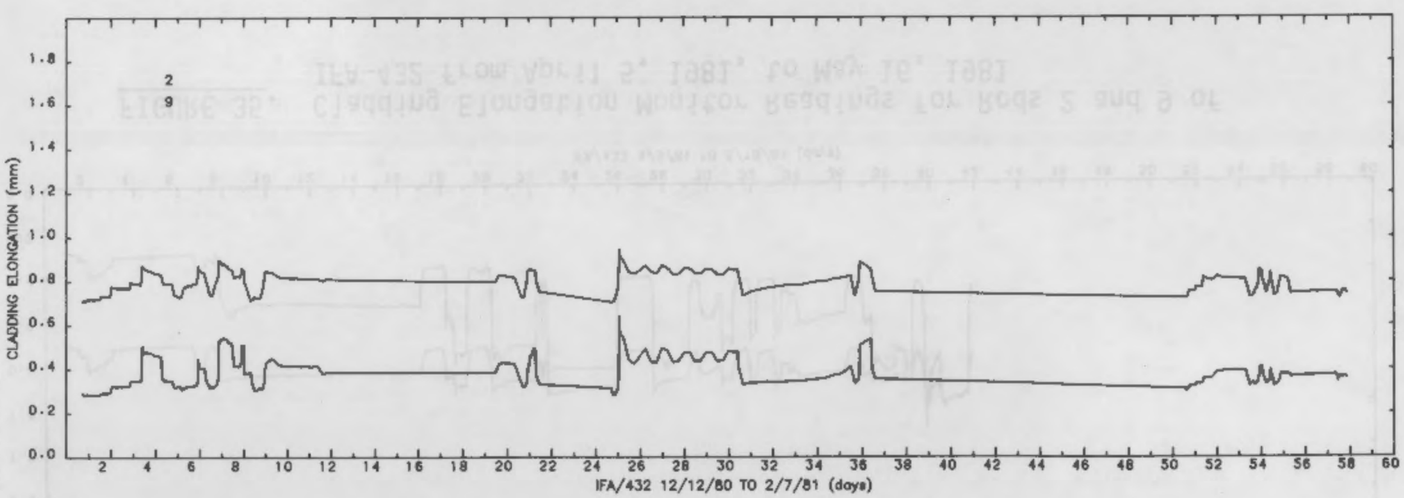

FIGURE 34. Cladding Elongation Monitor Readings for Rods 2 and 9 of IFA-432 from December 12, 1980, to February 7, 1981

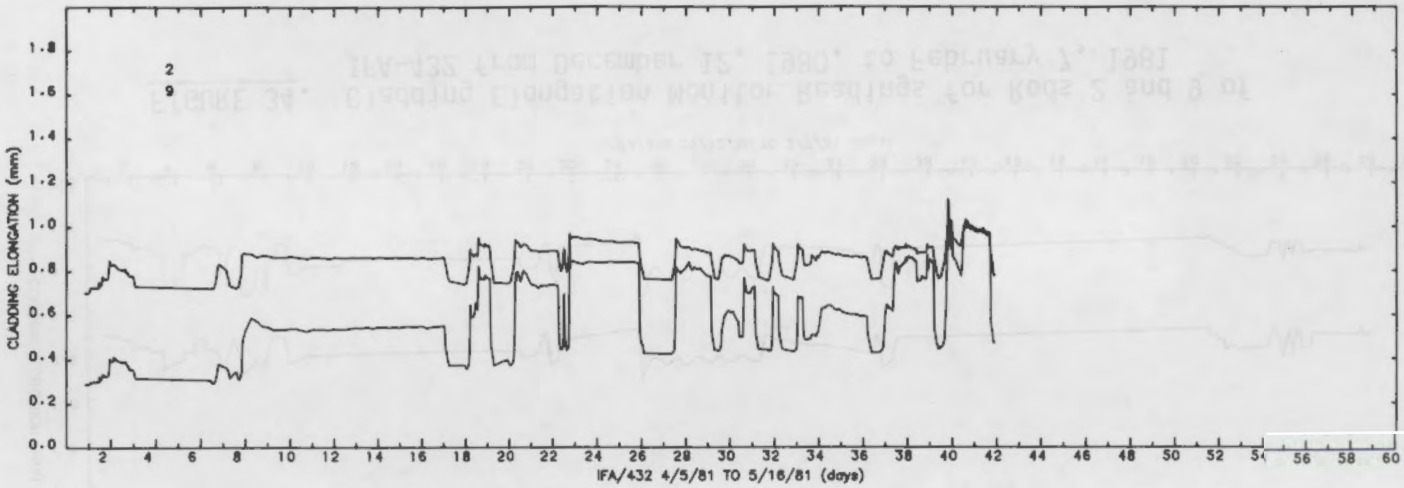

FIGURE 35. Cladding Elongation Monitor Readings for Rods 2 and 9 of IFA-432 from April 5, 1981, to May 16, 1981

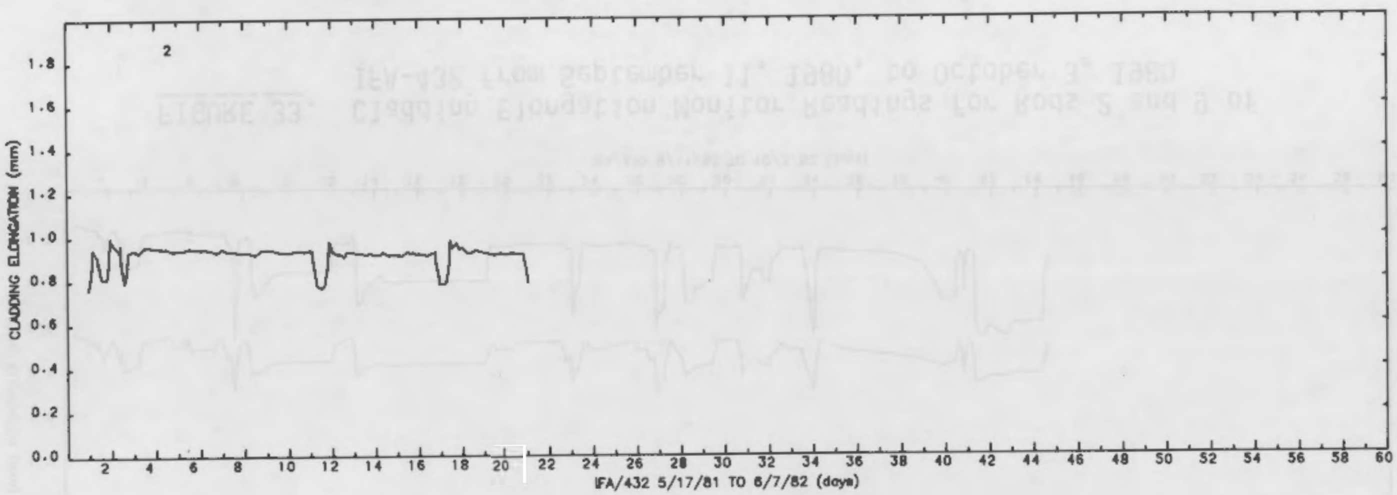

FIGURE 36. Cladding Elongation Monitor Readings for Rod 2 of IFA-432 from May 17, 1981, to June 7, 1981

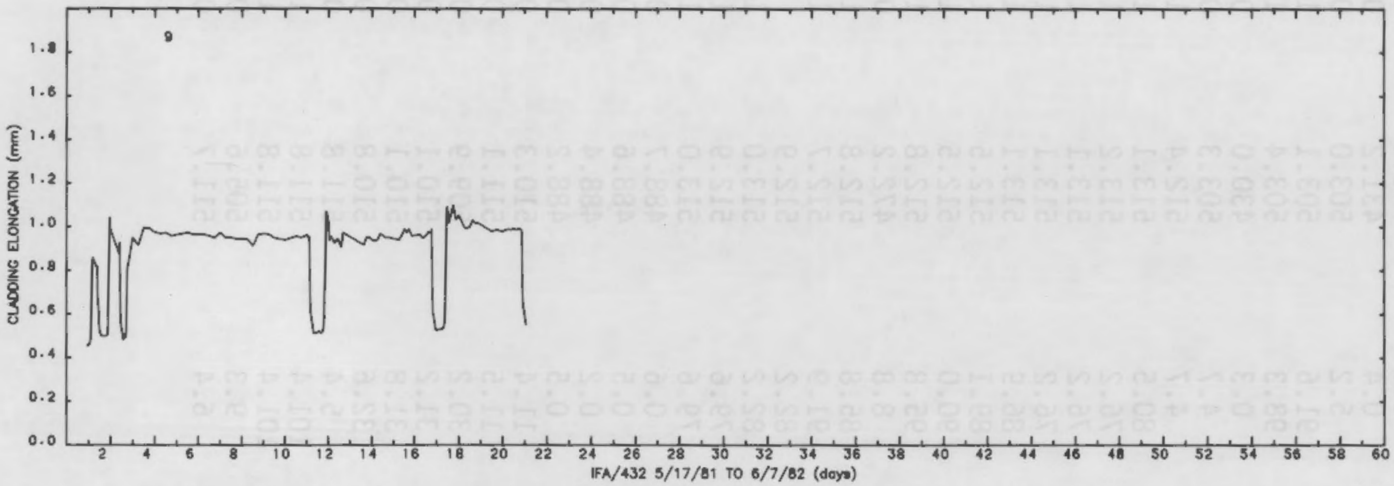

FIGURE 37. Cladding Elongation Monitor Readings for Rod 9 of IFA-432 from May 17, 1981, to June 7, 1981

BURNUP

Calculated burnups at each TC location for each rod are presented in Table 7. These are the local burnups at the end of each month of operation and were calculated by numerically integrating the depletion-corrected power history over time. There was good agreement between the results using this method and PIE data from rod 6 of IFA-431 (Nealley et al. 1979).

TABLE 7. Burnup Levels for IFA-432[a]

Month-Year	Location	Rod 1	Rod 2	Rod 3	Rod 9	Rod 5	Rod 6
4-80	Upper TC	2532.5	2446.4	2448.0	77.2	2545.1	2563.0
	Lower TC	1865.4	1835.4	1869.0	65.0	1928.8	1910.6
5-80	Upper TC	2558.1	2471.4	2471.9	104.7	2569.0	2588.1
	Lower TC	1886.1	1855.7	1889.2	87.8	1949.6	1931.6
7-80	Upper TC	2620.1	2531.8	2530.2	171.8	2627.4	2649.2
	Lower TC	1935.3	1903.7	1937.6	142.8	1999.8	1981.7
8-80	Upper TC	2658.1	2568.9	2565.9	212.8	2663.1	2686.6
	Lower TC	1966.4	1934.1	1968.2	177.6	2031.4	2013.4
9-80	Upper TC	2708.8	2618.4	2613.7	268.2	2711.3	2736.6
	Lower TC	2007.0	1974.0	2008.5	223.5	2073.2	2055.0
10-80	Upper TC	2715.2	2624.6	2619.8	275.2	2717.3	2742.9
	Lower TC	2012.2	1979.0	2013.7	229.3	2078.5	2060.2
12-80	Upper TC	2749.6	2658.6	2652.5	312.5	2749.5	2776.5
	Lower TC	2039.4	2005.9	2040.3	258.9	2105.2	2087.5
1-81	Upper TC	2783.4	2691.8	2684.4	348.8	2780.8	2809.3
	Lower TC	2065.6	2031.6	2065.7	287.2	2130.9	2113.6
2-81	Upper TC	2796.2	2704.3	2696.4	362.6	2792.8	2821.9
	Lower TC	2076.1	2041.9	2076.0	298.6	2141.3	2124.2
4-81	Upper TC	2860.2	2767.3	2758.2	434.5	2854.7	2885.5
	Lower TC	2128.0	2092.6	2125.4	353.1	2191.0	2175.5
5-81	Upper TC	2949.6	2855.5	2844.8	535.1	2941.1	2974.0
	Lower TC	2200.2	2163.4	2194.7	429.5	2260.5	2246.9
6-81	Upper TC	2972.4	2878.0	2867.0	560.8	2963.2	2996.6
	Lower TC	2218.4	2181.3	2212.3	448.9	2278.0	2264.9

(a) Values in GJ/kgU; to convert to GWd/MTU multiply by 0.0116.

TABLE 6. Pressure Data for IFA-432

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, K	Pressures, MPa		
					Rod 1	Rod 5	Rod 6
23 06 80	1455	0.20	0.6	340.4	0.568	0.921	---
1 07 80	1102	0.00	0.4	431.2	0.735	1.000	---
5 08 80	1438	2.00	5.2	503.0	0.951	1.401	---
11 08 80	822	14.40	91.6	503.1	1.460	2.362	---
14 08 80	1450	14.60	98.3	503.4	1.499	2.381	---
10 09 80	1550	0.00	0.3	430.0	0.657	0.960	---
11 09 80	837	1.60	4.7	503.3	0.941	1.431	---
11 09 80	846	1.60	4.7	512.4	1.000	1.529	---
12 09 80	1640	14.10	80.5	513.1	1.470	2.597	---
13 09 80	1115	13.60	76.2	513.2	1.401	2.479	---
13 09 80	1122	13.50	76.2	513.1	1.411	2.430	---
13 09 80	1140	13.50	76.2	513.1	1.411	2.430	---
22 09 80	1254	13.40	86.9	513.1	1.460	2.499	---
22 09 80	1305	13.50	89.1	512.5	1.480	2.509	---
22 09 80	2157	13.50	90.0	512.5	1.470	2.489	---
2 10 80	1232	14.40	95.8	512.8	1.509	2.509	---
12 12 80	750	0.90	8.8	472.2	0.931	1.401	---
15 12 80	1155	13.50	85.8	512.8	1.333	2.695	---
19 12 80	811	14.40	91.9	512.7	1.352	2.656	---
8 01 81	1137	13.70	82.2	512.9	---	2.342	---
8 01 81	1139	13.80	82.2	513.0	1.254	2.381	---
16 01 81	1158	12.70	79.6	512.9	1.235	2.293	---
16 01 81	1201	12.80	79.6	513.0	1.235	2.244	---
11 05 81	1235	0.60	0.6	488.7	0.764	1.019	1.470
11 05 81	1238	0.60	0.5	488.6	0.480	0.941	1.411
11 05 81	1240	0.50	0.2	488.4	0.519	0.921	1.509
11 05 81	1244	0.60	0.5	488.2	0.529	0.882	1.490
14 05 81	847	1.70	11.4	510.3	0.951	1.401	1.970
14 05 81	854	1.70	11.5	511.1	0.676	1.401	1.989
19 05 81	1450	4.60	30.2	509.9	0.911	1.735	2.352
19 05 81	1453	4.60	31.2	510.1	0.921	1.735	2.401
19 05 81	1455	4.60	31.8	510.1	0.921	1.754	2.401
19 05 81	1458	4.70	32.6	510.8	0.931	1.764	2.381
28 05 81	755	1.40	5.4	511.8	0.745	1.323	1.950
5 06 81	1544	14.00	101.4	511.8	1.529	2.509	3.401
5 06 81	1552	14.10	101.4	511.8	1.480	2.509	3.430
7 06 81	118	1.70	9.3	505.6	0.843	1.431	2.127
7 06 81	124	1.30	6.4	511.7	0.794	1.323	1.960

POSTIRRADIATION EXAMINATION OF ROD 8

Rod 8 of IFA-432 was removed from the assembly in January 1980 and was sent to Harwell for PIE and axial load compliance testing. The compliance tests were designed to quantify the mechanical behavior of cracked UO_2 fuel and were emphasized in these examinations. The results of the mechanical compliance tests along with profilometry and gamma scan data will be presented in a separate report. Ceramography and fission gas release data are presented in this section.

The fuel rod free volume and gas content were measured using standard Harwell gas sampling procedures. The gas composition was determined from mass spectrographic analyses of three individual gas samples. The results along with the calculated fission gas release fraction are presented below.[a] The fission gas release calculations are based on a fission energy of 190 MeV/fission and assume 0.30 fission gas atoms are produced for each fission.

Free Volume, cm^3	Internal, Pressure, MPa at 273K	Gas Composition, % He	N_2/CO	O_2	CO_2	A	Kr	Xe	Fission Gas Release, %
3.74	0.94 (138 psi)	13.3	2.4	0.6	0.05	0.02	11.7	72.3	9.0

The average burnup for the fuel in rod 8 was 1888 GJ/kgU (21.9 GWd/MTU); and the calculated fission gas release was 9%. This value agrees reasonably well with the estimated gas release for rod 1 at this burnup (6.5 $\pm$1%) that was derived from the in-pile internal pressure data. Both of these rods were of similar design and should have operated at similar temperatures. However, the end pellets in rod 1 were drilled to accommodate the TCs, which would reduce the centerline temperatures in rod 1 along 40% of its length and may explain the somewhat lower estimated gas release in rod 1.

After the compliance tests, a transverse section was cut from rod 8 for ceramography at an axial location corresponding to the location of the tip of the lower TC in rod 1. The sample was prepared by standard techniques and was examined before and after etching. An unirradiated archive fuel sample was also prepared and examined.

A photomicrograph of the polished cross section is shown in Figure 38. The fuel shows a typical crack pattern that is caused by differential thermal expansion of the fuel during reactor operation. Some additional fuel cracking may also have occurred during the compliance testing, but it is not readily apparent from the microstructure. The grain structure of the fuel was unaffected by the irradiation. The grain size in the irradiated pellet varied from

(a) The measured helium content in the irradiated fuel rod was 13% higher than estimated from the fabricated dimensions. This additional helium is thought to have been produced and released from the fuel during irradiation.

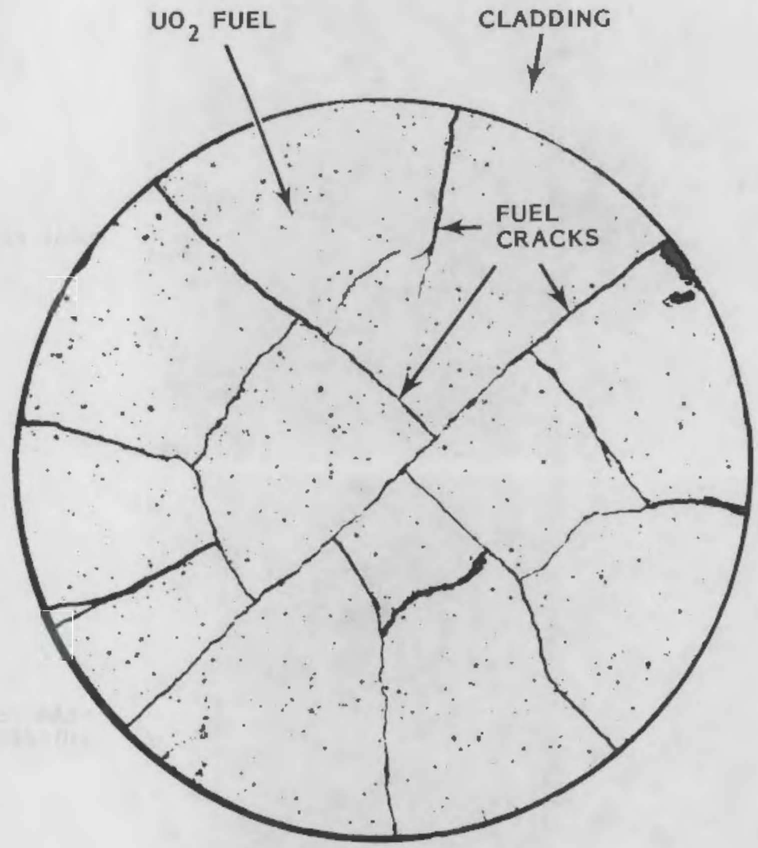

FIGURE 38. Polished Cross Section from Rod 8 of IFA-432

~20 μm near the pellet edge to ~70 μm at the pellet center; these values are nearly identical to those measured during the precharacterization of this fuel type (Hann et al. 1977). The fuel temperatures at this axial location were estimated to be below 1700K during irradiation, and little grain growth was expected at these temperatures.

Grain boundary porosity is the prominent feature of the irradiated microstructures as shown in Figure 39. Grain boundaries near the pellet center contain many fission gas bubbles, and some bubble interlinkage has occurred. The extent of grain boundary porosity decreased with increasing pellet radius because of the lower fuel operating temperatures. Near the midradius position, the bubbles at the grain boundaries were smaller than at the pellet center and only a few grain boundaries near the pellet edge exhibited this type of porosity.

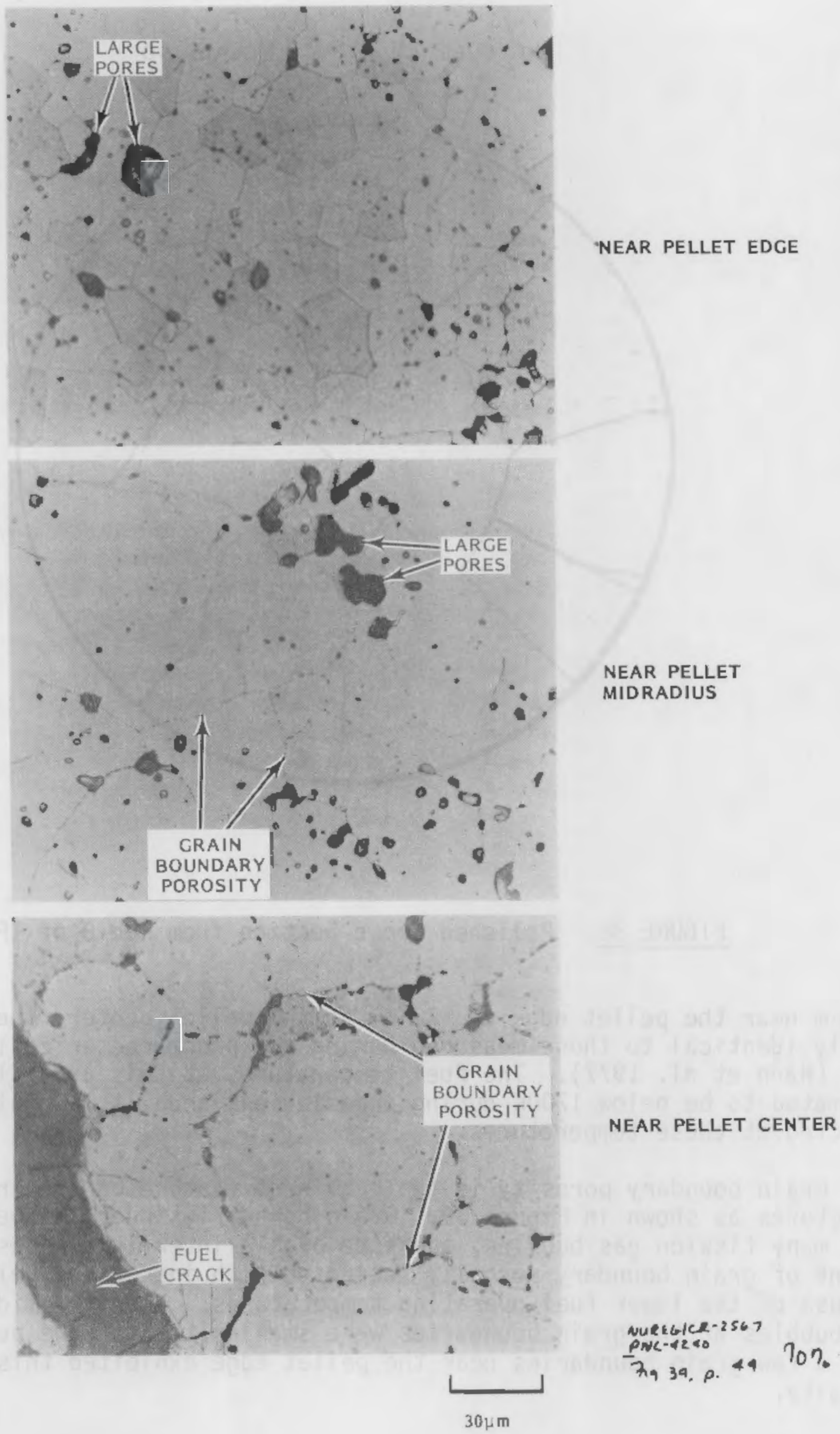

FIGURE 39. Typical Microstructures of Irradiated Fuel at Three Radial Locations from Rod 8 of IFA-432

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APPENDIX A

DATA PROCESSING

APPENDIX A

DATA PROCESSING

The data received from Halden on magnetic tape are processed as shown in Figure A.1. After the data tapes are received, they are translated from the Halden IBM/1800 language (EBCDIC) to the PDP11/70 language (ANSI); the translated version is then stored on tape. The tape is formatted so that all data from a particular time on a particular date are in one block; all data are simultaneously stored on a disk file.

Once the raw data are stored on disk, another program corrects the rod local heat ratings at the thermocouple (TC) locations for radial flux tilt across the assembly. Rod local and assembly powers are corrected for axial flux shape and heat losses to the moderator, and corrections for local mass distributions of fissile material are made for each rod.

While this is being done, other checks are made on the data. A total heat balance check is made for the assembly and rod average powers during application of the axial correction factor to account for the difference between the average and true mean of the axial flux distribution. The first attempt at this uses an axial profile that represents normal operating conditions. If the heat balance for this profile does not check, a second attempt is made with an axial flux shape that represents a disturbed flux profile. This occurs when a nearby control rod is partially inserted. After this step, another program corrects burnups and heat ratings for depletion of uranium-235.

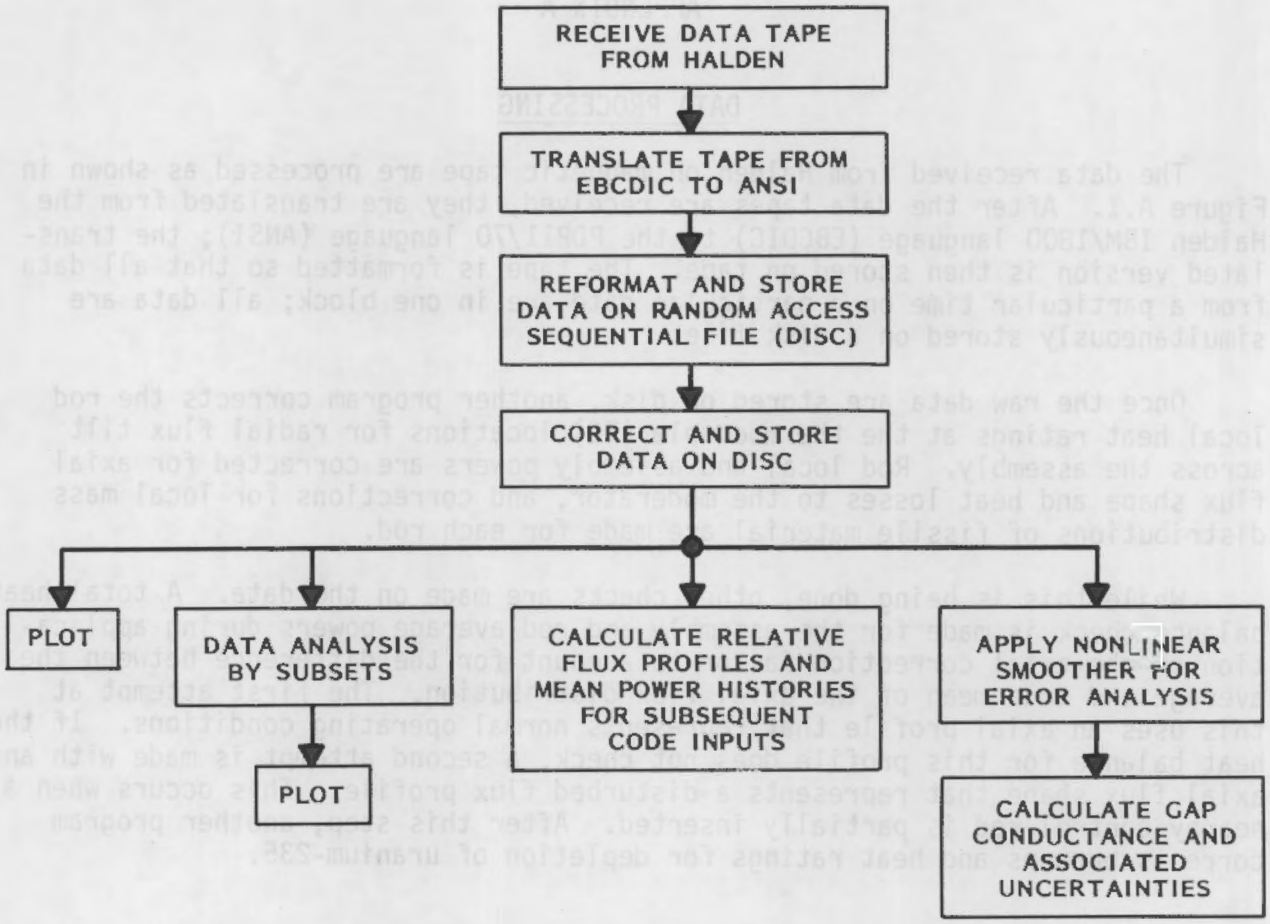

FIGURE A.1. Flow Diagram for Processing Halden Data

APPENDIX B

ASSEMBLY POWER CALIBRATION

APPENDIX B

ASSEMBLY POWER CALIBRATION

The data report for instrumented fuel assembly (IFA)-431 briefly explained the usual method for calibrating assemblies in the Halden reactor.[a] This procedure was not used for IFA-432 because the calibration flow valve (text Figure 2, p. 6) failed in the normal operating position, allowing only natural circulation. However, both assemblies were in the core at the startup of IFA-432, and IFA-432 was calibrated by comparisons of total assembly power and rod 3 (small gap) power. The uncertainty in assembly power for IFA-432 was estimated to be ±10% at a 3σ confidence level.

(a) Hann, C. R., et al. 1978. Data Report for the NRC/PNL Halden Assembly IFA-431. PNL-2494, Pacific Northwest Laboratory, Richland, Washington.

APPENDIX C

INSTRUMENT DESCRIPTIONS AND CALIBRATION

APPENDIX C

INSTRUMENT DESCRIPTIONS AND CALIBRATION

Instrumented fuel assembly (IFA)-432 was equipped with a comprehensive array of in-pile instrumentation to collect data (see text Figures 1 and 2, pp. 5 and 6). The most important of these instruments were:

- 6 vanadium beta emitter self-powered neutron detectors (SPNDs)
- 1 cobalt fast-response SPND
- 11 W-5% Re/W-26% Re sheathed fuel centerline thermocouples (TCs)
- 1 ultrasonic thermometer
- 6 linear variable differential transformer (LVDT) cladding elongation monitors
- 3 diaphragm-type rod internal pressure transducers.

Each of these instruments is briefly discussed below. The accuracy and uncertainty of their respective outputs is discussed more completely in Hann et al. (1977).

NEUTRON DETECTORS

IFA-432 was equipped with six vanadium self-powered beta current neutron detectors (Figure C.1) to monitor the power in the fuel assembly after the initial thermal-hydraulic calibration. Each detector was 100 mm (3.93 in.) long and was positioned so that the center of the detector and the TC junction were located on the same horizontal plane.

The neutron detectors used in IFA-432 were not calibrated. Their precisions were based on the results of the irradiation of 30 similar vanadium neutron detectors in the Studsvik R2-0 Reactor in Sweden. The 30 detectors were irradiated in a thermal neutron flux of 1.1×10^{14} n/m^2-s, and error limits for the outputs of the detectors were estimated to be $\pm 2.5\%$ at this flux. In addition, Halden has conducted long-term tests of similar neutron detectors in the Halden Boiling Water Reactor (HBWR) that have established that the detectors are reliable and accurate instruments with no measurable change in sensitivity at higher flux levels.

The sensitivities of the test assembly neutron detectors were calculated from the sensitivities of the calibrated detectors and the physical characteristics of the test assembly detectors supplied by the manufacturer. The gamma sensitivity was not measured and is considered to be negligible by Halden.

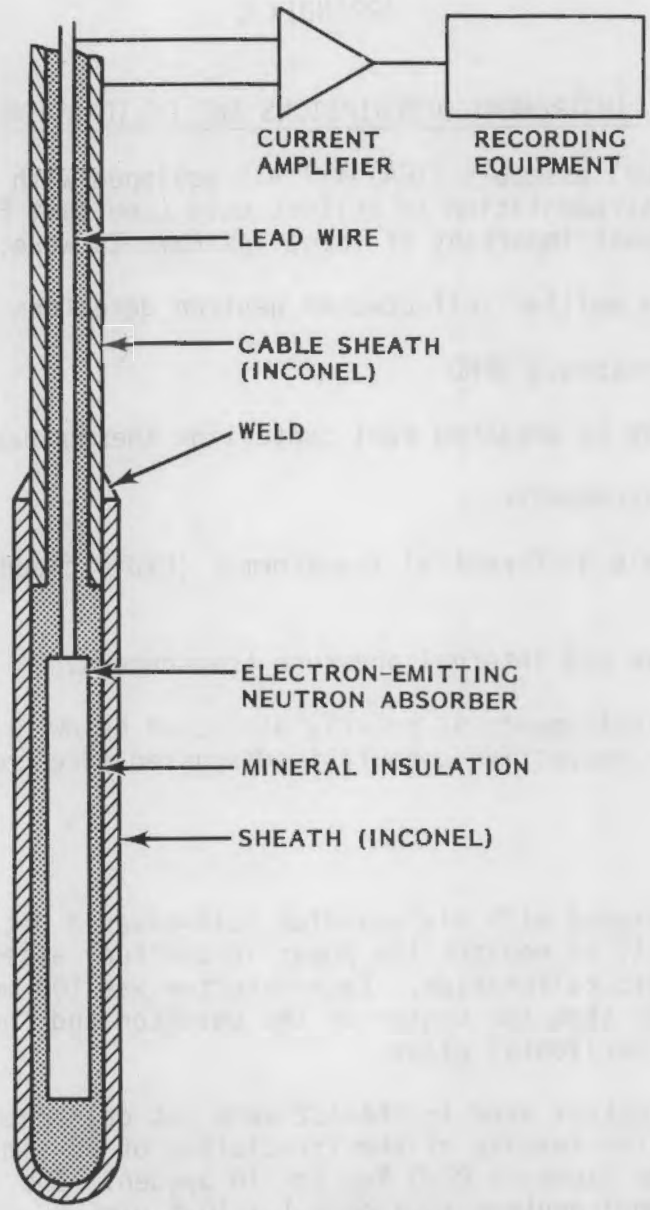

FIGURE C.1. Schematic of Self-Powered Beta Current Neutron Detector

The vanadium detectors had a calculated burnup rate of 0.013% per month at a neutron flux of 1×10^{17} n/m^2-s. Based on this rate, the neutron detector end-of-life (EOL) burnup for IFA-432 is 0.3%. Because of this low value, the neutron detector outputs were not corrected for burnup. However, it should be noted that during up and down power ramps a correction factor should be considered for the output values because of the slow response time of the vanadium detectors.[(a)] The cobalt detector, which is similar in appearance to the vanadium detector but 200 mm long, was placed in the center of the assembly to monitor average assembly power during transient tests (Lanning and Hann 1977).

FUEL THERMOCOUPLES

The 11 TCs in IFA-432, which measured central fuel temperatures, had grounded junctions with 1.575-mm (0.062-in.) outside diameter (OD) tungsten/22% rhenium sheaths and W-5% Re/W-26% Re seven-stranded TC wires with thorium oxide insulators (Figure C.2). The sensor in the top of rod 2 was an ultrasonic thermometer (Lynnworth et al. 1969) that failed immediately.

The TCs were fabricated and calibrated by the Idaho National Engineering Laboratory (INEL); the calibration curve for the tungsten-rhenium TCs is shown in Figure C.3. Since calibration of the TCs over the range of use produces a

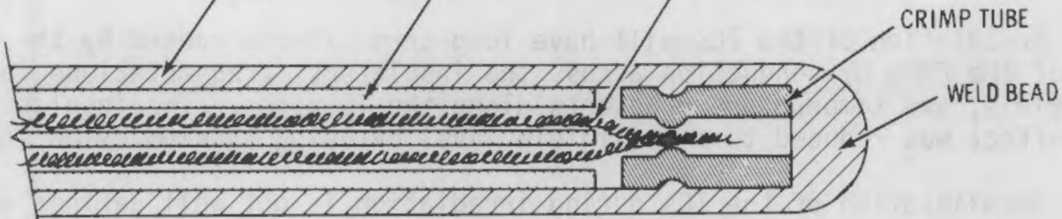

FIGURE C.2. Schematic of W-5% Re/W-26% Re Thermocouples with Grounded Junction

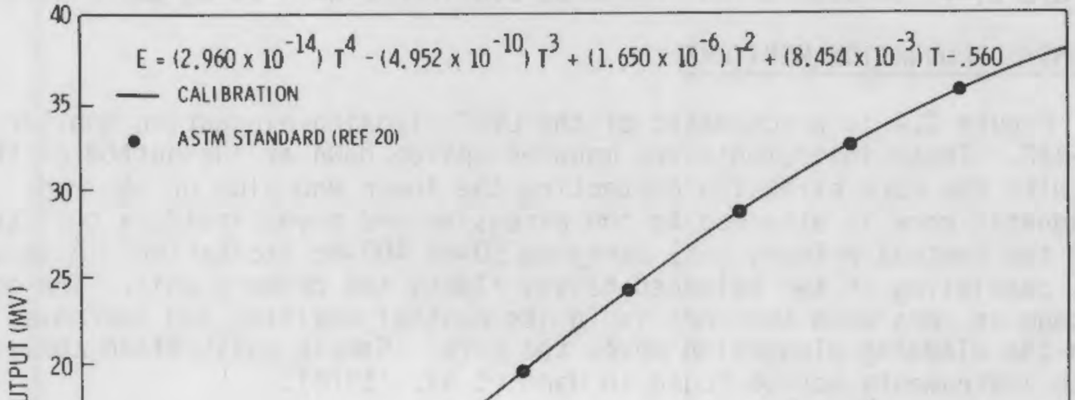

FIGURE C.3. Calibration Curve for W-5% Re/W-26% Re Thermocouples brittle assembly that is fragile and subject to breakage only one TC, which was not used in the in-reactor test, was calibrated. This tungsten-rhenium TC was calibrated against a reference TC of bare W-5% Re/W 26% Re and an optical pyrometer (as a second reference). The reference TC and the optical pyrometer agreed within 295K (40°F) up to 2477K (4000°F); but as the temperature approached 2755K (4500°F), the difference between the two widened. The optical pyrometer was thought to be more accurate since the 2755K temperature is above that given in most calibration tables for W/Re TCs. The calibrated TC had the following error limits:

- ambient to 811K (1000°F) = ±5.5K (10°F)
- 811 to 2477K (1000 to 4000°F) = ±1% of reading
- 2477 to 2755K (4000 to 4500°F) = ±2% of reading.

Irradiation of the TCs will have long-term effects caused by the shunting of the EMFs by conduction across the insulators, transmutations in the TC materials, and temperature gradients along the TC wires. The insulator shunting effect was reduced to a negligible level by using thorium oxide insulators.

Decalibration of the TCs during irradiation is not well defined at the present time. Experimental data from Halden and analysis of the IFA-432 transient data suggest possible decalibrations from nearly 0 up to 1% per 1000-MWd/MTU burnup. Consequently, the measured fuel temperatures could be 25% lower than the actual temperatures at the end of the current reporting period; and, therefore, TC decalibration should be considered when using these data.

CLADDING ELONGATION MONITORS

Figure C.4 is a schematic of the LVDT cladding elongation monitor used in IFA-432. These instruments are mounted upside down at the bottom of the assembly with the core extension contacting the lower end plug of the rod. The ferromagnetic core is attached to the extension and moves inside a coil system with the central primary coil carrying 50-mA 400-Hz excitation. A secondary coil consisting of two balanced halves flanks the primary coil. The output voltage is zero when the core is in its central position and increases linearly when the cladding elongation moves the core. Sample calibration curves for these instruments may be found in Hann et al. (1978).

FISSION GAS PRESSURE TRANSDUCERS

Figure C.5 shows a schematic of the diaphragm-type pressure transducer used to measure the internal rod pressures due to fission gas release during irradiation. It is essentially an on-off measurement. The thin platinum alloy membrane is exposed to the rod internal gases on one side, and the other other side is connected to an external pressure manifold. When the external pressure equals the internal pressure, the deflection of the membrane causes it to make an electrical contact. The step increase in voltage signals a null pressure balance. Over a range of 10 MPa (100 kg/cm^2), the sensitivity of the instrument is 0.01 MPa and the accuracy and repeatability are ±0.1 MPa and ±0.04 MPa, respectively.

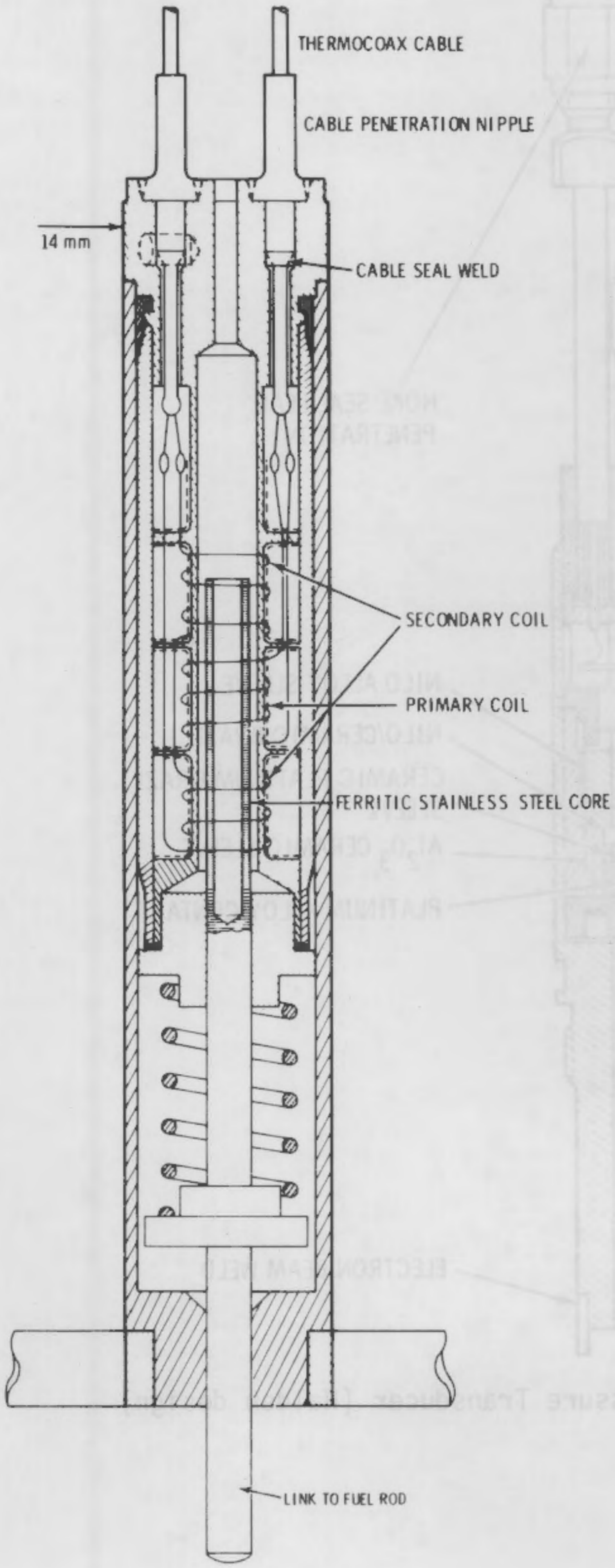

FIGURE C.4. Cladding Elongation Monitor (Halden design)

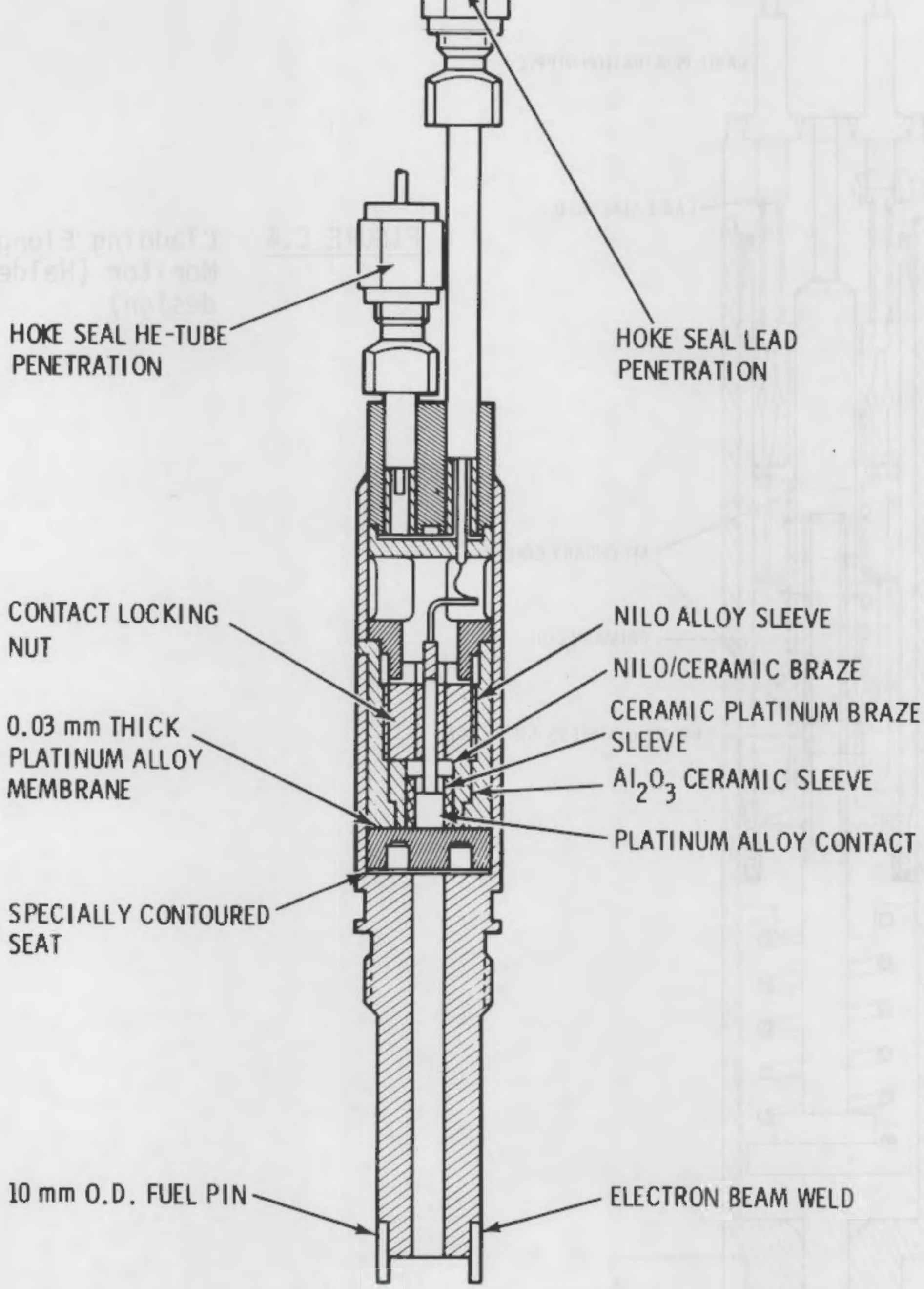

FIGURE C.5. Fission Gas Pressure Transducer (Halden design)

Pressure transducer calibration is done out of the reactor and consists of checking the deflection sensitivity of the membrane, which does not change appreciably with pressure level. The effects of irradiation or temperature on the membrane are not known but are assumed to be minimal by Halden. Halden has made no recommendation for temperature compensation for this instrument.

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APPENDIX D

CORRECTED CENTERLINE TEMPERATURE HISTORIES

APPENDIX D

CORRECTED CENTERLINE TEMPERATURE HISTORIES

Fuel centerline temperature histories for rods 1, 2, 3, and 5 were corrected for thermal neutron irradiation-induced decalibration of the thermocouples (TCs) (see Figures D.1 through D.7). The TC decalibration rate was taken as 1.75% per 10^{24} n/m^2,[a] and the corrected temperatures were obtained from the following equations:

$$corr = \frac{0.0175\ F}{10^{24}} \tag{1}$$

$$T_{corr} = (T_{orig} - 20)(1 + corr) + 20 \tag{2}$$

where corr = correction factor
F = thermal neutron fluence, n/m^2
T_{corr} = corrected temperature, K
T_{orig} = measured temperature, K.

The temperature histories shown in Figures E.1 through E.7 are the current estimates of the true temperature histories during the period from July 1980 to June 1981.

(a) Crouthamel, C. E., and M. D. Freshley. October 1980. Fuel Performance Improvement Program: Semiannual Progress Report, April 1980-September 1980. DOE/ET/34215-19.

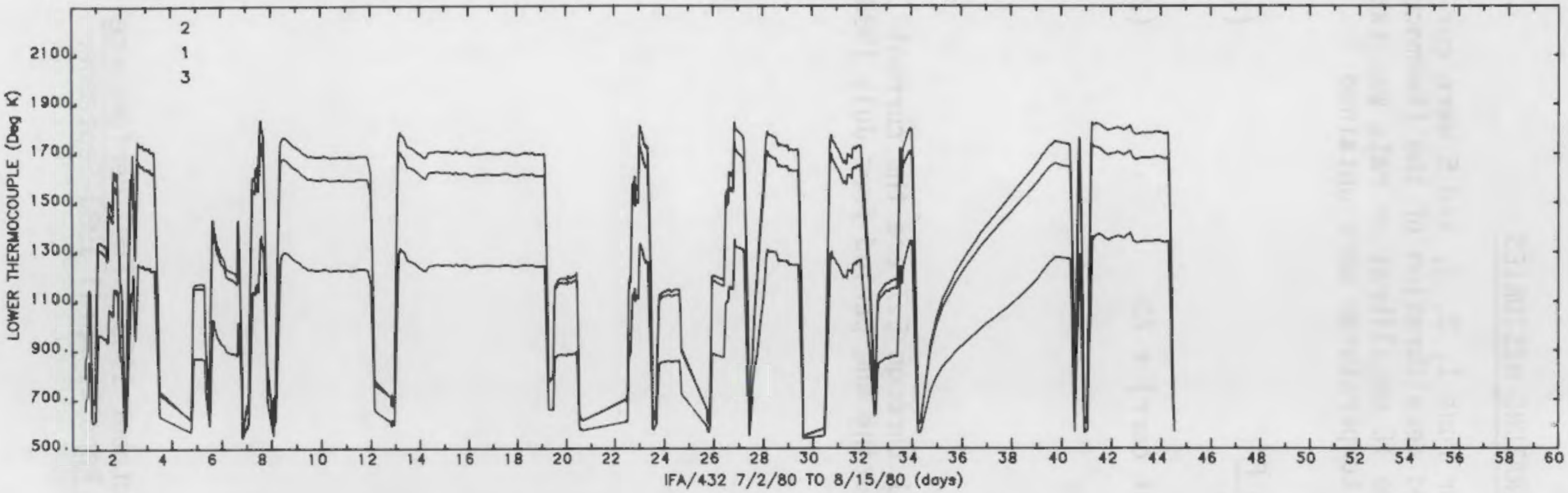

FIGURE D.1. Corrected Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from July 2, 1980, to August 15, 1980

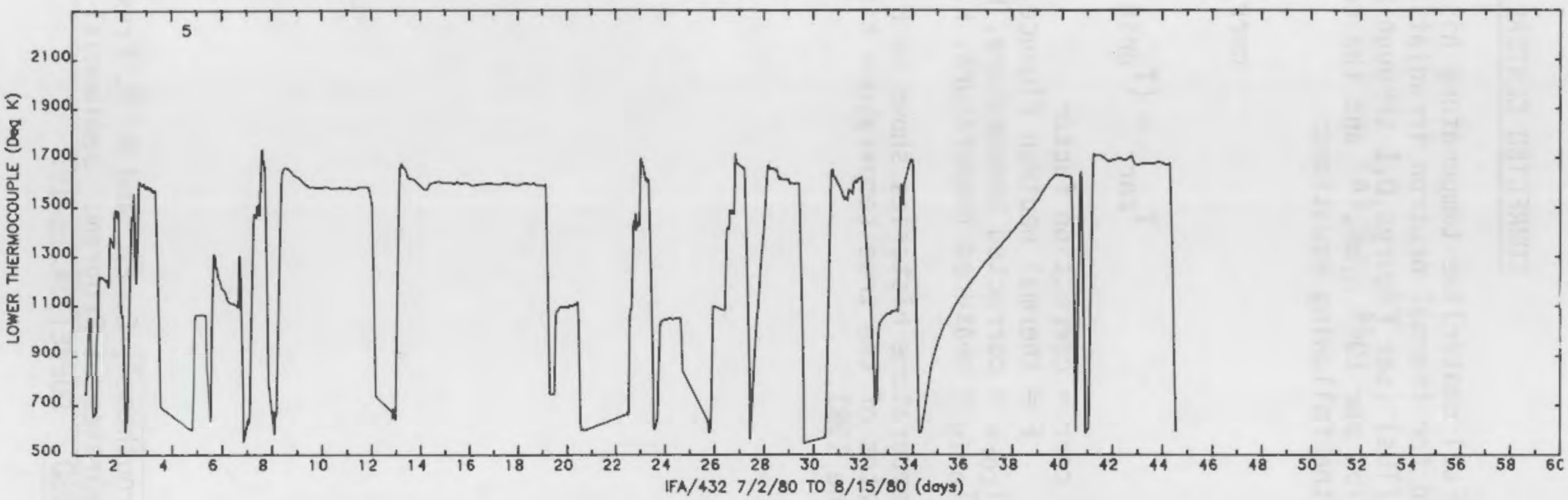

FIGURE D.2. Corrected Lower Thermocouple Readings for Rod 5 of IFA-432 from July 2, 1980, to August 15, 1980

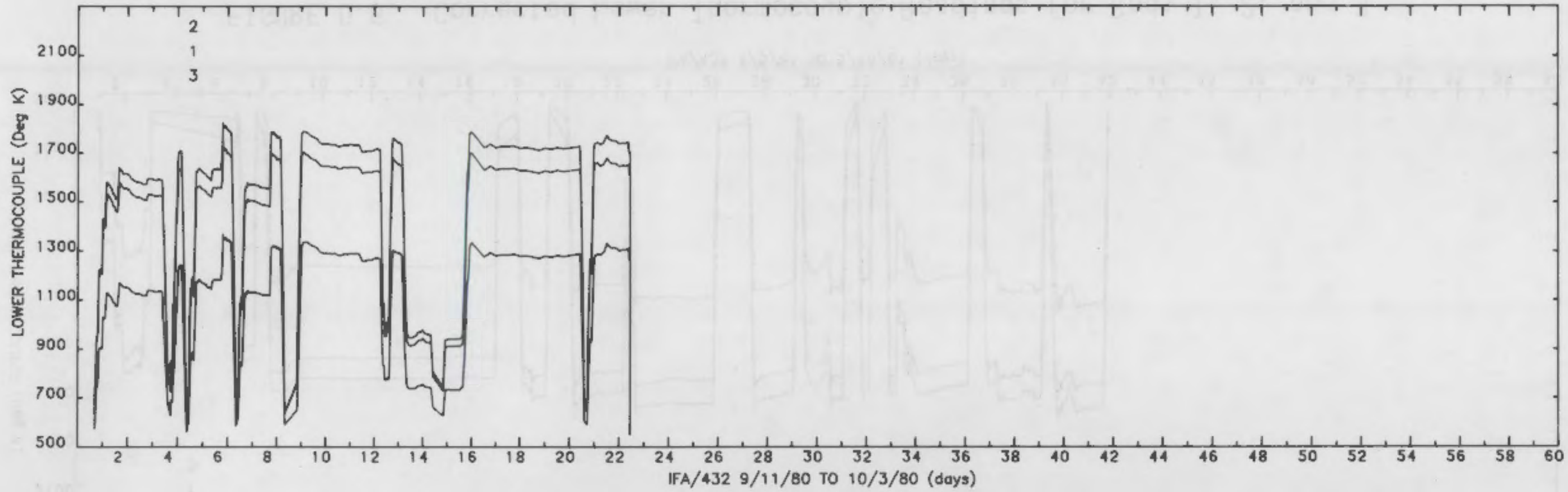

FIGURE D.3. Corrected Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from September 11, 1980, to October 3, 1980

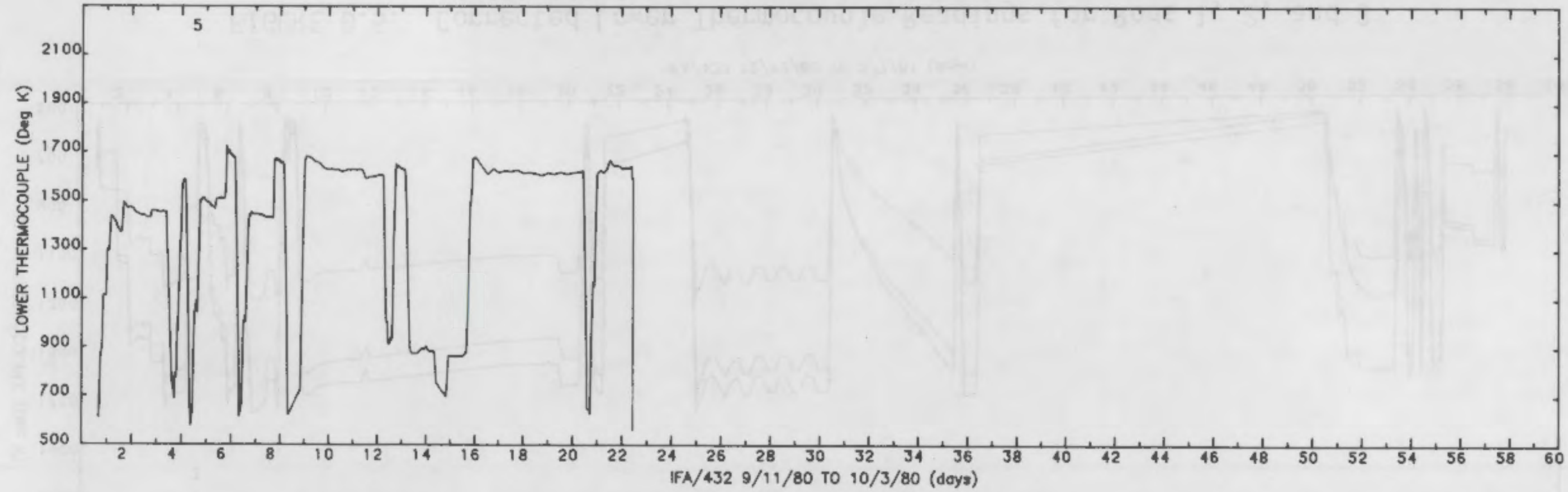

FIGURE D.4. Corrected Lower Thermocouple Readings for Rod 5 of IFA-432 from September 11, 1980, to October 3, 1980

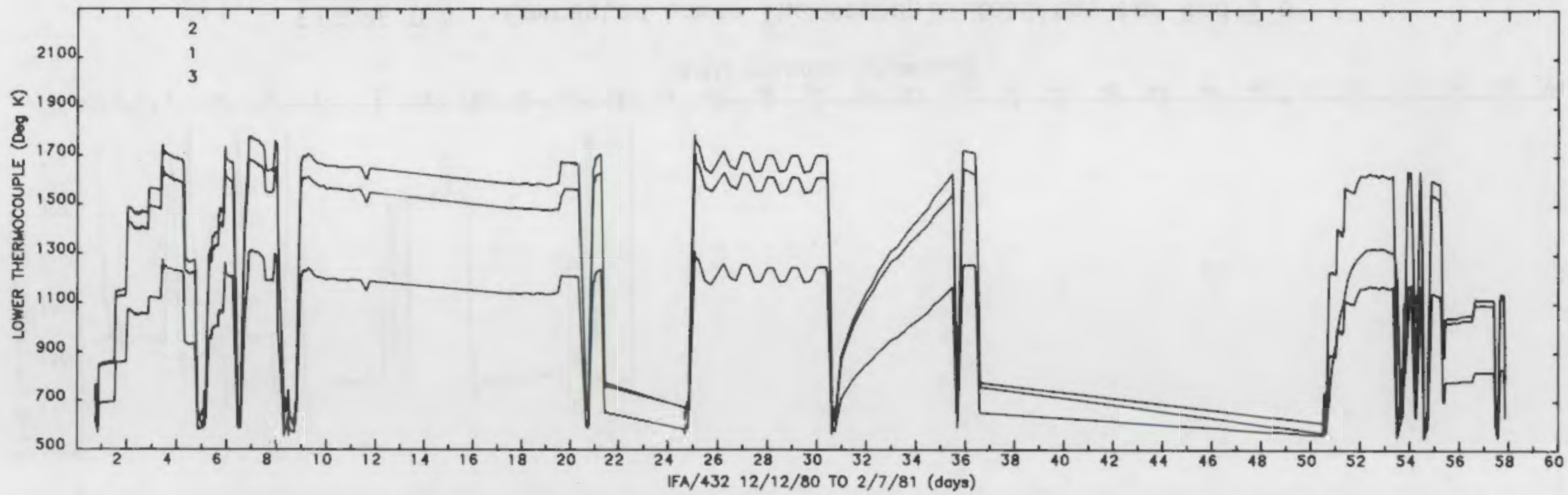

FIGURE D.5. Corrected Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from December 12, 1980, to February 7, 1981

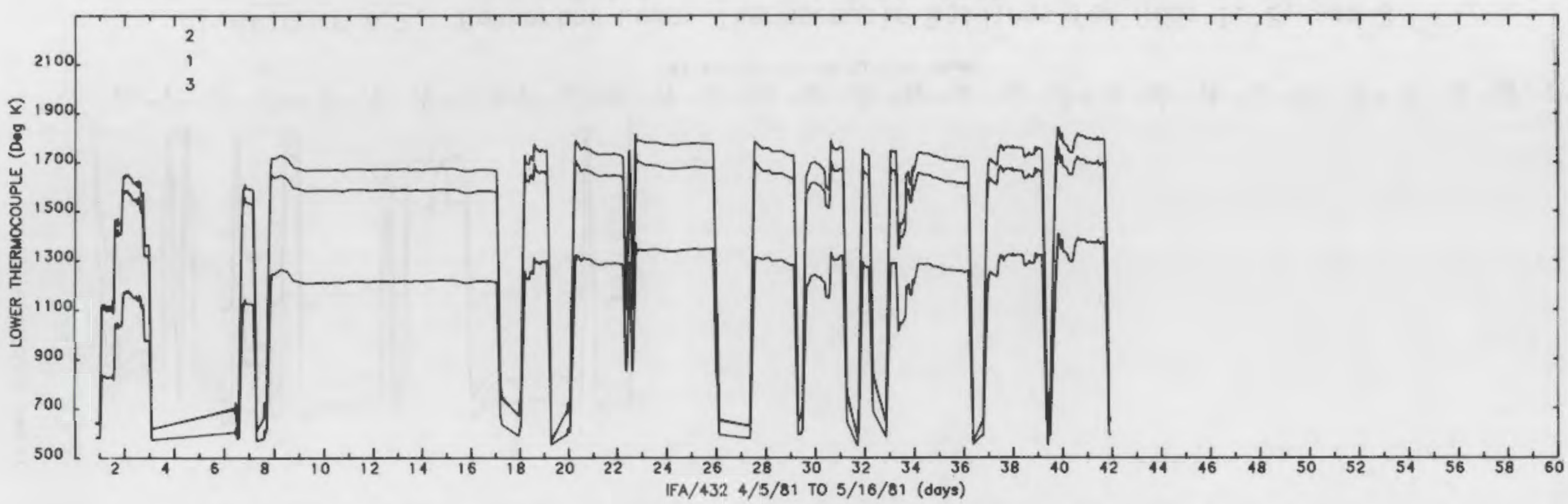

FIGURE D.6. Corrected Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from April 5, 1981, to May 16, 1981

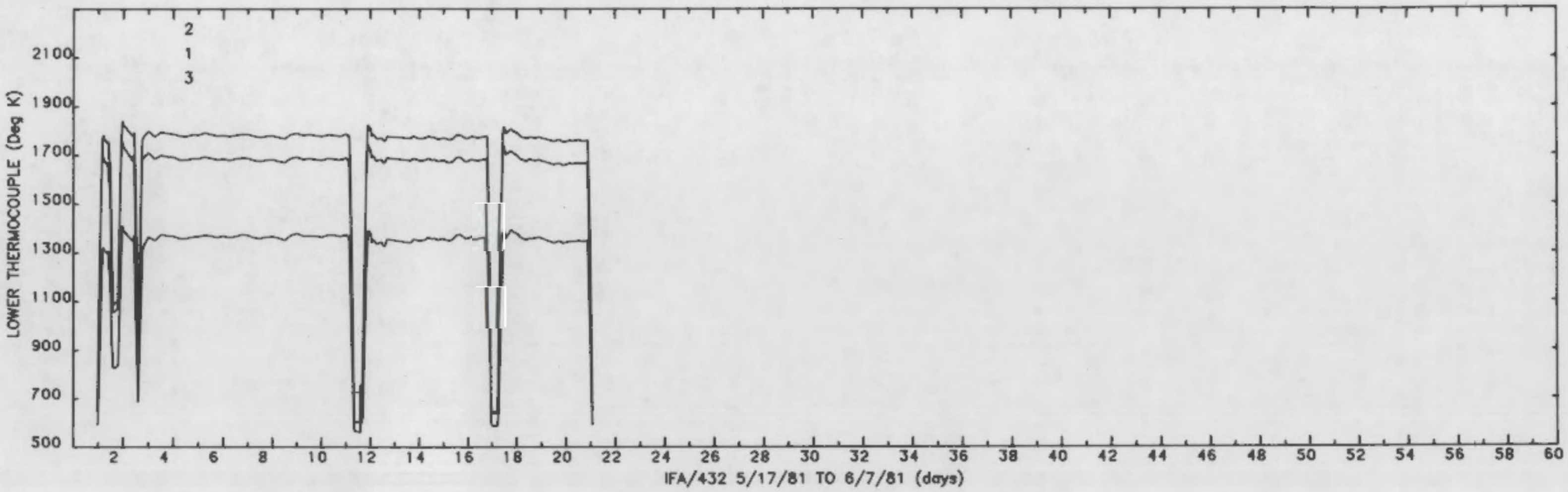

FIGURE D.7. Corrected Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from May 17, 1981, to June 7, 1981

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NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-2567
PNL-4240

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Final Data Report for the Instrumented Fuel Assembly (IFA)-432

5. DATE REPORT COMPLETED

MONTH	YEAR
May	1982

DATE REPORT ISSUED

MONTH	YEAR
June	1982

6. *(Leave blank)*

7. AUTHOR(S)
E.R. Bradley, M.E. Cunningham, D.D. Lanning

8. *(Leave blank)*

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Pacific Northwest Laboratory
Richland, WA 99352

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO.
B2043

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Accident Evaluation
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

13. TYPE OF REPORT
Topical Report

PERIOD COVERED *(Inclusive dates)*
June 1980 - June 1981

14. *(Leave blank)*

15. SUPPLEMENTARY NOTES

16. ABSTRACT *(200 words or less)*

This report presents the in-reactor data collected during the irradiation of the six-rod instrumented fuel assembly (IFA)-432 in the Halden (Norway) Boiling Water Reactor (HBWR) from June 1980 through June 1981. This assembly (designed by PNL) was one of a series of NRC sponsored tests to obtain data for the development and assessment of steady-state fuel performance computer codes. IFA-432 operated from December 1975 until June 1981, when it was removed from the reactor. Burnup levels in excess of 30,000 MWD/MTM were achieved. Data collected prior to June 1980 were reported in NUREG/CR-0560 and NUREG/CR-1950.

Fuel centerline temperatures, cladding elongations, internal fuel rod pressures and local powers were monitored during the irradiation. Detailed analysis of the data reported is not made.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

Irradiated
BWR fuel rods
Linear heat rate
Thermocouple measurements

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

20. SECURITY CLASS *(This page)*
Unclassified

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (11-81)

NRC FORM 335

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)

NUREG/CR-2567
PNL-4280

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)

Final Data Report for the Instrumented Fuel Assembly (IFA)-432

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)

E.R. Bradley, M.E. Cunningham, D.D. Lanning

5. DATE REPORT COMPLETED

MONTH	YEAR
May	1982

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Pacific Northwest Laboratory
Richland, WA 99352

DATE REPORT ISSUED

MONTH	YEAR
June	1982

6. (Leave blank)

8. (Leave blank)

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Division of Accident Evaluation
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Unlimited

19. SECURITY CLASS (This report)
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS (This page)
Unclassified

22. PRICE
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