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End-of-Irradiation Data Report for the Instrumented Fuel Assembly (IFA)-527

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

Pacific Northwest Laboratory
Operated by
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**U.S. Nuclear Regulatory
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We would like to thank the following Pacific Northwest Laboratory (PNL) personnel: R. D. Freeman for his ongoing review and advice; W. D. Bennett for her assistance with handling the data; and J. K. Eiler for her editing assistance.

ABSTRACT

This report presents data obtained during the irradiation of the six-rod instrumented fuel assembly (IFA)-527 in the Halden Boiling Water Reactor (HBWR), Halden, Norway. This assembly is the last in 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-527 contains five identical rods with high-density stable fuel pellets and 230- μ m diametral gaps and one rod with similar fuel pellets but with a 60- μ m diametral gap. All six rods were xenon-filled to simulate the effects of fission gas and to enhance the observable effects of fuel cracking and relocation on fuel temperatures. The assembly operated successfully from July 1, 1980, to August 15, 1980; the reactor was then shut down until September 10, 1980. During the shutdown, at least four of the six rods suffered pressure boundary failures. Irradiation of the assembly continued with the failed rods from September 10, 1980, until April 8, 1981; the assembly was then removed from the reactor. This report presents both pre- and postfailure data for IFA-527.

SUMMARY

Instrumented fuel assembly (IFA)-527 is the last in a series of U.S. Nuclear Regulatory Commission (NRC)-sponsored tests that are being conducted in the Halden Boiling Water Reactor (HBWR), Halden, Norway. IFA-527 carried six heavily instrumented test fuel rods that were initially filled with xenon gas rather than the standard helium gas. Xenon fill gas was used to provide test data related to the worst-case effects of fission gas on fuel temperatures and to enhance the observable effects of fuel cracking and relocation on fuel temperatures. Five of the six rods in IFA-527 were identical with high-density stable fuel pellets and 230- μm diametral gaps; the sixth rod was similar in design but had a 60- μm diametral gap. The purpose of having five rods of identical design was to assess the variability of fuel relocation and its consequences. Power was held below 18.0 kW/m to maintain fuel temperatures within the range of commercial light-water reactors (LWRs).

The startup, calibration, and first operating cycle for IFA-527 were highly successful. Temperatures from the nominal- and small-gap rods provided a bracket for the expected temperatures for fission gas-saturated rods; fuel relocation in the peak-power sections of the rods became evident in the first month of operation. Temperature, elongation, and gas pressure data indicated that the behavior of the five identical nominal-gap rods was very similar.

Unfortunately, at least four of the six rods in IFA-527 failed during the reactor shutdown from August 15 to September 10, 1980. These rod failures were indicated by high pressure readings and simultaneously low fuel temperatures representative of steam rather than xenon in the fuel rods. Despite the apparent water ingress, no fission product leakage was detected; and the thermocouples appeared to be functioning properly. The assembly remained in the reactor for two additional operating cycles. When it was removed from the reactor (April 1981), all of the rods had failed and four of the six rods had indicated temperatures that were extremely low.

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4	Instrument Failure for TFA-881
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INTRODUCTION

The six-rod instrumented fuel assembly (IFA)-527, which was sponsored by the U.S. Nuclear Regulatory Commission (NRC) and irradiated in the Halden Boiling Water Reactor (HBWR), Halden, Norway, is the final assembly designed and irradiated under the Pacific Northwest Laboratory (PNL)^(a) program "Experimental Support and Development of Single-Rod Fuel Codes." A principal objective of this program is the collection of well-qualified steady-state fuel rod data, which is being used to develop and verify fuel performance computer codes.

The first two tests in this program--IFA-431 and -432--were identical assemblies that contained the same variations of gap sizes and fuel pellet types but were operated at different power levels and to different burn-ups.⁽¹⁻⁴⁾ The third assembly--IFA-513--was jointly sponsored by NRC and the Halden Reactor Project.^(5,6) It had six identical rods (high-density stable UO₂ fuel and a nominal pellet-cladding diametral gap size of 230 μ m) with variations in fill gas composition and pressure.

The fourth and final assembly in the series--IFA-527--contained five identical rods with high-density stable fuel pellets and 230- μ m diametral gaps; the sixth rod had a much smaller gap--60 μ m. All six rods were filled with xenon gas to enhance the observable thermal effects of fuel pellet cracking and relocation.

IFA-527 was designed to:

- include, within the publicly documented NRC Halden Test Series, rods with pure xenon fill gas, thus bracketing the worst-case estimates of fuel temperature dependence on power
- study fuel cracking and relocation effects upon fuel temperatures without the added complication of fill gas thermal conductivity degradation due to fission gas release
- assess not only the speed but also the variability of fuel cracking and relocation effects by including several rods of identical design and operating them identically
- possibly confirm burnup enhancement of fission gas release by operating these rods to high burnup at constant power and (because of xenon fill gas) constant temperature.

The irradiation of IFA-527 began July 1, 1980; and the first cycle continued until August 15, 1980. Very consistent behavior was obtained from all rods during this cycle. The second operating cycle began September 10, 1981, and

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

continued until October 4, 1980. Several rods showed evidence of failure and water ingress at the start of this cycle, but the decision was made to continue the irradiation to collect operating data from failed rods. After the third cycle (December 12, 1980, to February 7, 1981), rod 1 was removed from the assembly. After the final cycle of operation (April 5 to April 8, 1981), the assembly was removed from the reactor.

This report presents a description of IFA-527 and its instrumentation, a presentation of the data obtained from the irradiation of IFA-527, and comments on the observed rod temperatures.

CONCLUSIONS

The NRC/PNL Halden assembly IFA-527 carried six heavily instrumented test fuel rods that were initially filled with xenon gas rather than the standard helium gas. The objectives of these xenon-filled rods were to provide test data related to the worst-case effects of fission gas on fuel temperatures and to enhance the observable effects of fuel cracking and relocation on fuel centerline temperatures. Five of the six rods were of identical design to assess the variability of fuel relocation and its consequences. Operating power was limited to 18 kW/m to maintain fuel temperatures within the range of commercial light-water reactors (LWRs).

The first operating cycle was successful in meeting the goal of bracketing fuel temperatures. Temperatures from rod 6 (small gap) were only slightly higher than those of a small-gap helium rod, which suggests that fuel-cladding heat flow for small-gap rods is dominated by fuel-cladding contact and that the fill gas composition has little effect. Temperatures from the standard-gap xenon-filled fuel rods were higher than those observed for similarly dimensioned helium-filled rods that had been irradiated to fission gas saturation. Thus, these xenon-filled fuel rods provide a guide to the upper bound of fuel temperatures under normal operating conditions.

The ingress of water to the fuel rods resulted in much lower temperatures when compared to first cycle temperatures. The lower temperatures are attributed to the higher thermal conductivity of steam as compared to xenon. The rods continued to operate at powers up to 18 kW/m with steam as the fill gas for several months. There was no indication of an increase in fuel temperatures due to the oxidation of UO_2 and subsequent decrease in fuel thermal conductivity. In fact, there was a further decrease in fuel temperatures following prolonged exposure to water. These extremely low temperatures may have been a result of either a change in fuel thermocouple response characteristics or a change in the fuel thermal response at low powers relative to "normal" conditions.

CONCLUSIONS

The RCAPB, Halden assembly IFA-527 carried six heavily instrumented test fuel rods that were initially filled with xenon gas rather than the standard helium gas. The objectives of these xenon-filled rods were to provide test data related to the worst-case effects of fission gas on fuel temperatures and to enhance the observable effects of fuel cracking and relocation on fuel centerline temperatures. Five of the six rods were of identical design to assess the variability of fuel relocation and its consequences. Operating power was limited to 18 kW to maintain fuel temperatures within the range of commercial light-water reactors (LWRs).

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The ingress of water to the fuel rods resulted in much lower temperatures when compared to first cycle temperatures. The lower temperatures are attributed to the higher thermal conductivity of steam as compared to xenon. The rods continued to operate at powers up to 18 kW with steam in the core for several months. There was no indication of an increase in fuel temperatures due to the oxidation of UO₂ and subsequent decrease in fuel thermal conductivity. In fact, there was a further decrease in fuel temperatures following prolonged exposure to water. These extremely low temperatures may have been a result of either a change in the thermopile response characteristics or a change in the fuel thermal response at low powers relative to normal conditions.

ASSEMBLY DESCRIPTION AND INSTRUMENTATION

IFA-527 (see Figure 1) was designed with six heavily instrumented test fuel rods that were filled with xenon gas and contained 95% theoretical density (TD), flat-ended UD_2 pellets enriched to 10% uranium-235. Each rod was instrumented with a pressure transducer, an elongation sensor, and fuel centerline thermocouples in both ends. Fabrication details and instrumentation calibration for the rods and the final assembly are described in the precharacterization report.⁽⁷⁾

Tables 1 and 2 contrast the design features and instrumentation of IFA-527 with those of previous assemblies.^(a) IFA-527 was very similar to IFA-513 in design; one major difference was the inclusion of a small-gap rod. Other design and operating differences are discussed below:

- IFA-527 operated at lower powers than IFA-513 (18 kW/m peak power as opposed to 42 kW/m).
- While all previous assemblies had peak power at or near the top end, IFA-527 was located in the top half of the reactor and had its peak-power location at the bottom end. The axial location of the assembly relative to the reactor flux shape is shown in Figure 2.
- The assembly was located at the extreme edge of the reactor core (see Figure 3) to assure the low powers needed to hold fuel temperatures within the range of commercial reactor experience. This extreme edge position (position 8-25) suffers a significant thermal neutron flux tilt across the assembly channel (edge-to-edge differences are in excess of 15%). This flux tilt was confirmed by vanadium neutron detector data and fuel centerline temperature data.
- Single-phase rather than two-phase flow was experienced during the assembly power calibration.^(b)

Except for the above differences, the assembly design, instrumentation, and operation of IFA-527 were very similar to those of IFA-513. Both assemblies had eight vanadium neutron detectors: three at both the upper and lower planes of fuel temperature measurement and two at equal intervals between those planes so that both the flux tilt at the thermocouple planes and the axial distribution of neutron flux might be confirmed.

In addition, both assemblies had a fast-reacting cobalt neutron detector at the axial and radial center to provide the relative variation of flux and hence power level during fast transients. This variation is especially useful in analyzing fuel thermocouple response to rapid linear and "step" power changes, both of which are helpful in confirming steady-state data trends and thermocouple reliability.^(9,10)

(a) IFA-431, IFA-432, and IFA-513.

(b) A description of the assembly power calibration may be found in Reference 8.

FIGURE 1. Schematic Design of Instrumented Fuel Assembly (IFA)-527. Neutron detectors 4 and 5 are in line with detectors 1 and 6.

TABLE 1. General Design Characteristics for NRC/PNL Instrumented Fuel Assemblies

<u>Characteristic</u>	<u>IFA-513 and -527</u>	<u>IFA-431 and -432</u>
Number of rods	6	6
Cladding:		
Material	Zircaloy-2	Zircaloy-2
Inside diameter, mm	10.9	10.9
Outside diameter, mm	12.8	12.8
Tube length, mm	824	610
Fuel Pellets:		
Material	UO ₂	UO ₂
Density, % TD	95	95, 92
Enrichment, % ²³⁵ U	9.9	10.0
Inside diameter, mm	1.75	1.75
Outside diameter, mm	10.67 (nominal)	10.67 (nominal)
Pellet length, mm	12.7	12.7
Pellet column length, mm	780 (nominal)	570 (nominal)
Poison pellet length (each end), mm	7	7
Fuel rod plenum length, mm	29	19

TABLE 2. Experimental Matrix for IFA-431, -432, -513, and -527

Assembly and Rod	Power, (a) kW/m	Fuel Type (b)	Diametral Gap, μm			Initial Fill Gas			Detectors (c)				
			50-75	230	380	100% He, 0.1 MPa	100% He, >0.1 MPa	Xe, 0.1 MPa	UTC	LTC	ES	PT	SPND
IFA-431	35/25												7
1		95S		X		X			X	X	X	X	
2		95S			X	X			X	X	X		
3		95S	X			X			X	X	X		
4		95S		X				100%	X	X	X		
5		92S		X		X			X	X	X	X	
6		92U		X		X			X	X	X	X	
IFA-432	50/35												7
1		95S		X		X			X	X	X	X	
2		95S			X	X			UT	X	X		
3		95S	X			X			X	X	X		
4		95S		X				100%	X	X	X		
5		92S		X		X			X	X	X	X	
6		92U		X		X			X	X	X	X	
IFA-513	40/28												9
1		95S		X		X			X	X	X	X	
2		95S		X			0.3 MPa		X	X	X	X	
3		95S		X		X			X	X	X	X	
4		95S		X				8%	X	X	X	X	
5		95S		X		X			X	X	X	X	
6		95S		X				23%	X	X	X	X	
IFA-527	13/18												9
1		95S		X				100%	X	X	X	X	
2		95S		X				100%	X	X	X	X	
3		95S		X				100%	X	X	X	X	
4		95S		X				100%	X	X	X	X	
5		95S		X				100%	X	X	X	X	
6		95S	X					100%	X	X	X	X	

(a) Linear power is given for upper and lower thermocouple positions, respectively.

(b) Three fuel types were used, and all were enriched to 10% ^{235}U : 95S = 95% TD, stable; 92S = 92% TD, stable; 92U = 92% TD, unstable.

(c) UTC = upper thermocouple; LTC = lower thermocouple; ES = elongation sensor; PT = pressure transducer; SPND = self-powered neutron detector; UT = ultrasonic thermometer.

FIGURE 2. Arrangement of Temperature Sensors, Neutron Detectors, and Fuel Relative to Reference Axial Thermal Flux Profile for Instrumented Fuel Assembly (IFA)-527

FIGURE 3. Instrumented Fuel Assembly (IFA)-527 Position in the Halden Boiling Water Reactor

DATA PRESENTATION

The irradiation of IFA-527 began on July 1, 1980; and the first operating cycle continued until August 15, 1980. At that time, the reactor was shut down until September 10, 1980. At the beginning of the second operating cycle, internal gas pressure sensors for rods 1, 4, and 5 registered pressures equal to the reactor coolant system pressure--an indication of rod failure. In addition, centerline fuel temperatures indicated the presence of steam rather than xenon in these rods. The pressure transducers for rod 2 also registered high initial pressures; however, these pressures decreased with time, signifying that the rod resealed and dried out. This pressure decrease was accompanied by a simultaneous increase in fuel temperature. Data from rods 3 and 6 gave no indication of failure at this time.

The decision was made to continue irradiating IFA-527 to collect data on the operation of failed rods. This decision was influenced by the fact that fission product detectors in the reactor were not picking up any releases from IFA-527 that would impact reactor operation. The second operating cycle for IFA-527 ran from September 10 until October 4, 1980; and the third, from December 12, 1980, to February 7, 1981. At the end of the third cycle, rod 1 was removed from the assembly for future postirradiation examination (PIE) on a failed fuel rod that had been irradiated for a shorter time than the other rods. During the fourth and final operating cycle (April 5 through 8, 1981), an increased amount of fission products in the coolant (from several sources) resulted in the decision to permanently remove all the failed fuel assemblies (including IFA-527) from the reactor. Calculated local burnups before and after rod failure are presented in Table 3 for each of the rods in IFA-527.

To provide an overview of the irradiation history, the reactor power during the irradiation period is presented in Figure 4 and the coolant outlet temperature and temperature change across the assembly are presented in Figure 5. There were no extended periods of constant power during the irradiation.

When considering the data presented in this report, several points are of importance. First, the coolant temperature is important because of its effect on the measured cladding elongation. The fuel rods are hung from the top of the assembly while the elongation sensors are mounted on the bottom of the assembly, which is supported by the stay rods. Thus, an increase in the length of the stay rods, independent of the fuel rods, will result in an indicated decrease in the length of the fuel rods. During periods of constant coolant conditions, measured cladding elongation is indicative of the elongation of only the fuel rods; however, under changing coolant conditions, the measured cladding elongation is a combination of the elongations of the fuel rods and the stay rods.

Second, the reported centerline temperatures are actually the temperatures of the fuel annulus where the thermocouple is located. The centerline temperatures will be higher for solid fuel pellets. The difference in radial

TABLE 3. Burnup for Instrumented Fuel Assembly (IFA)-527

Rod Number	Position ^(a)	Calculated Local Burnup, Mwd/MTM	
		Before Rod Failure (August 15, 1980)	End of Irradiation (April 8, 1981)
1	Upper	545	1338
	Lower	381	978
2	Upper	515	1265
	Lower	347	894
3	Upper	475	1168
	Lower	347	813
4	Upper	460	1135
	Lower	312	810
5	Upper	487	1202
	Lower	343	887
6	Upper	546	1345
	Lower	389	1003

(a) The designations "upper" and "lower" refer to the elevation of the upper and lower thermocouple tips.

temperature profile due to the thermocouple hole is illustrated in Figure 6. The difference in temperature with and without a thermocouple hole as a function of power is presented in Figure 7. For IFA-527, 75% of the fuel length did not contain a thermocouple hole.

Third, any measurement is subject to some degree of uncertainty. The estimated uncertainties for the data presented here are:⁽¹¹⁾ 2% for centerline temperature, 1% for cladding elongation, 2% for gas pressure, and 6% for local linear heat rate. These uncertainties are quoted at the 2σ (95%) confidence level. Thermocouple decalibration due to neutron irradiation-induced transmutations can increase the uncertainty of the temperature measurements; however, because of the low exposure of IFA-527, thermocouple decalibration should have had a negligible effect.

A previous report⁽⁸⁾ discussed the behavior of IFA-527 during the first cycle and the first few days of the second cycle. During the first cycle all rods behaved in what is considered to be a normal manner for xenon-filled fuel rods. There were no signals of impending fuel rod failure as indicated by anomalous readings from the fuel centerline thermocouples, cladding elongation sensors, and gas pressure sensors. Rods 1 through 5 (nominally identical except for peak power differences because of the flux tilt across the assembly) demonstrated consistent and highly similar behavior.

FIGURE 4. Halden Power History During the Irradiation of IFA-527

FIGURE 5. Coolant Temperature History During the Irradiation of IFA-527

FIGURE 6. Comparison of Calculated Radial Temperature Profile for Xenon-Filled Fuel Rods With and Without Thermocouple Hole at 20 kW/m

Rods 1 through 5 did experience a centerline temperature rise at constant linear heat rate during the first few days at power. This temperature increase was attributed to densification and can be somewhat correlated with the peak temperature attained by the rod. The observed increase in centerline temperature is plotted as a function of the initial attained centerline temperature in Figure 8.

The observed centerline temperature versus linear heat response was remarkably similar for rods 1 through 5 (see Figure 9). This consistency of behavior continued throughout the first operating cycle, with a slight increase in the band width by the end of the cycle. Since xenon fill gas was expected to amplify temperature differences caused by differences in fuel cracking and relocation, the observed narrow temperature band indicates that there was little difference in the effect of fuel cracking and relocation experienced by these rods. Similar behavior has been seen for helium-filled fuel rods.⁽¹²⁾

FIGURE 7. Calculated Increase in Centerline Temperature Relative to Measured Centerline Temperature for Fuel Regions with No Thermocouple Hole

Fuel cracking and relocation reduces fuel centerline temperatures by reducing the fuel-cladding gap and thereby improving the heat transfer from the fuel to the cladding. In contrast to previous xenon-filled rods operated at higher powers, fuel temperatures for rods 1 through 5 did not greatly decrease during the first operating cycle. Centerline temperatures as a function of local linear heat rate for rod 1 are plotted for two different dates in Figure 10. It is speculated that fuel relocation may prove to be more dependent on attained power level than has been previously assumed. This hypothesis is based on the observation that although the helium-filled and xenon-filled fuel rods operated at approximately the same centerline temperature the higher power helium-filled rods apparently underwent more fuel relocation as indicated by centerline temperature decreases at constant power.

Because of its small fuel-cladding gap, rod 6 had significantly lower centerline temperatures than rods 1 through 5. In fact, the observed temperatures were approximately the same as those for the small-gap helium-filled fuel rods

FIGURE 8. Temperature Increments Experienced During First Full-Power Hold Period Versus Initial Temperature at the Start of the Hold (1L, 1U, etc., refer to rod number and thermocouple position)

in IFA-431 and IFA-432,(2,3) which leads to the supposition that solid contact conductance is dominating heat transfer in both rod types and that the gas mixture has little effect under this condition.^(a) As would also be expected, the cladding elongation for rod 6 was much greater than that for rods 1 through 5.

(a) It is also possible that rod 6 suffered water ingress prior to the first cycle, which might explain both its pressure transducer failure and its low indicated fuel temperatures.

FIGURE 9. Example of Observed Similarity of Centerline Temperature for Rods 1 Through 5 of IFA-527

The remainder of this section presents the data on a rod-by-rod basis. Data for December 23-29, 1980, was lost because of a failure by the computer collecting the data, thus the smooth operating history seen for those days is not real. Instrument failure dates are summarized in Table 4.

FIGURE 10. Lower Thermocouple Readings for Rod 1 Versus Linear Heat Rating on July 8 and August 12, 1980

TABLE 4. Instrument Failures for IFA-527

Instrument	Date of Failure
Rod 2 upper thermocouple	After July 4, 1980
Rod 5 upper thermocouple	Beginning of second cycle (September 10, 1980)
Rod 5 pressure transducer	Beginning of third cycle (December 12, 1980)
Rod 6 pressure transducer	Beginning of first cycle (July 1, 1980)

ROD 1 DATA

Lower thermocouple data and the lower linear heat rate for rod 1 are presented in Figures 11 through 13; upper thermocouple data and the upper linear heat rate, in Figures 14 through 16; and cladding elongation data and the rod average linear heat rate, in Figures 17 through 19. Gas pressure data and the rod average linear heat rate for the first cycle are presented in Figure 20. Gas pressure data is not presented for cycles 2 and 3 because the pressure transducer continually registered reactor system pressure until apparent instrument failure near the end of the third cycle. Rod 1 was removed from the assembly at the end of the third cycle for future PIE.

Centerline temperatures were significantly lower at the start of the second cycle because of the presence of steam in the rod. The temperatures registered by the upper thermocouple were extremely low during the second and third cycles; even lower than rod 6 during the first cycle (compare Figures 15 and 72). The lower thermocouple of rod 1 also registered these extremely low temperatures during the third cycle. It has not been concluded if the recorded temperatures are truly representative of the centerline temperature.

An approximately step increase in zero-power cladding elongation was seen for rod 1 at the start of the first and second cycles. This elongation increased slightly during the first and second cycles.

FIGURE 11. Lower Centerline Temperature and Linear Heat Rate for Rod 1 of IFA-527 During First Operating Cycle

FIGURE 12. Lower Centerline Temperature and Linear Heat Rate for Rod 1 of IFA-527 During Second Operating Cycle

FIGURE 13. Lower Centerline Temperature and Linear Heat Rate for Rod 1 of IFA-527 During Third Operating Cycle

FIGURE 14. Upper Centerline Temperature and Linear Heat Rate for Rod 1 of IFA-527 During First Operating Cycle

FIGURE 15. Upper Centerline Temperature and Linear Heat Rate for Rod 1 of IFA-527 During Second Operating Cycle

FIGURE 16. Upper Centerline Temperature and Linear Heat Rate for Rod 1 of IFA-527 During Third Operating Cycle

FIGURE 17. Cladding Elongation and Rod Average Linear Heat Rate for Rod 1 of IFA-527 During First Operating Cycle

FIGURE 18. Cladding Elongation and Rod Average Linear Heat Rate for Rod 1 of IFA-527 During Second Operating Cycle

FIGURE 19. Cladding Elongation and Rod Average Linear Heat Rate for Rod 1 of IFA-527 During Third Operating Cycle

FIGURE 20. Gas Pressure and Rod Average Linear Heat Rate for Rod 1 of IFA-527 During First Operating Cycle

ROD 2 DATA

Lower thermocouple data and the lower linear heat rate for rod 2 are presented in Figures 21 through 23; upper thermocouple data and the upper linear heat rate for the first cycle, in Figure 24. The upper thermocouple failed after only four days of operation; Figures 25 and 26 indicate the upper linear heat rate for cycles 2 through 4. Cladding elongation data and the rod average heat rate are presented in Figures 27 through 29, and gas pressure data and the rod average linear heat rate are presented in Figures 30 through 32.

At the start of the second cycle, rod 2 registered an increased gas pressure and a decreased centerline temperature. In contrast to rod 1, the gas pressure did not reach system pressure and began decreasing after two days of operation; the pressure had returned to prefailure levels after four days. Correspondingly, the temperature registered by the lower thermocouple at the start of the second cycle was low compared to the first cycle but also returned to prefailure levels after four days. This behavior indicates that some steam was present at the start of the second cycle but that the rod dried out and resealed for the remainder of the cycle.

At the start of the third cycle, the rod gas pressure went to system pressure but decreased after four days of operation. Although the pressure continued to decrease, it never returned to prefailure levels. The rod pressure returned to system pressure at the end of the third cycle. Cladding elongation for rod 2 showed increases in the zero-power elongation at the start of each cycle. The recorded temperatures at the lower thermocouple during this cycle were indicative of the presence of steam in the rod. The centerline temperatures recorded by the lower thermocouple during the fourth cycle were very low--equivalent to those for rod 1.

Although rod 2 may have resealed at the start of the second cycle, the rod definitely failed prior to the third cycle.

FIGURE 21. Lower Centerline Temperature and Linear Heat Rate for Rod 2 of IFA-527 During First Operating Cycle

FIGURE 22. Lower Centerline Temperature and Linear Heat Rate for Rod 2 of IFA-527 During Second Operating Cycle

FIGURE 23. Lower Centerline Temperature and Linear Heat Rate for Rod 2 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 24. Upper Centerline Temperature and Linear Heat Rate for Rod 2 of IFA-527 During First Operating Cycle

FIGURE 25. Upper Linear Heat Rate for Rod 2 of IFA-527 During Second Operating Cycle

FIGURE 26. Upper Linear Heat Rate for Rod 2 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 27. Cladding Elongation and Rod Average Linear Heat Rate for Rod 2 of IFA-527 During First Operating Cycle

FIGURE 28. Cladding Elongation and Rod Average Linear Heat Rate for Rod 2 of IFA-527 During Second Operating Cycle

FIGURE 29. Cladding Elongation and Rod Average Linear Heat Rate for Rod 2 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 30. Gas Pressure and Rod Average Linear Heat Rate for Rod 2 of IFA-527 During First Operating Cycle

FIGURE 31. Gas Pressure and Rod Average Linear Heat Rate for Rod 2 of IFA-527 During Second Operating Cycle

FIGURE 32. Gas Pressure and Rod Average Linear Heat Rate for Rod 2 of IFA-527 During Third and Fourth Operating Cycles

ROD 3 DATA

Lower thermocouple data and the lower linear heat rate for rod 3 are presented in Figures 33 through 35; upper thermocouple data and the upper linear heat rate, in Figures 36 through 38; cladding elongation data and the rod average linear heat rate, in Figures 39 through 41; and rod gas pressure data and the rod average linear heat rate, in Figures 42 through 44.

Rod 3 demonstrated no unusual behavior during the second cycle, and there were no signals of failure and water ingress. Rod gas pressure was equivalent to that observed during the first cycle. Zero-power cladding elongation increased at the start of the first and second cycles.

The pressure transducer for rod 3 registered some significant increases in pressure at the start of the third cycle, with a return to first and second cycle levels after 10 days of operation. Centerline temperatures were reduced at the beginning of the third cycle with a return to first and second cycle values after the first three days. This same behavior occurred during the fourth cycle but with lower temperatures than in the third cycle. The elongation sensor demonstrated some erratic behavior at the end of the third cycle.

It has been concluded that some water ingress occurred prior to the third cycle, with dry-out and resealing during the third cycle. A significant amount of water entered prior to the fourth cycle, indicating a more definite failure. No extremely low temperatures (as seen for rods 1 and 2) were observed for rod 3.

FIGURE 33. Lower Centerline Temperature and Linear Heat Rate for Rod 3 of IFA-527 During First Operating Cycle

FIGURE 34. Lower Centerline Temperature and Linear Heat Rate for Rod 3 of IFA-527 During Second Operating Cycle

FIGURE 35. Lower Centerline Temperature and Linear Heat Rate for Rod 3 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 36i. Upper Centerline Temperature and Linear Heat Rate for Rod 3 of IFA-527 During First Operating Cycle

FIGURE 37. Upper Centerline Temperature and Linear Heat Rate for Rod 3 of IFA-527 During Second Operating Cycle

FIGURE 38. Upper Centerline Temperature and Linear Heat Rate for Rod 3 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 39. Cladding Elongation and Rod Average Linear Heat Rate for Rod 3 of IFA-527 During First Operating Cycle

FIGURE 40. Cladding Elongation and Rod Average Linear Heat Rate for Rod 3 of IFA-527 During Second Operating Cycle

FIGURE 41. Cladding Elongation and Rod Average Linear Heat Rate for Rod 3 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 12. Gas Pressure and Rod Average Linear Heat Rate for Rod 3 of IFA-527 During First Operating Cycle

FIGURE 43. Gas Pressure and Rod Average Linear Heat Rate for Rod 3 of IFA-527 During Second Operating Cycle

FIGURE 44. Gas Pressure and Rod Average Linear Heat Rate for Rod 3 of IFA-527 During Third and Fourth Operating Cycles

ROD 4 DATA

Lower thermocouple data and the lower linear heat rate for rod 4 are presented in Figures 45 through 47; upper thermocouple data and the upper linear heat rate, in Figures 48 through 50; cladding elongation data and the rod average linear heat rate, in Figures 51 through 53; and rod gas pressure data and the rod average linear heat rate, in Figures 54 through 56.

The gas pressure for rod 4 increased to reactor coolant system pressure at the beginning of the second cycle. Unlike rod 1, however, the gas pressure began decreasing after six days of operation and indicated a capability to follow power changes for the remainder of the cycle. Although the pressure decreased, it did not return to prefailure levels. The zero-power cladding elongation increased at the start of the second cycle and then remained consistent throughout the remainder of the cycle. Recorded centerline temperatures decreased to a level representative of a steam-filled rod during the second cycle.

At the beginning of the third cycle the gas pressure for rod 4 again returned to system pressure. Approximately halfway through the cycle the pressure transducer registered some ability to track power changes, thus indicating an overall drop in rod gas pressure. High gas pressures were also recorded during the fourth cycle. Cladding elongation at zero power again increased at the start of the third cycle, with no unusual behavior seen during the third and fourth cycles. As demonstrated in Figure 57, the centerline temperature for rod 4 increased slightly during the third cycle.

This rod had a definite failure prior to the second cycle and never returned to prefailure conditions.

FIGURE 45. Lower Centerline Temperature and Linear Heat Rate for Rod 4 of IFA-527 During First Operating Cycle

FIGURE 46. Lower Centerline Temperature and Linear Heat Rate for Rod 4 of IFA-527 During Second Operating Cycle

FIGURE 47. Lower Centerline Temperature and Linear Heat Rate for Rod 4 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 48. Upper Centerline Temperature and Linear Heat Rate for Rod 4 of IFA-527 During First Operating Cycle

FIGURE 49. Upper Centerline Temperature and Linear Heat Rate for Rod 4 of IFA-527 During Second Operating Cycle

FIGURE 50. Upper Centerline Temperature and Linear Heat Rate for Rod 4 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 51. Cladding Elongation and Rod Average Linear Heat Rate for Rod 4 of IFA-527 During First Operating Cycle

FIGURE 52. Cladding Elongation and Rod Average Linear Heat Rate for Rod 4 of IFA-527 During Second Operating Cycle

FIGURE 53. Cladding Elongation and Rod Average Linear Heat Rate for Rod 4 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 54. Gas Pressure and Rod Average Linear Heat Rate for Rod 4 of IFA-527 During First Operating Cycle

FIGURE 55. Gas Pressure and Rod Average Linear Heat Rate for Rod 4 of IFA-527 During Second Operating Cycle

FIGURE 56. Gas Pressure and Rod Average Linear Heat Rate for Rod 4 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 57. Change in Centerline Temperature for Rod 4 of IFA-527 During Third Operating Cycle

ROD 5 DATA

Lower thermocouple data and the lower linear heat rate for rod 5 are presented in Figures 58 through 60; upper thermocouple data and the upper linear heat rate for the first cycle are presented in Figure 61. The upper thermocouple failed after the first cycle; Figures 62 and 63 indicate the upper linear heat rate for cycles 2, 3, and 4. Cladding elongation data and the rod average linear heat rate are presented in Figures 64 through 66; rod gas pressure and the rod average linear heat rate for the first two cycles, in Figures 67 and 68. No reliable pressure transducer data was obtained after the second cycle.

The gas pressure for rod 5 increased to reactor coolant system pressure at the beginning of the second cycle but began decreasing after eight days and was then observed to follow power changes (similar to rod 4). The gas pressure did not return to prefailure levels, and the pressure transducer was unreliable for the third and fourth cycles. Centerline temperatures registered by the lower thermocouple during the second cycle were representative of a steam-filled rod. Cladding elongation at zero-power increased at the start of the second cycle and then remained consistent through the remainder of the cycle.

Data collected during the third cycle showed some unusual behavior. Beginning about December 29, 1980, the cladding elongation significantly increased; concurrently, the lower thermocouple registered a centerline temperature increase. It would therefore appear that the temperature increase was responsible for the elongation behavior. However, while the temperature increase is believed to have happened, there is no immediate explanation for it, and the other rods in the assembly did not increase. The centerline temperature returned to levels expected of a steam-filled rod on January 6, 1981. The very low temperatures seen for rod 1 were also observed for rod 5 during the fourth cycle.

Rod 5 definitely failed prior to the start of the second cycle as indicated by gas pressure and centerline temperature. The observed behavior during the third and fourth cycles is indicative that the failure may have become more severe.

FIGURE 58. Lower Centerline Temperature and Linear Heat Rate for Rod 5 of I-A-527 During First Operating Cycle

FIGURE 59. Lower Centerline Temperature and Linear Heat Rate for Rod 5 of IFA-527 During Second Operating Cycle

FIGURE 60. Lower Centerline Temperature and Linear Heat Rate for Rod 5 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 61. Upper Centerline Temperature and Linear Heat Rate for Rod 5 of IFA-527 During First Operating Cycle

FIGURE 62. Upper Linear Heat Rate for Rod 5 of IFA-527 During Second Operating Cycle

FIGURE 63. Upper Linear Heat Rate for Rod 5 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 64. Cladding Elongation and Rod Average Linear Heat Rate for Rod 5 of IFA-527 During First Operating Cycle

FIGURE 65. Cladding Elongation and Rod Average Linear Heat Rate for Rod 5 of IFA-527 During Second Operating Cycle

FIGURE 66. Cladding Elongation and Rod Average Linear Heat Rate for Rod 5 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 67. Gas Pressure and Rod Average Linear Heat Rate for Rod 5 of IFA-527 During First Operating Cycle

FIGURE 68. Gas Pressure and Rod Average Linear Heat Rate for Rod 5 of IFA-527 During Second Operating Cycle

ROD 6 DATA

Lower thermocouple data and the lower linear heat rate for rod 6 are presented in Figures 69 through 71; upper thermocouple data and the upper linear heat rate, in Figures 72 through 74; and cladding elongation data and the rod average linear heat rate, in Figures 75 through 77. No reliable pressure transducer data was obtained for this rod.

As was expected because of the smaller fuel-cladding gap size (60 μm), the centerline temperatures for rod 6 during the first cycle were considerably lower than those of the other five rods. In addition, the cladding elongation at power for rod 6 was much greater. As can be seen in Figure 75, cladding elongation decreased during steady-state operation, thus indicating a relaxing of fuel and/or cladding stresses.

The behavior of rod 6 changed at the start of the second cycle relative to the first cycle. Unfortunately, because of the pressure transducer failure, a change in gas pressure could not be used to signal rod failure. However, centerline temperatures during the second, third, and fourth cycles were lower than the first cycle (see Figure 78). The recorded temperatures also became as low as those seen for rods 1, 2, and 5. In addition, the cladding elongation for rod 6 was somewhat erratic during the second cycle. No unusual cladding elongation behavior was seen during the third and fourth cycles although the elongation was low compared to the first cycle.

Apparently there was some water ingress for rod 6 prior to the second cycle as indicated by decreased centerline temperatures and erratic cladding elongation.

FIGURE 69. Lower Centerline Temperature and Linear Heat Rate for Rod 6 of IFA-527 During First Operating Cycle

FIGURE 70. Lower Centerline Temperature and Linear Heat Rate for Rod 6 of IFA-527 During Second Operating Cycle

FIGURE 71. Lower Centerline Temperature and Linear Heat Rate for Rod 6 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 72. Upper Centerline Temperature and Linear Heat Rate for Rod 6 of IFA-527 During First Operating Cycle

FIGURE 73. Upper Centerline Temperature and Linear Heat Rate for Rod 6 of IFA-527 During Second Operating Cycle

FIGURE 74. Upper Centerline Temperature and Linear Heat Rate for Rod 6 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 75. Cladding Elongation and Rod Average Linear Heat Rate for Rod 6 of IFA-527 During First Operating Cycle

FIGURE 76. Cladding Elongation and Rod Average Linear Heat Rate for Rod 6 of IFA-527 During Second Operating Cycle

FIGURE 77. Cladding Elongation and Rod Average Linear Heat Rate for Rod 6 of IFA-527 During Third and Fourth Operating Cycles

FIGURE 78. Change in Centerline Temperature During Irradiation for Rod 6 of IFA-527

FIGURE 16. Change in Temperature Distribution during Investigation of RA-227, For a Lower to

COMMENTS ON OBSERVED TEMPERATURES

Comments on the low observed centerline temperatures are provided in this section; and the observed temperatures are placed in perspective with other centerline temperature data.

CENTERLINE TEMPERATURES AFTER FAILURE

Three distinct temperature versus linear heat rate behaviors were observed during the irradiation of the xenon-filled fuel rods in IFA-527. The first behavior occurred prior to the failure of the fuel rods and was characterized by high centerline temperatures as a function of linear heat rate. After the failures, centerline temperatures decreased substantially. This second temperature behavior is considered to be representative of a fuel rod where steam has replaced the initial xenon fill gas.^(a) The third temperature behavior, which occurred after the rods had been operated in the failed condition for some time, was characterized by extremely low temperatures. These three temperature behaviors are illustrated in Figure 79, where centerline temperature versus local linear heat rate is plotted for the lower thermocouple of rod 1.

The first two temperature behaviors are considered to be normal and representative of this rod design operating first with xenon fill gas and then with steam fill gas after the rod failure.⁽⁸⁾ The problem lies with the extremely low temperatures of the third behavior. Although the temperature was observed to follow power changes, the recorded temperatures were so low as to immediately arouse suspicion. This condition occurred for the upper thermocouple of rod 1 in cycles 2 and 3, the lower thermocouple of rod 1 in cycle 3, the lower thermocouples of rods 2 and 5 in cycle 4, and the lower and upper thermocouples of rod 6 in cycles 2, 3, and 4.

A prime check on the observed low temperatures is a simple calculation of whether or not the observed temperatures are considered possible. This calculation will be based on a centerline temperature of 873K at 20 kW/cm. First, the cladding inner surface temperature (T_{ci}) may be calculated by:

$$T_{ci} = T_w + \frac{q}{\pi d_{co} h_f} + \frac{q \ln (d_{co}/d_{ci})}{2\pi K_c} = 546K$$

where T_w = coolant temperature, 513K

q = linear heat rate, 20 kW/cm

d_{co} = cladding outer diameter, 12.789 mm

d_{ci} = cladding inner diameter, 10.909 mm

(a) Steam has a higher thermal conductivity than xenon and therefore improves heat flow within the fuel rod.

FIGURE 79. Change in Centerline Temperature Behavior with Time for Rod 1 of IFA-527

h_f = coolant-cladding heat transfer coefficient, $8.0 \times 10^4 \text{ W/(m}^2\text{-K)}$
 K_c = cladding thermal conductivity, 16.0 W/(m-K)

Second, the fuel surface temperature (T_s) may be calculated using the integral conductivity method:

$$\frac{qF}{4\pi} = \int_{T_s}^{T_{CL}} K_f(T) dT$$

where F = flux depression factor, 0.835

T_{CL} = centerline temperature, 873K

$K_f(T)$ = fuel thermal conductivity, W/(m-K) =

$$7.27 - 7.89 \times 10^{-3}(T - 273) + 4.21 \times 10^{-6}(T - 273)^2 - 7.57 \times 10^{-10}(T - 273)^3$$

This calculation results in a fuel surface temperature of 565K. Therefore, the observed temperatures are possible if a temperature change across the fuel-cladding gap of approximately 20K exists.

Although it would appear that the observed temperatures are possible, some other considerations need to be kept in mind.

- The observed temperatures are less than those for a small-gap, helium-filled rod--rod 3 of IFA-432.⁽³⁾
- Both the standard- and small-gap rods of IFA-527 displayed the same temperature versus power response.
- Oxidation of the UO_2 by steam has been observed to reduce the thermal conductivity of the fuel⁽¹³⁾ and should increase temperatures.
- Swelling of the fuel would cause closure of the gap and cracks and thus reduce fuel temperatures.
- It is possible that there was some kind of "common-mode" thermocouple malfunction that resulted in registering reduced temperatures.

One scenario that could explain the low temperatures would be as follows. Halden has suggested that PIE will show that the fuel-cladding gap and the fuel cracks are filled with U_3O_8 powder from oxidation of the UO_2 . If all available space were filled with UO_2 or U_3O_8 , this could explain the lower temperatures. The thermal conductivity of UO_2 , U_3O_8 , and possible fill gases are listed in Table 5. Although the thermal conductivity of U_3O_8 is less than that of UO_2 , it is still greater than that of the gases. Assuming intimate contact between UO_2 and U_3O_8 and between the U_3O_8 and the cladding, the lower temperatures observed in IFA-527 could have come about through this mechanism.

TABLE 5. Thermal Conductivity of Fuel and Gases at 773K

Material	Thermal Conductivity, W/(m-K)
UO_2	4.3
U_3O_8	1.3(a)
Helium	0.3
Steam	0.05
Xenon	0.0013

(a) Assuming thermal conductivity of U_3O_8 is 30% of UO_2 thermal conductivity.⁽¹³⁾

TEMPERATURE COMPARISON TO OTHER RODS

The data obtained from IFA-527 provides an opportunity to compare centerline temperatures from rods operating under different conditions. One such comparison is presented in Figure 80 where centerline temperature is plotted as a function of linear heat rate. The upper curve is for rod 1 of IFA-527 prior to failure (xenon fill gas), the lower curve is for rod 1 of IFA-527 after failure (steam fill gas), and the middle curve is obtained from rod 1 of IFA-432 at a rod average burnup of approximately 19,000 MWd/MTM (this rod was saturated with fission gas at that point). As would be expected from gas thermal conductivity, the steam-filled condition resulted in the lowest temperature

FIGURE 80. Comparison of Centerline Temperatures for Fuel Rods Filled with Xenon, Steam, and Fission Gas

for the observed low powers. The other point of note is that a fission gas-filled rod with substantial burnup was observed to operate at lower temperatures^(a) than a xenon-filled rod that had been operated to low burnup at lower power. Thus, the power/temperature relationships provide an upper bound to the expected temperatures for commercial fuel.

A second comparison involves the small-gap rods in IFA-527 (xenon-filled) and IFA-432 (helium-filled) at the beginning of life. The xenon-filled rod (see Figure 81) is at slightly higher temperatures at low power. As power

FIGURE 81. Comparison of Centerline Temperatures for Small-Gap Fuel Rods Filled with Xenon and Helium at Beginning of Life

(a) The thermocouple for rod 1 of IFA-432 had significant decalibration at this burnup, which is not accounted for in Figure 80. However, accounting for this effect will only raise temperatures about 100K at 20 kW/m.

increases, the two rods approach each other in temperature, which suggests sufficient fuel-cladding contact for heat flow that the fill gas has a negligible effect under these conditions.

A second comparison involves the small-gap rods in 1FA-523 (Xenon-filled) and 1FA-482 (Helium-filled) at the beginning of life. The xenon-filled rod (see Figure 81) is at slightly higher temperatures at low power. As power

FIGURE 81. Comparison of Centerline Temperatures for Small-Gap Fuel Rods Filled with Xenon and Helium at Beginning of Life

(a) The temperature for rod 1 of 1FA-482 had significant deviation at this burning, which is not accounted for in Figure 80. However, accounting for this effect will only raise temperatures about 100K at 100 kW.

REFERENCES

1. Hann, C. R., et al. November 1977. Test Design, Precharacterization, and Fuel Assembly Fabrication for Instrumented Fuel Assemblies IFA-431 and IFA-432. NUREG/CR-0332, BNWL-1988, Pacific Northwest Laboratory, Richland, Washington.
2. Hann, C. R., et al. April 1978. Data Report for the NRC/PNL Halden Assembly IFA-431. NUREG/CR-0318, PNL-2494, Pacific Northwest Laboratory, Richland, Washington.
3. Hann, C. R., et al. August 1978. Data Report for the NRC/PNL Halden Assembly IFA-432. NUREG/CR-0560, PNL-2673, Pacific Northwest Laboratory, Richland, Washington.
4. Bradley, E. R., et al. April 1981. Data Report for the NRC/PNL Halden Assembly IFA-432: April 1978-May 1980. NUREG/CR-1950, PNL-3709, Pacific Northwest Laboratory, Richland, Washington.
5. Bradley, E. R., et al. November 1979. Precharacterization Report for Instrumented Fuel Assembly IFA-513. NUREG/CR-1077, PNL-3156, Pacific Northwest Laboratory, Richland, Washington.
6. Bradley, E. R., et al. August 1981. Data Report for the Instrumented Fuel Assembly IFA-513. NUREG CR-1838, PNL-3637, Pacific Northwest Laboratory, Richland, Washington.
7. Cunningham, M. E., et al. July 1981. Precharacterization Report for Instrumented Fuel Assembly (IFA)-527. NUREG/CR-2168, PNL-3824, Pacific Northwest Laboratory, Richland, Washington.
8. Lanning, D. D. September 1981. Beginning-of-Life Data Report for the Instrumented Fuel Assembly (IFA)-527. NUREG/CR-2167, PNL-3831, Pacific Northwest Laboratory, Richland, Washington.
9. Lanning, D. D., B. O. Barnes, and R. E. Williford. January 1979. Manifestations of Nonlinearity in Fuel Center Thermocouple Steady-State and Transient Data: Implications for Data Analysis. NUREG/CR-0220, PNL-2692, Pacific Northwest Laboratory, Richland, Washington.
10. Lanning, D. D. 1979. Use of Time-Varying Fuel Thermocouple Data to Confirm Steady-State Data Trends. Master's Thesis, University of Washington, Seattle, Washington.
11. Hann, C. R., et al. February 1977. A Method for Determining the Uncertainty of Gap Conductance Deduced from Measured Fuel Centerline Temperatures. BNWL-2091, Pacific Northwest Laboratory, Richland, Washington.

12. Cunningham, M. E., and D. D. Lanning. "Predicted Versus Measured Scatter in Centerline Temperature from Replicate Fuel Rods." ANS Trans. 39:376-377, November 30-December 3, 1981, San Francisco, California.
13. Christensen, J. A. 1968. Stoichiometry Effects in Oxide Nuclear Fuels. 1. Power Rating Required for Melting and Oxygen Redistribution in Molten Center UO_{2+x} . BNWL-536, Pacific Northwest Laboratory, Richland, Washington.

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16. ABSTRACT (200 words or less) <p> This report presents data obtained during the irradiation of the six-rod instrumented fuel assembly (IFA)-527 in the Halden Boiling Water Reactor (HBWR), Halden, Norway. This assembly is the last in 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-527 contains five identical rods with high-density stable fuel pellets and 230-μm diametral gaps and one rod with similar fuel pellets but with a 60-μm diametral gap. All six rods were xenon-filled to simulate the effects of fission gas and to enhance the observable effects of fuel cracking and relocation on fuel temperatures. The assembly operated successfully from July 1, 1980, to August 15, 1980; the reactor was then shut down until September 10, 1980. During the shutdown, at least four of the six rods suffered pressure boundary failures. Irradiation of the assembly continued with the failed rods from September 10, 1980, until April 8, 1981; the assembly was then removed from the reactor. This report presents both pre- and postfailure data for IFA-527. </p>					
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