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PELLET-CLADDING MECHANICAL INTERACTION IN HALDEN ASSEMBLY IFA-226

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August 1978

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

DEPARTMENT OF ENERGY

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PELLET-CLADDING MECHANICAL INTERACTION
IN HALDEN ASSEMBLY IFA-226

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Date Published: August 1978

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operated by
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ABSTRACT

Instrumented Fuel Assembly (IFA) 226 was a (U,PU)O₂ fueled and zircaloy clad assembly which operated in the Halden Boiling Water Reactor, Halden, Norway, to a peak burnup in excess of 40 000 MWd/t. The assembly was instrumented with fuel rod elongation sensors to continuously monitor pellet-cladding mechanical interaction (PCMI) in selected fuel rods. The elongation behavior of these rods was evaluated to assess the influence of fabrication and irradiation variables on PCMI. The PCMI behavior at startup is contrasted with PCMI at high exposures. Observations on fuel fracture, fuel conditioning, axial ratcheting, ramp rate, strain relaxation, fuel swelling, and zircaloy growth, and their effects on PCMI, are included.

SUMMARY

Fuel rod behavior is being studied by the Thermal Fuels Behavior Program of EG&G Idaho, Inc., as part of the United States Nuclear Regulatory Commission's Water Reactor Safety Research Fuel Behavior Program. Experimental data for verification of analytical models developed to predict the behavior of light water nuclear fuel rods under normal and accident conditions are being obtained from a variety of reactor experiments, including those conducted in the Halden Boiling Water Reactor (HBWR) in Halden, Norway. Instrumented Fuel Assembly 226 (IFA-226) operated in the HBWR from November 1971 to October 1975. IFA-226 consisted of mixed oxide fuel clad in zircaloy-4 and assembled into two, six-rod clusters. Except for length, the rods were of pressurized water reactor design. The fuel rods operated for approximately four years in a heavy water temperature and pressure environment of 513 K and 3.45 MPa. The assembly, which attained a peak burnup of 41 000 MWd/t, was designed to study the long-term thermal response and pellet-cladding mechanical interaction (PCMI) of pressurized water reactor, plutonium recycle fuel rods. This report addresses the PCMI of the fuel rods as revealed by the response of fuel rod elongation sensors attached to selected fuel rods.

The onset of PCMI at low power in fuel rods with small fuel-to-cladding radial gaps is shown to be the result of UO_2 fracture, which begins at rod powers as low as 3 kW/m. Observations at startup show that the power level at the onset of PCMI decreases with decreasing fuel density. This behavior is attributed to the lower fuel fracture strength and thermal conductivity in low density fuel, which result in fuel fracture at lower heat ratings. Fuel rod density and gap fabrication variables, which govern fuel rod PCMI behavior at startup, are rapidly eliminated by the conditioning effect of power ramps.

Ramp rate is seen to have an effect on the onset of PCMI and on total rod elongation. PCMI occurs at increasingly lower power, and total rod elongation increases when higher ramp rates are encountered.

Cladding strain recovery at constant elevated power early in life is too large to be caused by thermally induced fuel creep. It is attributed to fission-induced fuel creep and fuel fragment relocation under the compressive load of the cladding. Strain recovery at constant power decreases to low levels after extended operation.

Fission-product-induced fuel swelling is observed in high power regions of the rod, but is generally obscured by fuel crack and gap formation. Cladding growth due to fast neutron irradiation may account for ~11% of the observed fuel rod permanent axial deformation at end of life.

At constant fuel density, the largest axial plastic strain occurs in fuel rods with the largest initial gap. This behavior is present from the first power ramp through the end of life. Apparently, it is easier for the fuel fragments to settle (relocate downward) during power reductions in large gap rods. Axial ratcheting, the incremental increase in rod elongation associated with each power ramp, has been noted to gradually decrease with time. A sudden increase in ratcheting for one rod following very high power operation reveals that the phenomenon likely results from the fracture and relocation of fuel chips into the radial gap, or from additional pellet fragment settling.

CONTENTS

ACKNOWLEDGMENTS	ii
ABSTRACT	iii
SUMMARY	iv
I. INTRODUCTION	1
II. EXPERIMENT DESCRIPTION	5
1. IFA-226	5
2. HALDEN BOILING WATER REACTOR	10
III. PELLET-CLADDING MECHANICAL INTERACTION IN IFA-226	11
1. EARLY LIFE PCMI	11
1.1 Ideal Fuel Rod Behavior	11
1.2 PCMI at Startup	13
1.3 Fuel Density and Gap Effects on the Onset of PCMI . .	15
1.4 Fuel Rod Conditioning	17
1.5 Strain Relaxation	21
2. PCMI DURING EXTENDED OPERATION	25
2.1 Time-Dependent PCMI	25
2.2 Effect of Ramp Rate	30
2.3 Fuel Swelling	30
2.4 Zircaloy Growth Due To Fast Neutron Irradiation	34
2.5 PCMI at High Burnup	35
IV. CONCLUSIONS	37
V. REFERENCES	39

FIGURES

1. Diagram of Instrumented Fuel Assembly 226	6
2. Schematic of a cladding elongation sensor	9

3.	PCMI in an ideal fuel rod	12
4.	PCMI in IFA-226, Rod AM	13
5.	Rod AM elongation versus power, first power ramp.	15
6.	Effect of fuel density on the onset of PCMI	16
7.	Effect of density on cladding elongation, first power ramp	17
8.	Effect of diametral gap on the onset of PCMI	18
9.	Fuel rod elongation curves, first and sixth power ramps	19
10.	Cladding elongation versus up-ramp number	20
11.	Fuel rod elongation curves, first and sixth power ramps	21
12.	Cladding elongation versus up-ramp number	22
13.	Cladding strain recovery in IFA-226, first power ramp	23
14.	Cladding strain recovery in IFA-226, December 1971	24
15.	Permanent elongation during extended operation	26
16.	Permanent axial deformation after the first power ramp	27
17.	Fuel pellet cocking in large gap rods	28
18.	Postirradiation neutron radiograph of Rod AC	29
19.	Effect of ramp rate on PCMI	31
20.	Effect of ramp rate on total fuel rod elongation	32
21.	Calculated cladding elongation as a result of fuel swelling . .	33
22.	Calculated cladding axial growth as a result of fast neutron irradiation	35
23.	PCMI behavior at high burnup	36

TABLE

I.	IFA-226 Test Rod Dimensions	7
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PELLET-CLADDING MECHANICAL INTERACTION

IN HALDEN ASSEMBLY IFA-226

I. INTRODUCTION

The Thermal Fuels Behavior Program of EG&G Idaho, Inc., is conducting fuel behavior irradiation tests for the United States Nuclear Regulatory Commission's (USNRC) Water Reactor Safety Research Fuel Behavior Program. These tests are intended to improve the understanding of nuclear fuel rod behavior under steady state and transient conditions. Experimental data from these tests are used for verification of analytical models developed to predict the behavior of nuclear fuel rods under normal and postulated accident conditions.

Steady state tests are being conducted in the Halden Boiling Water Reactor (HBWR) located in Halden, Norway. Instrumented Fuel Assembly 226 (IFA-226) is one of the USNRC experiments which was operated in the HBWR. IFA-226 was originally fabricated by Nuclear Fuel Services, Inc., of Rockville, Maryland. The experiment, which began irradiation in November 1971, was purchased by the USNRC in 1974 and continued operation until October 1975. IFA-226 was a mixed oxide fueled, zircaloy clad assembly consisting of twelve rods with varying initial pellet-to-cladding gap sizes and fuel densities. The experiment, which was instrumented with cladding elongation sensors, fuel centerline thermocouples, and rod internal pressure transducers, was designed to study:

- (1) Fuel pellet-cladding mechanical interactions as a function of power, burnup, fuel density, and initial pellet-to-cladding diametral gap
- (2) Fuel density, pellet-to-cladding gap, and temperature effects on fission gas release and rod internal pressure

- (3) Pellet-to-cladding gap, fuel density, and irradiation effects on the temperature of mixed oxide fuels
- (4) Fuel rod axial strains as functions of power and burnup.

This report presents an analysis of pellet-cladding mechanical interaction in the assembly, Items (1) and (4) above, during startup and later in life, as revealed by the cladding elongation behavior.

Fuel elements operating in a nuclear reactor deform in numerous and varied ways. The deformation may be measured after the fact by a thorough postirradiation examination (PIE). PIE techniques have been developed to measure the typical fuel rod deformations which occur during reactor operation; for example, diametral swelling, collapse, ovality, fuel rod lengthening or shortening, magnitude of local cladding ridges, and fuel rod bowing. In addition, neutron radiography techniques permit the examination of the intact fuel and cladding, and allow the measurement of fuel column lengthening, fuel cracking and restructuring, and cladding hydriding.

These postirradiation techniques are extremely accurate and quite useful, but in many instances the fuel rod dynamic behavior must also be understood. For example, fuel designers want to know if and when the initial fuel-to-cladding radial gap closes and what the closure mechanism(s) (fuel cracking, fuel expansion, cladding collapse, . . .) might be. Fuel rod behavior during and after a power ramp is also extremely important and the designer needs information which specifies how much elastic and plastic deformation (resulting from fuel-cladding contact) will result from a given power ramp, and how the deformation will be influenced by ramp rate, initial gap, burnup, and other variables.

These dynamic fuel rod deformations result from the action of the fuel on the cladding, from deformation of the cladding itself,

or, more commonly, from a synergistic combination of these effects. Fuel thermal expansion, on the order of 0.5% at normal operating temperatures, will not, by itself, produce fuel-cladding contact. The extreme temperature gradient in operating fuel will produce nonuniform thermal expansion. The hot interior core of the fuel expands more than the fuel periphery. The increased diametral expansion at the ends ("hourglassing") may produce cladding circumferential ridges at pellet interfaces in high power rods. Fuel swelling also contributes to closure of the pellet-to-cladding gap. Fuel swelling results from the formation of fission products, both solid and gaseous, within the fuel. The extent of swelling depends on numerous factors including the chemical and physical state of the fission products, the temperature and stress distributions within the fuel, and on irradiation parameters such as flux and exposure. Volume swelling will vary markedly, but is on the order of 1% near the end of life of commercial fuel.

However, fuel rod gap closure generally occurs at rod powers too low to be accounted for by fuel thermal expansion and swelling. This closure is assumed to be the result of the stochastic behavior of the fuel upon initial heatup. Thermal stresses in the fuel produce (primarily) radial cracks during initial reactor startup. These pellet fragments may relocate, move radially, and lock together. The general result is an increase in effective fuel volume which can continue with subsequent power increases.

The fuel rod cladding (zircaloy in light water reactors) itself deforms during irradiation. The axial and diametral thermal expansions are relatively small and are approximately 0.2% at typical cladding temperatures. If the cladding is subjected to a constant stress below the zircaloy yield point, for example, by fuel-cladding contact, the cladding will creep. The creep rate is a complex function of time, temperature, flux, structure, and stress level. Strains of several percent can be obtained due to creep. At stress levels above the yield point, the material plastically deforms. The deformation may be anisotropic as a result of the preferred

crystalline orientation (texture) introduced into the cladding during manufacturing.

Irradiation growth occurs in zircaloy as a result of lattice damage caused by fast neutron (>1 MeV) bombardment. Typical axial irradiation growth values are 0.3 to 0.4% at the end of the fuel rod life.

All of these effects influence PCMI to varying degrees, depending on fuel rod design and exposure. Early in life, the burnup effects are insignificant and the thermal-mechanical effects (thermal expansion, fuel cracking, and lockup) govern fuel rod behavior. On startup, the fuel is largely uncracked (except for some minor damage resulting from rod loading and handling). Fuel pellets are randomly stacked in the rods so that radial gaps vary azimuthally from zero to the diametral gap design value. Pellet cocking will be present, resulting from end face nonperpendicularity. Fuel column locking may even exist due to pellet cocking, gravity settling, and the fuel column weight. Thus, the PCMI behavior on startup and during the first several power ramps will depend, in part, on the random arrangement of fuel pellets in the rod. After some period of irradiation during which numerous power ramps have occurred, the fuel will have been conditioned; that is, thoroughly cracked and relocated. The burnup-dependent variables will then play an increasingly important role in PCMI.

This report interprets the dynamic pellet-cladding mechanical interaction behavior of the fuel rods in IFA-226. Section II is a brief description of IFA-226 and the Halden reactor. Section III uses the fuel rod elongation data of Halden Assembly IFA-226 to assess the causes of pellet-cladding mechanical interaction at various stages of irradiation.

II. EXPERIMENT DESCRIPTION

This section presents a general description of IFA-226 and the HBWR. A detailed discussion of the former may be found in Reference 1.

1. IFA-266

Instrumented Fuel Assembly 226 consisted of a vertically stacked train of two clusters of six fuel rods each. Each cluster was about 94.5 cm long and 7.1 cm in diameter. The rods within each cluster were arranged in a hexagonal lattice. A diagram of IFA-226 is presented in Figure 1. The dimensions, except length, are typical of pressurized water reactor fuel. Table I summarizes the test variables and the fuel rod instrumentation.

The test assembly was cooled by natural circulation of heavy water. When the assembly was operating at high, steady state power levels, the nominal inlet flow rate of the coolant was about 2 l/s, and the coolant inlet temperature was about 510 K. The nominal coolant pressure and the saturation temperature were 3.45 MPa and 513 K, respectively.

IFA-226 contained mixed oxide fuel (9.5 wt% PuO_2) which was pressed and sintered into pellet form and clad with zircaloy-4. Complete fabrication history and preirradiation characterization data may be found in Reference 2.

IFA-226 fuel rods were instrumented with three types of sensors: internal pressure transducers, fuel centerline thermocouples, and cladding elongation sensors. Table I summarizes the type of instruments used on each rod. This report addresses only the pellet-cladding mechanical interaction behavior of IFA-226 as measured primarily by the elongation sensors.

Fig. 1 Diagram of Instrumented Fuel Assembly 226.

TABLE I

IFA-226 TEST ROD DIMENSIONS

		Cladding ^[a]		Fuel ^[b]				Instrumentation			Comment
		Outside Diameter	Inside Diameter	Density ^[c] (% theoretical)	Diameter (mm)	Stack Length (mm)	Gap ^[d] (mm)	Fuel Centerline ^[e] Thermocouple	Pressure Transducer	Elongation Sensor	Elongation Sensor Failure Date
Upper Cluster	AA	10.69	9.51	92.06	9.30	653	0.21	X	X		
	AB	10.64	9.52	91.50	9.30	650	0.21		X		
	AC	10.66	9.51	90.58	9.30	650	0.21		X		
	AD	10.65	9.52	95.10	9.30	663	0.21		X		
	AE	10.68	9.51	91.64	9.26	655	0.25	X	X		
	AG	10.68	9.50	94.33	9.40	663	0.10			X	NA
Lower Cluster	AH	10.69	9.49	94.46	9.40	607	0.09			X	3/75
	AJ	10.67	9.50	94.43	9.35	610	0.14			X	NA
	AK	10.66	9.51	95.94	9.31	607	0.20	X		X	1/75
	At thermo-couple location	10.66	9.51	96.10	9.30		0.21				
	AM	10.68	9.50	94.52	9.30	607	0.20			X	7/73
	AO	10.66	9.52	92.11	9.30	615	0.21	X		X	5/72
	At thermo-couple location	10.66	9.51	92.04	9.30		0.21				
	B-5	10.76	9.51	94.18	9.26	612	0.25				

[a] All cladding made of zircaloy-4.

[b] Fuel pellet length = 15.0 mm
Dish depth = 0.33 mm
Dish shoulder = 1.70 mm
Dish spherical radius = 6.79 mm.

[c] The values listed here are measured geometric densities (that is, weight divided by volume). The density of a limited number of fuel samples was also measured by immersion techniques. The immersion (true) densities were found to be about 0.75% higher at all density levels.

[d] Backfill helium pressure = 1.01×10^4 N/m².

[e] Centerline thermocouple hole diameter = 1.73 mm.

The elongation sensors were linear variable differential transformers (LVDT) consisting of coaxial primary and secondary coils with a movable ferritic core as shown in Figure 2. The core, which was spring loaded against the end of the fuel rod, was displaced by axial cladding movement. Displacement of the core influenced the voltage induced in the secondary coils and provided a mV signal. The elongation sensors were concentrated in the bottom cluster because the neutron flux in IFA-226 was skewed toward the bottom of the core, and, therefore, the highest power and the hottest pellets were located in the lower cluster, near the top of the fuel stack. The HBWR data system automatically recorded the fuel rod instrumentation approximately every fifteen minutes and stored the data on magnetic type.

A complete description of all of the IFA-226 instrumentation, including preirradiation calibration results, is contained in Reference 2.

IFA-226 began irradiation in November 1971. Irradiation was terminated in October 1975 after average burnups of 24 886 MWd/t^[a] and 37 431 MWd/t had been reached in the upper and lower clusters, respectively. During the four-year irradiation period, the assembly underwent over 300 power ramps with assembly powers as high as 300 kW and rod average powers in excess of 40 kW/m.

Flux profiles within the assembly were determined by fitting a Fourier sine series to the eight axial and radial neutron detectors located within the assembly. Rod powers were then calculated using the cluster average neutron detector signal and the thermal-hydraulically measured power calibration constant. Fuel rod powers were corrected for burnup depletion effects, radial peaking, and variations in fuel rod design. The Halden data calculational techniques are described in detail in Reference 1.

[a] All burnup values in this report are in MWd/tonne of metal.

Fig. 2 Schematic of a cladding elongation sensor.

2. HALDEN BOILING WATER REACTOR

The HBWR operates with a water temperature of 513 K and a pressure of 3.45 MPa. The core consists of about 60 assemblies. Of these assemblies, 15 to 20 are either driver assemblies or nonfueled instrumented assemblies. The remaining are test assemblies fabricated by either the Halden Project or participating agencies.

Typical reactor steady state operating conditions are:

(1) Core power	12 MW
(2) Coolant pressure	3.45 MPa
(3) Coolant inlet temperature	510 K
(4) Coolant saturation temperature	513 K
(5) Natural circulation flow rate	0.5 to 2 ℓ/s
(6) Forced circulation flow rate	2 to 4 ℓ/s
(7) Axial power peaking factor	1.2 to 1.7
(8) Thermal flux	$3 \text{ to } 3.6 \times 10^{13} \text{ n/cm}^2 \cdot \text{s}$
(9) Fast flux ($> 1 \text{ MeV}$)	$5 \times 10^{11} \text{ n/cm}^2 \cdot \text{s}$
(10) Core active length	1.7 m
(11) Maximum test assembly outside diameter	73 mm
(12) Core configuration	Hexagonal lattice.

III. PELLET-CLADDING MECHANICAL INTERACTION IN IFA-226

This section discusses pellet-cladding mechanical interaction observations in IFA-226 during two distinct phases: early in life when dynamic fuel rod behavior is unique and depends to a large extent on the fabrication variables; and during the balance of the irradiation period, extending, in this case, to a peak burnup of $>40\ 000\ \text{MWd/t}$, when the rod has been conditioned by its prior history and when burnup effects become increasingly important.

1. EARLY LIFE PCMI

The onset of irradiation significantly alters the fuel rod environment. This section will discuss the influence of early life irradiation on pellet-cladding mechanical interaction.

1.1 Ideal Fuel Rod Behavior

The PCMI behavior of an ideal fuel rod at startup is an appropriate basis on which to begin an analysis. The fuel in an ideal rod is uniform, uncracked, and stacked in the cladding so that the average gap meets the design requirements. As the first ramp to power begins, the fuel and cladding expand individually according to their thermal expansions. At some power level, greater thermal expansion of the fuel causes fuel-cladding contact. Fuel-cladding slippage occurs until the steadily increasing frictional forces lock them together. The fuel then begins to elastically deform the cladding, and diametral expansion and axial lengthening of the fuel rod occur. If the power level continues to increase, plastic cladding deformation results. If the power ramp then levels off, the cladding elongation ceases and begins to decrease as the hot interior of the fuel plastically flows to relax the cladding stresses. A subsequent reduction to low power causes the fuel, cladding, and the fuel-cladding gap to recover. However, a permanent diametral and axial strain may remain in the cladding due to both the plastic deformation during the initial power increase and to creep during the steady (constant) power operation.

Figure 3 illustrates this ideal fuel rod behavior as it would be monitored by a cladding elongation sensor attached to the fuel rod. A similar plot for IFA-226 Rod AM during the first power cycle is shown in Figure 4. As can be seen, no sharp break in the elongation curve occurs to indicate the absolute onset of fuel-cladding interaction. When the cladding elongation data are compared with the expected elongation resulting from thermal expansion (Figure 4, dotted line), fuel-cladding contact is seen to initiate at ~ 10 kW/m rod peak power.

Fig. 3 PCMI in an ideal fuel rod.

Fig. 4 PCMI in IFA-226, Rod AM.

1.2 PCMI at Startup

Fuel cracking and consequent outward relocation is the reason for rapid fuel-cladding contact. Radial cracks are the primary fuel fracture mode. They result from tensile stresses (σ_θ) induced in the fuel pellet by differential thermal expansion caused by the parabolic temperature gradient. The maximum stress is^[3]

$$\delta r_{\max} = \frac{\alpha E P}{8\pi(1-\nu)k}$$

where, using typical values for UO_2 ^[a],

[a] For the purposes of this report, the differences in the properties of UO_2 and $(U,Pu)O_2$ fuels are assumed to be insignificant.

α = thermal expansion coefficient = $10^{-5}/\text{K}$

E = Young's modulus = 2×10^{11} Pa

P = linear power

ν = Poisson's ratio = 0.314

k = average UO_2 thermal conductivity = 0.03
W/cm $\cdot$ K.

Fuel centerline thermocouples indicated fuel temperatures of about 673 K at startup. The fracture strength of 94.5% dense polycrystalline UO_2 at 673 K is $\sim 1.09 \times 10^8$ Pa^[4]. Rearrangement of the previous equation and subsequent solution for the linear power, P , necessary to crack the UO_2 , results in $P = 3$ kW/m and clearly indicates that the fuel begins to fracture at low power.

Figure 4 reveals that fuel-cladding contact, as indicated by the cladding elongation behavior, began at ~ 10 kW/m rod peak power. This power is somewhat higher than the onset of fuel fracture (~ 3 kW/m) because the fuel pellets must increase their volume by $\sim 4\%$ by radial relocation of fuel fragments before their effective radius is large enough to induce PCMI.

Figure 5 reveals that cladding elongation does not follow the fuel thermal expansion immediately upon fuel-cladding contact. Rather, the elongation steadily increases until, at about 28 kW/m, the cladding elongation approximately follows the fuel thermal expansion. This behavior probably results from fuel-cladding slippage. At contact, the fuel-cladding interfacial pressure is too small to cause lockup. Instead, the fuel slides past the cladding, imparting only a portion of the fuel axial strain to the cladding. Finally, the interfacial pressure increases to such a degree that

Fig. 5 Rod AM elongation versus power, first power ramp.

lockup occurs. The fuel and cladding then begin to elongate together. The initial fuel-to-cladding contact may be termed "soft interaction," whereas when the interfacial pressure increases to the point at which the fuel-induced cladding elongation results, "hard interaction" is said to have developed.

1.3 Fuel Density and Gap Effects on the Onset of PCMI

The effect of fuel density on the onset of PCMI (soft interaction) during the first power ramp is shown in Figure 6. This effect is due to a small decrease in fuel fracture strength and a large decrease in fuel thermal conductivity with increased porosity. As a result, the thermal stresses necessary to crack the fuel and initiate PCMI are reached at a substantially lower linear heat generation rate in low density fuel.

Fig. 6 Effect of fuel density on the onset of PCMI.

The effect of density on cladding elongation during the first power ramp is shown in Figure 7. For Rods AM and AK, the slope (the axial strain rate) begins to decrease at 40 kW/m. However, the Rod AO fuel column shortening begins at ~ 35 kW/m. This behavior is generally attributed to creep; specifically the hot pellet interior plastically deforming to relieve cladding axial stresses. The behavior is known to be more pronounced at lower fuel densities. An alternate explanation for this behavior is discussed in Section 1.5 under strain relaxation.

The effect of fuel-cladding diametral gap on the onset of PCMI (soft interaction) is shown in Figure 8. As expected for fuel of similar density (that is, similar thermal conductivity, fracture, and creep properties), PCMI onset increases with gap size. However, it is not a linear function of gap. Apparently, when the gap is relatively

Fig. 7 Effect of density on cladding elongation, first power ramp.

large ($\sim 200 \mu\text{m}$), gap closure requires not only fuel fracture (which occurs at quite low power), but also some radial relocation which does not occur until higher powers are attained. Whereas, when the diametral gap is small ($\sim 100 \mu\text{m}$), the slight relocation which accompanies fuel fracture is adequate to produce gap closure. Another possible reason for the very early onset of PCMI in the small gap rod (Rod AH), fuel loading interference, is discussed in Section 1.4.

1.4 Fuel Rod Conditioning

After several power ramps, the behavior of fuel rods with similar densities and varying gaps illustrates how the extent of fuel-cladding interaction stabilizes and the cladding strain rate decreases markedly. This effect is shown in Figure 9 where the elongation

Fig. 8 Effect of diametral gap on the onset of PCMI.

curves for the sixth power ramp are compared with those for the first ramp. Approximately 600 h elapsed between the first and the sixth power ramps, during which the rod powers were ramped both up and down and the rods were operated approximately 300 h at steady state high power (35.0 to 37.5 kW/m).

During the first power ramp the small gap rod (Rod AH) experienced PCMI essentially immediately. If thermal expansion only (no fuel cracking) is assumed, PCMI would not have initiated until ~ 30 kW/m. Fabrication records indicate that, due to the small gap size, some interference was encountered in loading the fuel in Rod AH. This interference may have lead to fuel pellet jamming, cocking, or cracking during loading, with consequent very early onset of PCMI.

Though the preirradiation fuel-cladding gap varies significantly in these rods (90 to 200 μm), and this difference is reflected in the

Fig. 9 Fuel rod elongation curves, first and sixth power ramps.

elongation behavior during the first power ramp, the elongation behavior after extended operation is nearly identical. This nearly identical behavior is a result of fuel rod conditioning. Extended operation results in significant fuel cracking and radial relocation of the fragments. The effect is to homogenize initial fabrication gap variables. In Figure 9 the permanent cladding axial strain (as indicated by the starting elongation for the sixth power ramp) is virtually identical for each rod, indicating that PCMI behavior at low power is similar in each case. A measure of how rapidly fuel restructuring neutralizes fabrication variables is seen in Figure 10. Here, a plot of cladding elongation for Rods AH, AJ, and AM at an arbitrary peak rod power of 35 kW/m is plotted versus the power ramp number (up ramp).

The effect of density variations on PCMI are also rapidly eliminated as shown in Figure 11. At the sixth ramp, differences

Fig. 10 Cladding elongation versus up-ramp number.

still persist in the amount of permanent axial deformation (as reflected in the starting cladding elongation for the sixth ramp), but the variations in PCMI onset have largely disappeared. Figure 12 also illustrates that density variations are eliminated as fuel conditioning proceeds, but comparison with Figure 10 reveals that density effects persist longer than gap effects. Figure 12 also reveals that elongation increases with decreasing density during the first power ramp, but during the third and sixth ramps this changes, and axial elongation is actually smallest in the low density rod, largest in the intermediate density rods, and intermediate in the high density rod. Apparently, stochastic cracking and relocation combined with the extensive creep of the very low density fuel overcomes the density variations very early in life.

Fig. 11 Fuel rod elongation curves, first and sixth power ramps.

1.5 Strain Relaxation

Strain relaxation or recovery, as evidenced by a steady decrease in cladding elongation at constant elevated power, is a common observation^[5]. The effect is often attributed to fuel thermal creep, fission enhanced fuel creep, or hot pressing of the UO_2 fuel under the compressive load of the cladding. The hot plastic central cores of the fuel pellets, which are in contact, deform to relieve the cladding tensile stress and result in a gradual cladding length decrease. Presumably, if creep is the mechanism, the fuel expands radially or deforms into cracks to accommodate the fuel column length decrease.

Strain recovery in the IFA-226 fuel rods is shown in Figure 13. The figure shows the elongation behavior of Rods AK, AM, and AO

Fig. 12 Cladding elongation versus up-ramp number.

(similar initial gap, varying fuel density) during the constant power period which followed the first power ramp. The data points are at constant power for each rod to within 0.1 kW/m. The cladding strain steadily decreases with time until basically a steady state is reached after approximately 12 h. The maximum strain rates shown in Figure 13 are $\sim 1 \times 10^{-5} \text{ h}^{-1}$ for the upper curves (Rods AK and AM) and $\sim 5 \times 10^{-4} \text{ h}^{-1}$ for Rod AO. The measured fuel centerline temperatures were high enough (1950 to 2100 K) for thermally induced fuel creep to produce these strain recoveries. However, these rods contained dished pellets; thus, only the pellet shoulders, at temperatures substantially less than 2000 K, were in contact.

The strain recovery of Figure 13 could be due to temperature independent fission-induced creep, which is a low temperature deformation mechanism depending only on the induced stress in the fuel and on

Fig. 13 Cladding strain recovery in IFA-226, first power ramp.

the fission rate. This mechanism generally produces strain rates of approximately 10^{-5} h^{-1} . In addition to creep, the strain recovery of Figure 13 may be aided by fuel relocation. At constant power some of the fuel fragments may squeeze back together due to local plastic (or creep) deformation of the fragment surfaces. As the fuel fragments reform, the fuel crack volume and the fuel-to-cladding contact pressure (friction) will be reduced, allowing slippage of the fuel column with respect to the cladding. The reduced strength and increased porosity of the low density fuel in Rod AO would explain the more pronounced relaxation in this rod compared with that of Rods AK and AM.

Strain recovery curves taken about one month later, after the seventh major power ramp, are shown in Figure 14. The steady state power level is about 8% lower than that of the first ramp; therefore,

Fig. 14 Cladding strain recovery in IFA-226, December 1971.

the rod strain at the onset of relaxation is less. Less than 0.05 mm relaxation is observed, which is substantially less than that of Figure 13. The relaxation observed in Figure 14 may result, in part, from fuel creep or hot pressing. Again, fuel relocation may also contribute to the strain recovery. Similar small relaxation (< 0.05 mm) continued to be measured throughout life during periods of constant power operation.

2. PCMI DURING EXTENDED OPERATION

This section extends the discussion of PCMI in IFA-226 to late in life; that is, to high burnups. Some of the effects observed late in life may be attributed to burnup (fission) dependence. Other effects, however, may be more subtle. For example, the relationship, discussed in Section III-1.4, between ramp number and the onset of PCMI was attributed to the large number of fuel cracks which developed during or after each power ramp. Such behavior is not, in the strictest sense, burnup dependent. Instead, it depends on the thermal history of the fuel. Thus, this section should be viewed as discussing time-dependent fuel rod PCMI behavior.

2.1 Time-Dependent PCMI

Fuel rod elongation measured until the end of life, or elongation sensor failure, is shown in Figure 15. Contrary to what might be expected, the greatest permanent axial strain, at constant pellet density, is found in the fuel rod with the largest initial gap. This behavior probably results from the propensity of the fuel to crack during operation, with subsequent movement of fuel chips which lock the fuel to the cladding. Fuel relocation probably occurs after a power ramp when the fuel fragments settle, by gravity. Subsequent power ramps produce additional fuel-cladding lockup and axial elongation and this ratcheting process continues. Figure 16 reveals that this behavior (greatest permanent deformation occurring in large gap rods) is present even during the first power ramp.

Another possible explanation of the greatest strain occurring in the largest gap rod is that of fuel pellet cocking. This explanation suggests that in a large gap rod more room exists for the pellets to stack offset during loading, shipping, and handling as shown in Figure 17. Thus, the mass of the pellet stack and the force exerted

Fig. 15 Permanent elongation during extended operation.

Fig. 16 Permanent axial deformation after the first power ramp.

on the pellet stack by the plenum spring combine with the small area of fuel-cladding contact (pellet shoulders only in dished pellets) to lock the fuel stack in a position in which numerous pellets are cocked and wedged against the cladding. As a consequence, hard interaction occurs shortly after startup. However, PCMI behavior during the initial ramp disputes this conclusion. Figure 9 reveals that PCMI occurs at lower power and is larger in the smaller gap rods. Similar behavior was found in another study^[6] in which 95% dense fuel with initial gaps of 51 and 380 μm were tested. PCMI, as indicated by cladding elongation, occurred in the small gap rod at ~ 3 kW/m and in

INEL-A-6320

Fig. 17 Fuel pellet cocking in large gap rods.

the large gap rod did not occur until ~ 2600 MWd/t at a power of ~ 25 kW/m. Fuel cocking and jamming occur during loading, but apparently, except under certain circumstances (such as the difficulty with fuel loading in Rod AH mentioned in Section III-1.4), pellet cocking itself did not lock the fuel column at startup in IFA-226. Instead, the lockup and subsequent PCMI resulted primarily from the tendency of the fuel to crack and relocate. Figure 18, a postirradiation neutron radiograph, reveals the extent to which fuel cracking and relocation can develop in an operating fuel rod.

The fuel rod behavior shown in Figure 15 reveals that axial ratcheting of the fuel rod is occurring. The ratcheting behavior shown in Figure 15 occurs with each high power ramp when cladding axial tensile stresses, induced by fuel-cladding contact, either exceed the zircaloy yield point or are high enough to produce significant cladding creep. This results in both elastic and plastic cladding deformation and, consequently, axial cladding growth. Only the elastic component of growth is recovered after the ramp. The result

Fig. 18 Postirradiation neutron radiograph of Rod AC.

is an incremental increase in length with each high power ramp. The ratcheting rate levels off with time (as shown in Figure 15), either as a result of the expanding cladding tube reducing fuel-cladding friction, or the effects of fuel swelling or fuel-to-cladding bonding maintaining the fuel-cladding contact and preventing gravity settling of the fuel during a power decrease. If pieces of fuel continued to relocate following each power down ramp, the ratcheting rate would not decrease, but would remain approximately steady throughout life. The fact that the ratcheting rate does decrease means that limits exist on fuel fracture (perhaps because the hot plastic core remains intact, or perhaps crack healing occurs during extended high temperature operation) and on fuel settling. Figure 18 shows an intact fuel pellet adjacent to a badly fractured one. The individual pellet delineations (after four years of irradiation) are clear despite extensive fracturing, clearly indicating that fuel fracture does not proceed without limit. Also visible are large interpellet gaps, indicating that gravity settling of the fuel, and consequently, ratcheting, is also limited.

In Figure 15 the cladding elongation rate in fuel Rod AH, which had begun to stabilize, begins to increase in July 1974. This time coincides with a power ramp which resulted in the highest peak linear heat rating (49.9 kW/m) attained by Rod AH during the irradiation history up to that time. Even though both the up and down ramps were gradual and the peak linear heat rating exceeded the previous maximum by <1 kW/m, it is believed that as a result of additional fuel fragment settling or a large fuel chip(s) relocating into the gap during

or after this ramp, the elongation rate (ratcheting) began to increase. The ratcheting rate again began to stabilize, as suggested by the last two data points before the failure of the elongation sensor in March 1975.

2.2 Effect of Ramp Rate

The rate of power increase affects cladding elongation behavior once the fuel has been conditioned by extended operation. Three up ramps (occurring in July 1973) of varying average ramp rates were chosen for analysis. The elongation behavior of Rods AJ, AH, and AK, the only lower cluster rods whose elongation sensors were still operating at this time, were compared. Rod average burnups were 17 000 to 18 000 MWd/t.

Figure 19 reveals that increasing the ramp rate results in a substantial decrease in the power at which the onset of PCMI (soft interaction) occurs. With a lower ramp rate, the transition time between soft interaction and hard interaction is longer, and the fuel may be able to slip and relocate more, thereby delaying somewhat the cladding axial deformation.

Figure 20 reveals that ramp rate also has an effect on the total fuel rod elongation. This effect is a consequence of the decrease in PCMI onset with increasing ramp rate as described previously. After hard contact, the cladding follows the fuel elongation. If hard contact occurs at a lower power, the net elongation clearly depends on the power range over which hard contact occurs; that is, on the ramp rate.

2.3 Fuel Swelling

As exposure increases in IFA-226, cladding length changes caused by fission-product-induced fuel swelling must be considered. Fuel swelling results from the presence of gaseous fission products dispersed in the fuel matrix as small bubbles or as larger voids formed

Fig. 19 Effect of ramp rate on PCMI.

by bubble coalescence at defects or pores, or swelling occurs as a result of solid fission products dispersed in the (U,Pu) O_2 lattice or present as free metallic inclusions. Fuel swelling data have been assessed^[4] and a temperature independent model has been developed. The results show an average volumetric swelling rate of

$$0.7\% \frac{\Delta V}{V} \text{ per } 10^{20} \text{ fissions/cm}^3.$$

For an assumed average fuel density of 94%,

$$1 \text{ MWd/t} = 2.3 \times 10^{16} \text{ fissions/cm}^3 [8].$$

Thus, the average volumetric swelling rate, S_r , is

$$1.61 \times 10^{-4}\% \text{ per MWd/t.}$$

Fig. 20 Effect of ramp rate on total fuel rod elongation.

Therefore, the linear swelling rate, $\frac{S_r}{3}$, is

$$5.37 \times 10^{-5} \% \text{ per MWd/t} = S_l.$$

The fuel pellets in IFA-226 were fabricated with pellet end dishes to accommodate fuel swelling. Postirradiation neutron radiographs^[7] revealed that the dishes were only partially filled in the peak power region of the rods. If this calculation was extrapolated to high burnup, assuming no dish volume filling and that all of the axial fuel growth was imparted to the cladding, the results would be as shown in Figure 21. The elongation values of Figure 21 are substantially larger than observed in IFA-226. Several assumptions of Figure 21 are invalid. The calculation assumes no fuel creep; that is, all swelling is translated into cladding deformation,

Fig. 21 Calculated cladding elongation as a result of fuel swelling.

which is clearly incorrect. Swelling has been observed to be less in restrained fuel than in unrestrained fuel because of fuel creep and "pressure sintering" resulting from the compressive load on the fuel. Also, fuel cracks and gaps certainly form when the fuel settles during reactor shutdown, and their volume is large enough to accommodate significant amounts of fuel swelling, as can be seen by comparing the effective volume of the left-most pellet in Figure 18 with the cracked pellet adjacent to it (by comparing their respective areas in the neutron radiograph). The fractured pellet represents a 4% volume increase. Such pellets in the fuel rod would accommodate fission product swelling and effectively prevent swelling-induced cladding deformation.

2.4 Zircaloy Growth Due to Fast Neutron Irradiation

Fast neutrons (≥ 1 MeV) have a two-fold effect on zircaloy cladding. Neutron bombardment will produce vacancy — interstitial pair defects in the metal lattice. While the interstitials rapidly anneal out, the vacancies apparently form vacancy-rich zones and vacancy loops which result in cladding growth. Secondly, neutron irradiation alters the cladding mechanical properties so that the cladding mechanical response, say to fuel contact, changes during irradiation.

Zircaloy axial growth data have been investigated and an axial growth model has been developed^[4]. The axial growth is given by

$$\frac{\Delta L}{L} = A \left[\exp(240.8/T) \right] \left[\Phi t \right]^{1/2} \left[1 - 3 f_z \right] \left[1 + 2.0 CW \right]$$

where

$$A = 1.407 \times 10^{-16} (\text{neutrons/m}^3)^{-1/2}$$

$$T = \text{cladding temperature} = 513 \text{ K (assumed equal to coolant temperature)}$$

$$\Phi = \text{fast neutron flux} = 5 \times 10^{15} \text{ n/m}^2 \cdot \text{s} \\ (\text{from Section II-2})$$

$$t = \text{time} = 6.1 \times 10^7 \text{ s (actual operating time)}$$

$$f_z = \text{texture factor} = 0.05$$

$$CW = \text{cold work factor} = 0.1.$$

The results of applying this equation to IFA-226 are shown in Figure 22. The indication is that fast neutron growth can account for ~11% of the observed elongation at the end of life.

Fig. 22 Calculated cladding axial growth as a result of fast neutron irradiation.

2.5 PCMI At High Burnup

The elongation behavior of IFA-226 at high burnup provides a measure of fuel rod PCMI. Diametral interactions and local effects such as ridging are not directly measured, however. Thus, elongation is probably only an indication of rod average PCMI.

Elongation behavior at high burnup for fuel Rod AJ is shown in Figure 23. The onset of PCMI, taken as the point of departure from

Fig. 23 PCMI behavior at high burnup.

linearity of the low power portion of the curve, does not vary significantly from 6000 to 35 000 MWd/t. Similar behavior was observed for the remaining fuel rods, although there was no consistent relationship between initial fuel density or gap and the onset of PCMI at high burnup. Thus, PCMI behavior appears to be governed by stochastic fuel relocation processes which develop fairly early in life and persist to high burnup. Other fuel rods would occasionally exhibit a power ramp for which PCMI occurred earlier than previous or subsequent ramps. An effort was made to correlate these observations with reactor shutdowns, which others have surmised will advance the onset of PCMI due to extensive fuel relocation. However, no positive correlation could be made for IFA-226.

IV. CONCLUSIONS

Mixed oxide assembly IFA-226 operated in the Halden Boiling Water Reactor to peak burnups of $\sim 41\ 000$ MWd/t. The assembly was instrumented with fuel rod elongation sensors which permitted the monitoring of pellet-cladding mechanical interaction (PCMI) during irradiation.

The fabrication variables of density and radial gap influenced the magnitude and onset of PCMI. PCMI occurred earlier in small gap rods, but the magnitude of the permanent strains were larger in large gap rods. Early onset of PCMI was observed in low density fuel, presumably due to the decreased fracture strength and thermal conductivity of the fuel.

Fuel fracture and relocation during operation appear to play a major role in the PCMI behavior of fuel rods, tending to condition the rods, rapidly eliminating the effects of fabrication variables. The elongation behavior of the rods becomes nearly identical after a period of irradiation during which several large power ramps have occurred.

Fuel relocation continues to influence fuel rod behavior as exposure increases; however, the stochastic nature of relocation can alter PCMI trends as evidenced by random changes in ratcheting behavior and elongation after a shutdown and by the effects of ramp rate on PCMI. Cladding strain recovery is large during periods of early life steady state operation, presumably as a result of fuel creep, although fuel relocation under a compressive load may also be responsible. After fuel rod conditioning, however, strain recovery is small and probably results from fuel creep.

Fuel rod elongation from fission-product-induced fuel swelling is potentially significant, but, except in the high power region of the fuel rod, appears to be obscured by the presence of accommodating fuel cracks and gaps and by the ability of cladding strains to recover

during periods of reactor shutdown. Fast neutron growth of the zircaloy cladding can account for only ~11% of the observed elongation at the end of life of IFA-226.

The power at which PCMI occurred tended to remain stable at high burnup. The power could not be related to the fabrication variables of fuel density and gap, and appeared to be a result of stochastic fuel relocation processes.

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