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Data Report for the NRC/PNL Halden Assembly IFA-432: April 1978-May 1980

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

Pacific Northwest Laboratory
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ABSTRACT

This report presents the in-reactor data collected from the U.S. Nuclear Regulatory Commission (NRC)/Pacific Northwest Laboratory (PNL) Halden test assembly IFA-432 for the period from April 1978 through May 1980. The irradiation test is part of an experimental program entitled "Experimental Support and Development of Single-Rod Fuel Codes" sponsored by the Fuel Behavior Research Branch of the NRC. The purpose of this program is to reduce the uncertainties of predicting the thermal and mechanical behavior of an operating nuclear fuel rod.

Fuel centerline temperatures, cladding elongation, internal fuel rod pressures, and local powers at the thermocouple (TC) positions are shown as a function of time. 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 also presented.

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC)/Pacific Northwest Laboratory (PNL) Halden test assembly IFA-432 has operated since December 1975 and has reached peak burnups in excess of 2560 GJ/kgU (29,600 MWd/MTM) as of May 1980. Data are currently being obtained from six neutron detectors, four fuel thermocouples (TCs), three cladding extensometers, and two pressure transducers. These data are providing valuable information regarding fuel performance at high burnups. The assembly will be removed from the reactor in mid-1981 with projected peak burnups in excess of 3000 GJ/kgU (35,000 MWd/MTM).

This report presents in-reactor data collected from IFA-432 for the period from April 1978 through May 1980. Data collected prior to April 1978 were presented in a previous report (Hann et al. 1978b). Fuel temperatures, power levels, and elongation data are presented in the form of plots of the variables versus time while internal pressure data and calculated burnups are tabulated.

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 (ECC) 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 many 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) have been designed by the Pacific Northwest Laboratory (PNL)[a] and are being irradiated in the boiling water reactor (BWR) at Halden, Norway. The first two tests in the series are IFA-431 and IFA-432, which are identical 6-rod assemblies containing the same variations of gap size and fuel type but operating at different power levels. IFA-513 is the third assembly in the series and contains six identical rods except for fill gas composition and pressure. The fourth assembly, IFA-527, uses xenon for the 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 (20,000 MWd/MTM) in late 1978. However, since most of the instruments in IFA-432 were still functioning properly at that time, it was left in the Halden 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

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

(Bradley et al. 1979a; Bradley et al. 1979b) and thermal and mechanical fuel rod performance (Lanning, Barnes, and Williford 1979; Lanning, Barnes, and Sheffler 1980; Williford and Hann 1977; Cunningham, Williford, and Hann 1979; Hann and Marshall 1977; and Williford et al. 1980) and for estimating error propagation in stored energy calculations (Cunningham et al. 1978). 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 through January 1978 were reported previously by Hann et al. (1978b). This report presents the experimental data collected from April 1978 through May 1980.[a]

(a) The reactor was shut down from January 1978 to April 1978.

TEST DESCRIPTION

Experimental verification of computer codes provides a means to quantify uncertainties in simulating the conditions for an operating nuclear fuel rod. 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 in order 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

(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 IFA-432

IFA-432 [Peak Power - 492 W/cm (15 kW/ft)]

Rod No.	Diameter Pellet mm	Diameter Pellet in.	Cold Diametral Gap[a] mm	Cold Diametral Gap[a] in.	Fill Gas	Fuel Density, % TD	Fuel Type[b]	Instrumentation: Temperature Upper	Instrumentation: Temperature Lower	Instrumentation: Pressure	Instrumentation: Cladding Length
1	10.681	0.4205	0.229	0.009	He	95	Stable	TC[c]	TC	PT[d]	ES[e]
2	10.528	0.4145	0.381	0.015	He	95	Stable	UT[f]	TC	--	ES
3	10.833	0.4265	0.076	0.003	He	95	Stable	TC	TC	--	ES
4	10.681	0.4205	0.229	0.009	Xe	95	Stable	TC	TC	--	ES
5	10.681	0.4205	0.229	0.009	He	92	Stable	TC	TC	PT	ES
6	10.681	0.4205	0.229	0.009	He	92	Unstable	TC	TC	PT	ES
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) Cladding for all rods has an OD of 12.789 mm (0.5035 in.) and an ID of 10.909 mm (0.4295 in.). Diametral gap is cladding ID minus pellet diameter.
(b) With respect to in-reactor densification.
(c) TC = Thermocouple
(d) PT = Pressure Transducer
(e) ES = Elongation Sensor
(f) UT = Ultrasonic Thermometer

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 Halden BWR (Figure 1) was also considered during the design. The top of the rods was placed at the peak, which forced the bottom of the rods to operate at 70-80% 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 staff were able to develop a workable design. TCs in both ends allow modelers to check the ability of various codes to extrapolate over a short power range within the same rod. If a code cannot perform these calculations adequately, calculations of the temperature distribution over a $\sim$4-m fuel length are also suspect.

Reference points with 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 EG&G-Idaho National Engineering Laboratory (INEL) for use in their Halden tests. Both programs (PNL and INEL) also 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 (Figure 2): 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

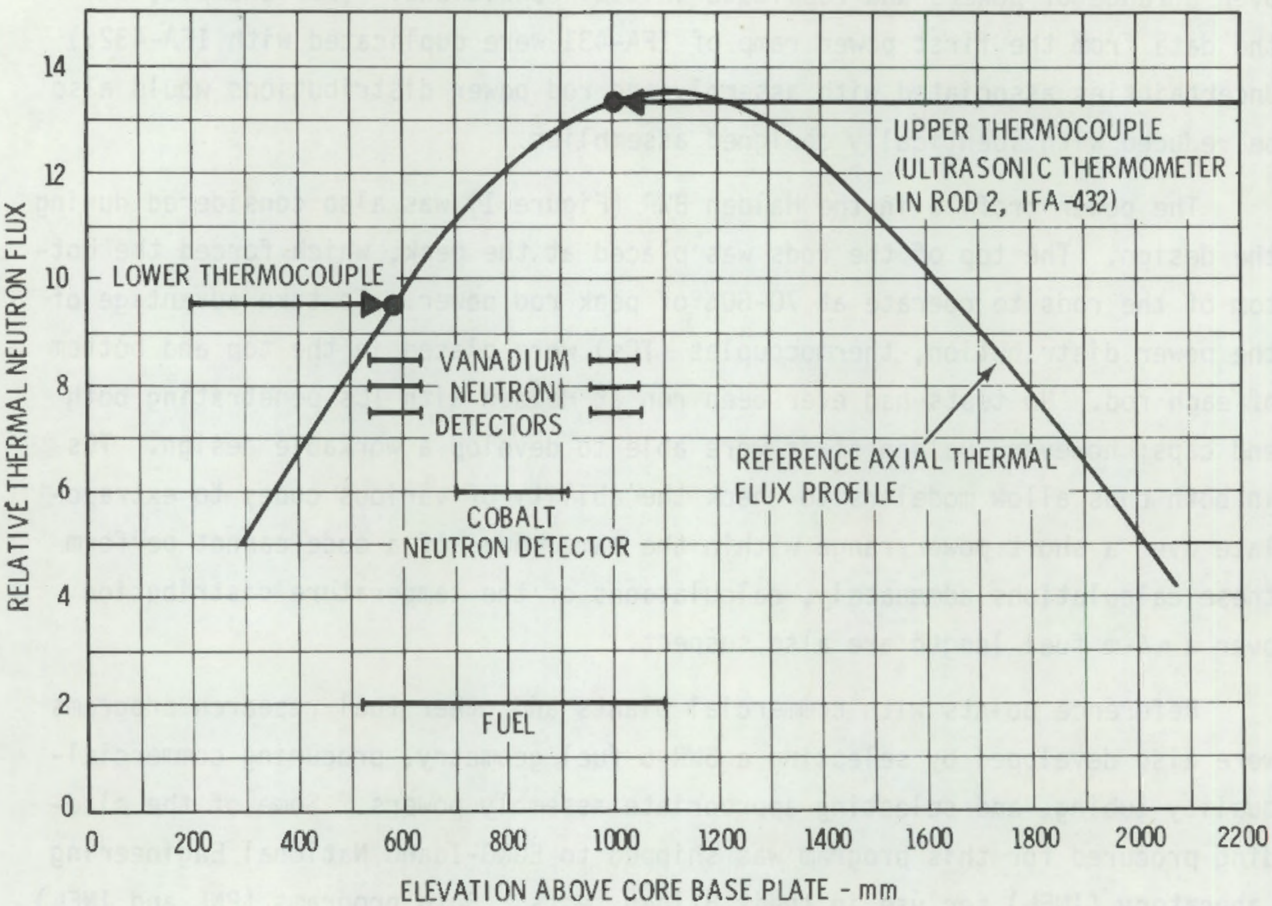


FIGURE 1. Arrangement of Temperature Sensors, Neutron Detectors, and Fuel Relative to Reference Axial Thermal Flux Profile

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. Each rod has a cladding elongation sensor; rods 1, 5, and 6 also have null balance fission gas pressure transducers (PXDs).

Table 2 illustrates the amount of cross-correlation that is possible. 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.

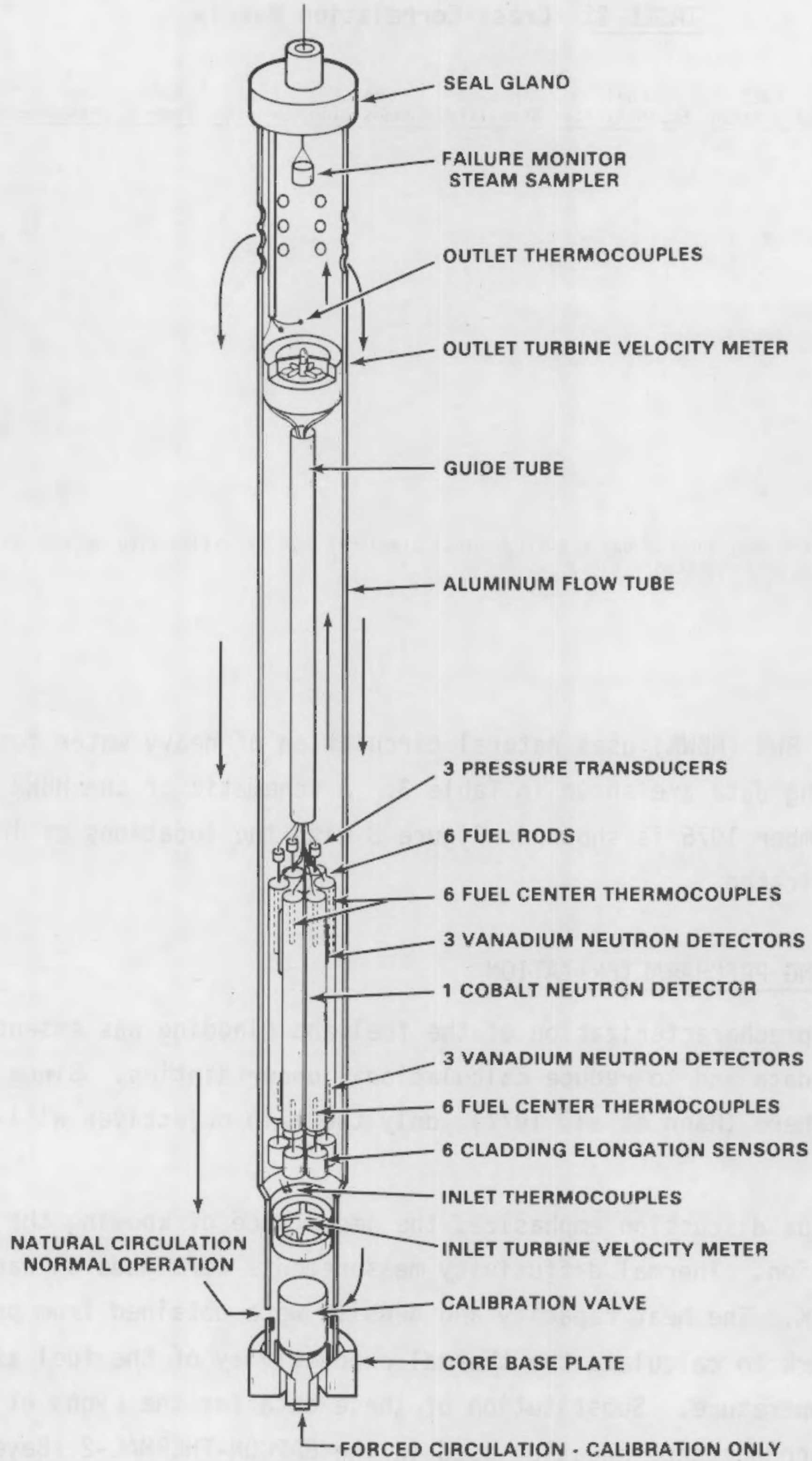


FIGURE 2. Schematic of Instrumented Fuel Assembly (IFA)-432

TABLE 2. Cross-Correlation Matrix

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

(a) (9-He-95-S) indicates that the rod has a 9-mil nominal diametral gap, is filled with helium, has a 95% theoretical density, and has stable fuel.

TEST FACILITY

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

FUEL AND CLADDING PRECHARACTERIZATION

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

The previous discussion emphasized the importance of knowing the correct power distribution. 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. Substitution of these data for the Lyons et al. (1964) thermal conductance equation used in the GAPCON-THERMAL-2 (Beyer et al. 1975) pretest predictions improved the power calibration calculation using

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

rod 3. However, after the first rise to power that produces fuel cracking, the Lyons formulation for the thermal conductance is believed to be more applicable.

Establishing the initial dimensions and void volumes within the pins was also an essential part of assessing all thermal calculations; 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 (see Appendix A). 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 densities. These data were used in two ways: in the correction to rod powers caused by differences in mass distribution and in the verification of U.S. Nuclear Regulatory Commission (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

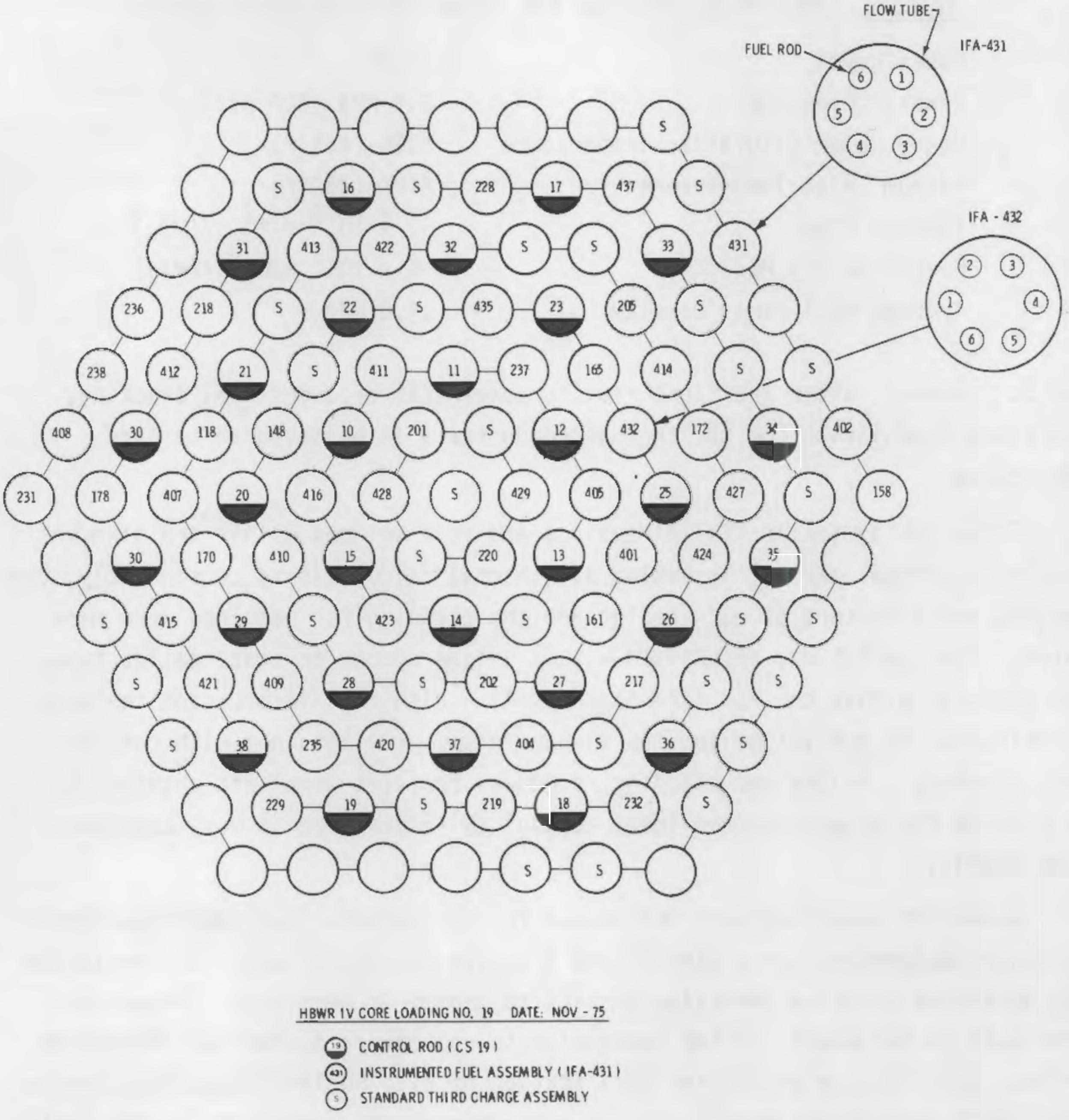


FIGURE 3. IFA-431 and IFA-432 Arrangements in the Flow Channel

used in these experiments were measured prior to irradiation to assure that the desired response to irradiation would be achieved. Both fuel densities and pore-size distribution will be measured during postirradiation examination (PIE) for rods 1, 5, and 6 at Harwell, UK. Archive pellets from each fuel type were retained to provide a means of reducing variances associated with potential differences in examination techniques used in the pre- and post-test measurements.

DATA PRESENTATION

In-reactor data collected from IFA-432 by the Halden IBM/1800 on-line computer data acquisition system for the period from April 1978 through May 1980 are presented in this section. Linear heat generation rates, fuel temperatures, and cladding elongation data are plotted as a function of time. In each plot, the rod number for each curve appears in the upper left-hand corner. The relative position of the rod number corresponds to the relative position of the curve in each figure. Rod 8 is a noninstrumented rod that replaced rod 4 following its removal at the end of February 1976. Rod 8 was replaced by rod 9 in February 1980.

Internal pressure data were taken manually and are presented in tabular form along with the moderator temperature and the reactor and assembly power levels. All of the pressure data taken since the initial startup (December 1975) are presented. The calculated burnup of the upper and lower TC locations are also given on a monthly basis.

POWER HISTORIES

Power histories for the upper and lower TC locations for all six rods are presented in Figures 4 through 39. 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 B).

Corrections were also made for the burnup-dependent depletion of ^{235}U. The correction that was used (-0.66% per 1000 MWd/MTM) was taken from depletion calculations performed at Halden.

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.

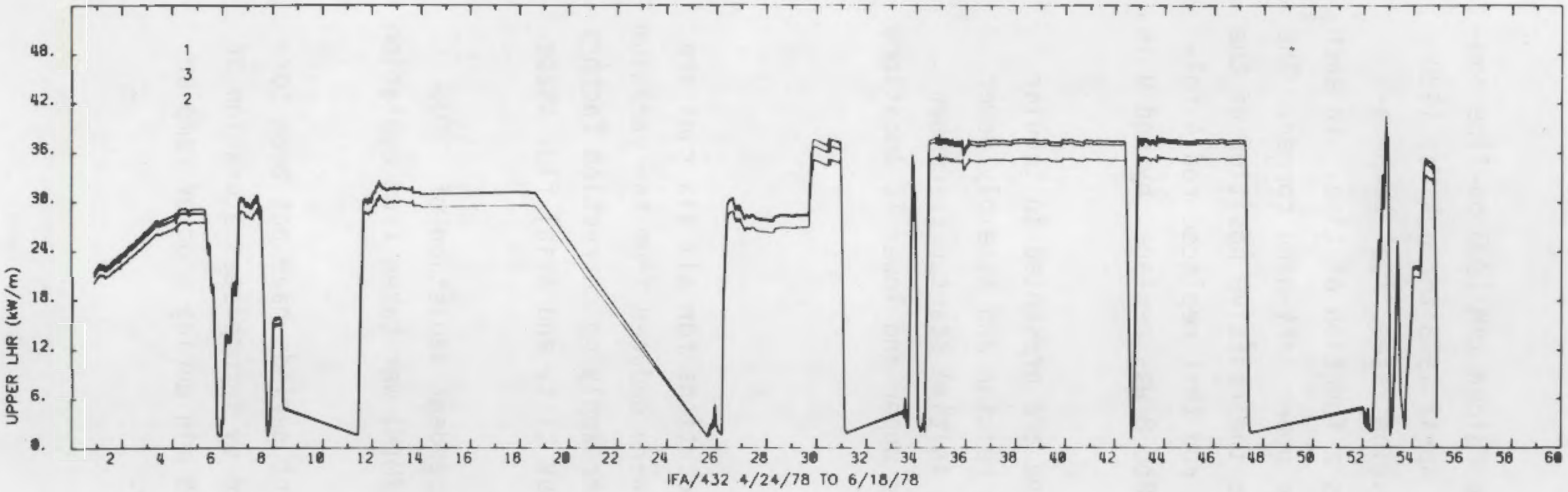


FIGURE 4. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from April 24, 1978, to June 18, 1978

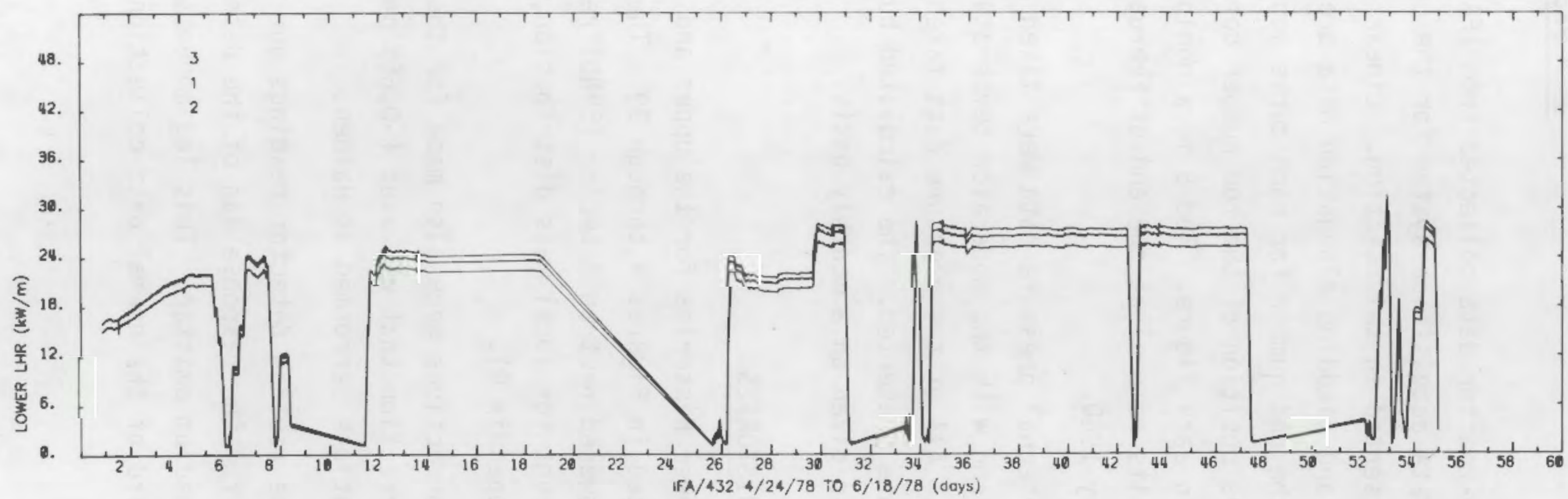


FIGURE 5. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2 and 3 of IFA-432 from April 24, 1978, to June 18, 1978

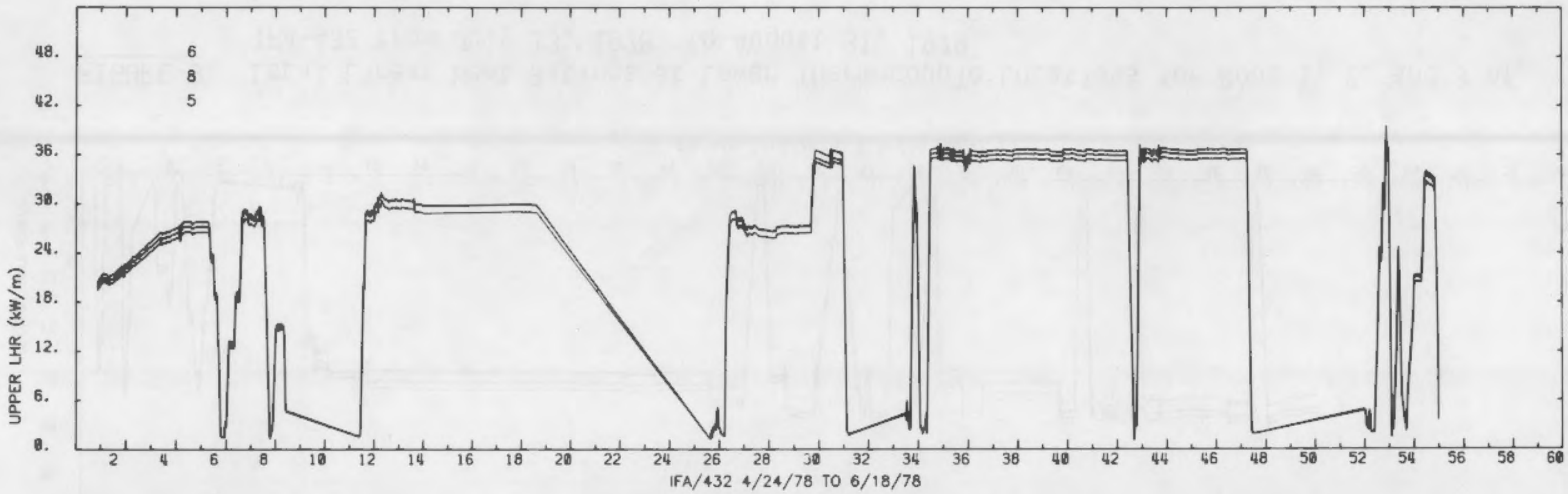


FIGURE 6. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from April 24, 1978, to June 18, 1978

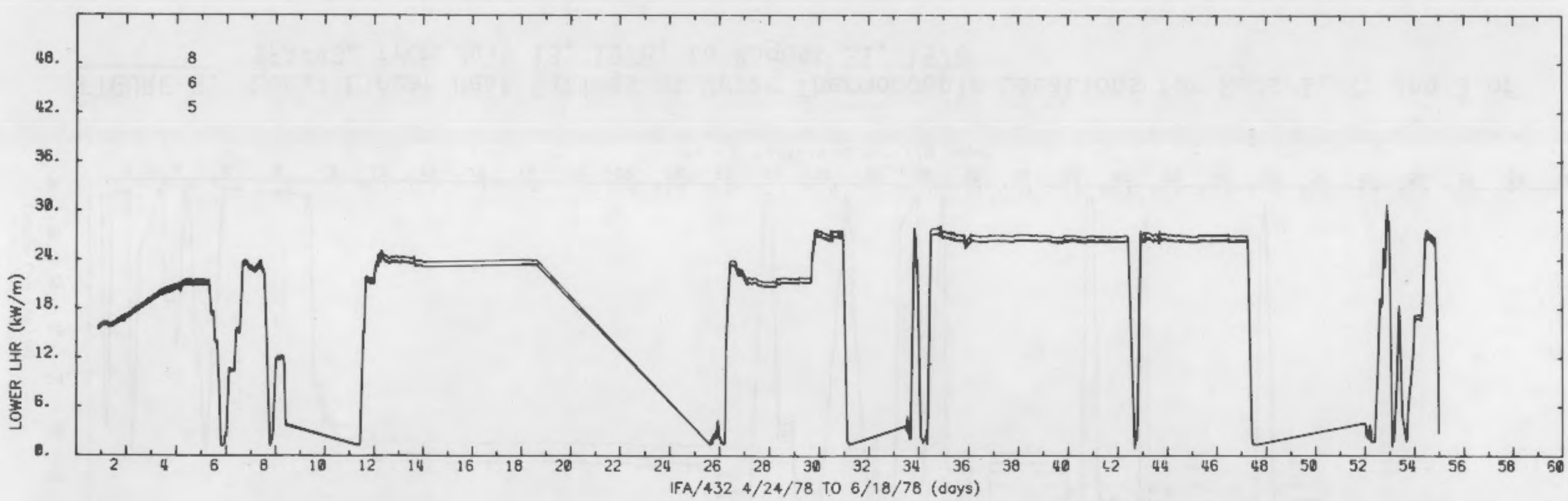


FIGURE 7. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from April 24, 1978, to June 18, 1978

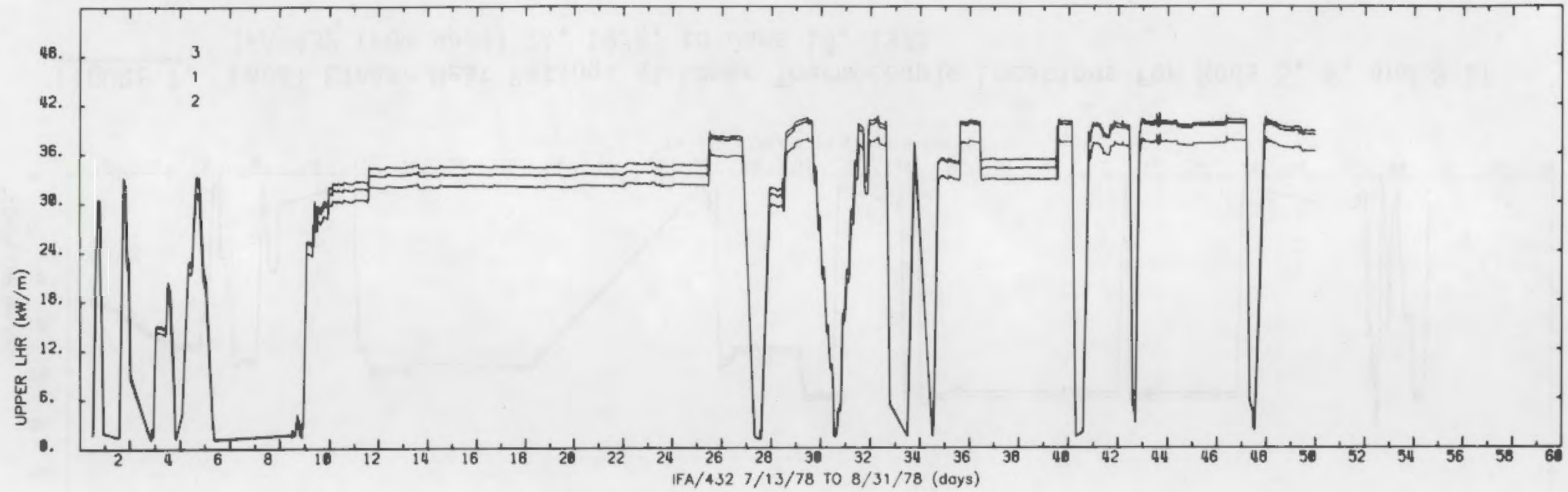


FIGURE 8. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from July 13, 1978, to August 31, 1978

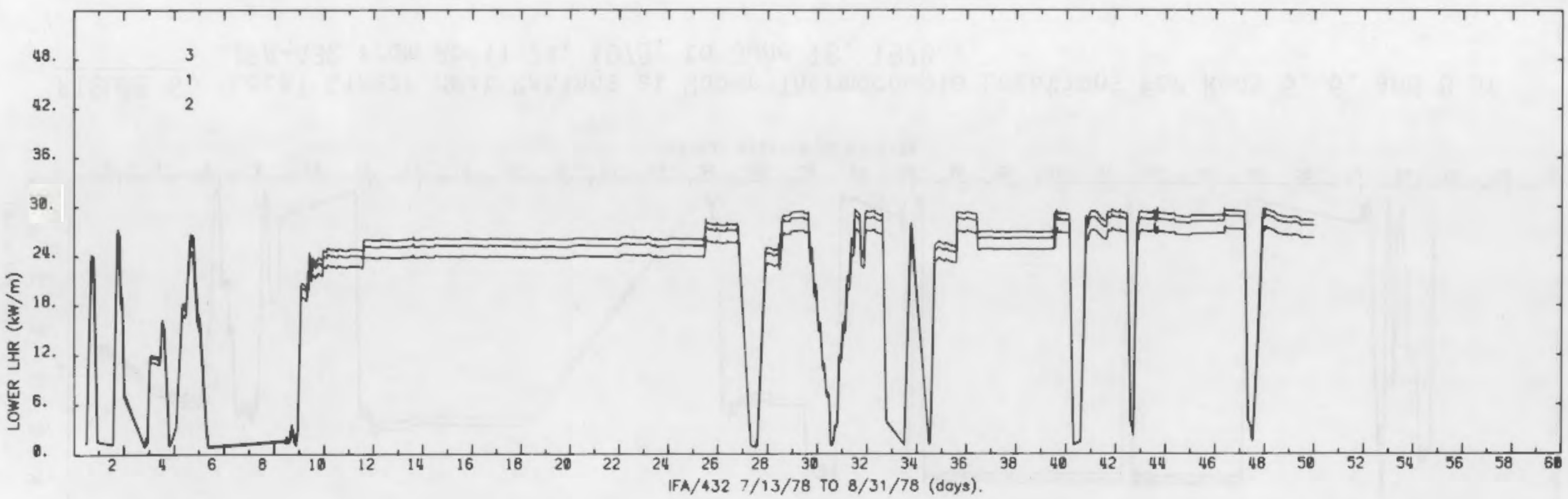


FIGURE 9. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from July 13, 1978, to August 31, 1978

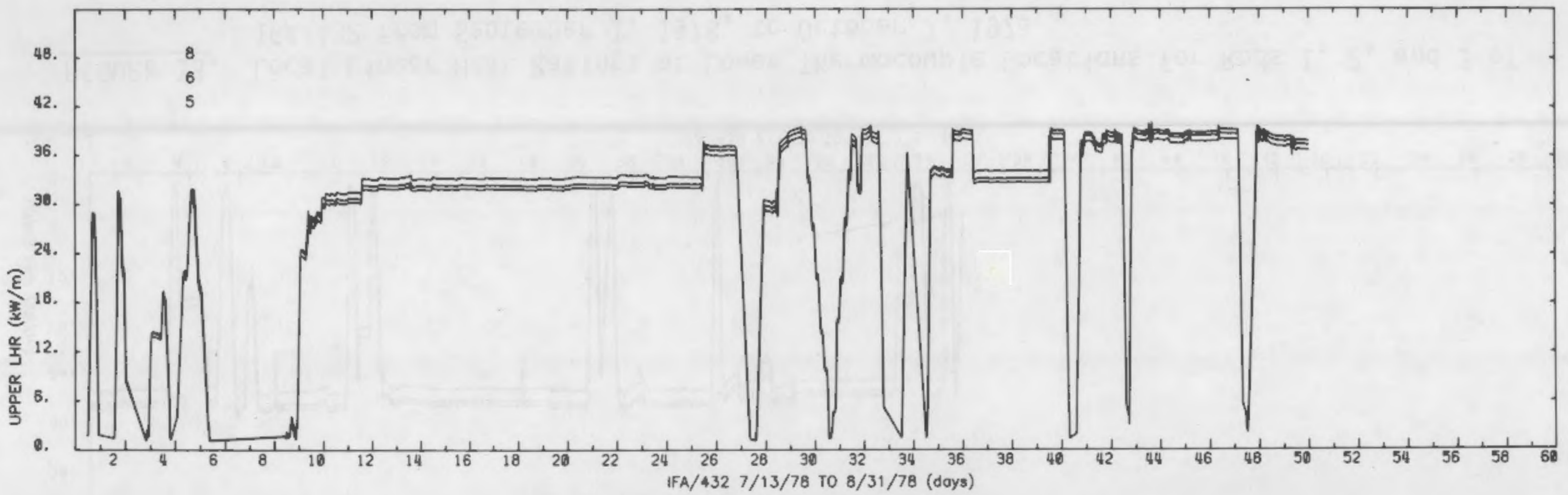


FIGURE 10. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from July 13, 1978, to August 31, 1978

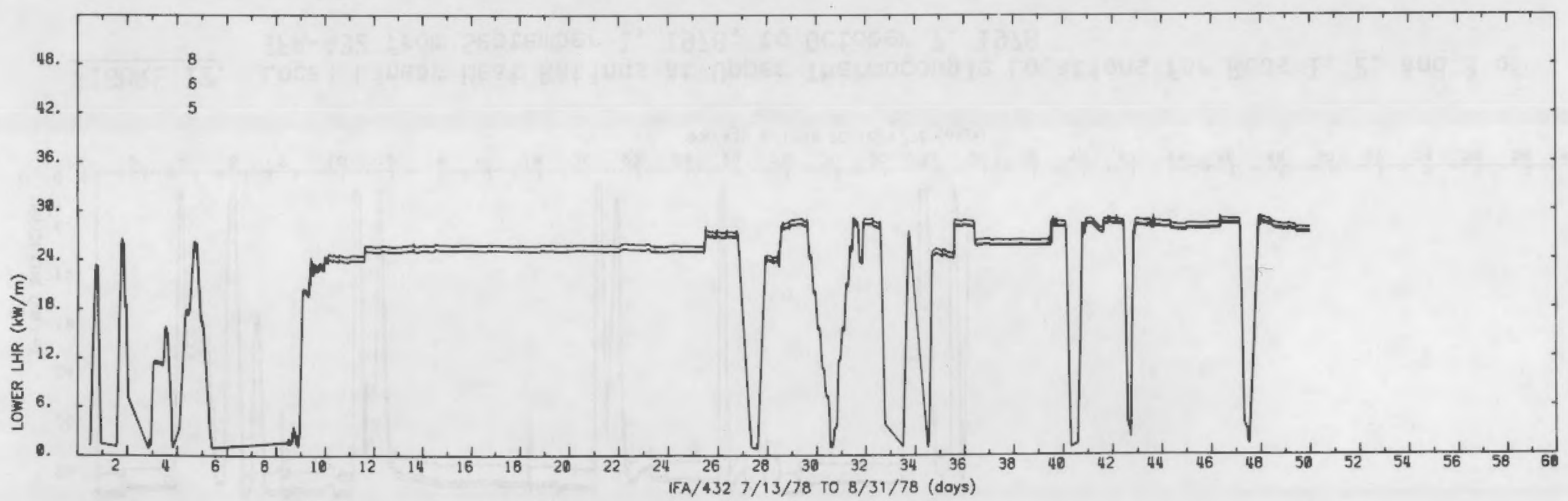


FIGURE 11. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from July 13, 1978, to August 31, 1978

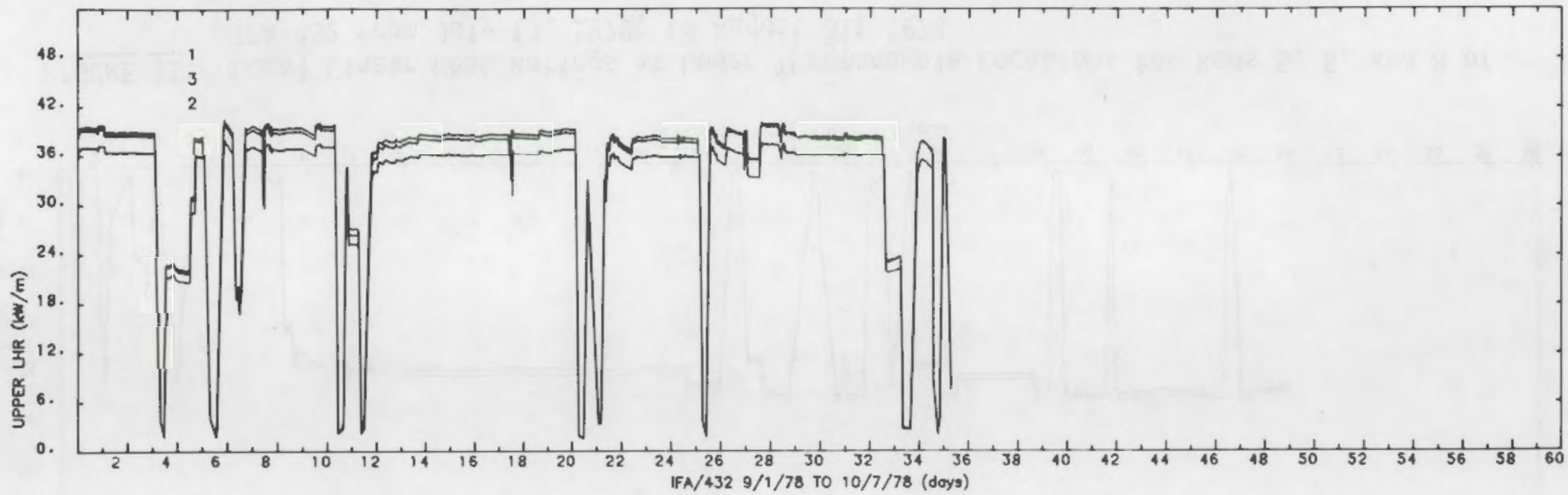


FIGURE 12. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from September 1, 1978, to October 7, 1978

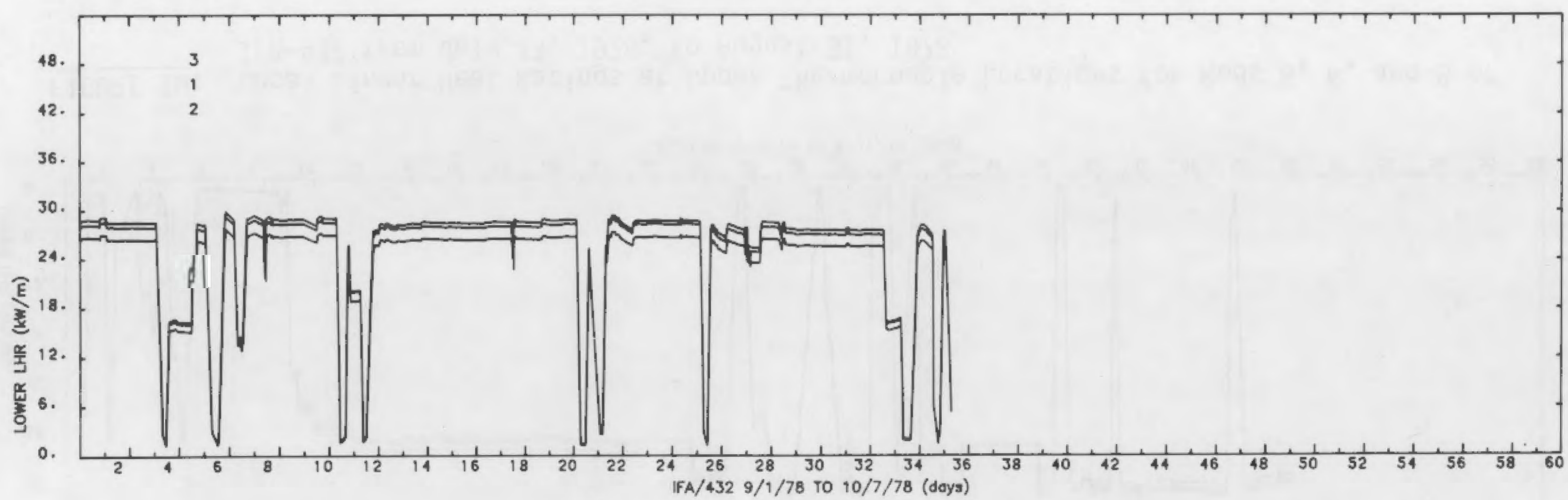


FIGURE 13. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from September 1, 1978, to October 7, 1978

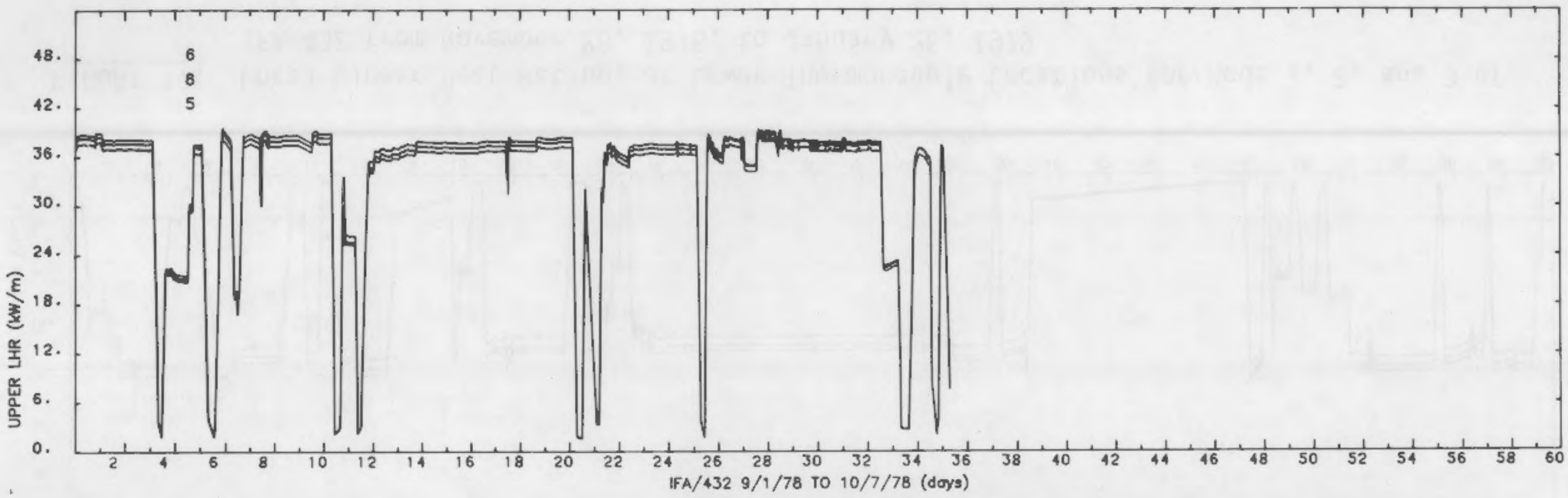


FIGURE 14. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from September 1, 1978, to October 7, 1978

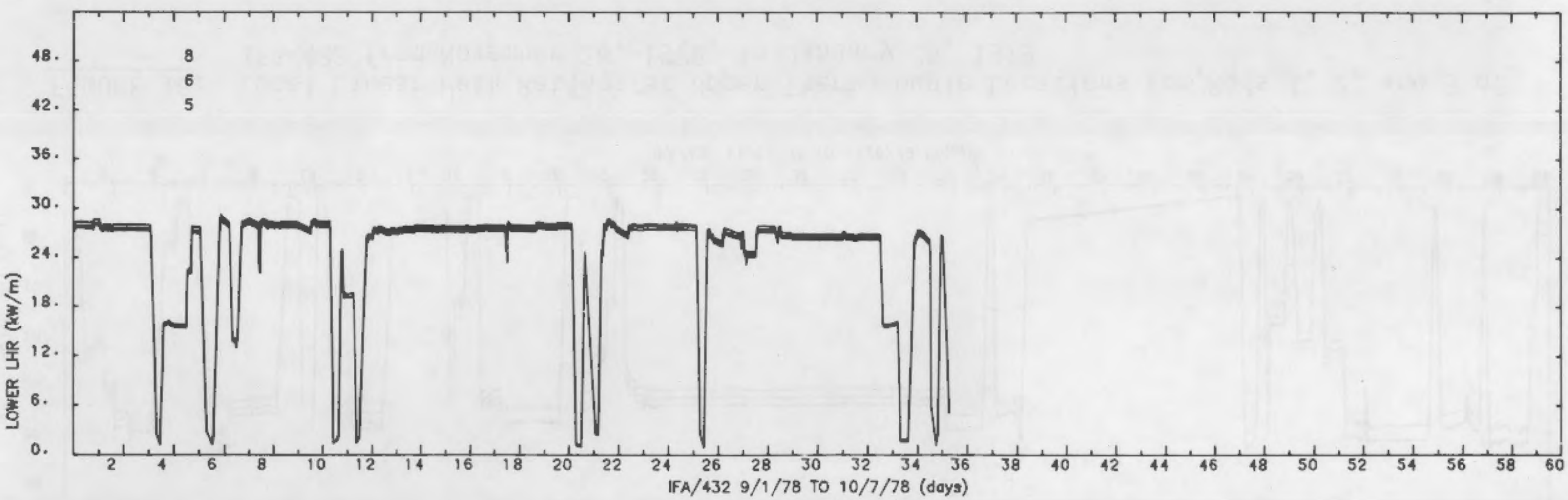


FIGURE 15. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from September 1, 1978, to October 7, 1978

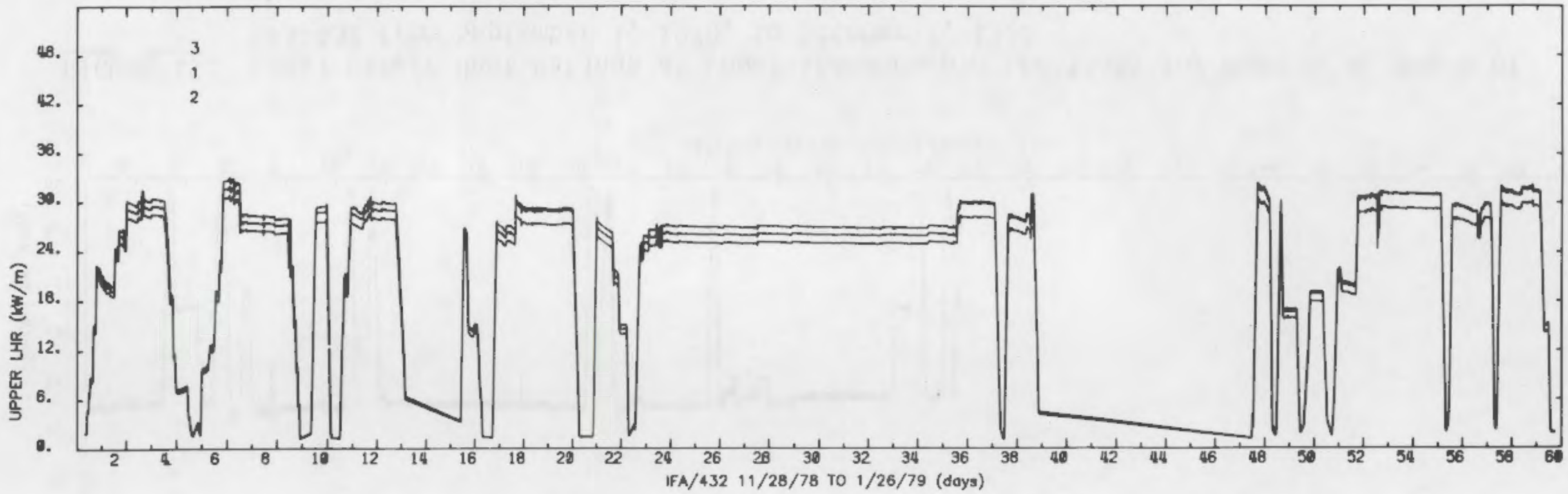


FIGURE 16. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from November 28, 1978, to January 26, 1979

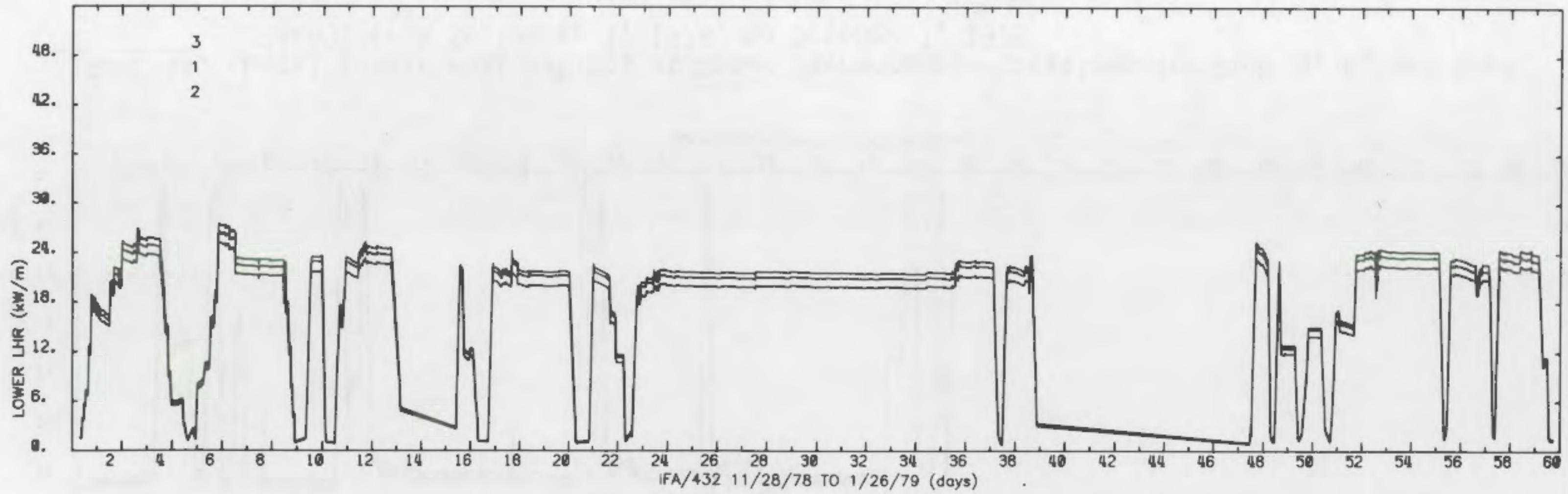


FIGURE 17. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from November 28, 1978, to January 26, 1979

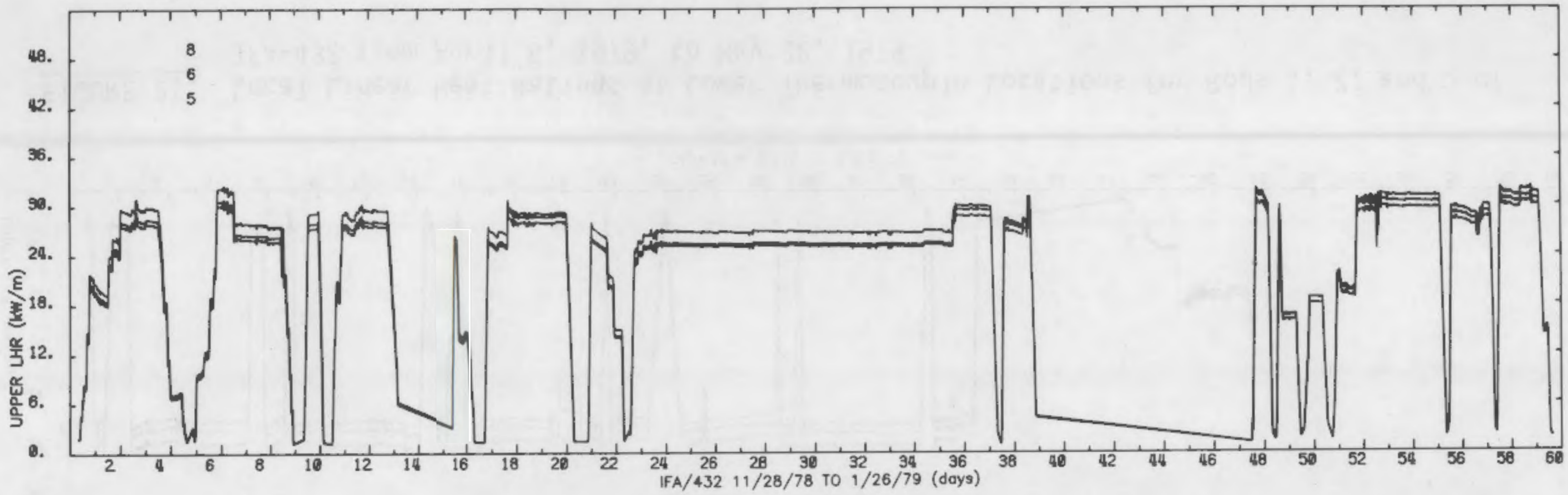


FIGURE 18. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from November 28, 1978, to January 26, 1979

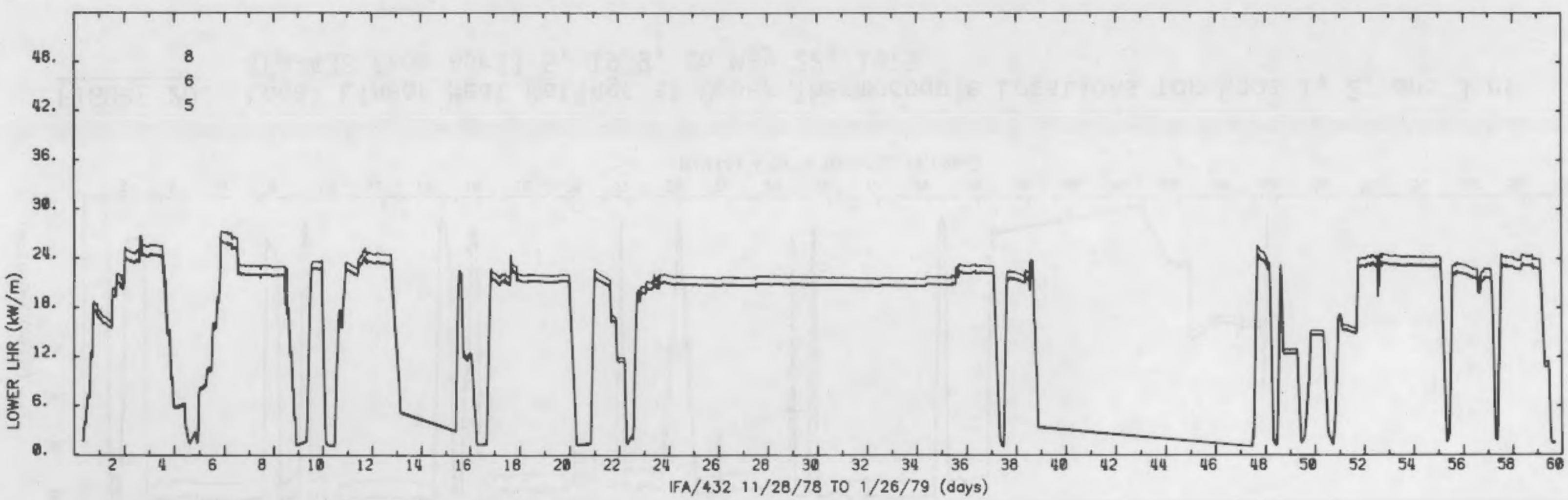


FIGURE 19. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from November 28, 1978, to January 26, 1979

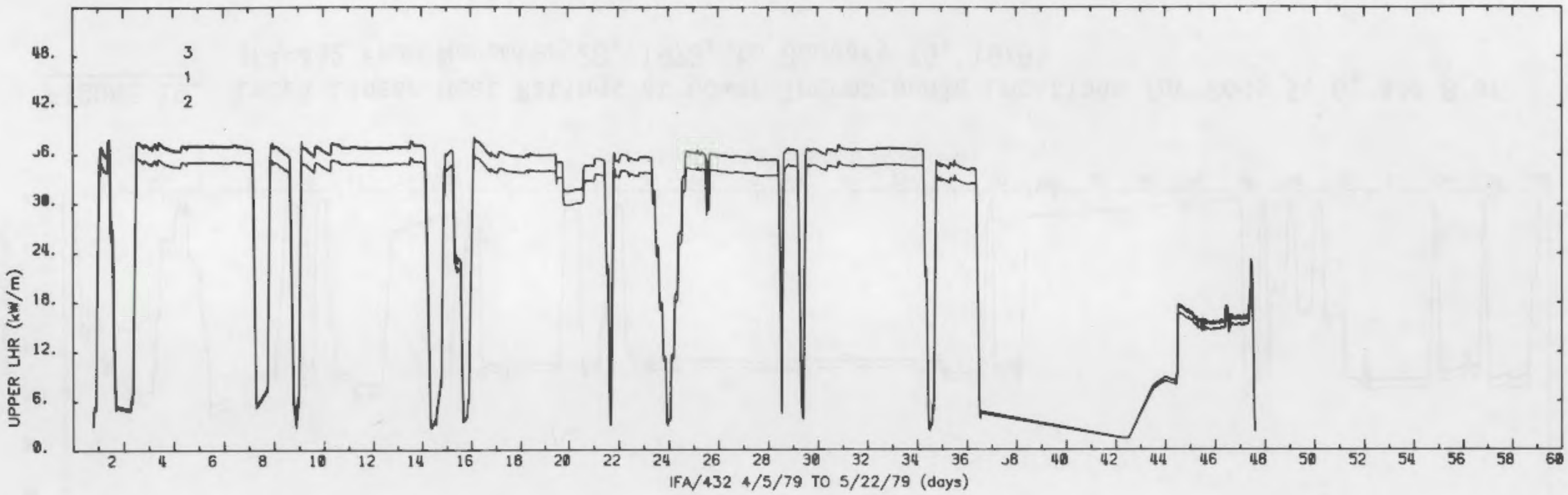


FIGURE 20. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from April 5, 1979, to May 22, 1979

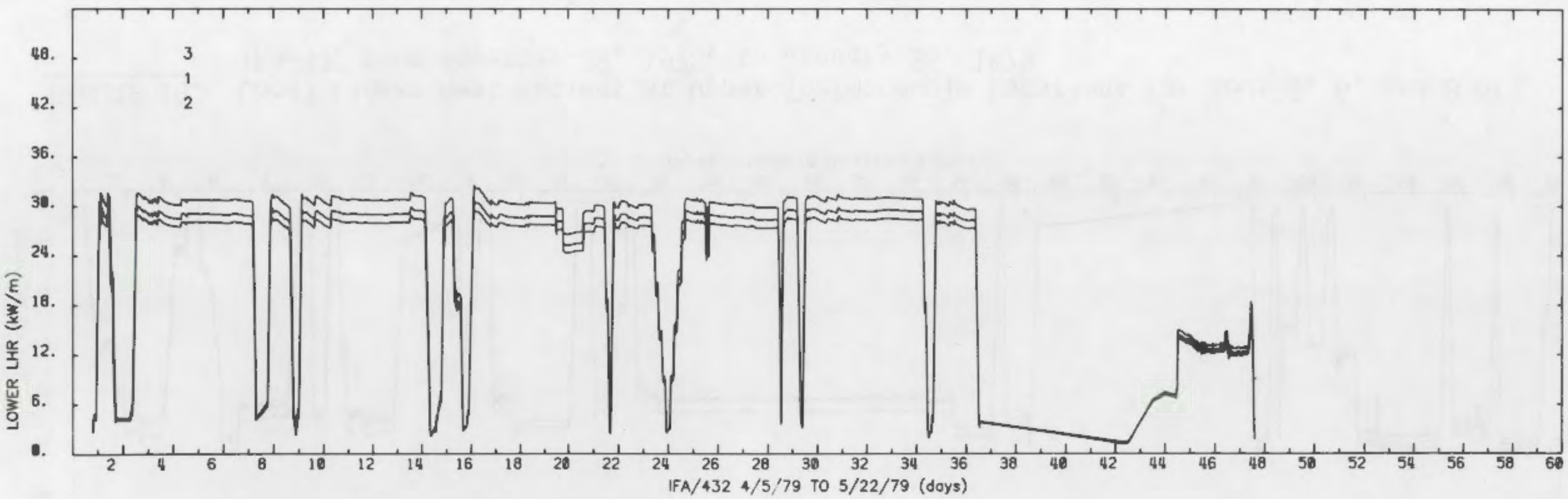


FIGURE 21. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from April 5, 1979, to May 22, 1979

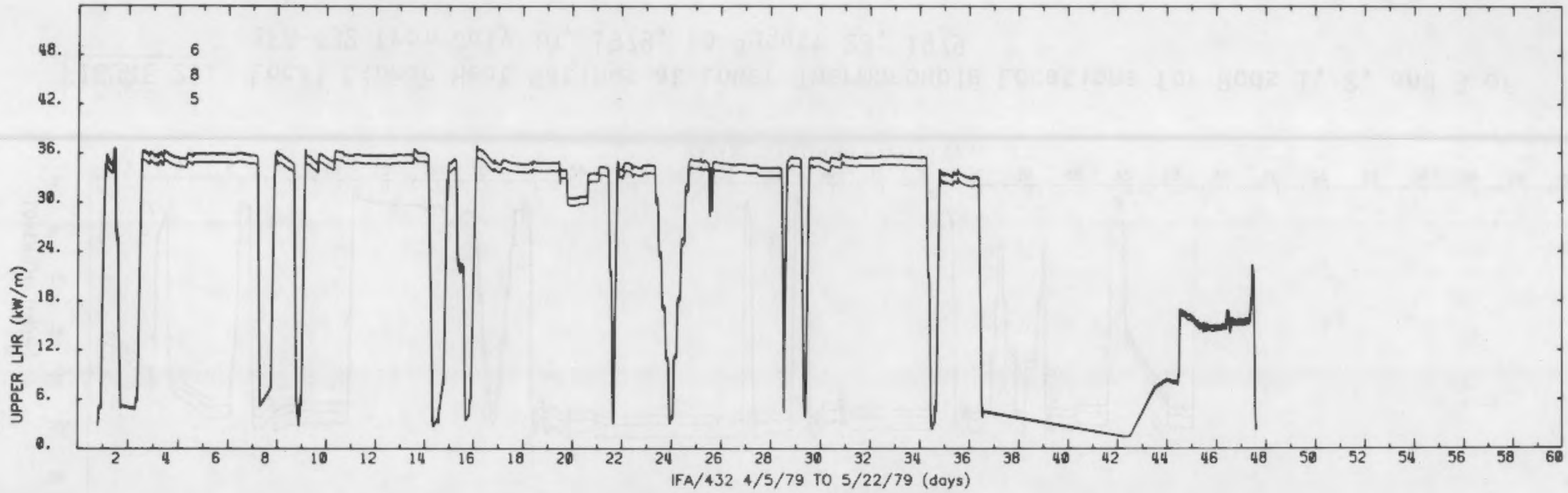


FIGURE 22. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from April 5, 1979, to May 22, 1979

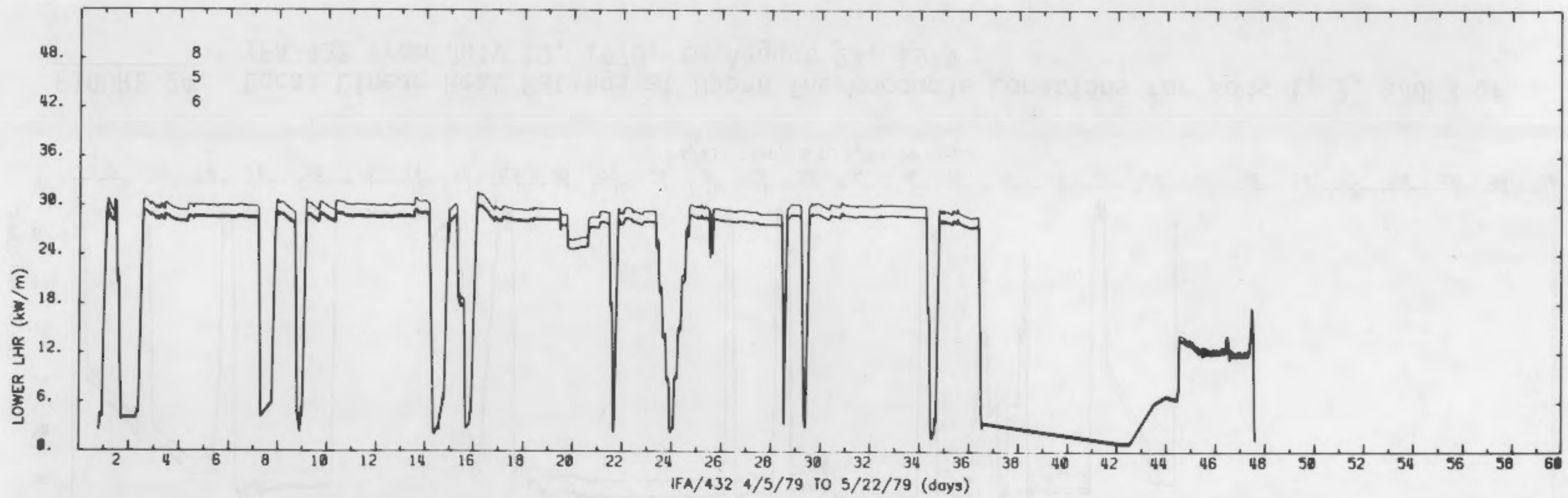


FIGURE 23. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from April 5, 1979, to May 22, 1979

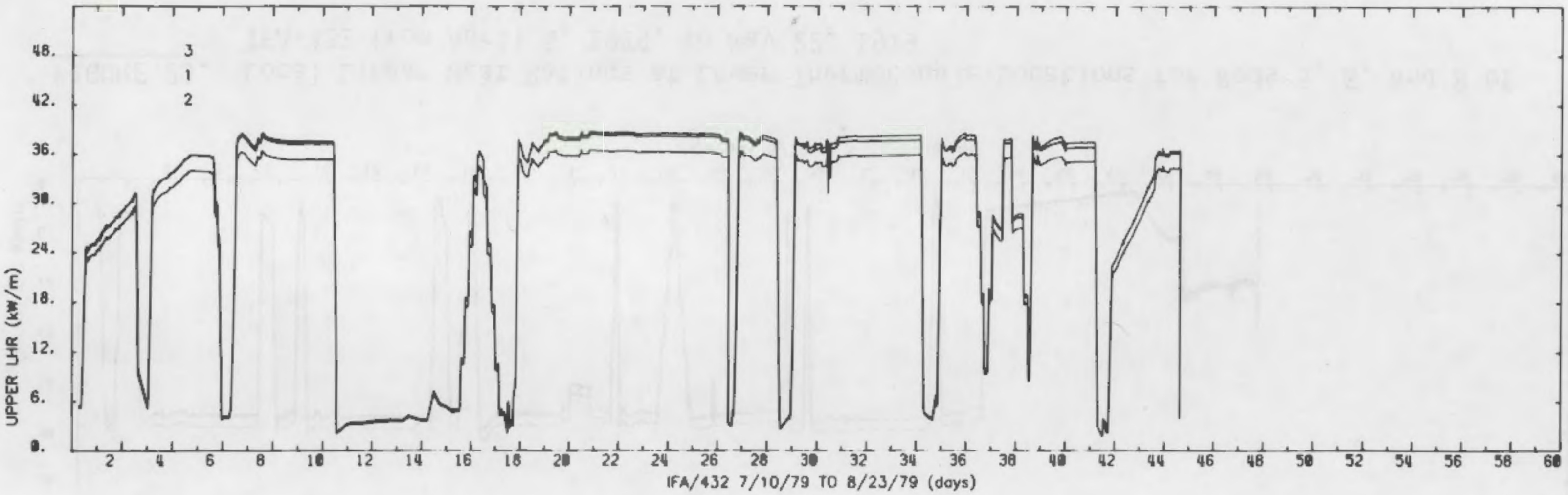


FIGURE 24. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from July 10, 1979, to August 23, 1979

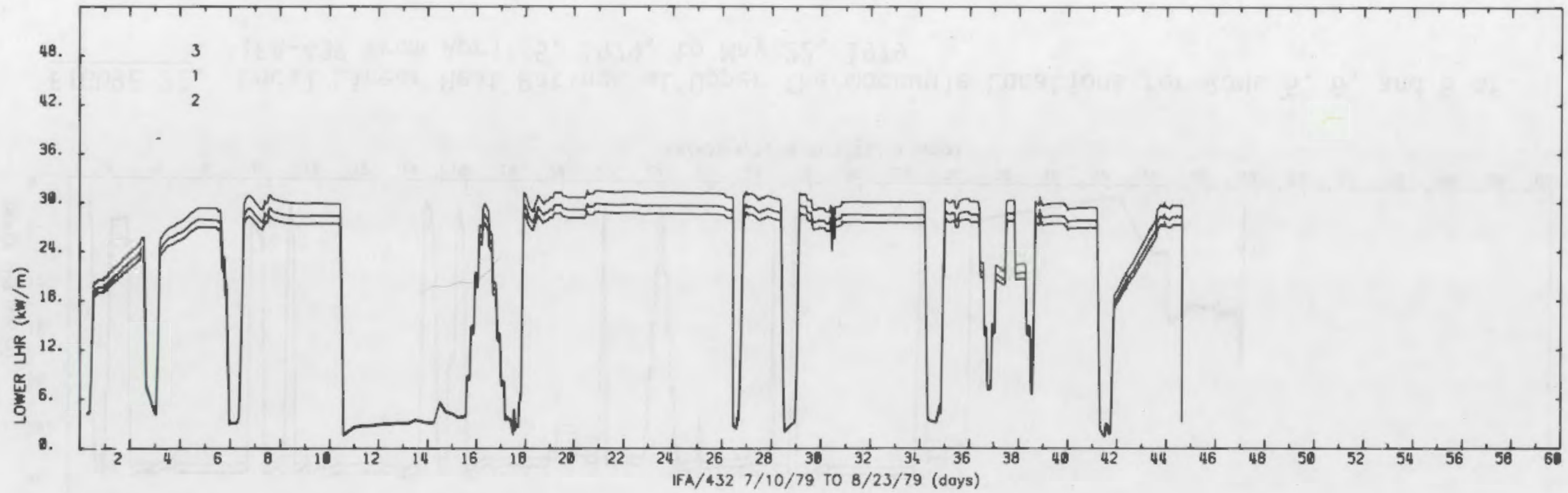


FIGURE 25. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from July 10, 1979, to August 23, 1979

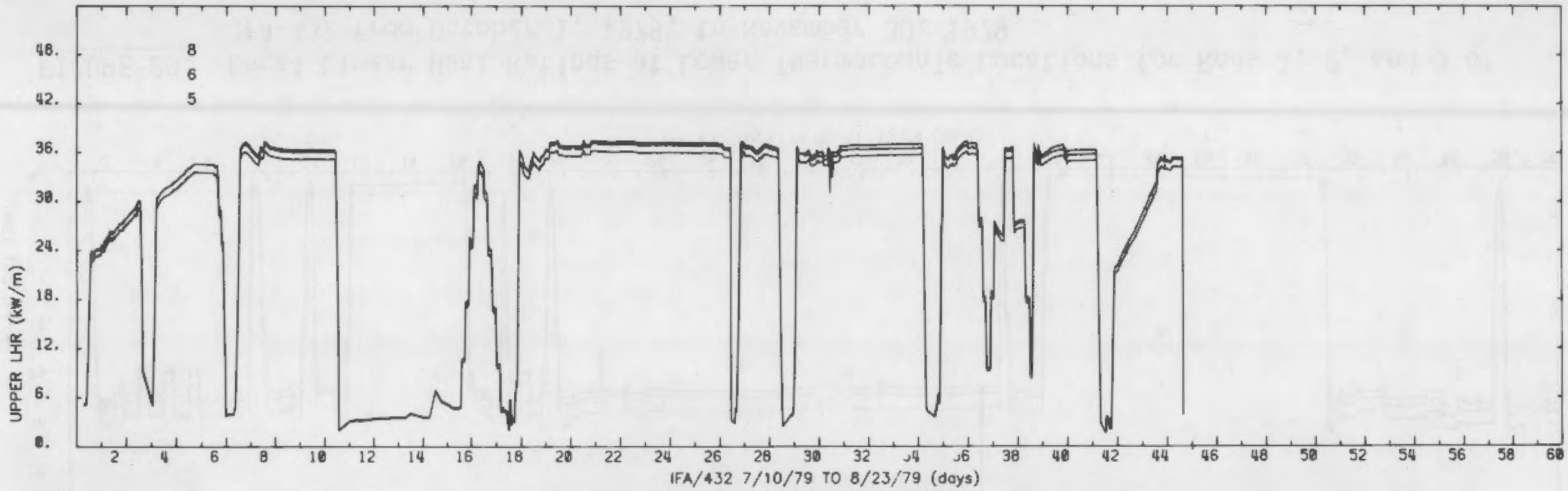


FIGURE 26. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from July 10, 1979, to August 23, 1979

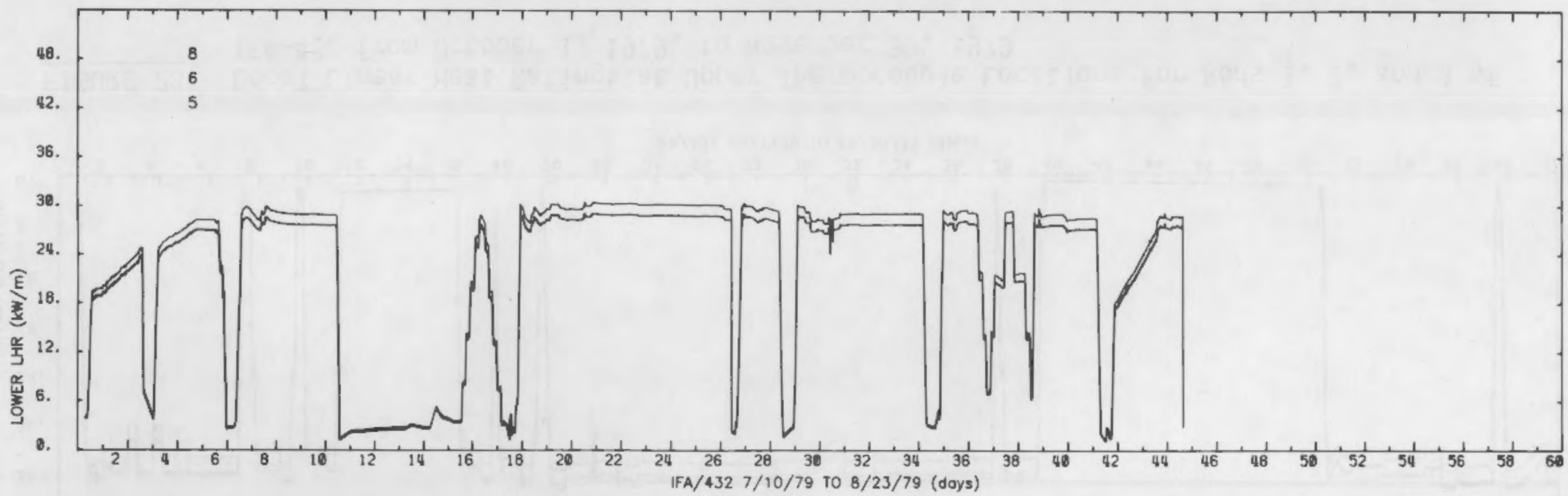


FIGURE 27. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from July 10, 1979, to August 23, 1979

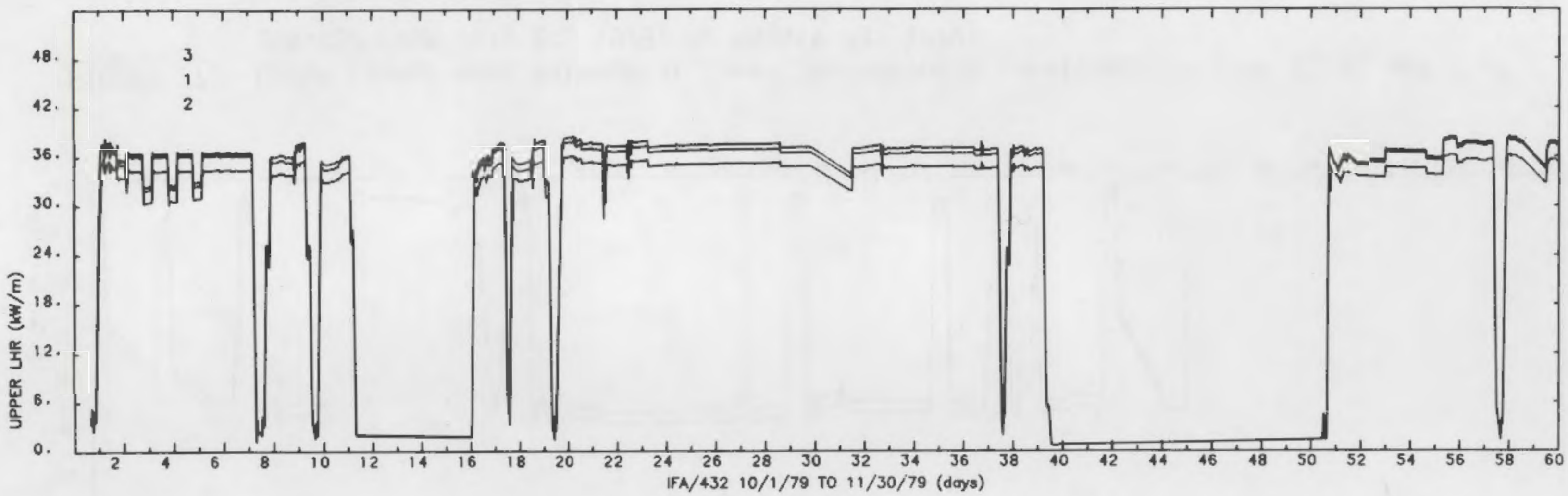


FIGURE 28. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from October 1, 1979, to November 30, 1979

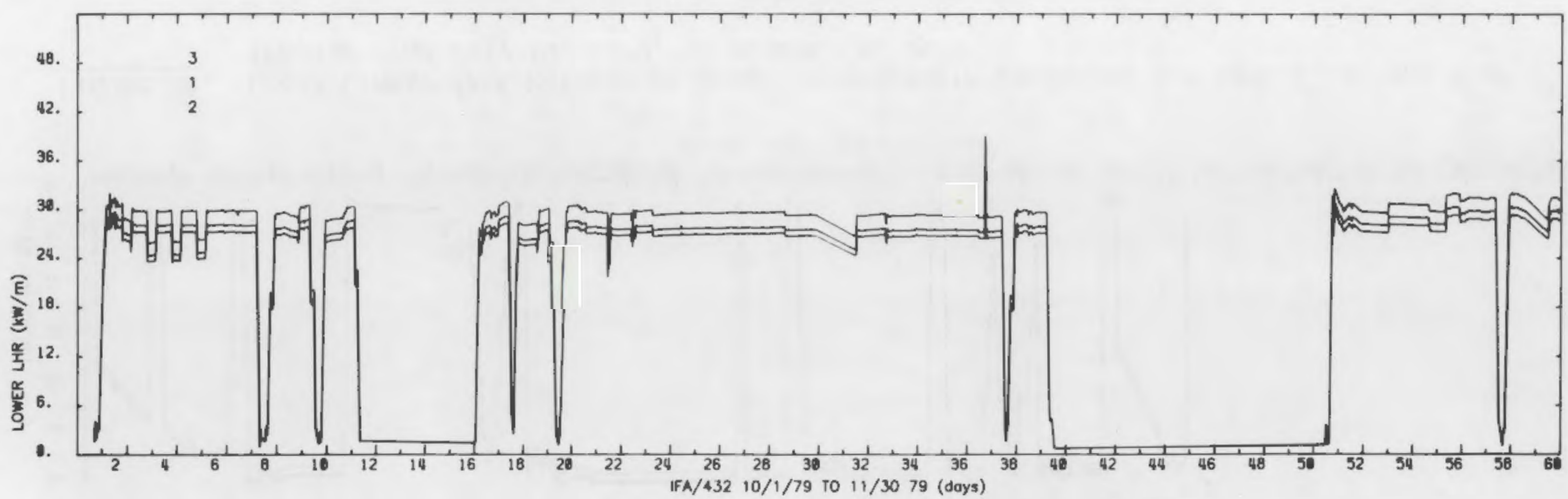


FIGURE 29. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from October 1, 1979, to November 30, 1979

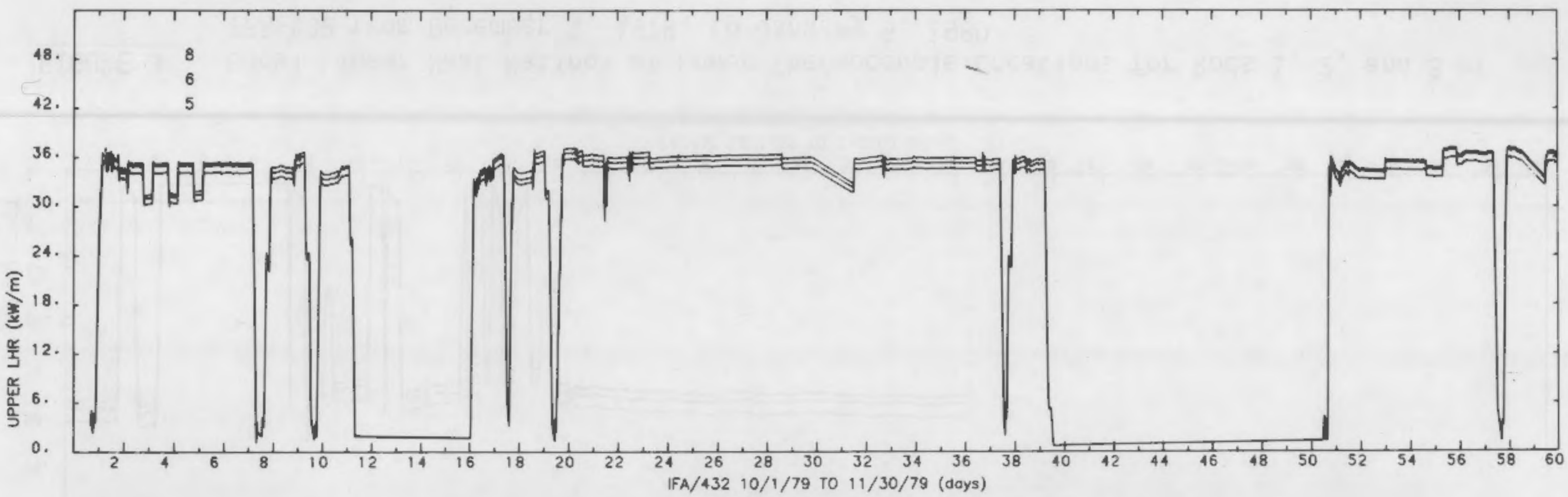


FIGURE 30. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from October 1, 1979, to November 30, 1979

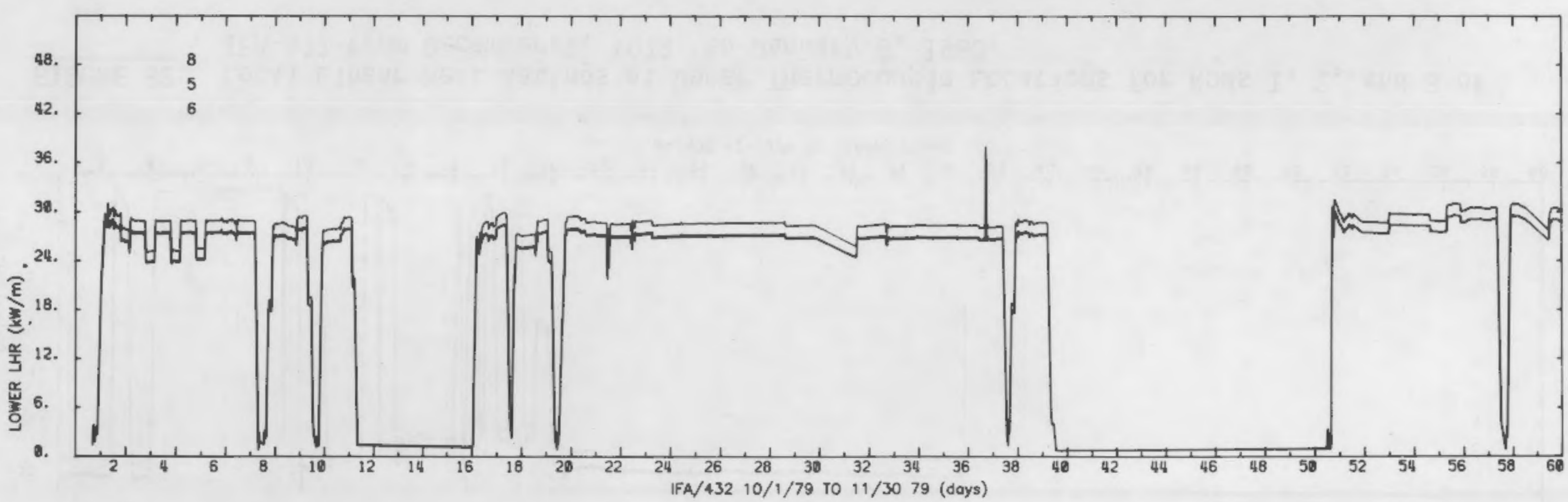


FIGURE 31. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from October 1, 1979, to November 30, 1979

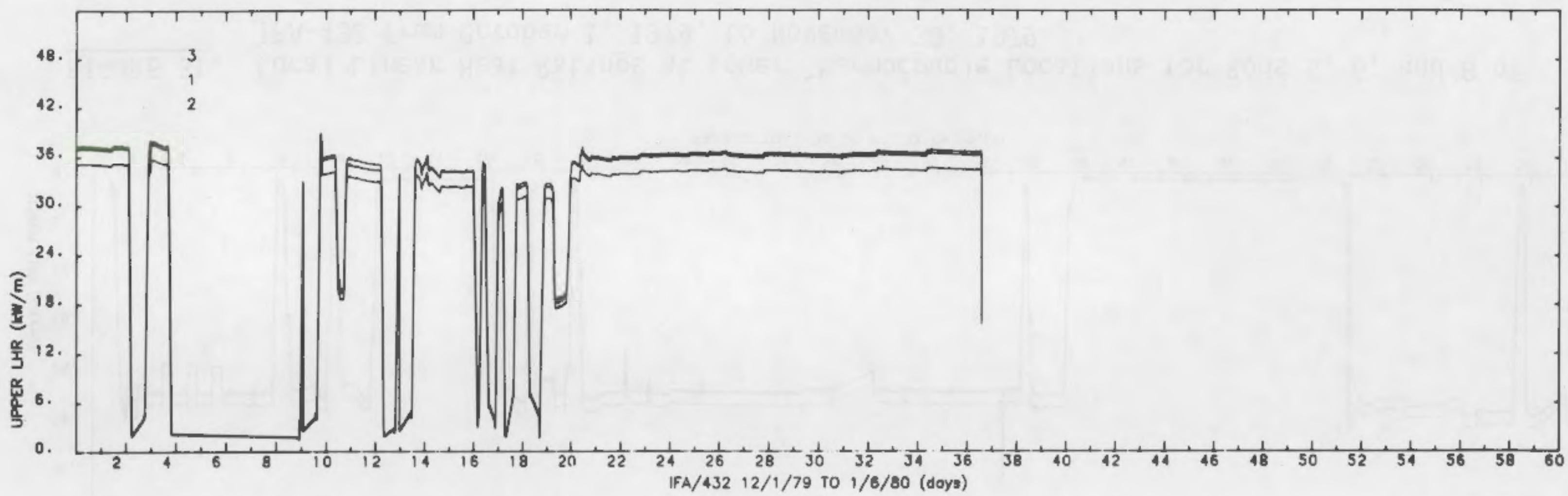


FIGURE 32. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from December 1, 1979, to January 6, 1980

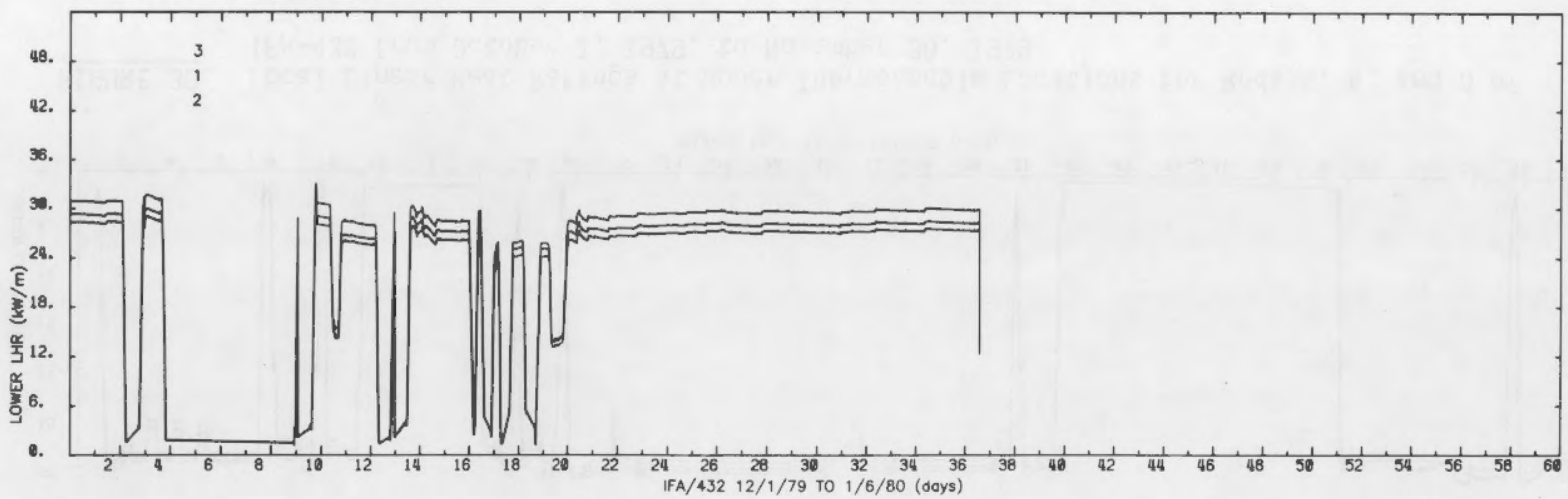


FIGURE 33. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from December 1, 1979, to January 6, 1980

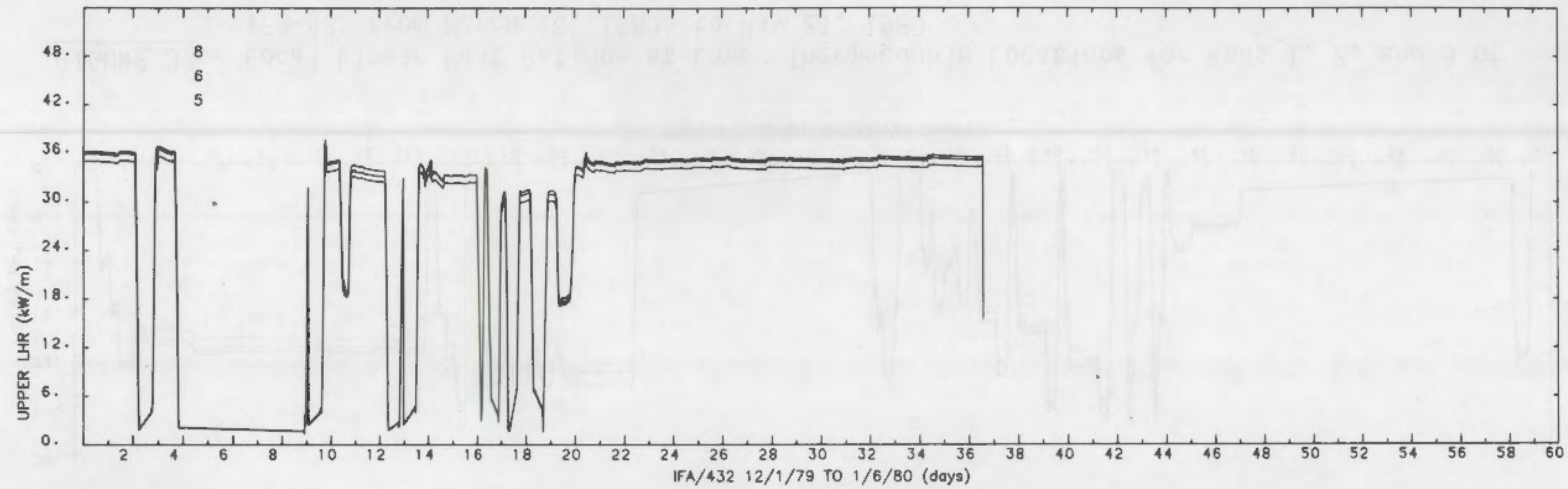


FIGURE 34. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from December 1, 1979, to January 6, 1980

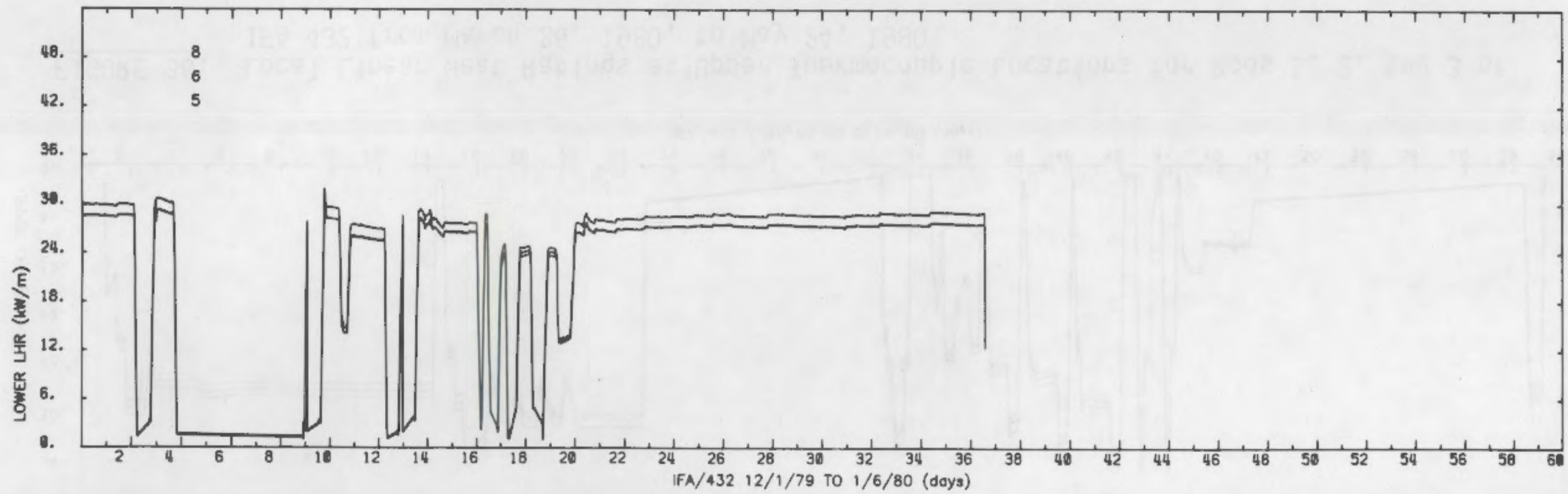


FIGURE 35. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 8 of IFA-432 from December 1, 1979, to January 6, 1980

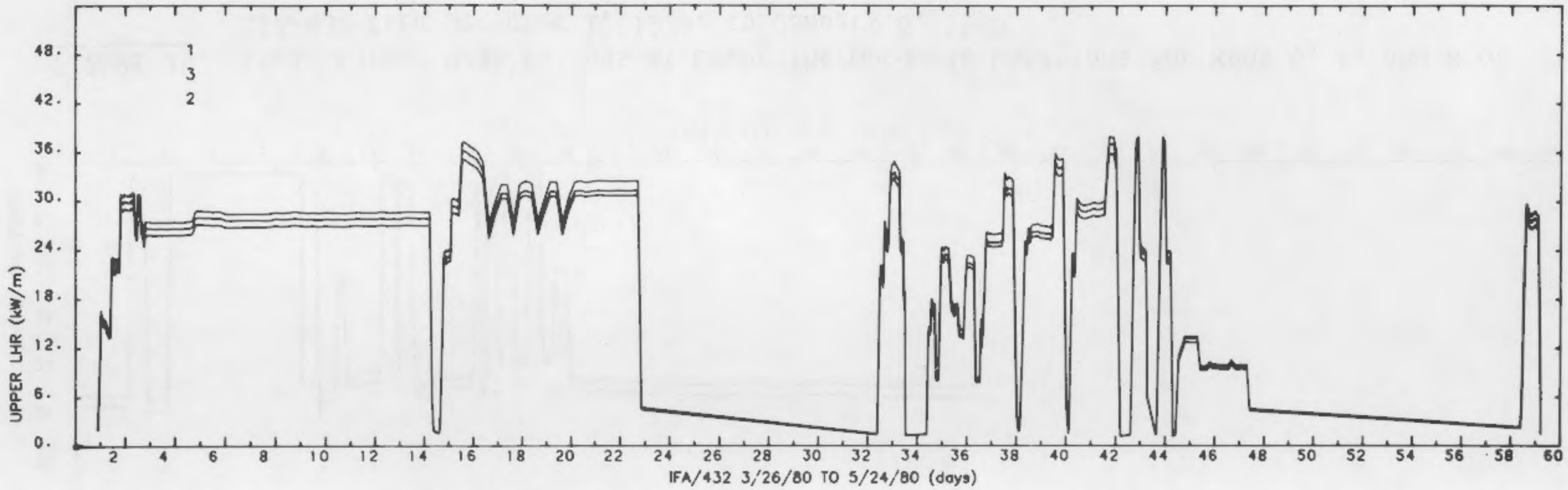


FIGURE 36. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from March 26, 1980, to May 24, 1980

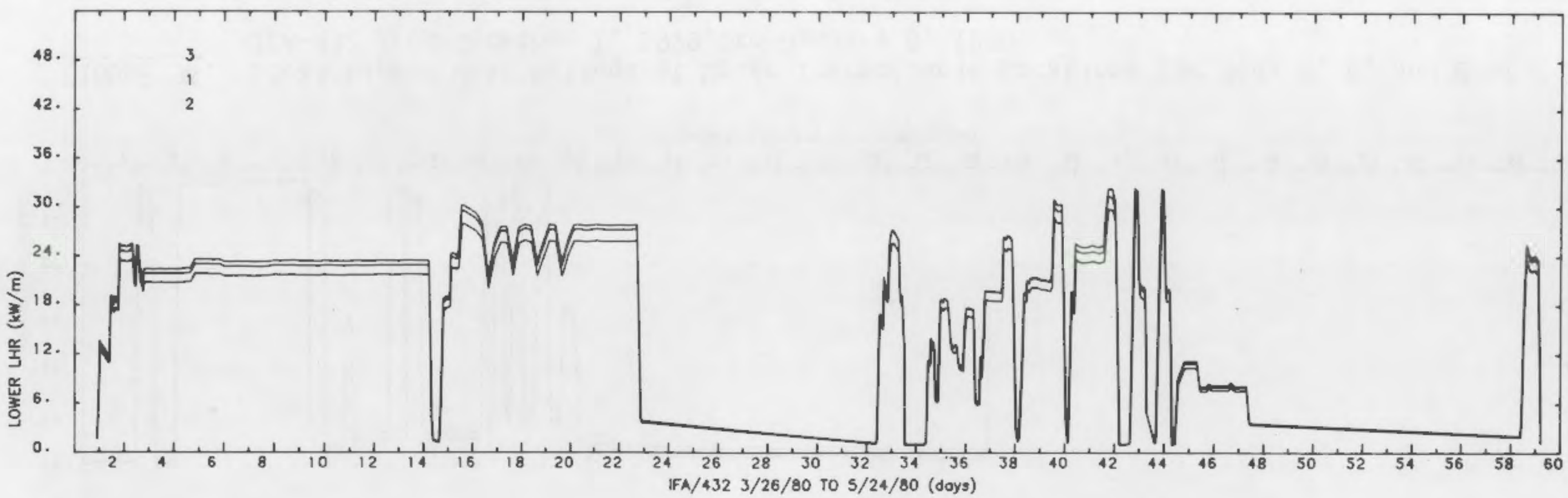


FIGURE 37. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 1, 2, and 3 of IFA-432 from March 26, 1980, to May 24, 1980

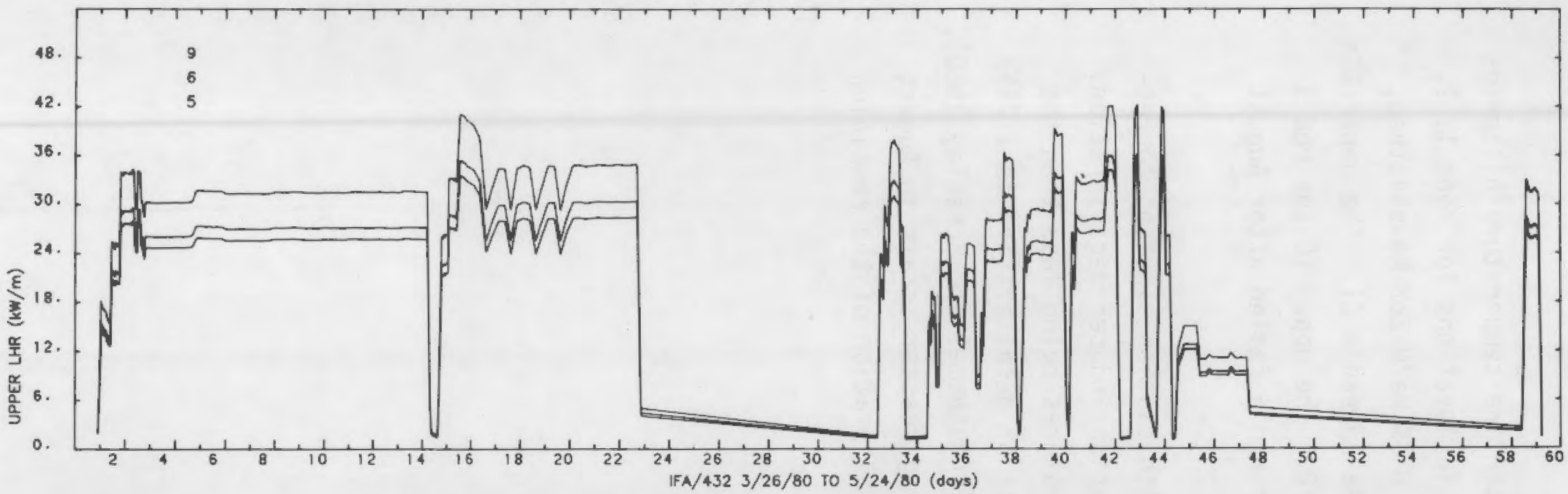


FIGURE 38. Local Linear Heat Ratings at Upper Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from March 26, 1980, to May 24, 1980

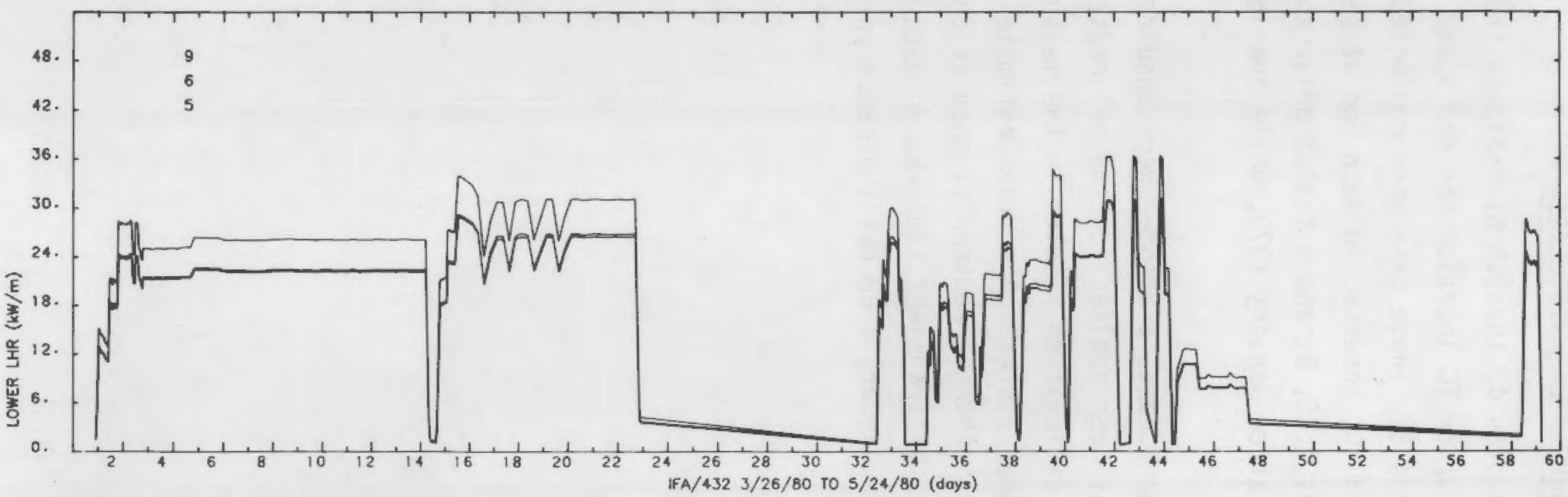


FIGURE 39. Local Linear Heat Ratings at Lower Thermocouple Locations for Rods 5, 6, and 9 of IFA-432 from March 26, 1980, to May 24, 1980

FUEL TEMPERATURE HISTORIES

Figures 40 through 61 indicate the fuel centerline temperature histories at the upper TC location for rod 3 and the lower TC locations for rods 1, 2, 3, 5, and 6. These 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. The upper TC for rod 3 failed after January 1979, while the lower TC for rod 6 failed after August 1979.

The TC data presented here should be used with caution since no correction has been applied for thermal neutron irradiation-induced decalibration. This decalibration results in the measured temperatures being less than the true temperatures. A current estimate of the rate of decalibration is 1.75%/ 10^{24} n/m^2 thermal neutron fluence at the TC tip (Crouthamel and Freshley 1980). Analysis of transient temperature data taken during reactor scrams in August 1979 and January 1980 has indicated up to 20% decalibration of the remaining TCs.

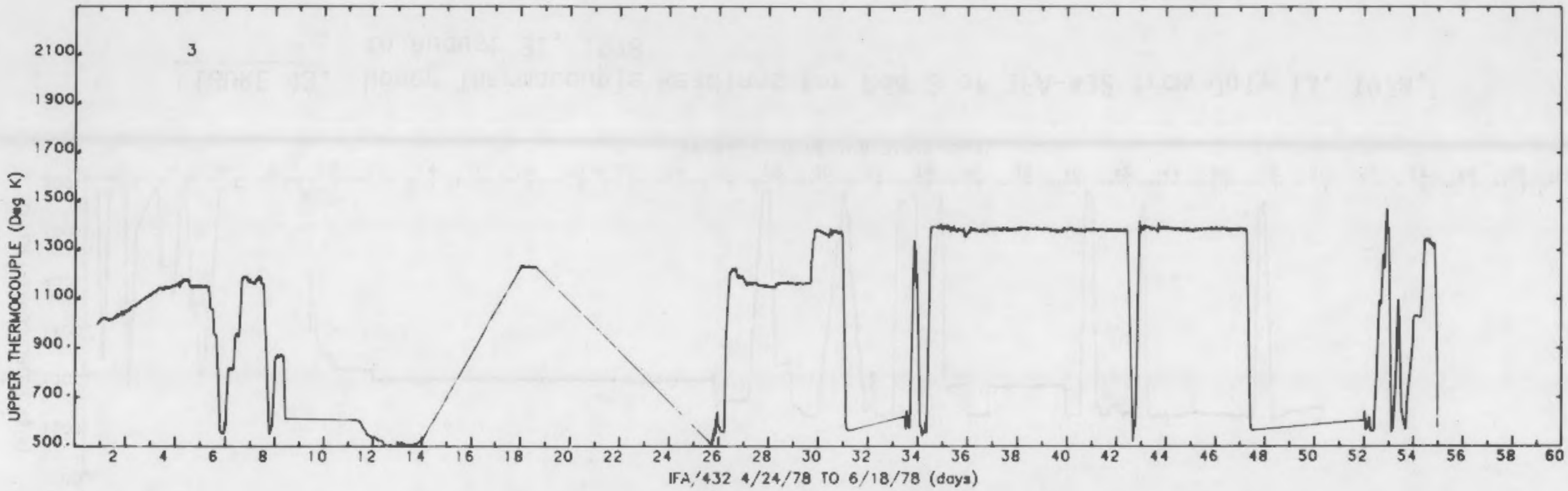


FIGURE 40. Upper Thermocouple Readings for Rod 3 of IFA-432 from April 24, 1978, to June 18, 1978

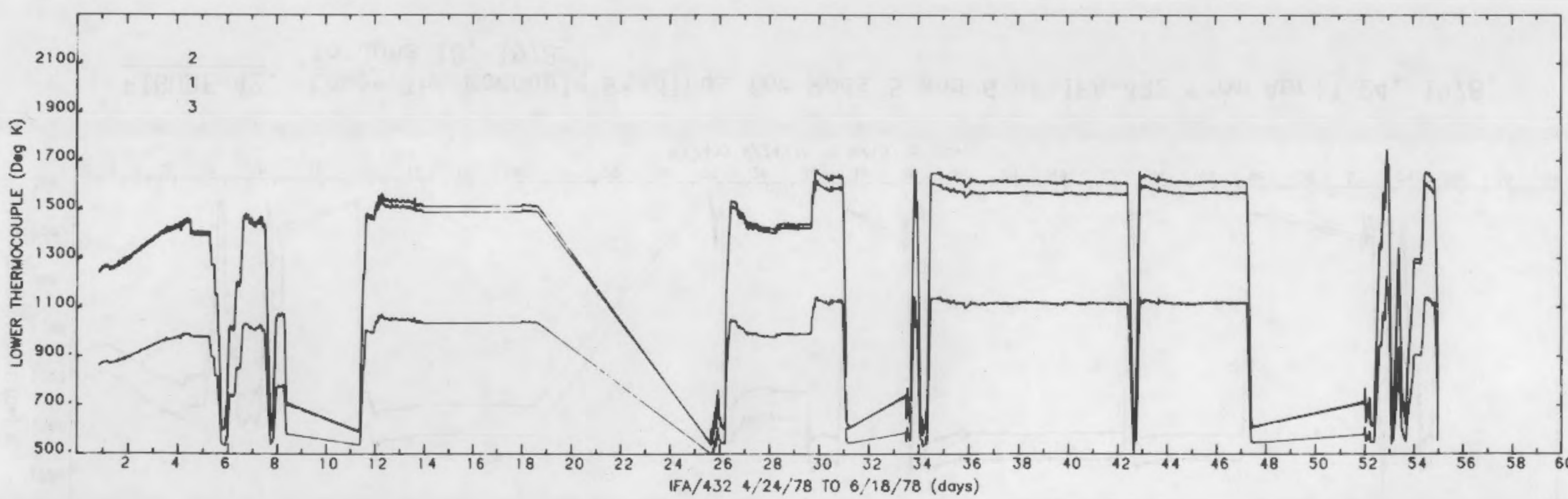


FIGURE 41. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from April 24, 1978, to June 18, 1978

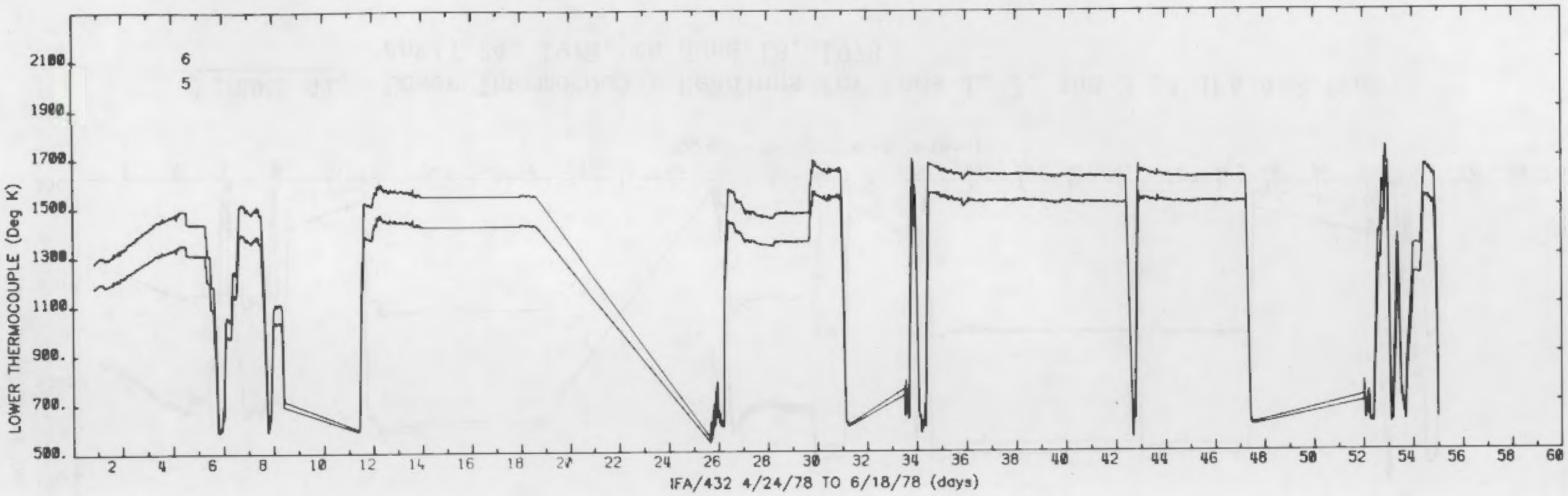


FIGURE 42. Lower Thermocouple Readings for Rods 5 and 6 of IFA-432 from April 24, 1978, to June 18, 1978

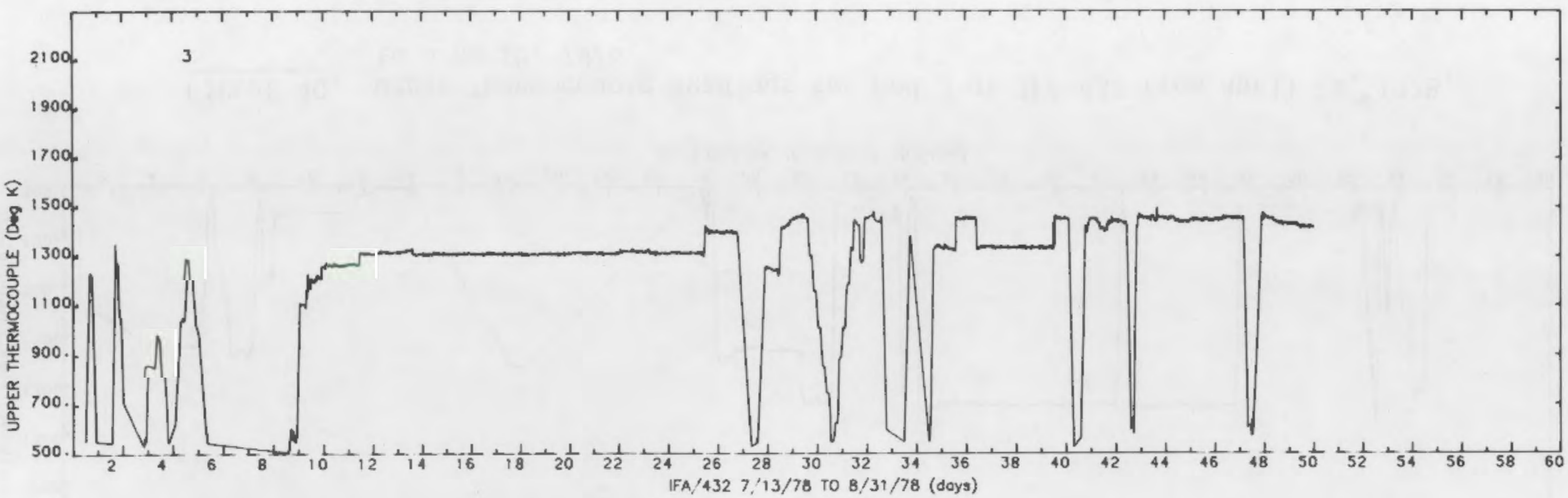


FIGURE 43. Upper Thermocouple Readings for Rod 3 of IFA-432 from July 13, 1978, to August 31, 1978

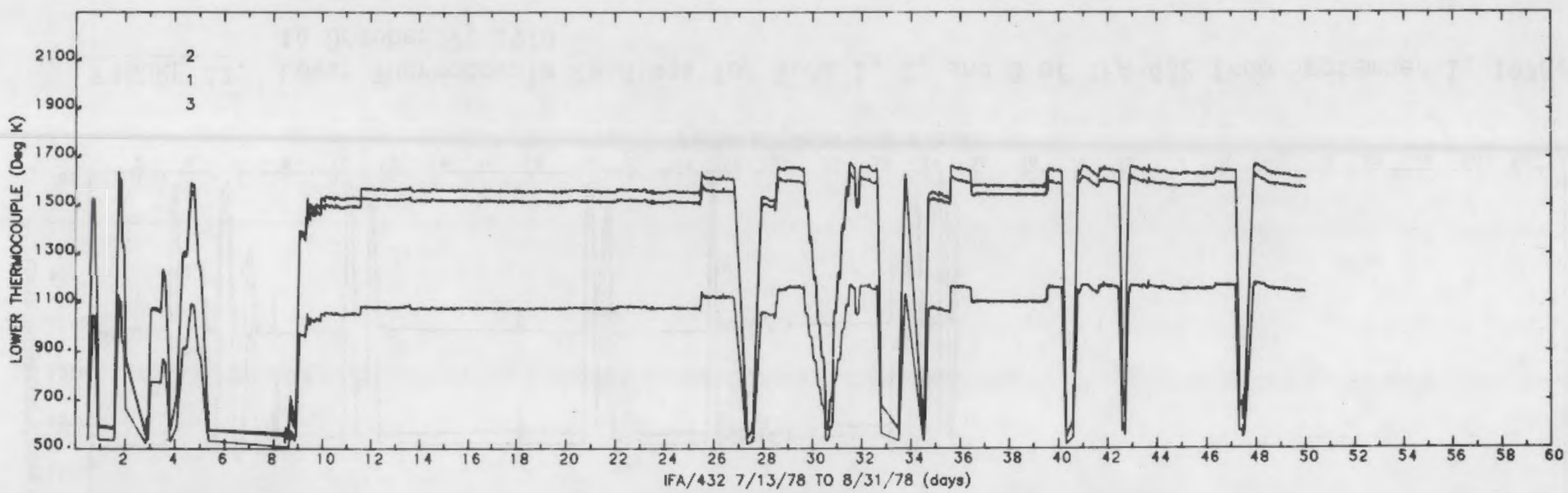


FIGURE 44. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from July 13, 1978, to August 31, 1978

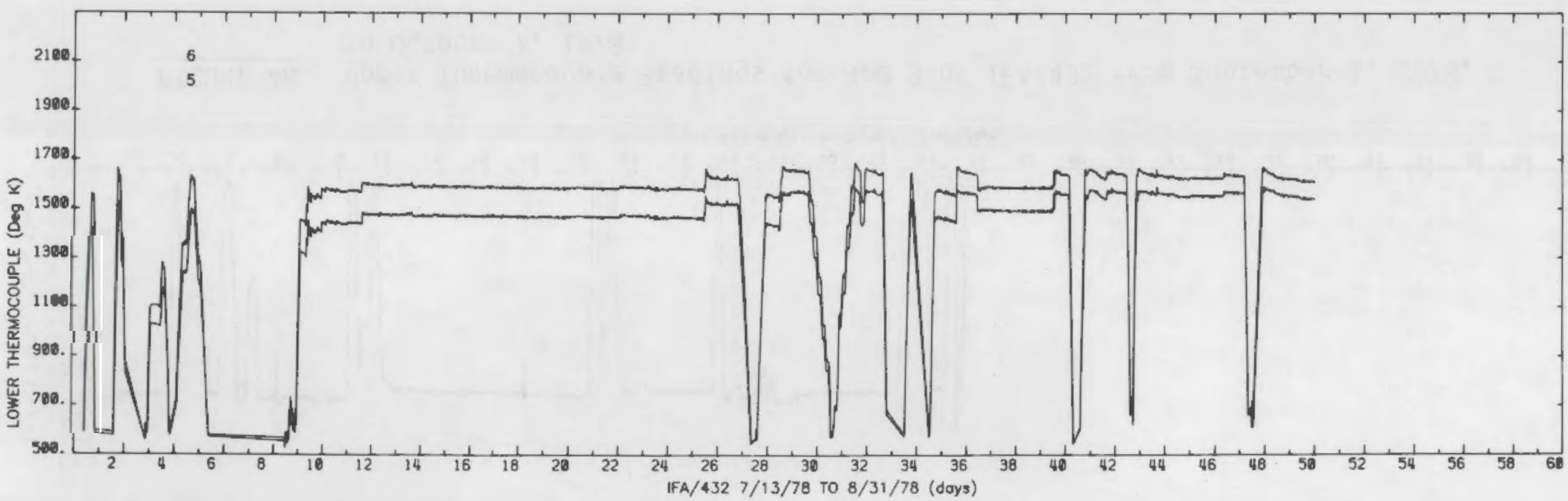


FIGURE 45. Lower Thermocouple Readings for Rods 5 and 6 of IFA-432 from July 13, 1978, to August 31, 1978

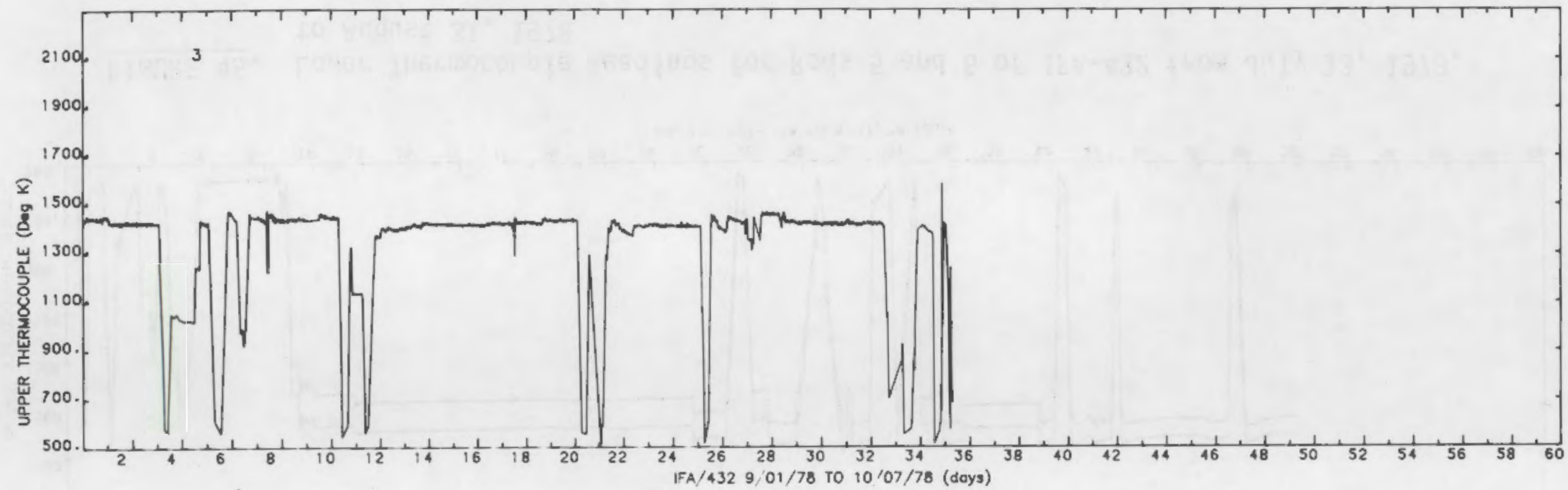


FIGURE 46. Upper Thermocouple Readings for Rod 3 of IFA-432 from September 1, 1978, to October 7, 1978

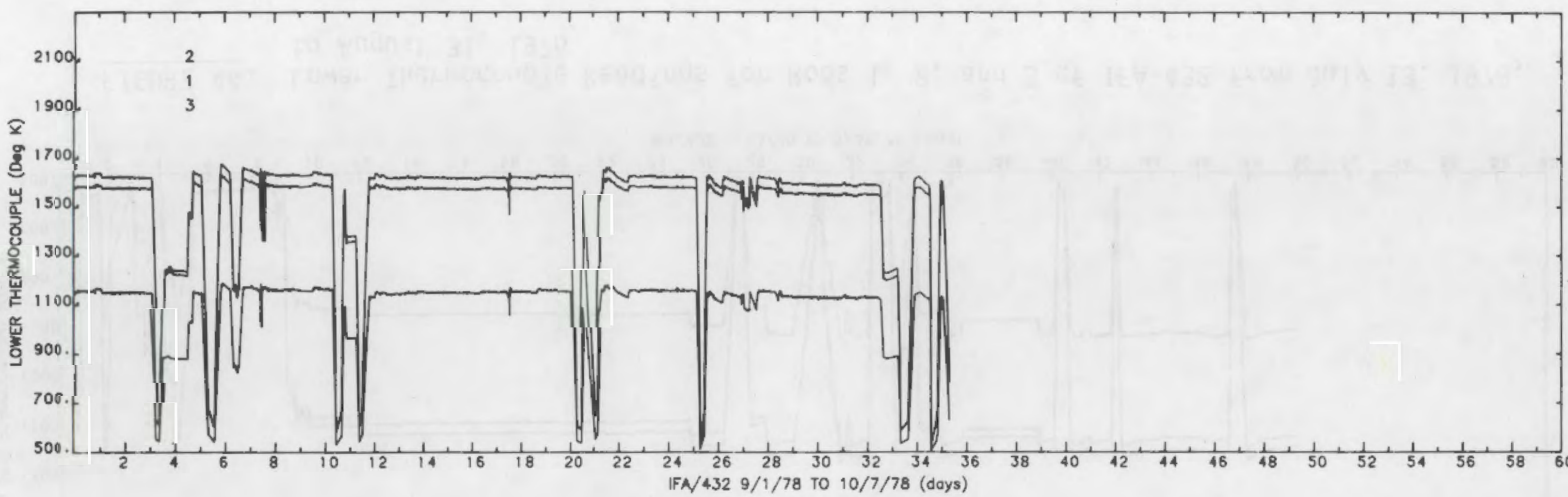


FIGURE 47. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from September 1, 1978, to October 7, 1978

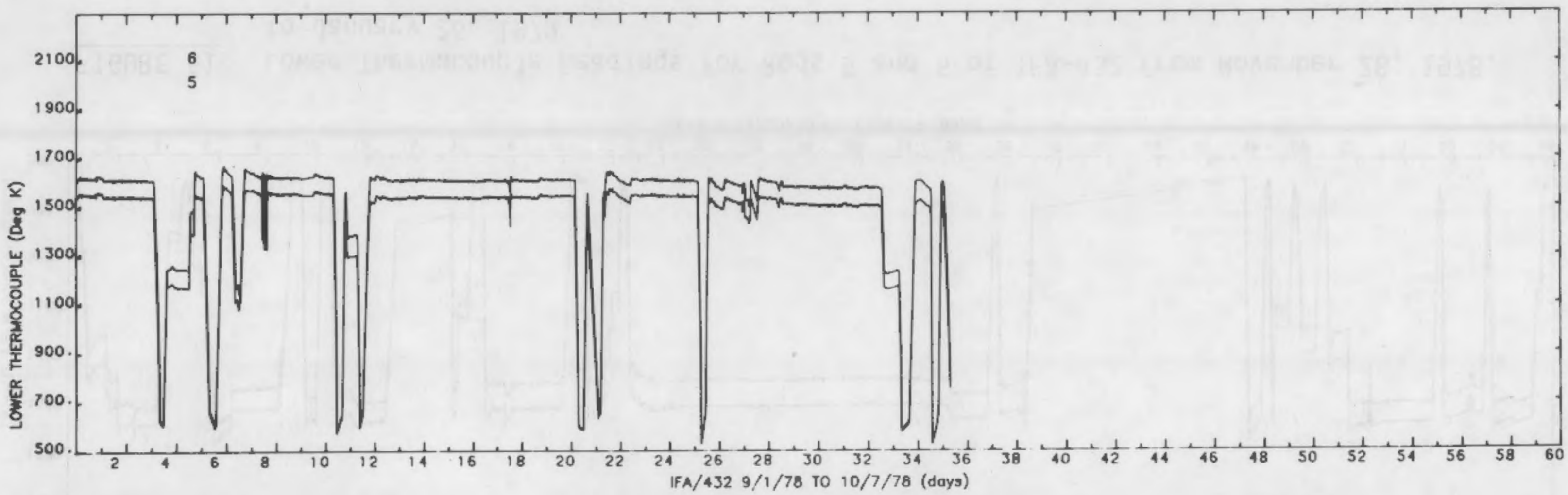


FIGURE 48. Lower Thermocouple Readings for Rods 5 and 6 of IFA-432 from September 1, 1978, to October 7, 1978

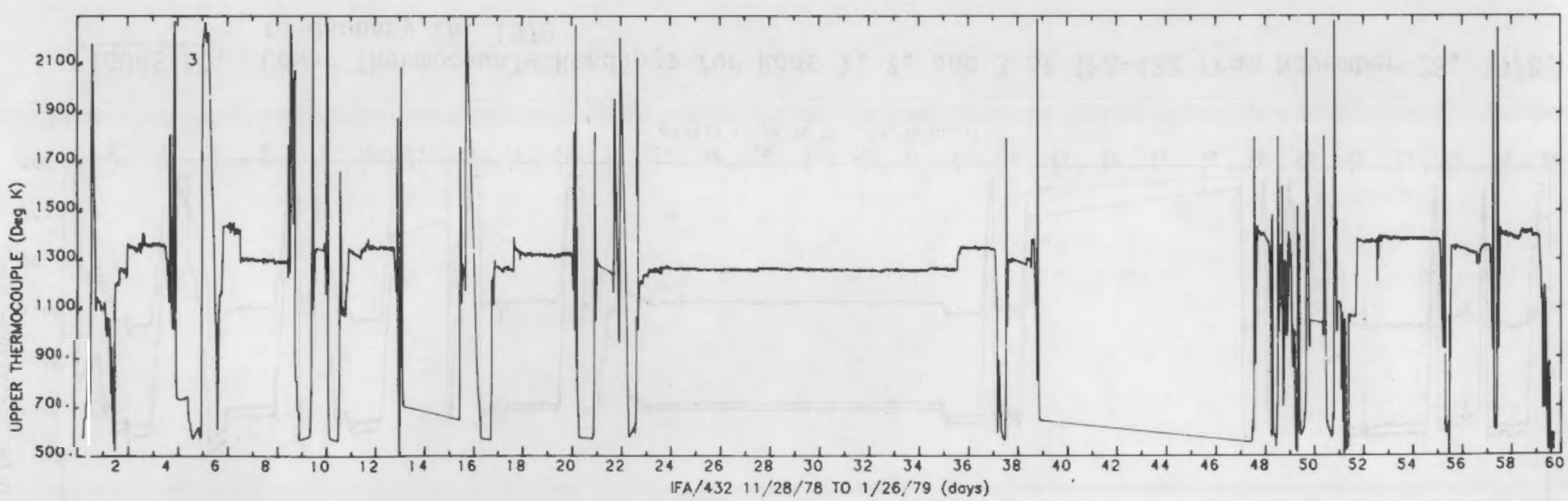


FIGURE 49. Upper Thermocouple Readings for Rod 3 of IFA-432 from November 28, 1978, to January 26, 1979

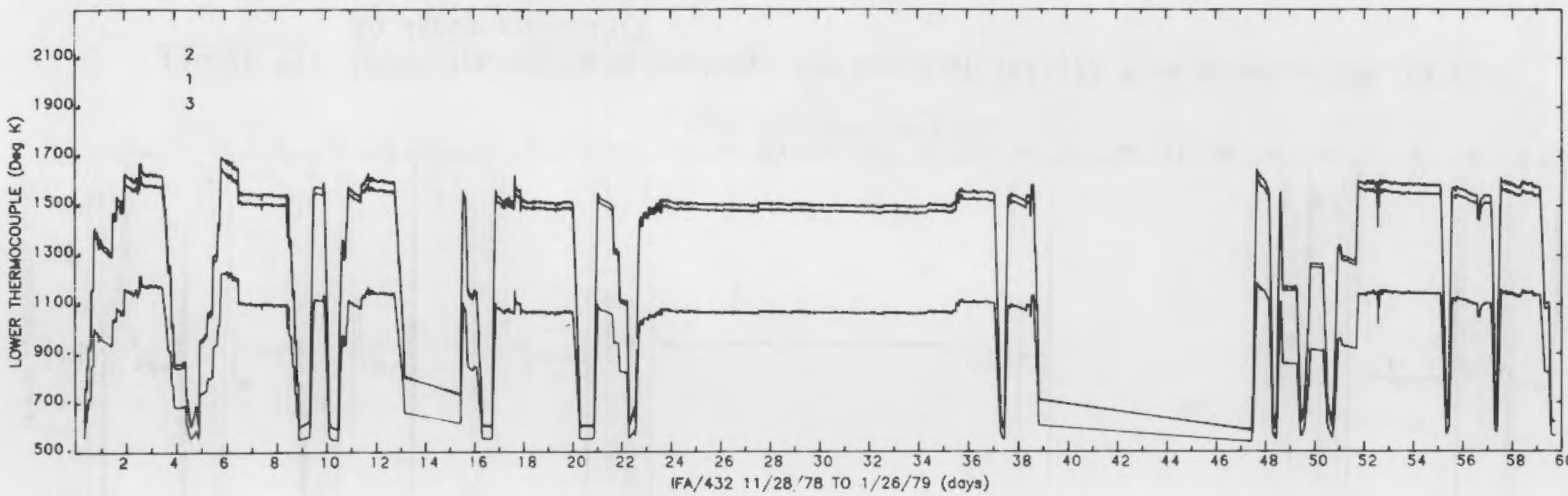


FIGURE 50. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from November 28, 1978, to January 26, 1979

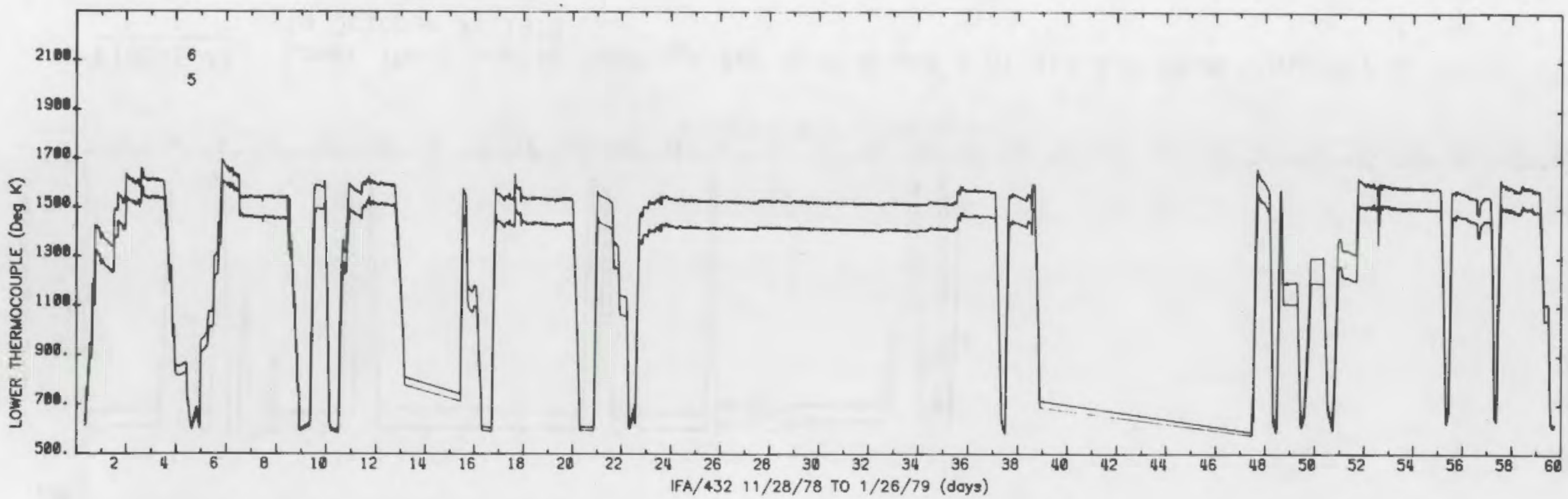


FIGURE 51. Lower Thermocouple Readings for Rods 5 and 6 of IFA-432 from November 28, 1978, to January 26, 1979

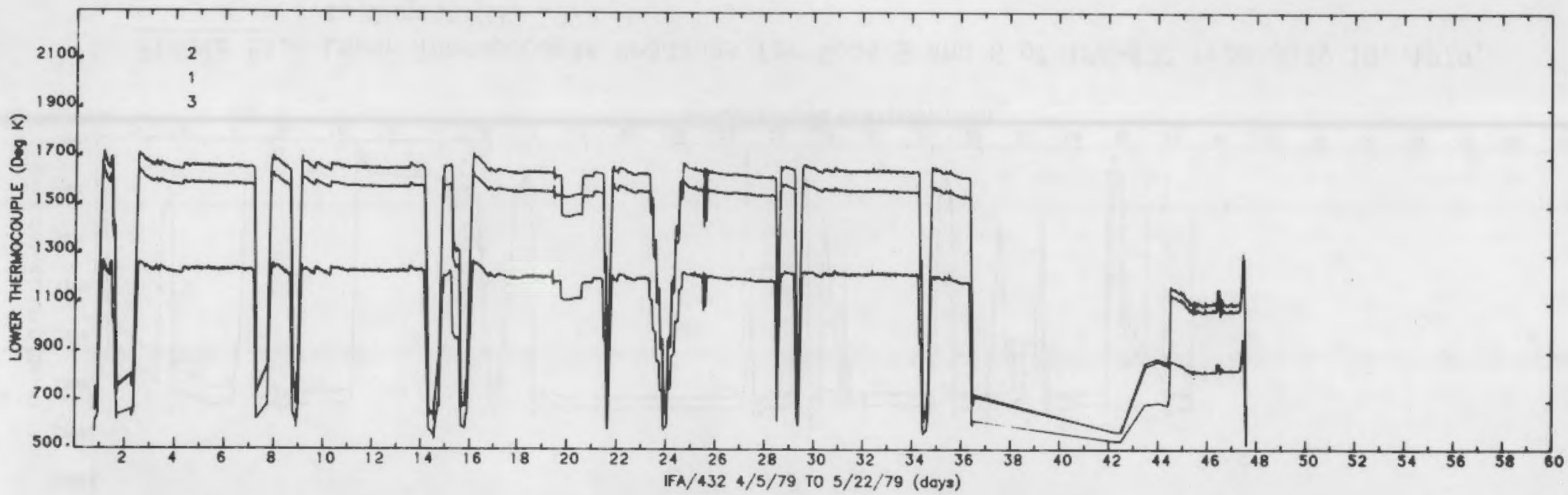


FIGURE 52. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from April 5, 1979, to May 22, 1979

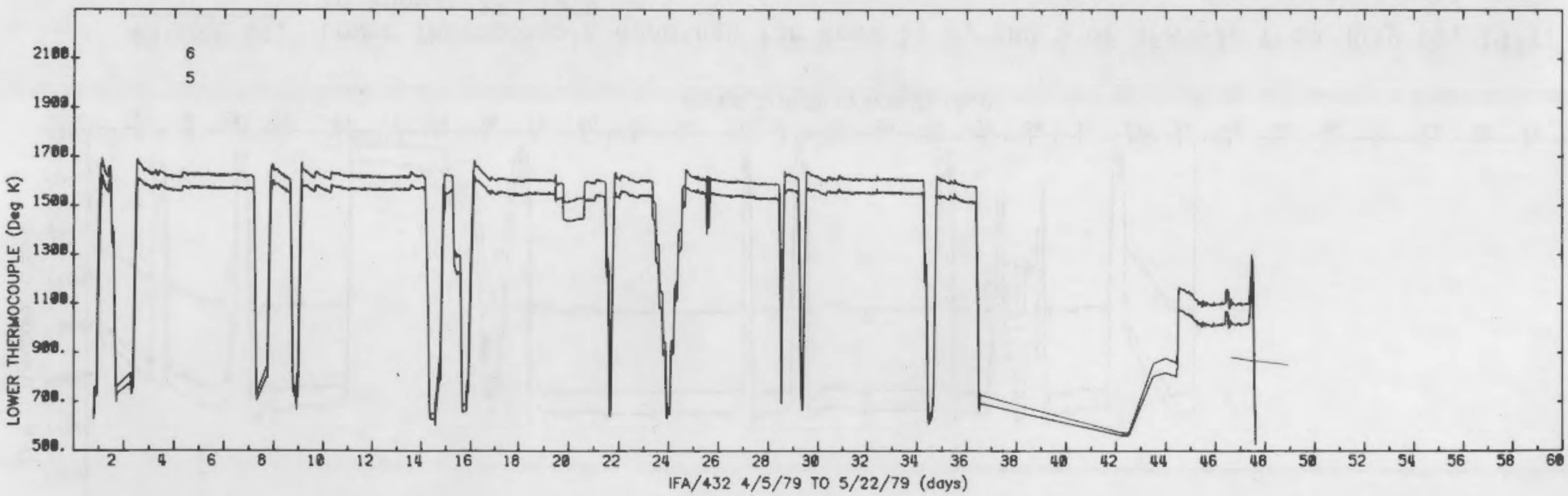


FIGURE 53. Lower Thermocouple Readings for Rods 5 and 6 of IFA-432 from April 5, 1979, to May 22, 1979

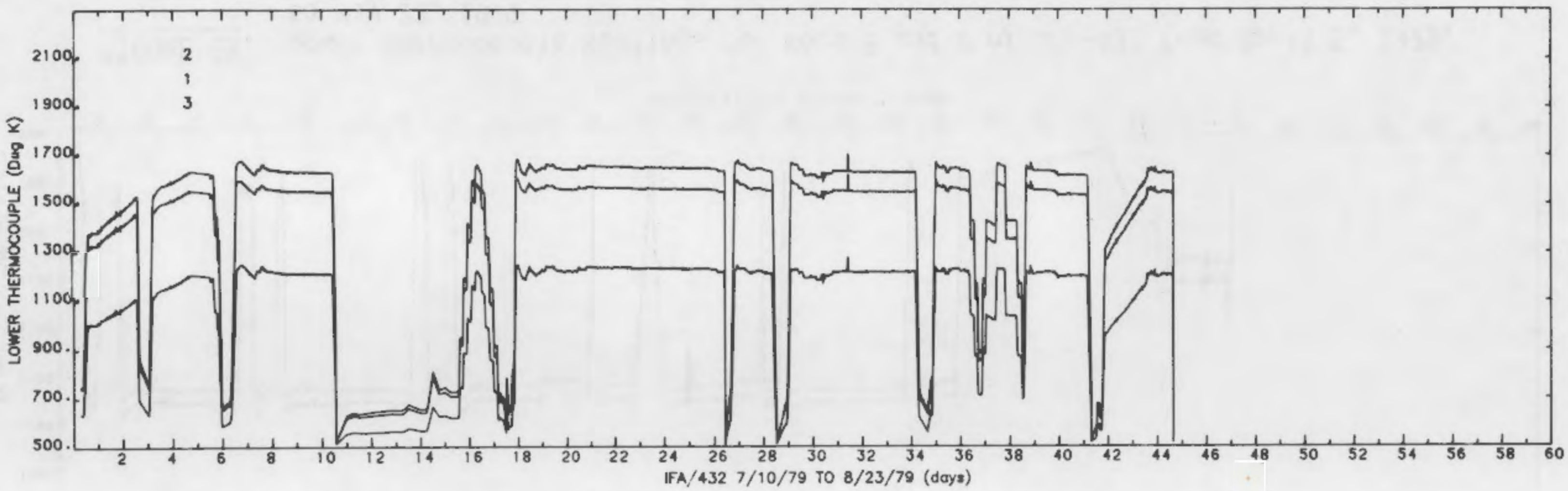


FIGURE 54. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from July 10, 1979, to August 23, 1979

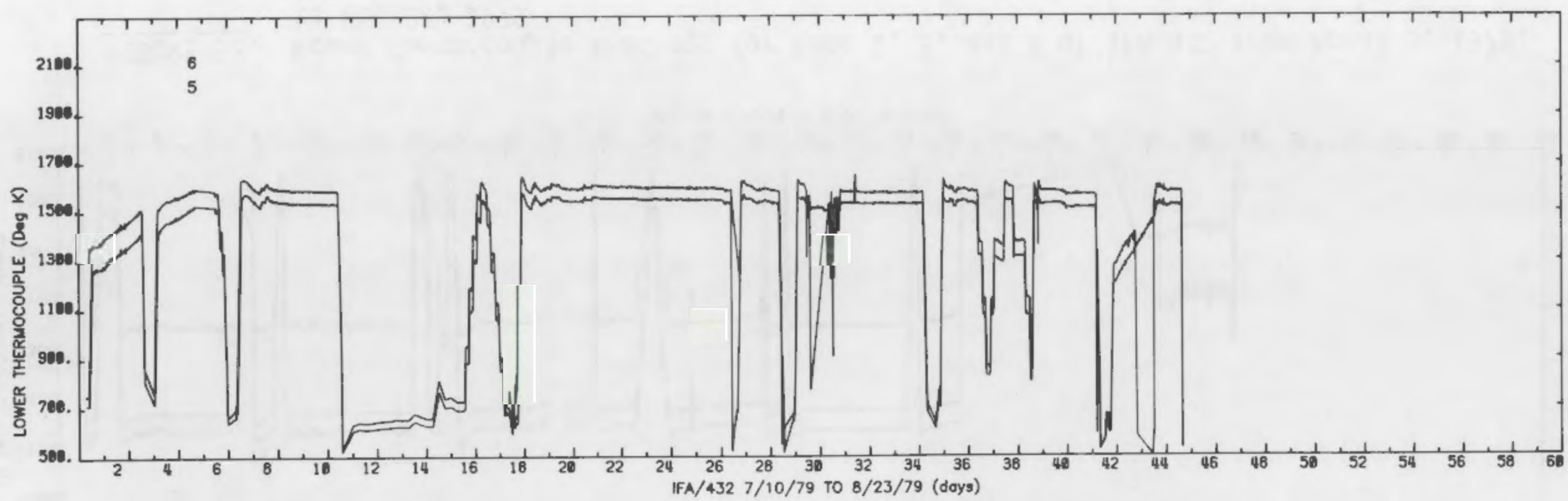


FIGURE 55. Lower Thermocouple Readings for Rods 5 and 6 of IFA-432 from July 10, 1979, to August 23, 1979

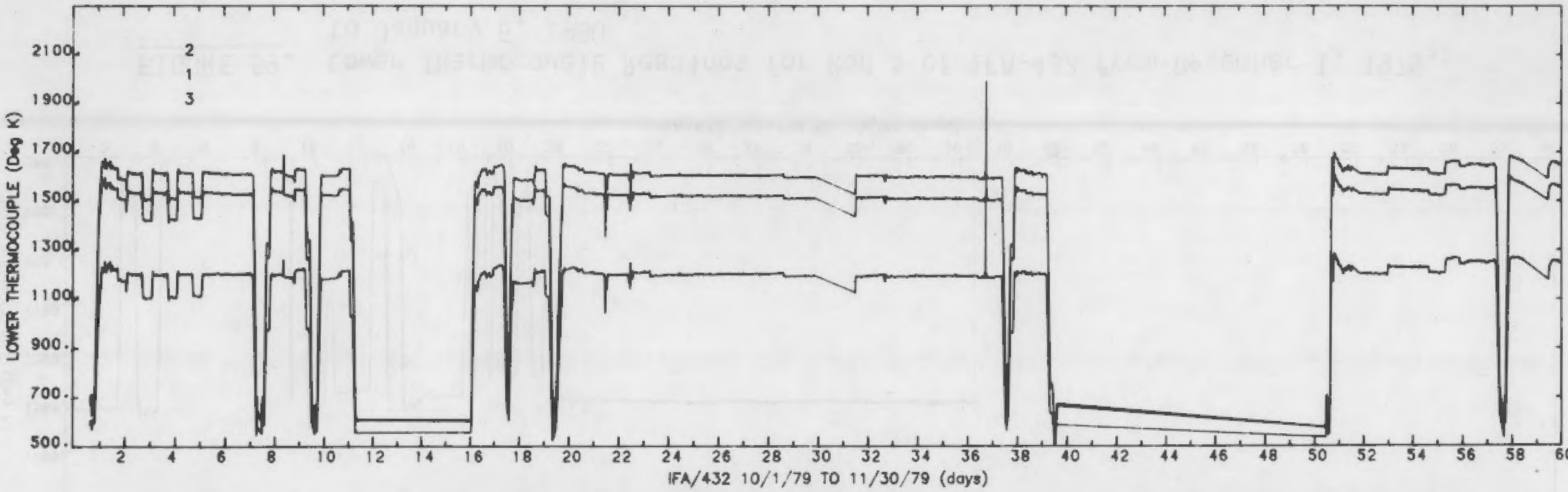


FIGURE 56. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from October 1, 1979, to November 30, 1979

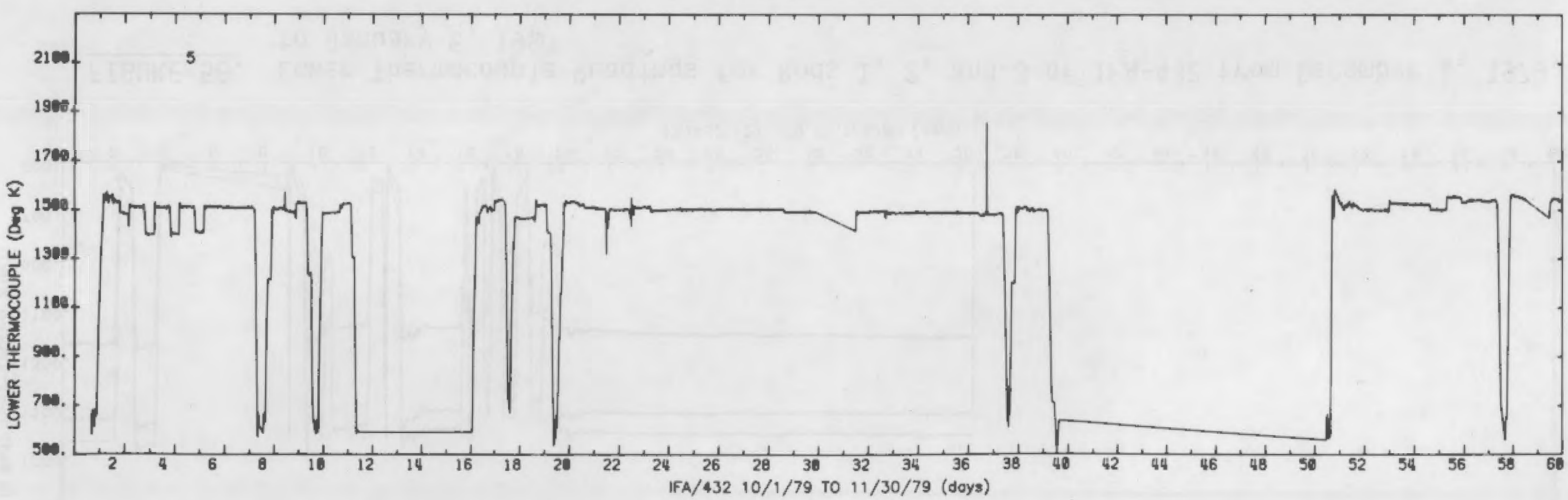


FIGURE 57. Lower Thermocouple Readings for Rod 5 of IFA-432 from October 1, 1979, to November 30, 1979

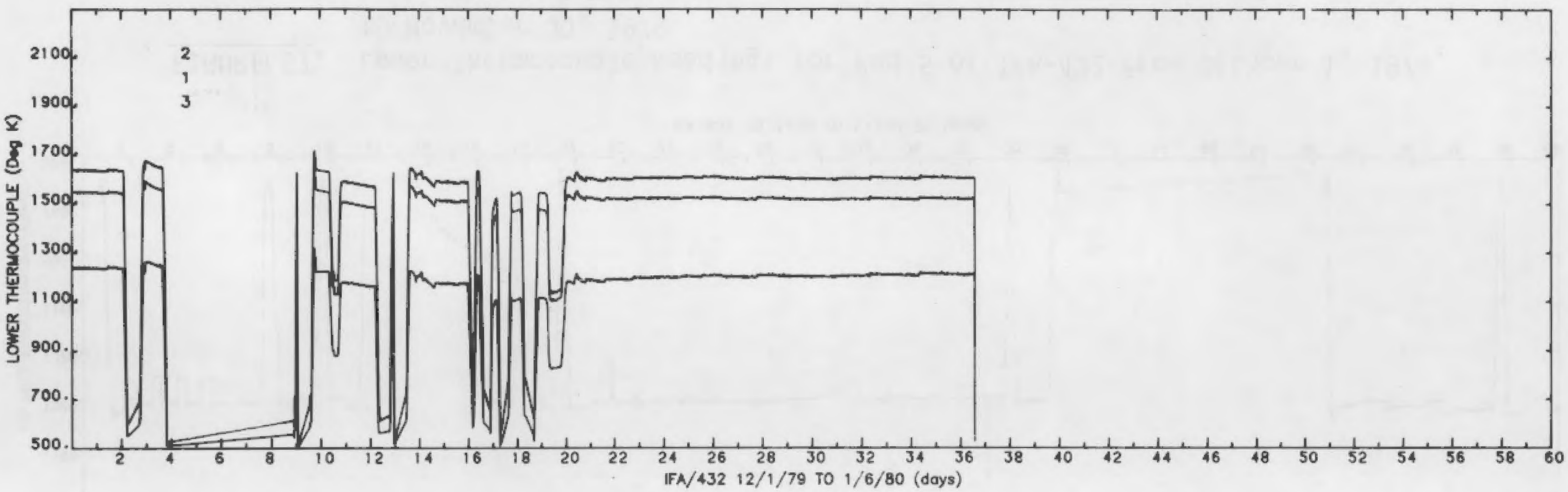


FIGURE 58. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from December 1, 1979, to January 6, 1980

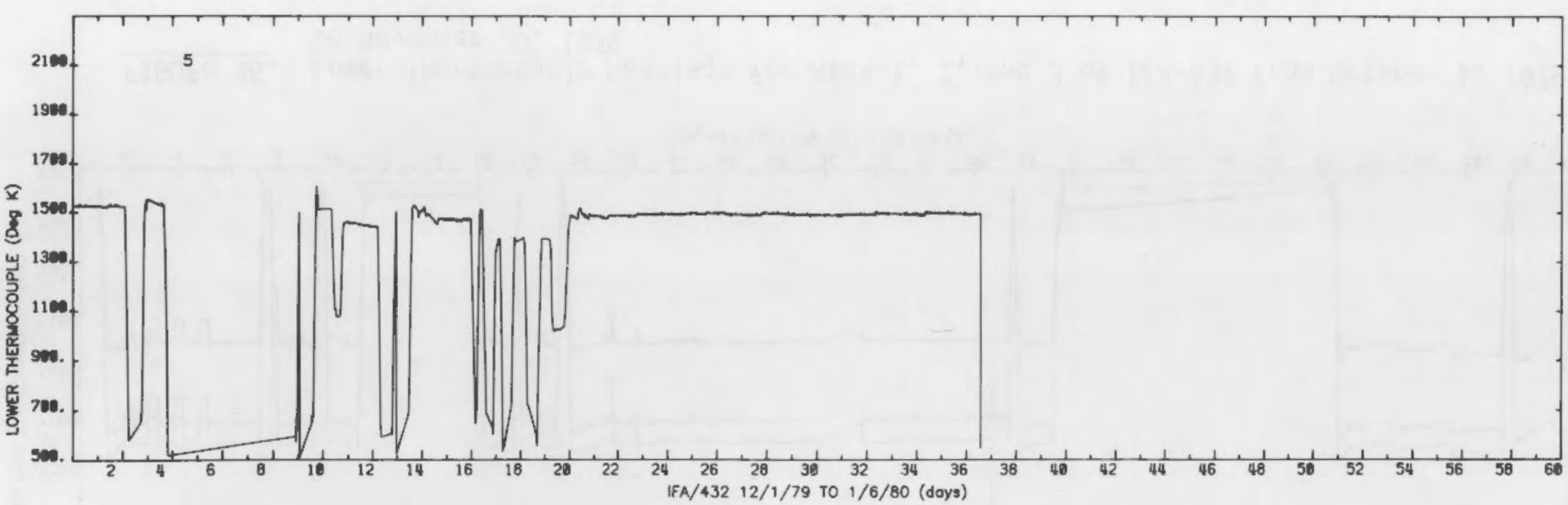


FIGURE 59. Lower Thermocouple Readings for Rod 5 of IFA-432 from December 1, 1979, to January 6, 1980

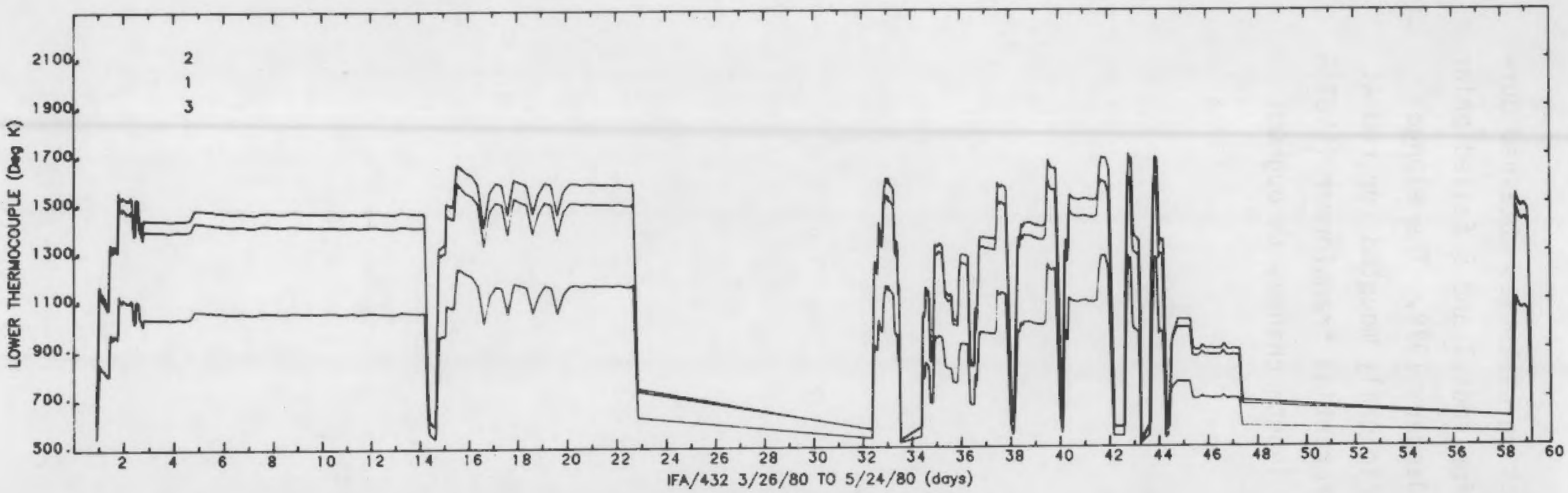


FIGURE 60. Lower Thermocouple Readings for Rods 1, 2, and 3 of IFA-432 from March 26, 1980, to May 24, 1980

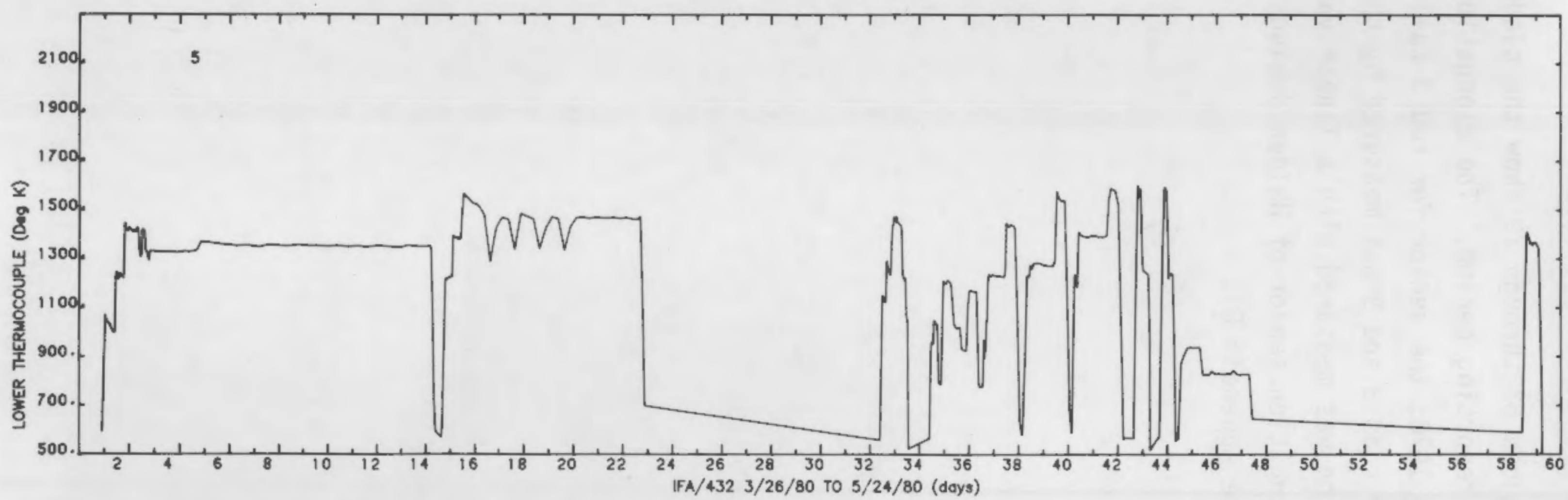


FIGURE 61. Lower Thermocouple Readings for Rod 5 of IFA-432 from March 26, 1980, to May 24, 1980

CLADDING ELONGATION HISTORIES

Figures 62 through 79 show the cladding elongation histories obtained during the reporting period. The elongation sensors for rods 1 and 5 failed prior to April 1978; the sensor for rod 3 failed during January 1979. The elongation for rods 8 and 9 was measured by the sensor originally mounted for rod 4. Elongation was measured with a linear variable differential transformer (LVDT)-type elongation sensor of Halden design to monitor length changes throughout life (see Appendix C).

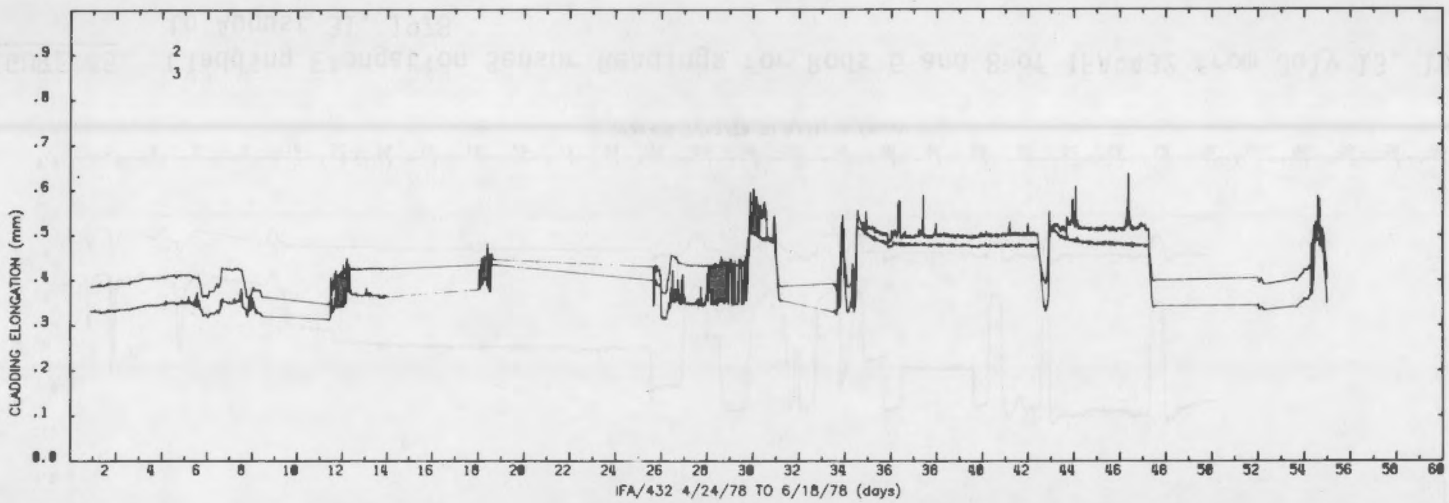


FIGURE 62. Cladding Elongation Sensor Readings for Rods 2 and 3 of IFA-432 from April 24, 1978, to June 18, 1978

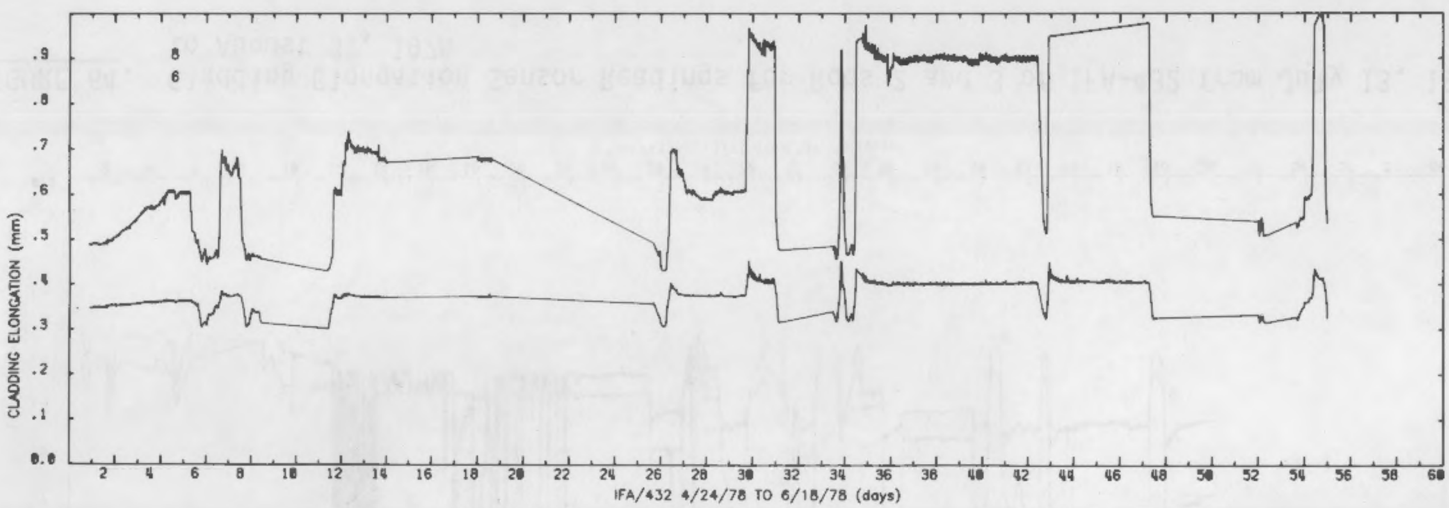


FIGURE 63. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from April 24, 1978, to June 18, 1978

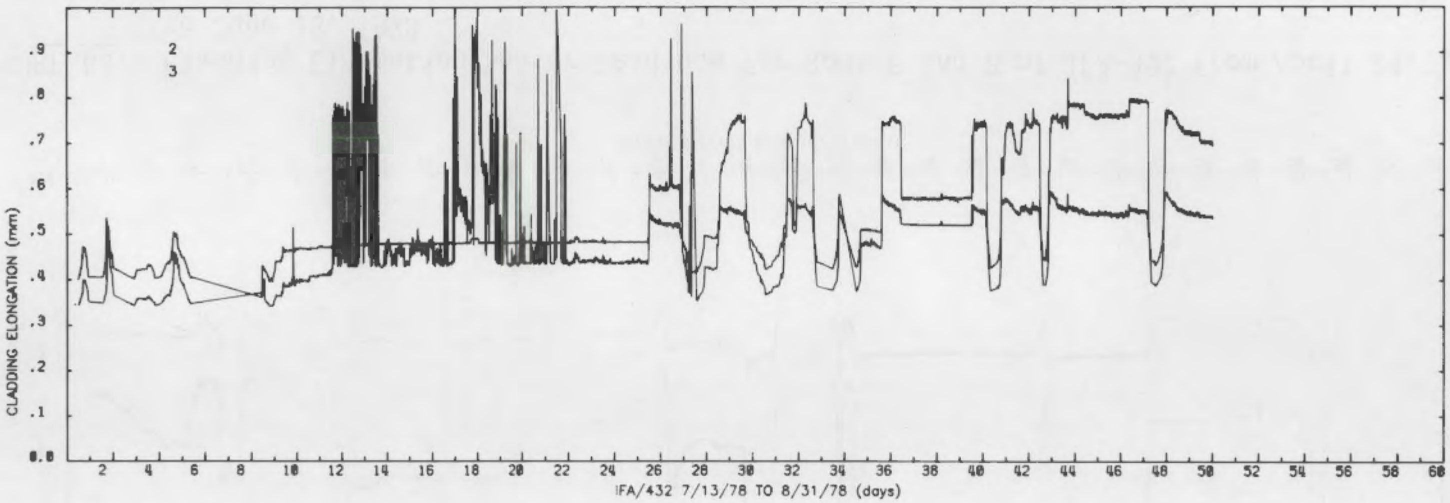


FIGURE 64. Cladding Elongation Sensor Readings for Rods 2 and 3 of IFA-432 from July 13, 1978, to August 31, 1978

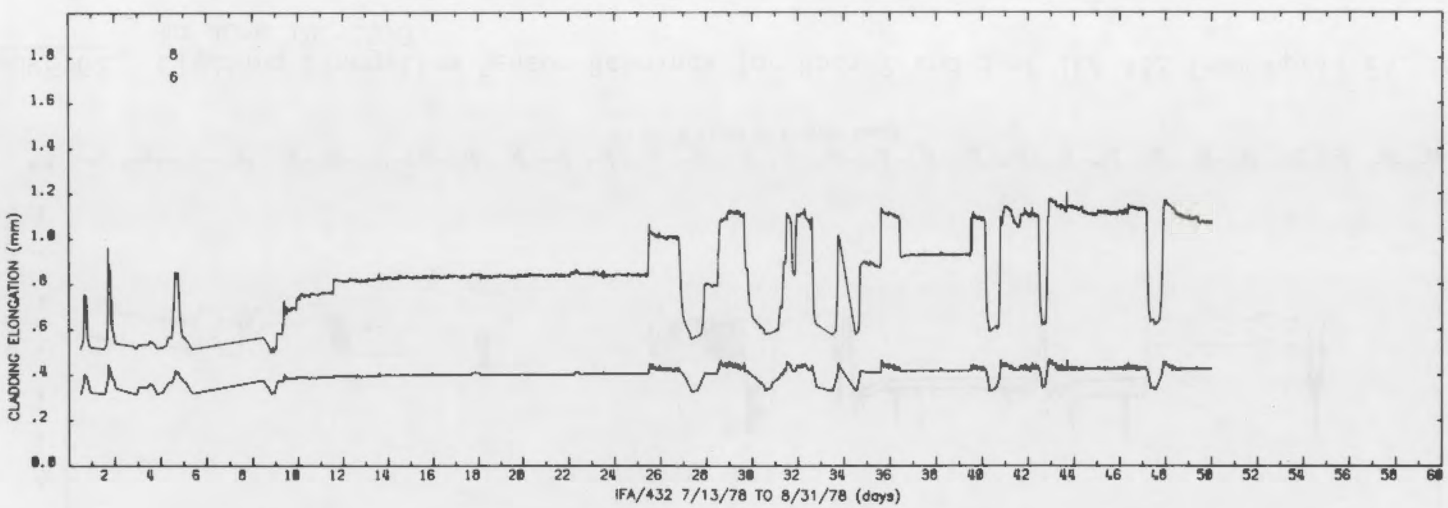


FIGURE 65. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from July 13, 1978, to August 31, 1978

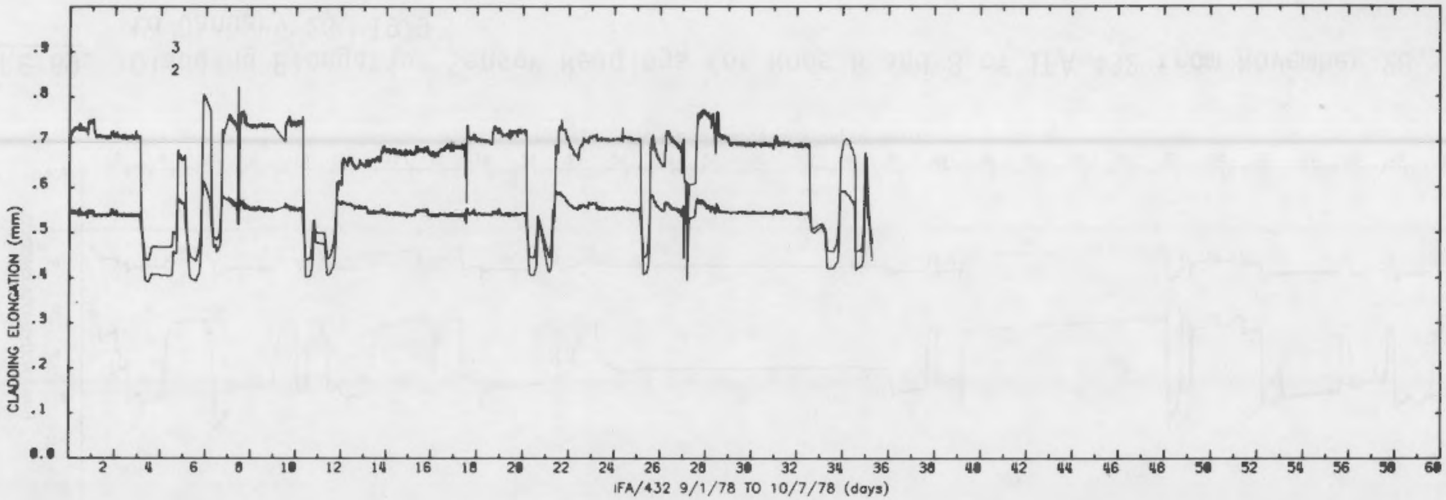


FIGURE 66. Cladding Elongation Sensor Readings for Rods 2 and 3 of IFA-432 from September 1, 1978, to October 7, 1978

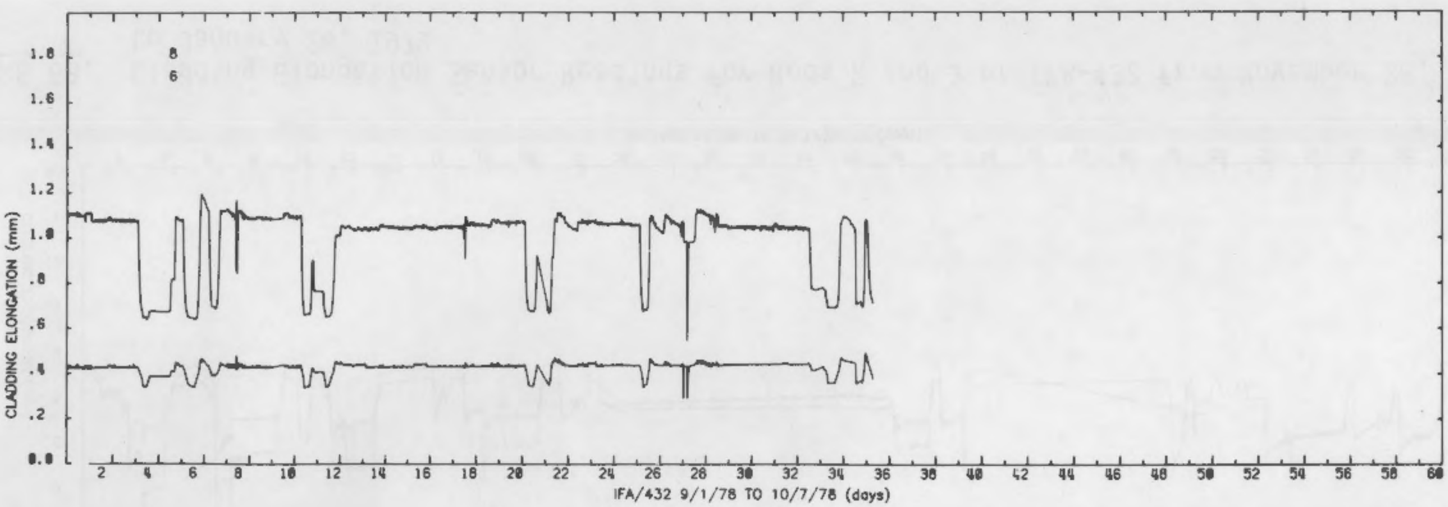


FIGURE 67. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from September 1, 1978, to October 7, 1978

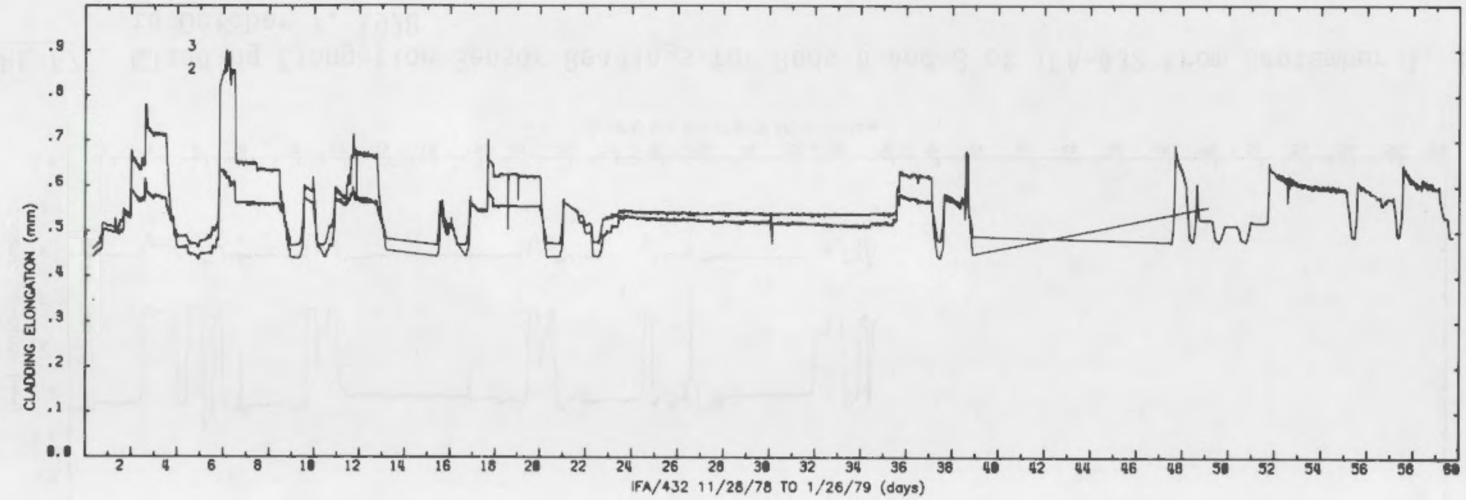


FIGURE 68. Cladding Elongation Sensor Readings for Rods 2 and 3 of IFA-432 from November 28, 1978, to January 26, 1979

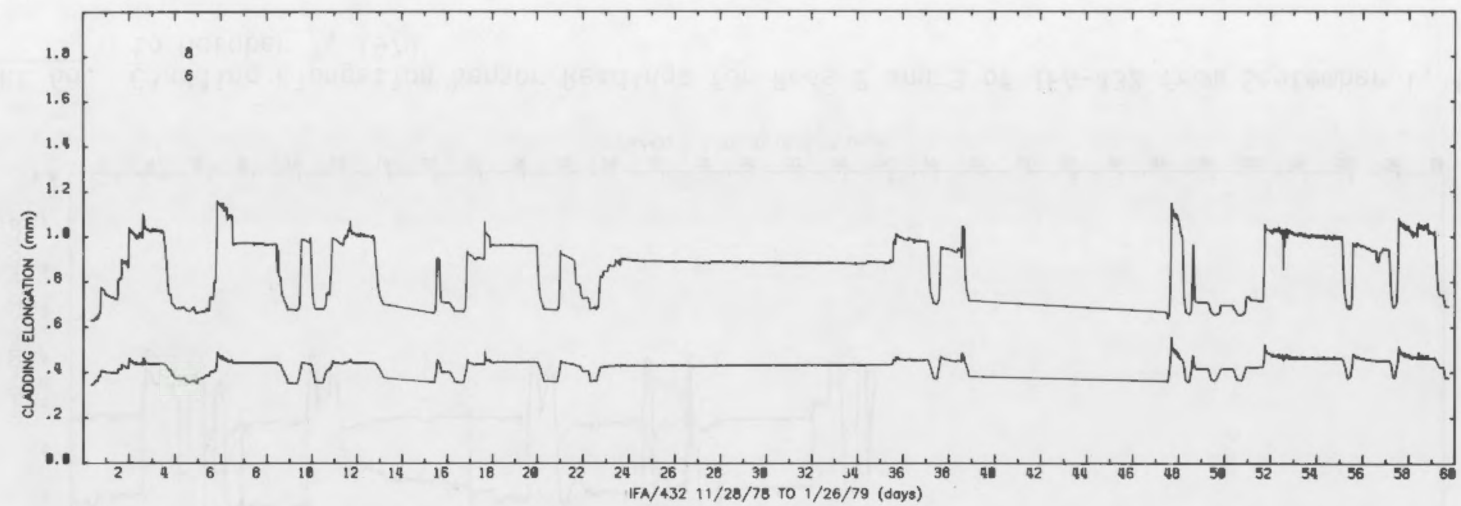


FIGURE 69. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from November 28, 1978, to January 26, 1979

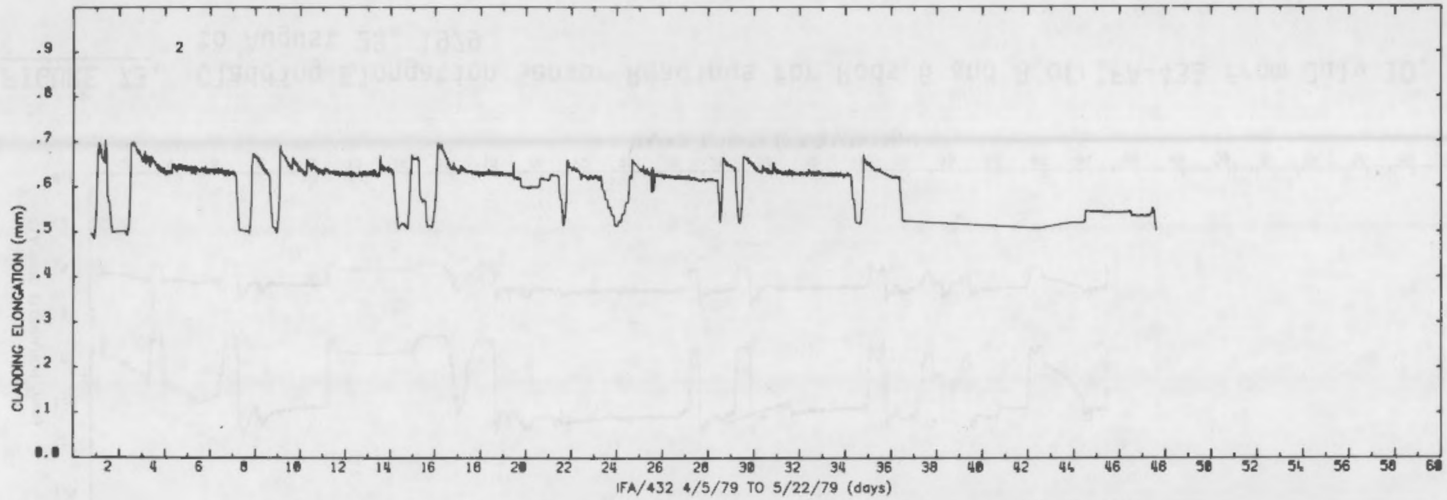


FIGURE 70. Cladding Elongation Sensor Readings for Rod 2 of IFA-432 from April 5, 1979, to May 22, 1979

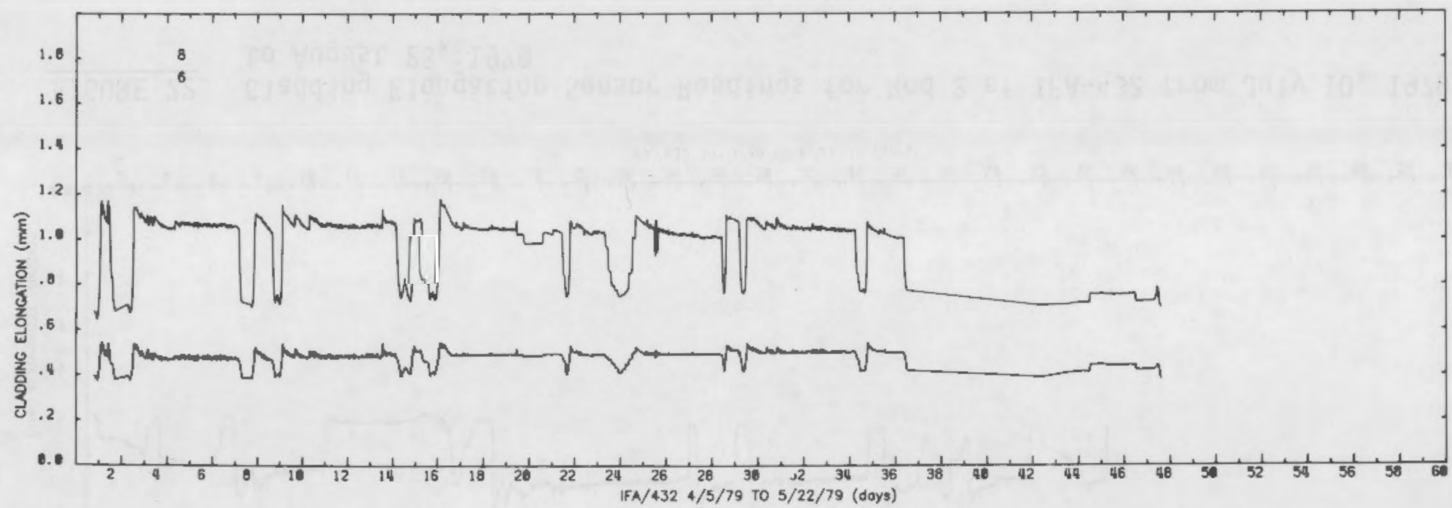


FIGURE 71. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from April 5, 1979, to May 22, 1979

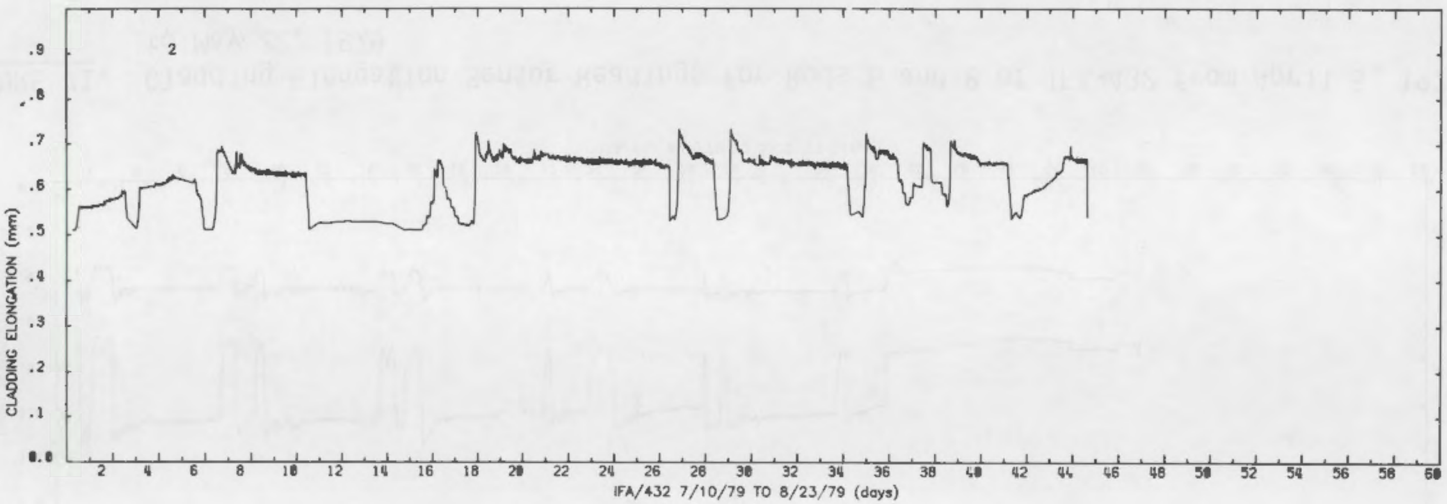


FIGURE 72. Cladding Elongation Sensor Readings for Rod 2 of IFA-432 from July 10, 1979, to August 23, 1979

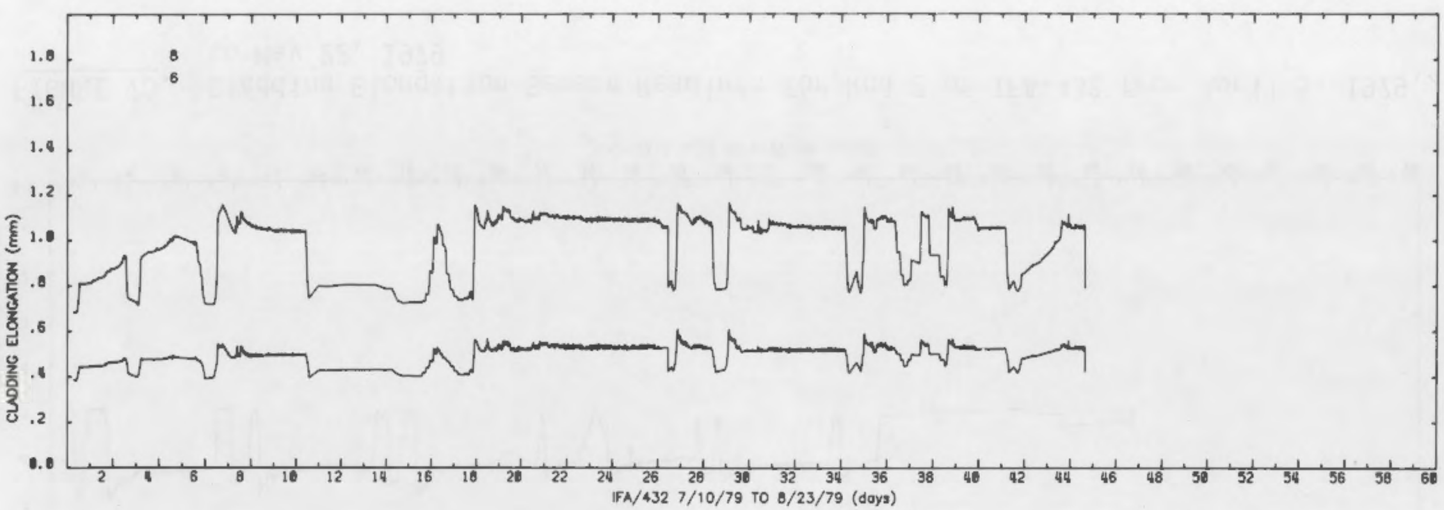


FIGURE 73. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from July 10, 1979, to August 23, 1979

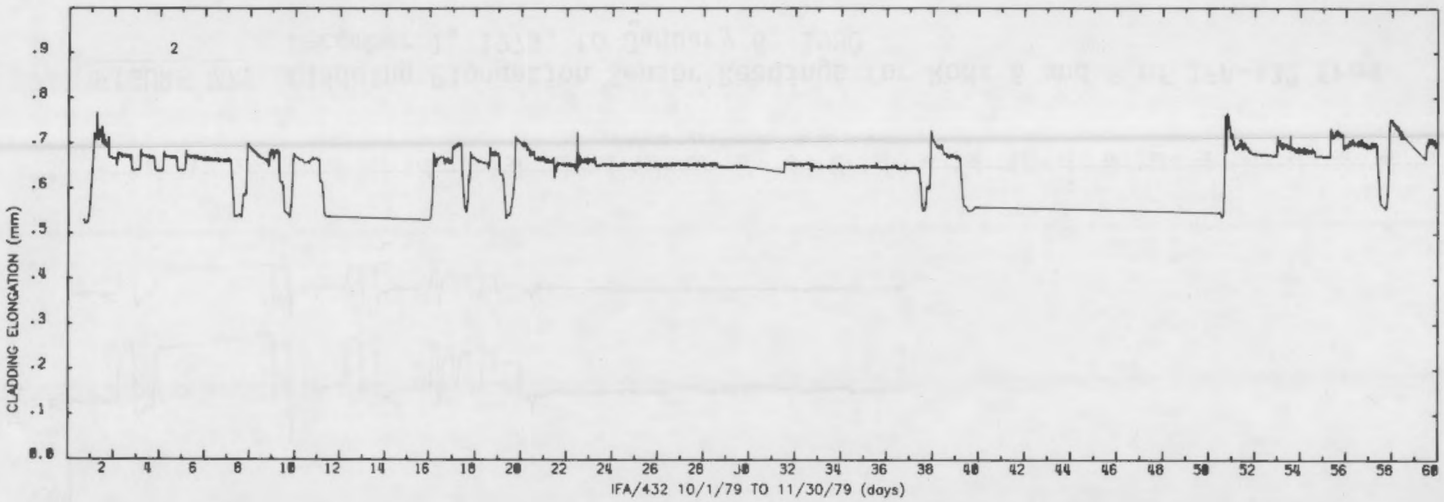


FIGURE 74. Cladding Elongation Sensor Readings for Rod 2 of IFA-432 from October 1, 1979, to November 30, 1979

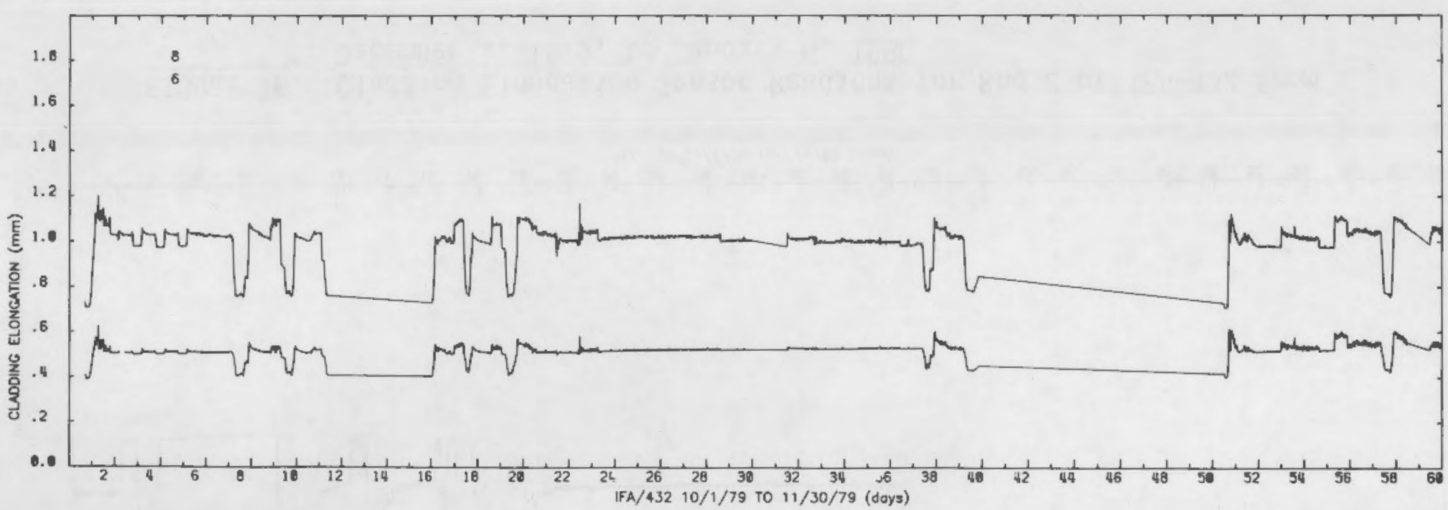
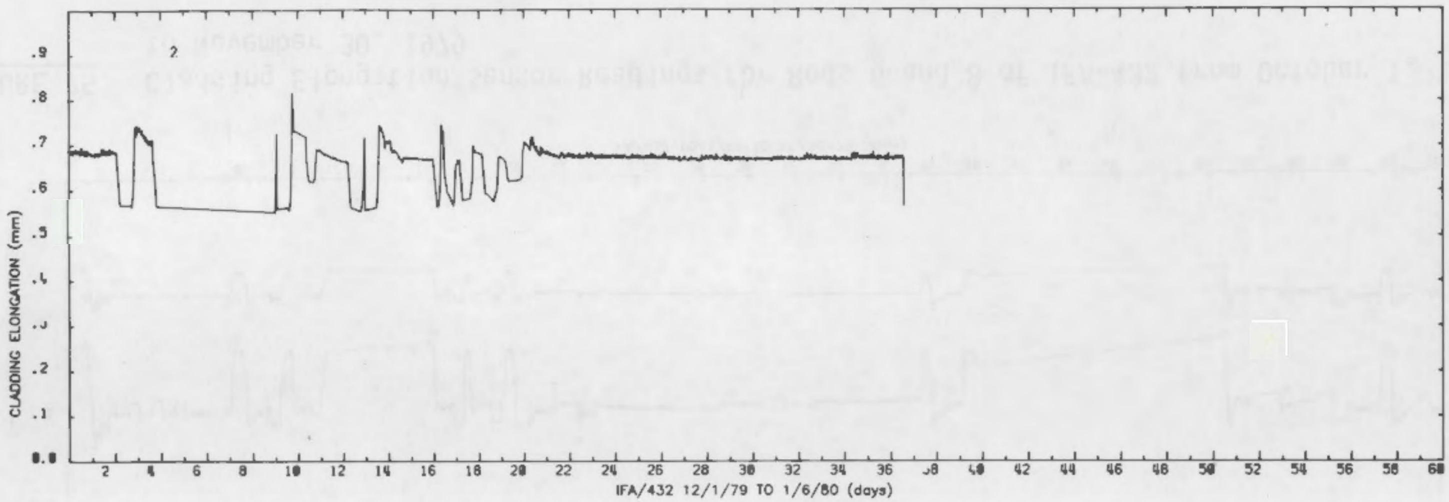
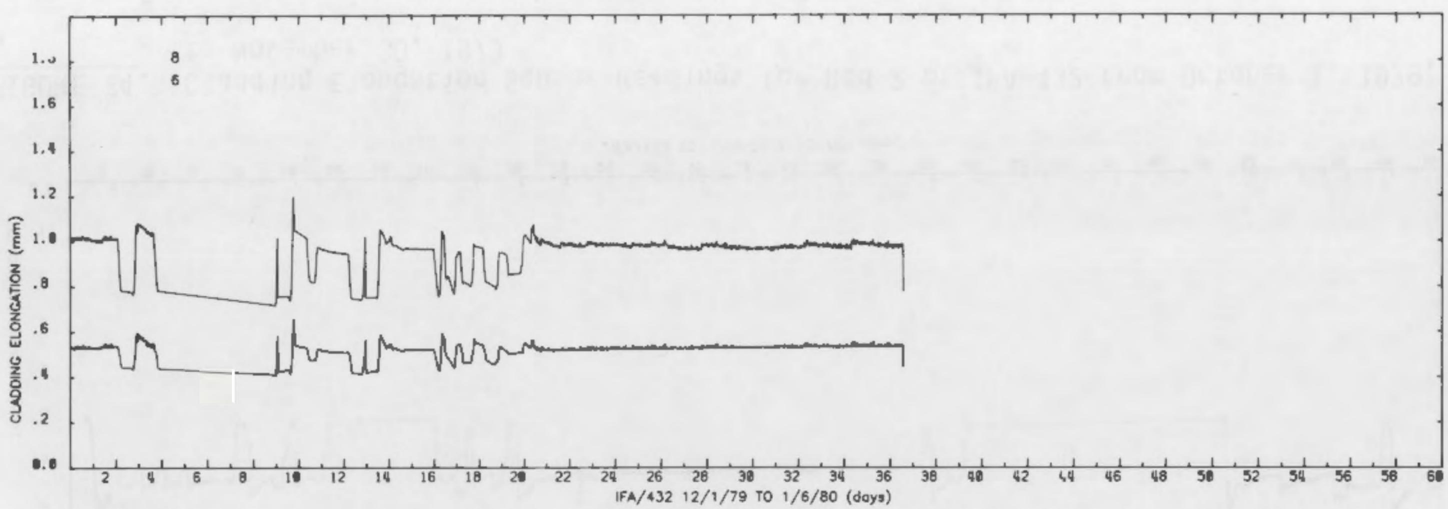


FIGURE 75. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from October 1, 1979, to November 30, 1979



<u>FIGURE 76</u>. Cladding Elongation Sensor Readings for Rod 2 of IFA-432 from December 1, 1979, to January 6, 1980



<u>FIGURE 77</u>. Cladding Elongation Sensor Readings for Rods 6 and 8 of IFA-432 from December 1, 1979, to January 6, 1980

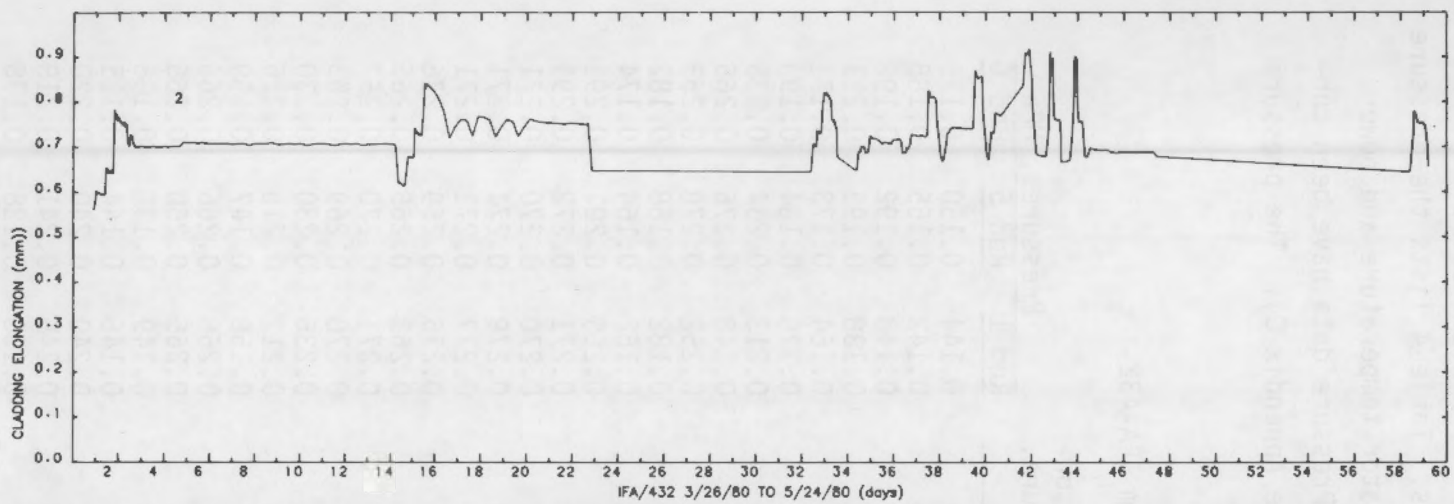


FIGURE 78. Cladding Elongation Sensor Readings for Rod 2 of IFA-432 from March 26, 1980, to May 24, 1980

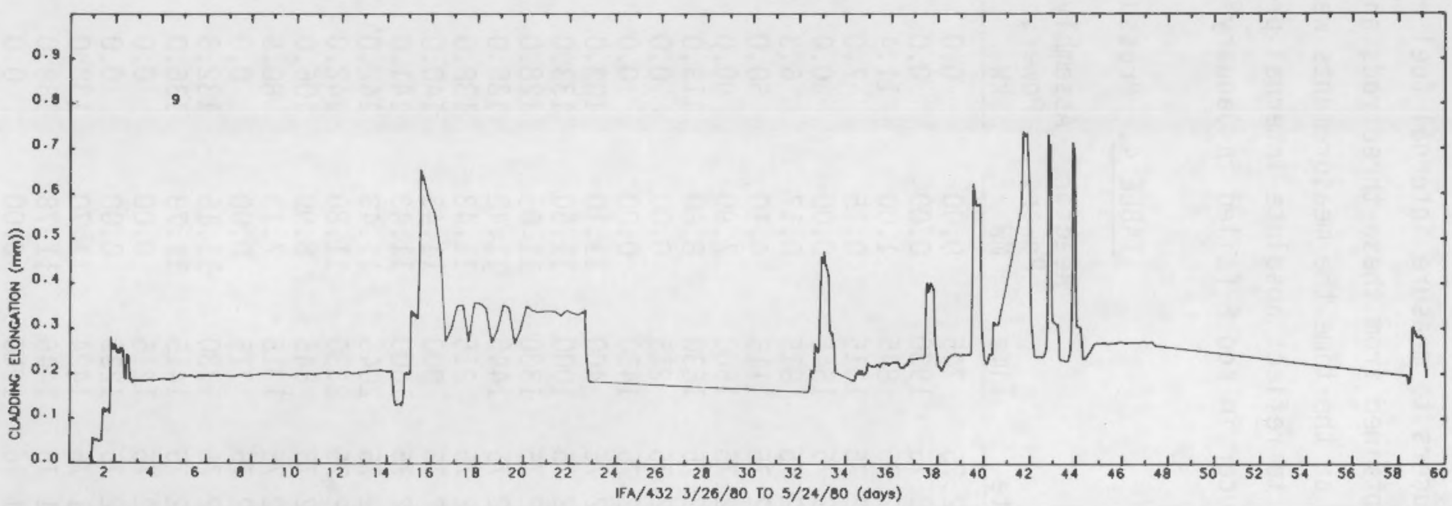


FIGURE 79. Cladding Elongation Sensor Readings for Rod 9 of IFA-432 from March 26, 1980, to May 24, 1980

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 4 lists the pressure data obtained from these three rods and the moderator temperature and power levels at the time the measurements were taken. Pressure data have been corrected to reflect absolute internal pressures (see Appendix C). The pressure transducer in rod 6 failed in January 1979.

TABLE 4. Pressure Data from IFA-432

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
1 12 75	745	0.00	0.0	150.0	0.144	0.150	0.175
1 12 75	1930	0.00	0.0	153.6	0.142	0.155	0.188
2 12 75	845	1.00	11.4	108.0	0.148	0.192	0.192
7 12 75	1315	0.15	7.0	240.0	0.138	0.193	0.211
7 12 75	1545	0.00	0.0	235.0	0.164	0.173	0.171
8 12 75	815	0.17	6.3	240.0	0.178	0.184	0.193
8 12 75	1115	4.10	50.0	238.0	0.213	0.234	0.235
8 12 75	1500	7.90	99.0	238.0	0.248	0.276	0.265
8 12 75	1630	8.80	113.0	232.0	0.252	0.278	0.253
9 12 75	215	0.00	0.0	221.0	0.182	0.168	0.182
13 12 75	1430	0.00	0.0	231.0	0.162	0.164	0.174
15 12 75	400	11.10	133.0	237.0	0.265	0.291	0.291
15 12 75	1000	11.50	133.0	237.0	0.271	0.272	0.291
15 12 75	1330	11.05	128.0	237.0	0.270	0.270	0.271
15 12 75	1445	11.42	135.0	237.0	0.279	0.271	0.271
16 12 75	215	11.42	138.0	237.0	0.277	0.271	0.271
16 12 75	900	11.43	140.0	237.0	0.275	0.269	0.276
16 12 75	1600	11.43	141.0	237.0	0.263	0.265	0.265
16 12 75	2045	11.43	142.0	237.0	0.271	0.270	0.257
16 12 75	2330	11.80	142.0	236.0	0.270	0.269	0.253
17 12 75	845	8.99	106.0	236.0	0.236	0.230	0.230
17 12 75	1315	7.13	80.5	237.0	0.217	0.216	0.216
18 12 75	715	0.00	0.0	229.0	0.158	0.147	0.159
19 12 75	730	11.15	132.3	237.0	0.266	0.266	0.266
19 12 75	1415	11.78	136.0	236.0	0.265	0.250	0.265
30 12 75	1215	0.00	0.0	222.0	0.149	0.138	0.108
30 12 75	1230	0.00	0.0	222.0	0.146	0.144	0.115
2 01 76	1424	11.70	138.0	236.0	0.240	0.240	0.220
2 01 76	1446	11.78	138.0	236.0	0.245	0.241	0.218
7 01 76	1000	0.00	0.0	222.0	0.138	0.128	0.138
7 01 76	1015	0.00	0.0	220.0	0.142	0.136	0.142
23 01 76	2130	0.00	1.1	226.0	0.128	0.119	0.108

TABLE 4. (contd)

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
24 01 76	1630	11.30	138.0	236.0	0.210	0.210	0.210
24 01 76	1645	11.20	138.0	236.0	0.210	0.210	0.210
5 02 76	1600	0.00	0.0	232.0	0.108	0.108	0.108
6 02 76	1440	9.30	113.0	235.0	0.179	0.189	0.199
13 02 76	530	7.20	84.0	235.0	0.179	0.189	0.189
15 02 76	1500	0.00	0.0	232.0	0.119	0.119	0.119
25 06 76	730	0.00	0.0	227.0	0.131	0.125	--
28 06 76	1500	12.21	140.0	240.0	0.157	0.185	0.190
6 08 76	1340	11.82	140.0	321.0	0.169	0.270	0.231
10 08 76	1615	11.50	139.0	230.0	0.199	0.311	0.271
18 08 76	730	0.00	0.0	225.0	0.128	0.210	0.199
18 08 76	800	0.00	0.0	225.0	0.103	0.180	0.172
22 09 76	1915	0.00	0.0	234.0	0.128	0.270	0.229
7 10 76	1300	0.00	0.0	227.0	0.138	0.280	0.240
21 10 76	1300	12.30	140.0	238.0	0.280	0.695	0.523
29 10 76	1015	11.82	128.0	239.0	0.237	0.688	0.517
29 10 76	1930	0.00	0.0	233.0	0.138	0.356	0.311
7 12 76	505	2.90	9.3	238.7	0.167	0.412	0.343
7 12 76	1005	4.70	41.6	240.0	0.206	0.519	0.441
7 12 76	1333	9.20	109.2	240.1	0.265	0.686	0.510
8 12 76	2004	3.00	13.1	239.5	0.147	0.451	0.353
9 12 76	1938	11.60	123.5	240.4	0.274	0.715	0.539
10 12 76	1933	12.50	146.4	239.9	0.274	0.735	0.559
3 01 77	1434	12.40	144.0	240.1	0.304	0.843	0.706
4 01 77	626	0.00	0.0	222.7	0.157	0.402	0.372
17 01 77	1033	11.80	135.9	239.7	0.314	0.862	0.755
19 01 77	1038	3.00	33.9	237.0	0.186	0.637	0.578
19 01 77	1511	8.50	96.3	240.4	0.216	0.823	0.735
27 01 77	930	11.90	144.1	239.3	0.333	0.921	0.843
4 02 77	1324	4.10	66.3	235.0	0.284	0.794	0.735
9 02 77	1023	11.10	139.9	239.9	0.363	0.804	0.892
25 03 77	1628	1.50	4.0	203.1	0.216	0.500	0.480
26 03 77	1233	6.50	54.8	239.3	0.314	0.784	0.715
26 03 77	1728	12.00	116.6	239.7	0.314	0.951	0.862
10 04 77	1034	11.70	136.6	239.4	0.392	0.862	0.970
14 04 77	1458	11.50	118.6	239.2	0.382	0.892	0.970
18 04 77	1437	12.20	143.2	239.0	0.421	0.931	0.882
3 05 77	1313	6.20	85.8	238.7	0.441	0.931	0.902
11 05 77	1331	12.00	138.3	238.9	0.568	1.205	1.117
16 05 77	1354	5.30	77.8	239.1	0.529	1.107	1.058
19 05 77	2308	12.50	143.1	239.3	0.627	1.313	1.245
19 05 77	2334	12.40	143.2	239.3	0.637	1.303	1.254
20 05 77	1300	12.40	143.0	239.5	0.627	1.303	1.245
20 05 77	1500	12.30	144.0	239.4	0.637	1.303	1.235
21 05 77	28	0.00	0.0	199.6	0.314	0.637	0.657
21 05 77	218	0.00	0.0	198.9	0.294	0.647	0.657

TABLE 4. (contd)

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
21 05 77	1840	0.00	0.0	68.8	0.186	0.441	0.421
21 05 77	2056	0.00	0.0	70.4	0.176	0.431	0.392
22 05 77	1950	0.00	0.0	70.6	0.245	0.461	0.500
16 06 77	1931	0.00	0.0	79.0	0.245	0.421	0.441
19 06 77	2250	0.00	0.0	183.3	0.255	0.568	0.588
22 06 77	1556	0.00	0.0	73.0	0.235	0.421	0.421
23 06 77	555	0.00	0.0	153.1	0.284	0.519	0.519
24 06 77	921	2.90	2.1	215.7	0.412	0.862	0.853
24 06 77	1543	0.00	0.0	213.5	0.333	0.637	0.647
28 06 77	2234	2.50	21.5	200.2	0.372	0.853	0.833
29 06 77	953	3.50	21.8	238.6	0.451	0.951	0.941
29 06 77	1510	6.00	91.6	238.5	0.559	1.088	1.019
7 07 77	1209	0.00	0.0	145.2	0.274	0.529	0.549
8 07 77	1954	4.60	46.0	225.2	0.480	0.882	0.862
9 07 77	54	11.20	127.2	235.6	0.647	1.186	1.156
9 07 77	1543	11.50	139.7	237.7	0.627	1.235	1.196
11 07 77	1151	12.10	138.0	238.4	0.647	1.245	1.225
13 07 77	1204	12.40	159.6	234.8	0.627	1.245	1.225
15 07 77	1309	0.00	0.0	206.5	0.265	0.627	0.657
1 08 77	1046	12.00	160.1	234.6	0.745	1.303	1.372
3 08 77	1020	12.30	162.8	233.9	0.774	1.343	1.421
3 08 77	1420	0.00	0.0	216.1	0.392	0.706	0.784
3 08 77	1816	1.60	9.1	211.6	0.431	0.804	0.872
5 08 77	1428	0.00	0.0	215.1	0.392	0.696	0.794
17 08 77	1415	12.10	158.6	234.0	0.872	1.382	1.509
10 08 77	1026	12.00	161.3	233.9	0.882	1.352	1.490
22 08 77	1200	12.00	160.4	234.0	0.892	1.372	1.568
22 08 77	1314	11.90	161.0	233.8	0.902	1.362	1.548
22 08 77	1410	11.90	160.8	234.0	0.902	1.382	1.539
26 08 77	954	12.00	164.2	234.0	0.931	1.392	1.588
26 08 77	1607	2.50	42.9	220.6	0.706	0.970	1.147
26 08 77	1659	1.80	15.7	220.1	0.598	0.951	0.960
14 10 77	1622	9.20	106.7	238.4	0.862	1.352	1.568
15 10 77	1434	10.00	128.7	238.4	0.911	1.441	1.646
16 10 77	1443	11.80	138.4	238.2	0.921	1.460	1.676
16 10 77	1508	11.90	133.8	238.1	0.666	1.156	1.480
17 10 77	752	12.00	131.6	238.5	0.911	1.421	1.646
17 10 77	1449	4.00	25.0	233.4	0.617	0.941	1.127
18 10 77	1208	12.70	0.7	234.3	0.549	0.833	0.990
19 10 77	2038	3.90	16.6	237.8	0.598	0.931	1.098
19 10 77	2101	0.00	6.3	236.8	0.480	0.725	0.911
24 10 77	937	11.40	109.2	238.6	0.882	1.372	1.597
29 10 77	13	10.70	118.8	239.0	0.902	1.392	1.646
31 10 77	808	15.20	170.5	238.6	0.941	1.519	1.784
31 10 77	2028	15.30	174.4	238.8	0.960	1.529	1.803
1 11 77	745	1.70	4.3	238.0	0.559	0.794	1.000

TABLE 4. (contd)

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
1 11 77	832	1.70	0.6	236.0	0.500	0.715	0.941
4 11 77	853	1.60	3.0	237.5	0.529	0.764	0.970
4 11 77	902	1.70	3.2	237.7	0.529	0.784	0.970
4 11 77	911	1.70	3.3	237.3	0.529	0.794	0.970
4 11 77	919	1.60	3.6	237.1	0.539	0.784	0.980
4 11 77	926	1.60	3.7	237.2	0.529	0.794	0.980
4 11 77	934	1.70	3.7	237.2	0.539	0.784	0.980
7 12 77	1312	0.00	0.1	159.4	0.431	0.588	0.745
7 12 77	2224	0.00	0.2	180.8	0.421	0.627	0.804
10 12 77	831	2.40	5.7	238.7	0.559	0.804	0.911
10 12 77	916	2.20	13.6	206.2	0.559	0.823	0.872
10 12 77	924	2.20	14.2	207.5	0.559	0.853	0.872
11 12 77	2105	12.40	152.4	239.2	0.862	1.411	1.666
13 12 77	2033	2.60	8.1	238.7	0.568	0.833	0.941
13 12 77	2044	2.60	7.9	238.9	0.559	0.843	0.951
13 12 77	2056	2.60	7.5	238.8	0.568	0.823	0.951
16 12 77	904	2.50	5.9	238.2	0.549	0.794	0.921
17 12 77	859	11.60	137.8	238.7	0.960	1.382	1.646
21 12 77	1022	11.90	137.4	238.6	0.941	1.333	1.646
21 12 77	1220	11.90	137.4	238.6	0.960	1.372	1.646
4 01 78	1013	11.30	156.3	238.9	0.951	1.372	1.735
5 01 78	935	11.20	155.8	238.7	0.882	1.372	1.715
5 01 78	954	11.30	155.8	238.8	0.970	1.382	1.735
5 01 78	1050	11.30	153.8	238.8	0.970	1.372	1.725
5 01 78	1117	11.40	153.5	230.0	0.970	1.382	1.725
5 01 78	1151	11.40	153.7	239.0	0.970	1.392	1.725
5 01 78	1211	11.30	153.7	238.9	0.872	1.382	1.725
5 01 78	1224	11.20	153.5	238.7	0.970	1.382	1.735
7 01 78	931	11.60	155.6	239.0	0.872	1.372	1.725
11 01 78	848	1.50	7.3	238.5	0.578	0.823	0.970
12 01 78	1313	11.60	158.0	239.4	0.882	1.372	1.754
12 01 78	1331	11.40	154.6	239.2	0.882	1.382	1.744
13 01 78	1452	1.70	8.5	238.6	0.568	0.861	1.000
13 01 78	1507	1.70	8.5	238.6	0.568	0.843	1.000
7 07 78	1630	0.00	0.4	72.4	0.402	0.490	0.608
7 07 78	1638	0.00	0.3	72.4	0.372	0.490	0.588
7 07 78	1644	0.00	0.3	72.2	0.372	0.470	0.598
7 07 78	1651	0.00	0.3	72.4	0.372	0.470	0.617
10 07 78	911	0.00	0.2	200.2	0.557	0.706	0.931
10 07 78	918	0.00	0.1	200.3	0.578	0.706	0.931
10 07 78	924	0.00	0.2	200.1	0.578	0.725	0.931
10 07 78	931	0.00	0.1	200.3	0.568	0.813	0.902
10 07 78	938	0.00	0.2	200.1	0.568	0.706	0.931
10 07 78	2032	2.10	5.0	239.9	0.647	0.853	1.117
10 07 78	2030	2.10	4.8	239.5	0.617	0.843	1.049
10 07 78	2039	2.10	5.2	239.4	0.608	0.833	1.049

TABLE 4. (contd)

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
10 07 78	2050	2.00	5.1	239.1	0.617	0.872	1.049
13 07 78	1250	3.30	11.8	239.9	0.706	0.960	1.235
13 07 78	1324	4.30	22.2	238.4	0.784	1.068	1.372
13 07 78	1407	6.20	36.2	239.4	0.892	1.205	1.529
13 07 78	1430	7.20	43.0	239.2	0.960	1.254	1.588
13 07 78	1457	8.20	49.6	238.5	0.951	1.294	1.637
13 07 78	1511	8.70	52.3	238.9	0.970	1.313	1.676
13 07 78	1531	9.40	56.9	239.0	0.970	1.382	1.695
13 07 78	1558	10.30	63.7	238.9	1.000	1.372	1.754
13 07 78	1619	11.00	71.1	239.4	1.029	1.421	1.793
13 07 78	1640	11.90	78.1	239.5	1.049	1.460	1.842
24 07 78	945	12.30	83.1	239.9	1.049	1.490	1.852
1 08 78	1355	12.50	87.4	235.5	1.058	1.509	1.891
4 08 78	1450	12.70	88.8	235.7	0.980	1.519	1.891
4 08 78	1505	12.70	88.5	235.8	1.078	1.519	1.911
4 08 78	1512	12.70	88.3	235.8	1.058	1.529	1.911
7 08 78	1822	12.30	97.0	235.7	1.088	1.519	1.950
7 08 78	1829	12.40	96.8	235.9	1.078	1.539	--
9 08 78	747	2.00	5.5	239.7	0.598	0.833	1.186
9 08 78	755	2.00	4.5	239.5	0.627	0.833	1.127
9 08 78	1059	1.50	0.2	229.9	0.559	0.745	1.098
14 08 78	1107	12.60	100.9	235.0	1.117	1.597	1.999
14 08 78	1115	12.60	100.6	235.0	1.156	1.588	1.989
15 08 78	1440	5.20	44.7	235.3	0.921	1.294	1.666
22 08 78	1634	3.90	48.7	234.9	0.951	1.303	1.686
22 08 78	1650	4.70	50.3	235.2	0.960	1.333	1.715
22 08 78	1706	6.30	56.2	235.3	1.019	1.372	1.784
28 08 78	746	12.30	101.7	233.7	1.254	1.539	2.019
28 08 78	758	12.40	101.6	233.5	1.147	1.578	2.019
31 08 78	751	12.60	100.3	235.1	1.176	1.568	2.038
3 09 78	2119	12.20	99.7	235.1	1.186	1.568	2.048
3 09 78	2138	12.10	99.7	235.3	1.186	1.558	2.048
3 09 78	2144	12.20	99.8	235.3	1.176	1.539	2.019
4 09 78	743	1.70	4.1	234.6	0.676	0.843	1.147
4 09 78	750	1.70	3.1	234.6	0.696	0.853	1.147
8 09 78	1048	12.40	96.8	235.5	1.186	1.558	1.999
13 09 78	759	11.20	98.3	235.0	1.196	1.548	1.989
13 09 78	808	11.60	98.3	234.9	1.196	1.558	2.009
15 09 78	753	11.90	99.7	235.1	1.196	1.558	1.999
20 09 78	758	11.80	100.9	235.2	1.196	1.519	1.999
20 09 78	808	11.80	100.9	234.9	1.205	1.539	1.999
25 09 78	755	11.70	101.0	235.0	1.215	1.539	1.960
26 09 78	923	1.30	0.4	227.4	0.676	0.784	1.039
26 09 78	930	1.30	0.4	227.8	0.666	0.833	1.039
3 10 78	749	11.90	90.0	235.2	1.225	1.539	2.058
3 10 78	756	11.90	90.0	235.3	1.235	1.548	2.038

TABLE 4. (contd)

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
5 10 78	803	10.50	96.5	235.1	1.254	1.597	2.058
23 11 78	1636	0.00	0.0	71.7	0.529	0.598	0.725
23 11 78	1649	0.00	0.0	72.1	0.529	0.627	0.725
27 11 78	838	2.70	7.1	239.5	0.637	0.755	0.784
27 11 78	1244	2.80	7.0	238.7	0.794	0.715	0.921
28 11 78	745	2.60	6.0	238.8	0.902	0.833	1.117
28 11 78	2056	8.90	65.9	239.9	1.186	1.480	1.891
30 11 78	1027	13.10	89.0	240.2	1.264	1.597	1.960
30 11 78	1115	13.30	90.8	240.3	1.313	1.656	2.038
30 11 78	1404	13.00	89.6	240.4	1.264	1.646	2.029
3 12 78	1020	6.10	37.8	238.1	1.009	1.254	1.578
4 12 78	1029	13.60	96.1	240.3	1.284	1.656	2.078
7 12 78	745	2.20	5.7	239.9	0.872	0.833	1.117
9 12 78	1719	13.00	90.5	239.9	1.264	1.607	2.019
14 12 78	754	1.70	4.4	239.5	0.862	0.833	1.127
18 12 78	750	1.60	4.3	239.2	0.882	0.843	1.147
20 12 78	745	1.60	4.8	238.9	0.882	0.960	1.147
21 12 78	805	12.20	78.4	240.3	1.264	1.607	2.038
2 01 79	810	12.30	80.4	239.6	--	1.548	1.960
3 01 79	920	11.90	89.5	240.1	1.264	1.607	2.097
3 01 79	939	12.10	89.6	240.3	1.264	1.607	2.097
3 01 79	953	12.00	89.4	240.3	1.284	1.617	2.107
3 01 79	1008	12.00	89.5	240.3	1.274	1.607	2.097
3 01 79	1021	11.90	89.4	240.0	1.284	1.627	2.097
3 01 79	1049	12.00	89.6	240.0	1.284	1.617	2.087
3 01 79	1131	12.10	89.5	240.3	1.284	1.627	2.097
3 01 79	1203	11.90	89.6	239.9	1.284	1.607	2.097
3 01 79	1247	12.00	89.8	240.3	1.284	1.627	2.087
3 01 79	1308	12.00	89.7	239.9	1.284	1.617	2.097
3 01 79	1318	12.00	89.7	240.0	1.294	1.637	2.097
3 01 79	1418	12.00	89.6	240.3	1.274	1.617	2.097
3 01 79	1436	12.00	89.7	240.3	1.284	1.617	2.097
4 01 79	845	1.60	4.9	239.1	0.862	0.804	1.166
4 01 79	944	1.50	4.0	238.6	0.882	0.833	1.205
5 01 79	1647	9.50	74.4	239.5	1.205	1.519	1.999
5 01 79	1647	9.50	74.4	239.5	1.205	1.519	1.999
15 01 79	755	1.30	6.0	239.4	0.902	0.804	0.794
16 01 79	2015	7.50	57.6	240.1	1.098	1.372	--
9 01 79	755	11.90	93.9	239.5	1.264	1.607	--
22 01 79	755	1.50	6.1	239.2	0.902	0.853	--
25 01 79	756	12.00	92.9	240.1	1.284	1.607	--
2 04 79	827	2.20	5.0	238.9	0.970	0.931	--
2 04 79	848	2.30	5.2	238.8	0.843	0.970	--
6 04 79	750	10.70	97.0	239.4	1.401	1.774	--
7 04 79	821	2.60	16.2	238.9	0.862	1.049	--
7 04 79	1253	11.10	99.4	239.6	1.421	1.774	--

TABLE 4. (contd)

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
17 04 79	746	11.10	97.8	235.0	1.352	1.666	--
20 04 79	1553	2.20	9.4	229.7	0.794	0.902	--
22 04 79	1811	10.80	95.7	230.0	1.352	1.637	--
26 04 79	756	10.70	94.8	229.8	1.343	1.617	--
27 04 79	858	10.60	95.8	230.6	1.411	1.695	--
27 04 79	0	10.50	95.6	230.3	0.098	0.098	--
27 04 79	931	10.50	95.6	230.3	1.421	1.695	--
27 04 79	1208	10.50	95.1	230.4	1.411	1.666	--
30 04 79	754	10.70	97.4	229.9	1.431	1.686	--
30 04 79	0	10.70	96.5	230.1	0.098	0.098	--
30 04 79	1057	10.50	96.4	229.7	1.431	1.686	--
4 05 79	746	1.40	16.2	229.5	0.911	1.009	--
11 05 79	753	9.60	92.7	230.1	1.431	1.646	--
11 05 79	1025	4.80	51.1	229.8	1.225	1.392	--
15 05 79	802	0.00	1.0	71.1	0.608	0.627	--
21 05 79	800	4.30	41.9	224.2	1.166	1.294	--
27 05 79	1043	0.00	1.1	74.2	0.412	0.431	--
9 07 79	1831	2.60	16.8	237.1	1.117	1.235	--
10 07 79	635	2.20	14.3	239.6	1.078	1.196	--
13 07 79	1400	10.10	88.1	238.9	1.656	1.921	--
15 07 79	1322	11.00	94.4	238.6	1.686	1.950	--
16 07 79	806	1.70	12.5	238.9	1.058	1.176	--
16 07 79	1934	12.10	101.3	239.3	1.695	1.980	--
25 07 79	1242	1.70	13.0	238.7	1.000	1.107	--
26 07 79	750	11.80	97.2	239.4	1.695	1.960	--
2 08 79	814	11.40	102.5	225.0	1.627	1.891	--
7 08 79	748	11.40	100.3	225.3	1.627	1.862	--
10 08 79	754	11.70	100.5	234.7	1.666	1.931	--
15 08 79	925	11.30	97.0	234.8	1.666	1.882	--
21 08 79	758	7.60	68.4	235.1	1.568	1.764	--
23 08 79	1433	11.80	97.9	235.7	1.637	1.940	--
23 08 79	1448	11.60	97.8	235.2	1.646	1.931	--
23 08 79	1525	1.40	2.2	222.7	0.657	0.735	--
24 08 79	1225	1.90	5.4	233.8	0.960	1.058	--
5 10 79	1623	11.80	97.7	239.6	1.470	1.911	--
5 10 70	1750	11.80	98.2	239.9	1.568	1.940	--
10 10 79	933	8.40	66.8	239.8	1.421	1.754	--
12 10 79	823	2.20	5.6	238.7	0.911	1.078	--
18 10 79	6	11.90	99.7	239.9	1.548	1.989	--
25 10 79	835	12.40	98.9	239.9	1.617	2.009	--
25 10 79	1220	12.30	99.1	239.9	1.617	2.029	--
26 10 79	1010	12.30	99.2	239.7	1.617	2.009	--
2 11 79	1203	12.40	99.2	239.8	1.588	1.989	--
8 11 79	815	12.10	99.9	239.5	1.607	1.980	--
24 11 79	859	11.80	97.9	239.3	1.568	2.019	--
30 11 79	1821	12.20	99.8	239.8	1.519	1.950	--

TABLE 4. (contd)

Date	Time	Reactor Power, MW	Assembly Power, kW	Moderator Temperature, °C	Pressures, MPa Rod 1	Rod 5	Rod 6
10 12 79	929	0.00	0.4	205.8	0.902	0.813	--
13 12 79	816	2.60	5.9	239.8	0.853	1.058	--
16 12 79	1930	11.60	92.9	239.2	1.539	1.931	--
19 12 79	1229	4.50	33.1	210.1	1.088	1.362	--
23 12 79	1805	12.10	96.5	239.7	1.539	2.009	--
27 12 79	325	12.20	98.4	239.4	1.548	2.087	--
2 01 80	2121	12.10	98.5	239.2	1.529	1.960	--
14 03 80	1826	0.00	0.5	84.4	0.666	0.951	--
16 03 80	1834	0.70	6.2	239.6	0.755	1.274	--
16 03 80	2054	0.00	0.3	228.3	0.833	1.068	--
28 03 80	1405	11.50	83.9	240.0	1.470	2.323	--
2 04 80	1037	10.20	76.5	239.4	1.499	2.362	--
5 04 80	2159	10.40	77.0	239.3	1.480	2.303	--
10 04 80	2055	13.10	98.6	240.0	1.529	2.411	--
13 04 80	957	13.20	88.2	240.0	1.470	2.293	--
16 04 80	923	13.40	88.6	239.7	1.499	2.303	--
25 04 80	1835	0.10	0.8	74.2	0.412	0.892	--
27 04 80	927	1.70	4.0	234.4	0.804	1.274	--
30 04 80	806	8.00	47.8	234.7	1.274	1.960	--
5 05 80	1207	12.20	81.8	234.7	1.441	--	--
5 05 80	2121	12.10	81.3	234.9	1.480	2.313	--
8 05 80	1835	8.40	62.4	229.4	1.411	2.156	--
9 05 80	1027	2.00	3.4	229.7	0.960	1.235	--
18 05 80	1840	1.80	3.3	210.4	0.843	1.078	--

BURNUP

Calculated burnups at each TC location for each rod are presented in Table 5. 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 these results using this method and PIE data from rod 6 of IFA-431 (Nealley et al. 1979).

TABLE 5. Burnup in GJ/kgU[a]

Month-Year	Location	Rod 1	Rod 2	Rod 3	Rod 8	Rod 5	Rod 6
1-78	UTC[b]	1456.1	1393.8	1408.1	1186.1	1484.2	1481.7
	LTC[c]	1044.9	1024.5	1047.1	877.9	1091.0	1077.3
4-78	UTC	1471.6	1409.0	1422.8	1200.4	1499.0	1497.1
	LTC	1056.6	1035.9	1058.4	889.1	1102.7	1089.2
5-78	UTC	1515.0	1451.2	1464.3	1241.5	1541.4	1540.5
	LTC	1088.8	1067.4	1090.3	921.4	1135.9	1122.2
6-78	UTC	1556.1	1491.1	1503.6	1280.7	1581.9	1582.0
	LTC	1118.4	1096.3	1119.6	951.0	1166.4	1152.5
7-78	UTC	1596.2	1530.7	1543.2	1319.9	1621.8	1622.2
	LTC	1149.0	1126.3	1150.1	981.7	1198.0	1183.9
8-78	UTC	1699.8	1631.8	1643.4	1420.1	1724.9	1727.0
	LTC	1224.8	1200.4	1225.1	1057.4	1276.0	1261.6
9-78	UTC	1803.1	1731.7	1741.8	1518.6	1827.2	1831.5
	LTC	1299.4	1273.4	1298.3	1130.7	1351.6	1337.5
10-78	UTC	1818.8	1746.9	1756.8	1533.7	1842.7	1847.4
	LTC	1310.4	1284.0	1309.1	1141.5	1362.7	1348.7
11-78	UTC	1825.1	1753.3	1763.1	1540.0	1849.0	1853.7
	LTC	1315.7	1289.3	1314.4	1146.9	1368.2	1354.1
12-78	UTC	1890.7	1818.2	1828.0	1606.2	1914.7	1919.8
	LTC	1368.3	1340.8	1366.5	1200.4	1422.2	1407.9
1-79	UTC	1933.3	1859.8	1869.2	1648.4	1957.2	1963.0
	LTC	1400.5	1372.4	1398.6	1233.2	1455.4	1440.9
4-79	UTC	2015.4	1940.1	1948.4	1727.0	2037.9	2045.4
	LTC	1465.8	1437.5	1465.0	1299.6	1522.5	1507.1

TABLE 5. (contd)

Month-Year	Location	Rod 1	Rod 2	Rod 3	Rod 8	Rod 5	Rod 6
5-79	UTC	2059.5	1983.3	1991.0	1769.2	2081.2	2089.6
	LTC	1501.5	1473.0	1501.1	1335.3	1558.6	1543.0
7-79	UTC	2111.3	2034.1	2041.2	1819.0	2132.1	2141.5
	LTC	1541.6	1513.1	1541.6	1375.3	1599.0	1583.3
8-79	UTC	2188.2	2109.6	2116.0	1893.3	2208.0	2218.6
	LTC	1600.8	1572.2	1601.6	1434.8	1659.0	1642.9
10-79	UTC	2274.0	2194.1	2200.0	1976.7	2293.0	2304.7
	LTC	1666.4	1638.1	1668.9	1501.7	1726.2	1709.1
11-79	UTC	2338.2	2257.2	2262.6	2039.0	2356.6	2369.2
	LTC	1716.5	1688.3	1720.0	1552.4	1777.2	1759.5
12-79	UTC	2414.9	2332.4	2337.0	2113.0	2432.3	2446.2
	LTC	1776.3	1747.9	1780.6	1612.5	1837.8	1819.8
1-80	UTC	2436.3	2353.4	2357.9	2133.8	2453.5	2467.8
	LTC	1792.9	1764.6	1797.5	1629.4	1854.8	1836.6
3-80	UTC	2450.1	2366.8	2370.8	14.8[d]	2466.4	2481.3
	LTC	1804.0	1775.4	1808.4	12.3	1866.0	1847.8
4-80	UTC	2508.6	2423.8	2425.3	77.1	2520.9	2538.6
	LTC	1851.4	1821.9	1855.0	64.8	1913.9	1895.9
5-80	UTC	2534.1	2448.7	2449.2	104.5	2544.8	2563.6
	LTC	1872.1	1842.1	1875.3	87.6	1934.7	1916.8

(a) To convert to MWd/MTM multiply by 11.6.
(b) Upper thermocouple.
(c) Lower thermocouple.
(d) At this time rod 8 was replaced by rod 9.

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*These reports are available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and/or the National Technical Information Service, Springfield, VA. 22161.

APPENDIX A

FUEL ROD AND FUEL COLUMN SCHEMATICS FOR IFA-432

APPENDIX A

FUEL ROD AND FUEL COLUMN SCHEMATICS FOR IFA-432

This appendix illustrates the fuel rod and fuel column schematics for instrumented fuel assembly (IFA)-432, which is being irradiated in the Halden boiling water reactor (HBWR) in Halden, Norway.

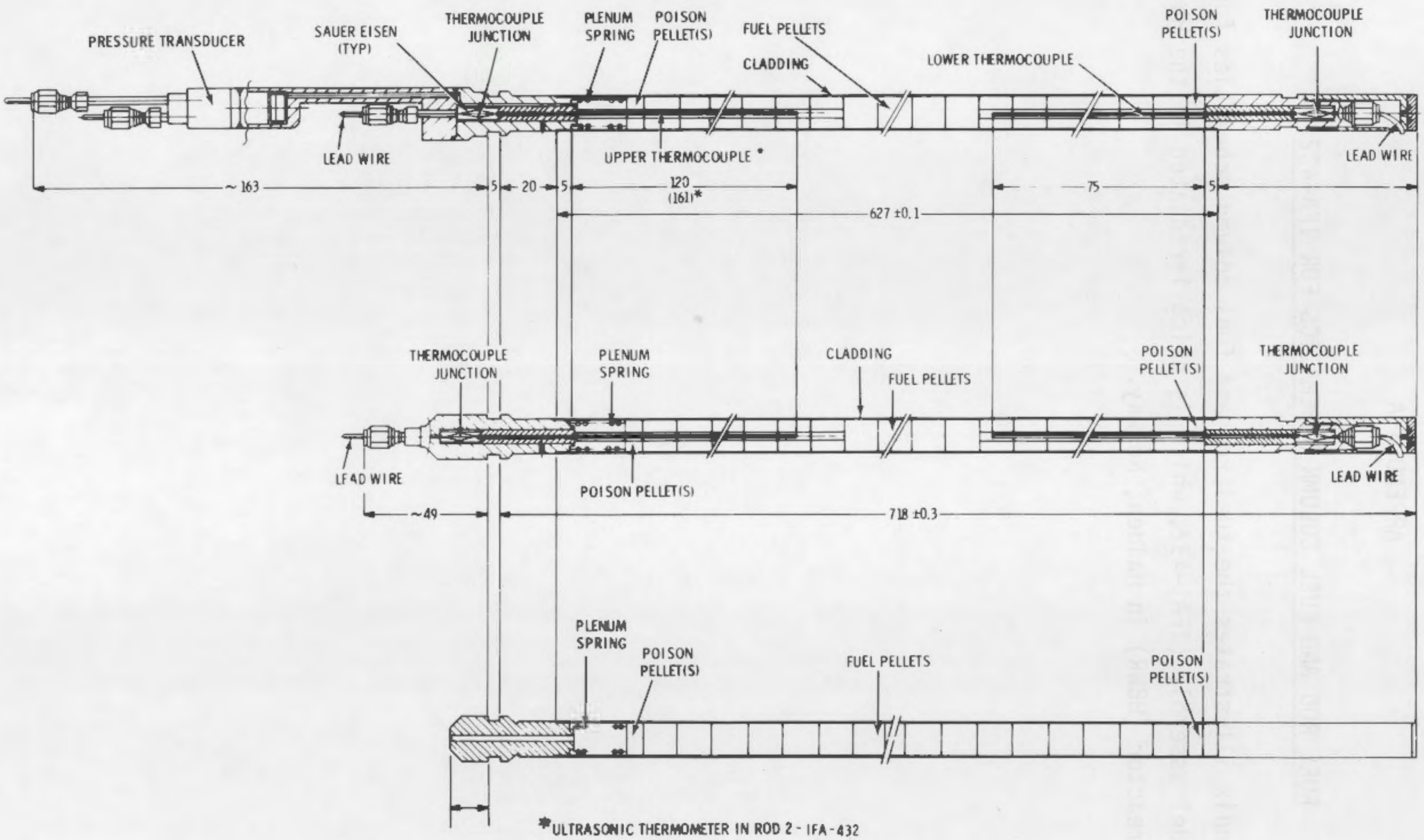
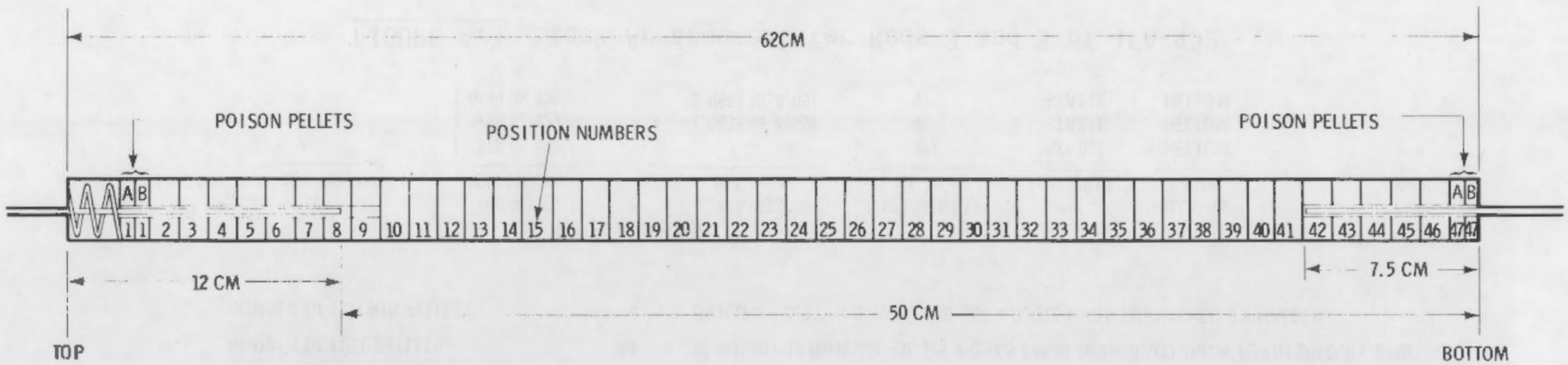


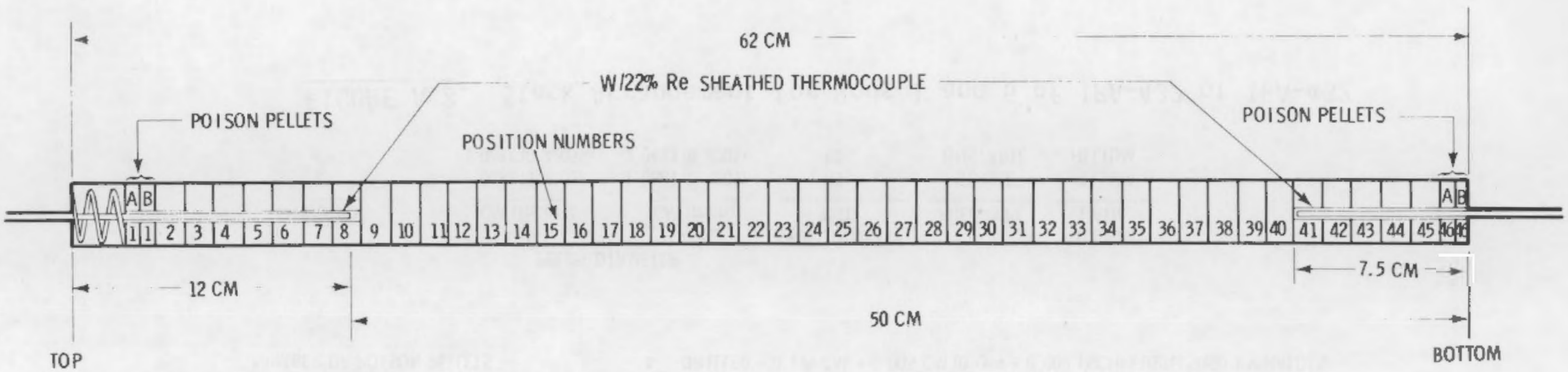
FIGURE A.1. Schematic Arrangement of Fuel Rods for IFA-432



TOTAL NUMBER OF PELLETS IN EACH STACK -	49	
NUMBER OF FUEL PELLETS -	45	32 SOLID, 13 DRILLED - 0.175 CM ± 0.005 CM (0.069 ± 0.002 INCH) FURNISHED BY BNW
NUMBER OF POISON PELLETS -	4	DRILLED - 0.175 CM ± 0.005 CM (0.069 ± 0.002 INCH) FURNISHED BY HALDEN

ROD NO.	PELLET DIAMETER IFA-431 CM (inch)	PELLET DIAMETER IFA-432 CM (inch)	FUEL DENSITY % TD	FUEL TYPE	FILL GAS 1 ATM
1	1.0681 (0.4205)	1.0681 (0.4205)	95	STABLE	HELIUM
6	1.0681 (0.4205)	1.0681 (0.4205)	92	UNSTABLE	HELIUM

FIGURE A.2. Stack Arrangement for Rods 1 and 6 of IFA-432 of IFA-432



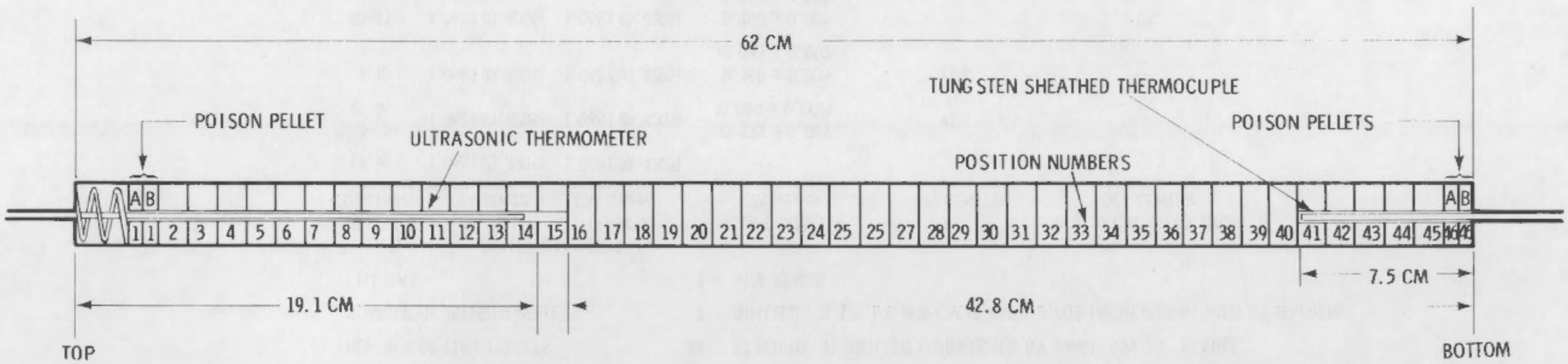
TOTAL NUMBER OF PELLETS IN EACH STACK - 48

NUMBER OF FUEL PELLETS - 44 32 SOLID, 12 DRILLED - 0.175 ± 0.005 CM (0.069 ± 0.002 INCH) FURNISHED BY BNW

NUMBER OF POISON PELLETS - 4 DRILLED - 0.175 ± 0.005 CM (0.069 ± 0.002 INCH) FURNISHED BY HALDEN

	PELLET DIAMETER				
ROD NO.	IFA-431 CM (inch)	IFA-432 CM (inch)	FUEL DENSITY % TD	FUEL TYPE	FILL GAS 1 ATM
2	1.0528 (0.4145)	--	95	STABLE	HELIUM
3	1.0858 (0.4275)	1.0833 (0.4265)	95	STABLE	HELIUM
5	1.0681 (0.4205)	1.0681 (0.4205)	92	STABLE	HELIUM

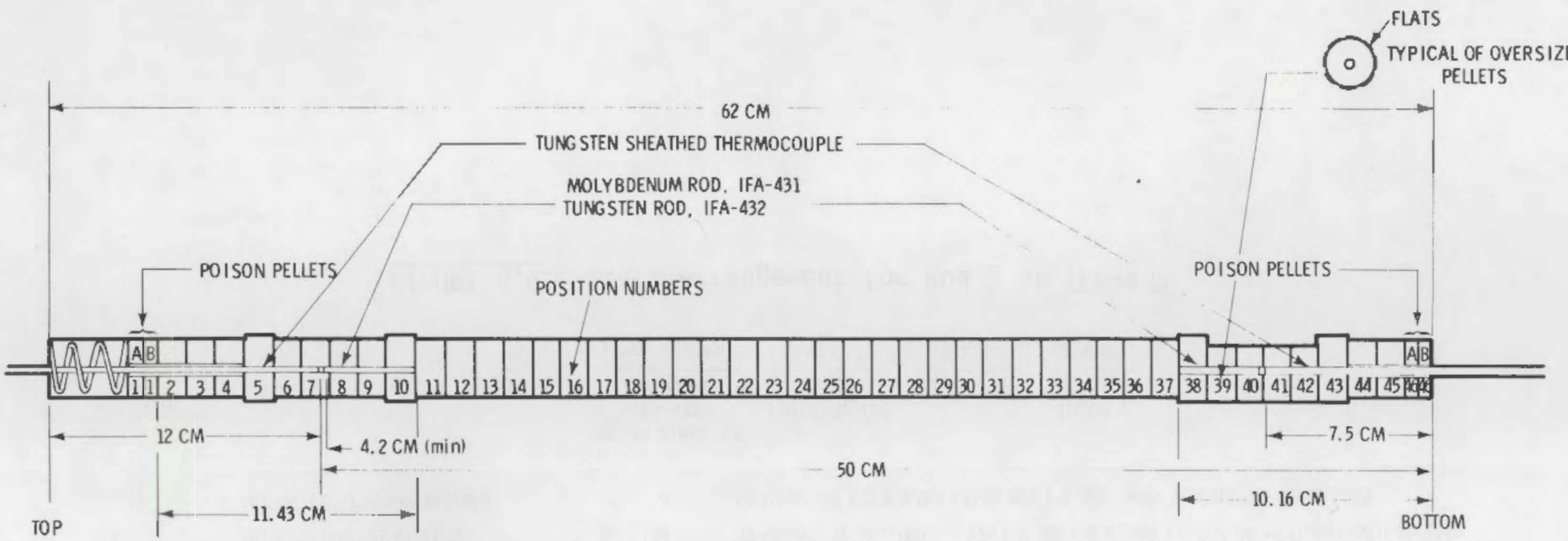
FIGURE A.3 Stack Arrangement for Rods 3 and 5 of IFA-432



TOTAL NUMBER OF PELLETS IN EACH STACK -	48	
NUMBER OF FUEL PELLETS -	44	32 SOLID, 12 DRILLED - 0.175 ± 0.005 CM (0.069 ± 0.002 INCH) FURNISHED BY BNW
NUMBER OF POISON PELLETS -	4	DRILLED - 0.175 ± 0.005 CM (0.069 ± 0.002 INCH) FURNISHED BY HALDEN

ROD NO.	PELLET DIAMETER IFA-432 CM(INCH)	FUEL DENSITY % TD	FUEL TYPE	FILL GAS 1 ATM
2	1.0528 (0.4145)	95	STABLE	HELIUM

FIGURE A.4. Stack Arrangement for Rod 2 of IFA-432



TOTAL NUMBER OF PELLETS IN STACK - 48

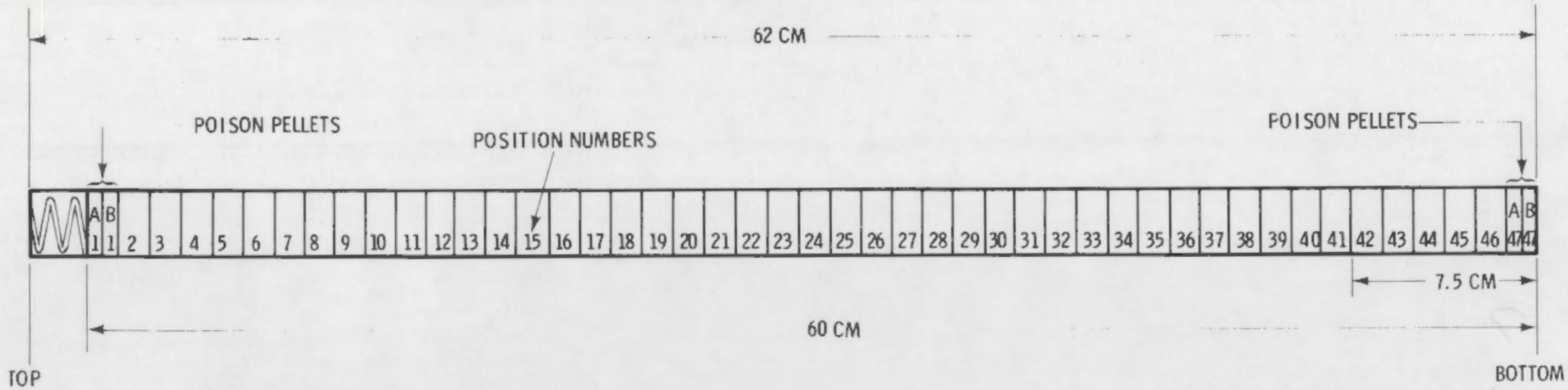
NUMBER OF FUEL PELLETS - 44 27 SOLID, 17 DRILLED (FURNISHED BY BNW), 95% TD, STABLE

NUMBER OF POISON PELLETS - 4 DRILLED - 0.175 ± 0.005 CM (0.069 ± 0.002 INCH) (FURNISHED BY HALDEN)

FILL GAS - 1 ATM XENON

PELLET POSITION	PELLET DIAMETER IFA-431 CM (inch)	IFA-432 CM (inch)	HOLE DIAMETER CM (inch)	ON CENTER	0.0127 CM (0.005 INCH) OFF CENTER
12-38	1.0681 (0.4205)	1.0681 (0.4205)	--	--	--
3- 5 45-46	1.0681 (0.4205)	1.0681 (0.4205)	0.175 ± 0.005 (0.069 ± 0.002)	YES	--
7-10	1.0681 (0.4205)	1.0687 (0.4205)	0.160 ± 0.005 (0.063 ± 0.002)	YES	--
40-43	1.0681 (0.4205)	1.0681 (0.4205)	0.160 ± 0.005 (0.063 ± 0.002)	--	YES
6, 11, 39, 44	1.0858 (0.4275)	1.0833 (0.4265)	0.160 ± 0.005 (0.063 ± 0.002)	YES	--

FIGURE A.5. Stack Arrangement for Rod 4 of IFA-432 (Xenon Fill Gas)



TOTAL NUMBER OF PELLETS IN EACH STACK -	49	
NUMBER OF FUEL PELLETS -	45	FURNISHED BY BNW
NUMBER OF POISON PELLETS -	4	FURNISHED BY HALDEN

ROD NO.	PELLET DIAMETER IFA-432 CM (INCH)	FUEL DENSITY % TD	FUEL TYPE	FILL GAS 1 ATM
7	1.0528 (0.4145)	95	STABLE	HELIUM
8	1.0681 (0.4205)	95	STABLE	HELIUM
9	1.0732 (0.4225)	95	STABLE	HELIUM

FIGURE A.6. Stack Arrangement for Noninstrumented Replacement Rods 7, 8, and 9 of IFA-432

APPENDIX B

DATA PROCESSING

APPENDIX B

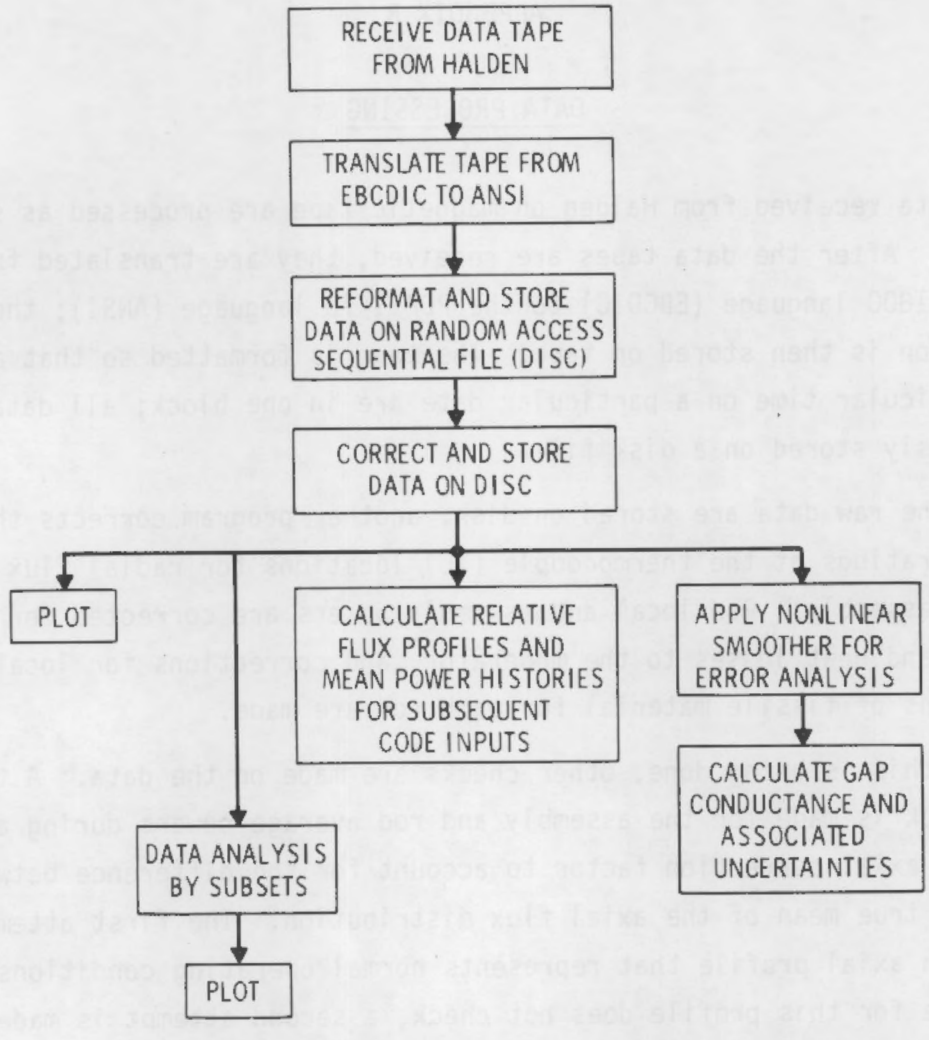
DATA PROCESSING

The data received from Halden on magnetic tape are processed as shown in Figure B.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 for each rod are made.

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 ^{235}U.



<u>FIGURE B.1.</u> Flow Diagram for Processing Halden Data

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. 6 and 7). 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 is briefly disccused below. The accuracy and uncertainty of their respective outputs is discussed more completely in Hann et al. (1977).

NEUTRON DETECTORS

IFA-432 is 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 is 100 mm (3.93 in.) long and is positioned so that the center of the detector and the TC junction are located on essentially the same 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. The error limits for the outputs of the detectors were estimated to be $\pm 2.5\%$ at a neutron flux of 1.1×10^{14} n/m^2-s.

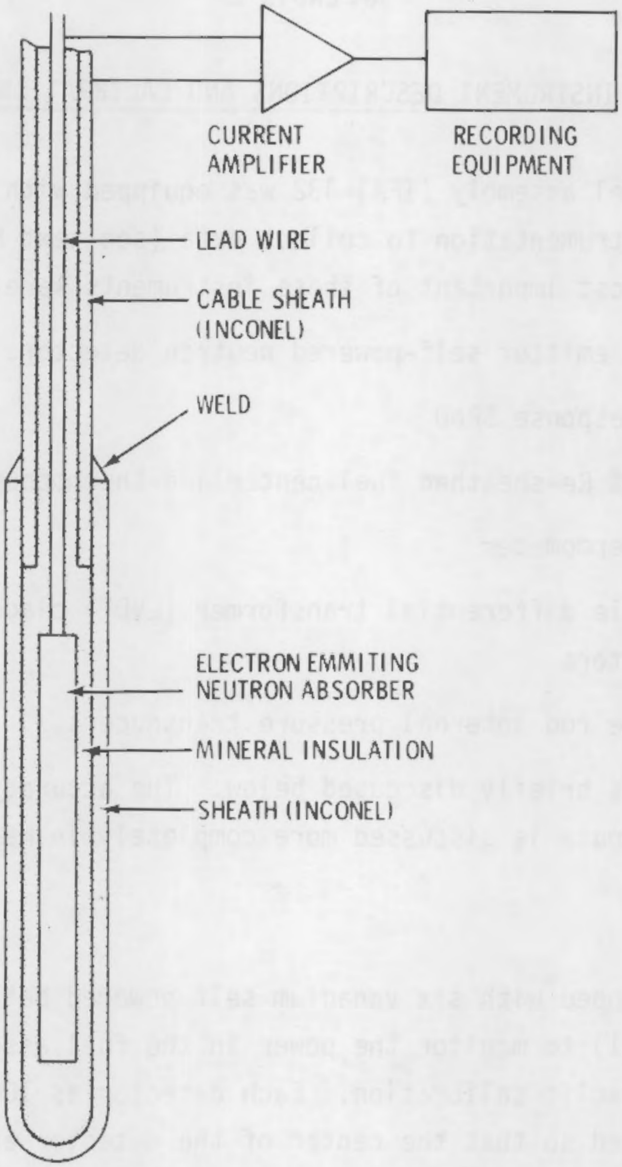


FIGURE C.1. Schematic of Self-Powered Beta Current Neutron Detector

In addition to correlating the detector outputs to the neutron flux in the Studsvik Reactor, Halden has conducted long-term tests of similar neutron detectors in the Halden boiling water reactor (HBWR). These tests have established the detectors as reliable and accurate instruments without a measurable change in sensitivity at the 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.

The vanadium detectors have 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 that were used in IFA-432 to measure the 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.

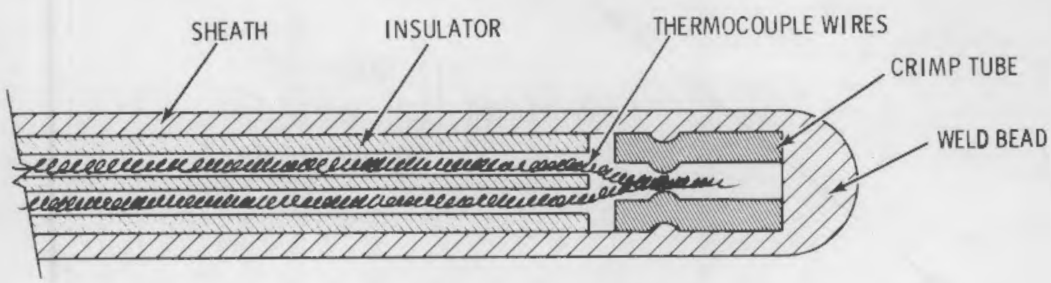


FIGURE C.2. Schematic of W 5% Re/W 26% Re Thermocouples with Grounded Junction

(a) 5.5 min, 0 to 63%.

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. Calibration of the TCs over the range of use produces a brittle assembly that is fragile and subject to breakage; consequently, only one TC, which was not used in the in-reactor test, was calibrated.

The 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 closer since the 2755K temperature is above that given in most calibration tables for W/Re TCs. The calibrated TC had the following limits of error:

- 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.

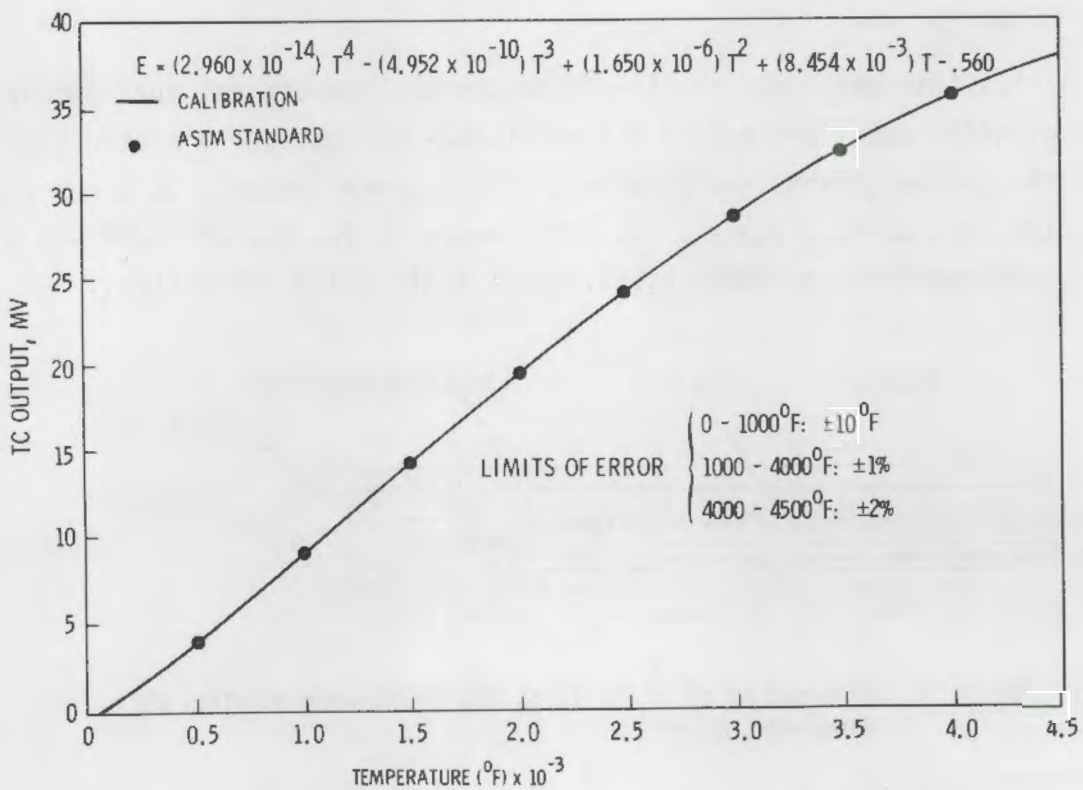


FIGURE C.3. Calibration Curve for W 5% Re/W 26% Re Thermocouples

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 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 decalibration of up to 1%/100 GJ/kgU burnup. Consequently, the measured fuel temperatures could be 20% 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 sensors 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, while the 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 (0.1 kg/cm^2) and the accuracy and repeatability are $\pm$0.1 MPa ($\pm$1 kg/cm^2) and $\pm$0.04 MPa ($\pm$0.4 kg/cm^2), respectively.

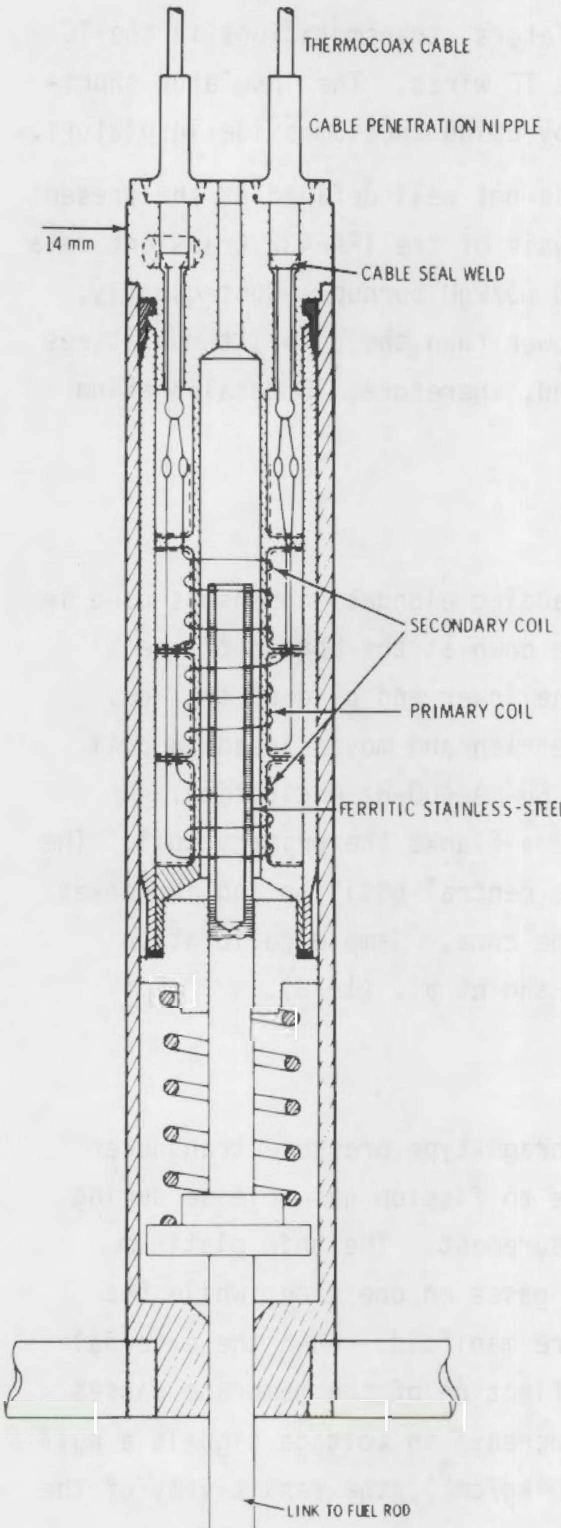


FIGURE C.4. Cladding Elongation Monitor (Halden Project Design)

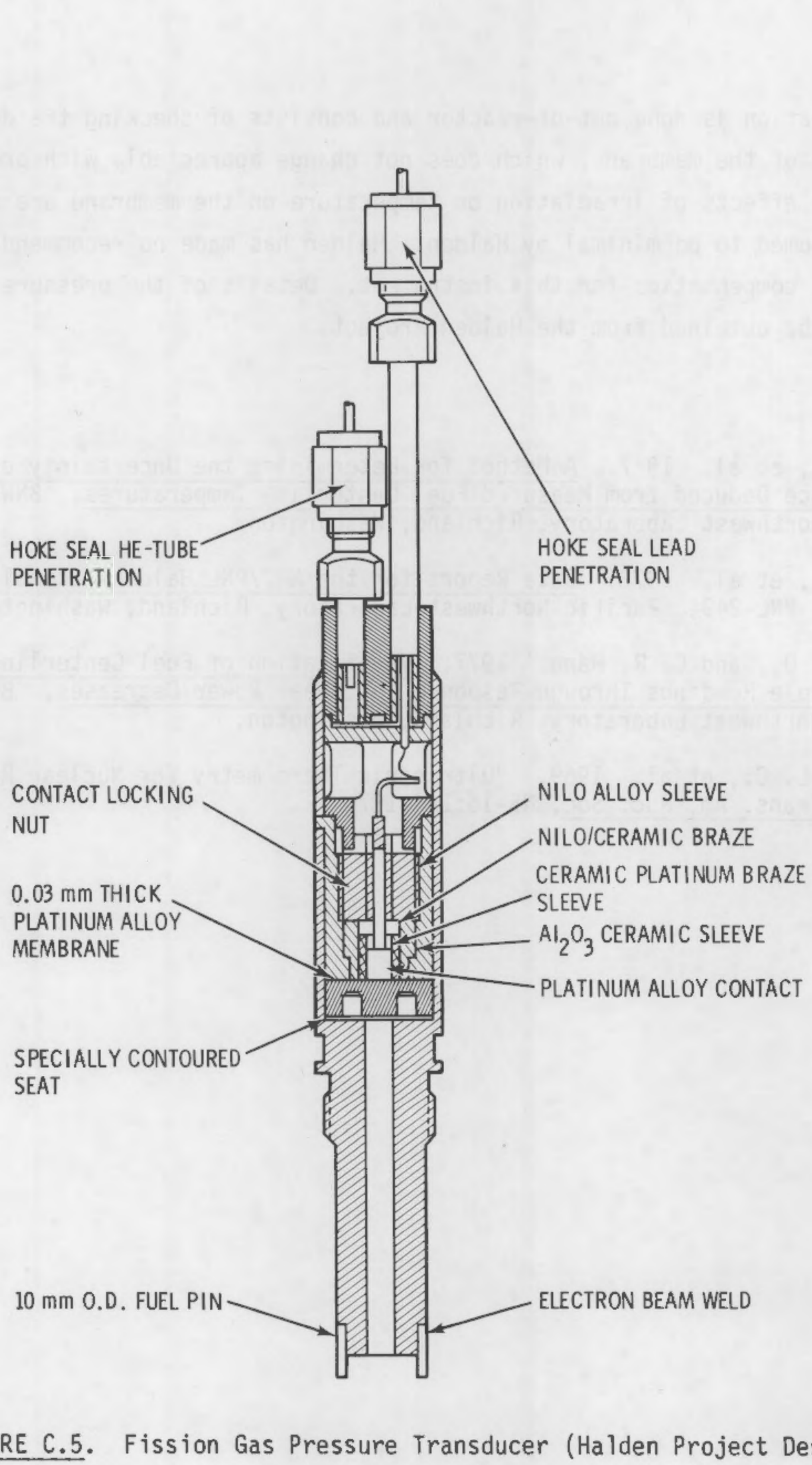


FIGURE C.5. Fission Gas Pressure Transducer (Halden Project Design)

Calibration is done out-of-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. Details of the pressure transducer must be obtained from the Halden Project.

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APPENDIX D

ASSEMBLY POWER CALIBRATION

APPENDIX D

ASSEMBLY POWER CALIBRATION

The data report for the instrumented fuel assembly (IFA)-431 briefly explained the usual method for calibrating assemblies in the Halden reactor.[a] This procedure was not used in the case of IFA-432 because the calibration flow valve (text Figure 2, p. 7) failed in the normal operating position, allowing only natural circulation. However, both assemblies were in the core simultaneously at the time of IFA-432 startup. The second assembly 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 ±6%.

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

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This report presents the in-reactor data collected from the PNL Halden test assembly IFA-432. The irradiation test is part of an experimental program entitled "Experimental Support and Development of Single-Rod Fuel Codes" sponsored by the Fuel Behavior Research Branch of the NRC. The purpose is to reduce the uncertainties of predicting the thermal and mechanical behavior of an operating nuclear fuel rod. Fuel centerline temperatures, cladding elongation, internal fuel rod pressures, and local powers at the thermocouple (TC) positions are shown as a function of time. 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 also presented.

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