IRRADIATION PERFORMANCE OF FULL-LENGTH METALLIC IFR FUELS*

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ANL/CP--74935

DE92 041095

July 1992

To be submitted to the Intl. Conf. on Design and Safety of Advanced Nuclear Power Plants (ANP '92), Oct. 25-29, 1992, Tokyo, Japan.

*Work supported by the U.S. Department of Energy, Office of Technology Support Programs, under Contract W-31-109-Eng-38.

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ABSTRACT

An assembly irradiation of 169 full-length U-Pu-Zr metallic fuel pins was successfully completed in FFTF to a goal burnup of 10 at.%. All test fuel pins maintained their cladding integrity during the irradiation. Postirradiation examination showed minimal fuel/cladding mechanical interaction and excellent stability of the fuel column. Fission-gas release was normal and consistent with the existing data base from irradiation testing of shorter metallic fuel pins in EBR-II.

1. INTRODUCTION

The Integral Fast Reactor (IFR) is an advanced liquid-metal-cooled reactor (ALMR) with improved reactor safety, fuel cycle economy, and environment protection[1,2,3]. These advantages are derived principally from using a metallic U-Pu-Zr alloy as the fuel material.

The test bed for the IFR concept is the Experimental Breeder Reactor-II (EBR-II) at Argonne National Laboratory. To date, over 2,500 U-Pu-Zr metallic fuel pins have been successfully irradiated, many to very high burnups.[4] The 343-mm core height for EBR-II, however, is only approximately one-third of that of the ALMR.[5] In order to apply the

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substantial EBR-II data base for the ALMR, it is imperative to establish that the difference in core height, i.e., fuel column length, will not adversely affect fuel pin performance. The central issue is whether the increased height and weight of the ALMR fuel column would cause (1) fuel compaction leading to enhanced localized fuel/cladding mechanical interaction (FCMI), or (2) impeded fission gas release leading to, again, enhanced FCMI from increased fuel swelling. To address this issue, an irradiation experiment with pins with 914-mm-long fuel was conducted in the Fast Flux Test Facility (FFTF). The results of this experiment, designated IFR-1, are the subject of this paper.

2. TEST DESCRIPTION

The IFR-1 irradiation test consisted of 169 fuel pins: 18 with U-19wt.%Pu-10wt.%Zr fuel, 19 with U-8wt.%Pu-10wt.%Zr fuel and the remainder 132 pins with U-10wt.%Zr fuel. To attain approximately the same linear power, different ^{235}U enrichments were used for the three fuel types. The cladding for all pins was 20% cold-worked D9, with a 6.86-mm OD and a 0.56-mm wall. The fuel column, 914-mm long, consisted of three stacked, equal-height, solid fuel slugs produced by injection casting. Immediately above and below the fuel column were depleted U-10wt.%Zr blanket slugs, each 165-mm long. The nominal diameter of the as-cast fuel and blanket slugs was 4.98 mm, yielding a smeared density of 75%, the same as in the reference IFR fuel design. To facilitate heat transfer across the as-built fuel/cladding gap, the slugs were sodium-bonded to the cladding. (The U-Pu-Zr metallic fuel is chemically compatible with sodium.) Because of the sodium bond, which is solid at ambient temperature, a hold-down spring was eliminated in the IFR-1 pin design. The plenum/fuel volume ratio of the pins was 1.0 at the reactor operating temperature after expansion of the sodium. Pin-to-pin spacing in the assembly was maintained by 1.37-mm-dia., 20% cold-worked D9 wires wrapped on the fuel pins at a 152-mm axial pitch.

The irradiation vehicle for the IFR-1 test was a standard FFTF open core assembly with a D9 duct. An orifice incorporated in the vehicle provided the desired sodium coolant flow. There was no built-in instrumentation in the assembly: however, the above-core instrumentation tree in FFTF provided data on the assembly coolant outlet temperature during the irradiation.

3. IRRADIATION HISTORY

The goal burnup for the IFR-1 test was 10 at.%. It was successfully achieved in five consecutive FFTF cycles, from 9A through 10B, in 620 effective-full-power days (EFPDs) without cladding breaching or anomalous events.

There was no interim reconstitution between the cycles: the same row-4 core position and assembly orientation was maintained throughout the irradiation. The peak linear power of the pins at beginning-of-life (BOL) ranged from 44 to 48 kW/m, depending on the pin's radial position. At end-of-life (EOL), the linear power decreased to 29 -33 kW/m due to fissile burnout. The peak pin burnup and fast fluence (E>0.1 MeV) at EOL were 10.1 at.% and 15.6 x 10^{22} n/cm^2, respectively. Due to the chopped-cosine flux distribution in FFTF, the power, burnup, and fluence at the ends of the fuel column were approximately half the peak values. The peak cladding ID temperatures were, respectively, 608 and 533°C at BOL and EOL.

4. TEST RESULTS

4-1. Assembly Performance

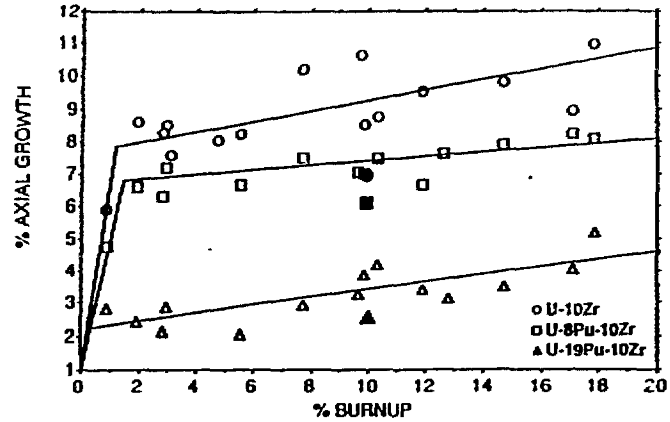
Following the reactor discharge, the IFR-1 vehicle and fuel-element bundle was examined to evaluate the condition of the assembly. Duct dilation was small, 2.7% max. flat-to-flat and 1.0% max. corner-to-corner, and within the FFTF data base.[6] There was essentially no bow or elongation of the duct. The pull force required to remove the duct from the fuel-pin bundle was minimal, indicating insignificant bundle/duct interaction. Visual inspection of the fuel-pin bundle confirmed the condition of the bundle to be excellent, with no discernible signs of distortion or cladding wear.

4-2. Fuel Column Stability

Fuel column stability, the key figure-of-merit for the IFR-1 test, was evaluated with postirradiation neutron radiography and axial gamma scanning. Ancillary evidence of fuel column elongation during the irradiation was obtained from the measured assembly outlet temperature data.

Eighteen of the 169 test pins were radiographed; all showed a mild elongation of the fuel column. The average elongations were 7.0, 6.1 and 2.5% for the 0, 8, and 19 wt.%Pu fuel types, respectively. These magnitudes are more modest than those of their respective EBR-II siblings,[4] as shown in Fig. 1.

Fig. 1. Fuel column elongation in the IFR-1 Fuel Pins. Closed symbols: IFR-1; open symbols: EBR-II Data Base.



Both radiographic and gamma scan data show the axial fissile distribution within the fuel column to be uniform, except at the very top where the fuel density is noticeably lower. The reduced density at the fuel top was apparently the result of a less-restricted growth into the plenum. None of the pins examined showed any signs of deleterious local fuel compaction. The ^{106}Ru scan data, which mimic the fission density profile, for a representative U-19wt.%Pu-10wt.%Zr pin 181193, is shown in Fig. 2.

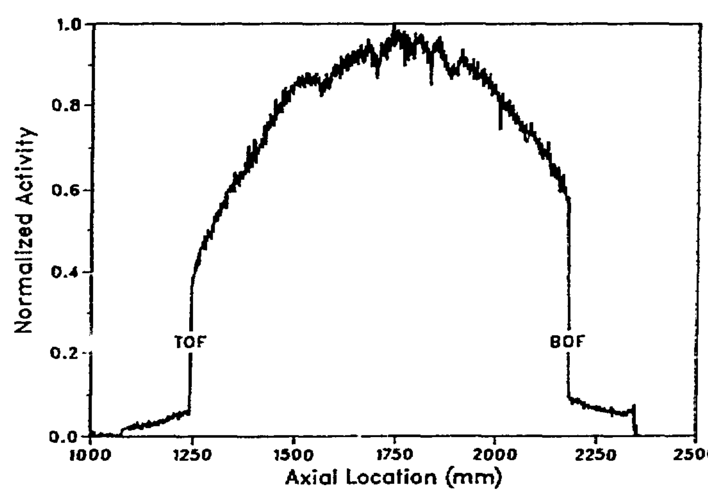


Fig. 2. Distribution of ^{106}Ru Activity in IFR-1 Pin 181193 showing no evidence of localized fuel compaction. (TOF and BOF refer to top of fuel and bottom of fuel, respectively.)

Because of the fuel column elongation, the top end of the fuel was raised above the core where the fission densities were markedly lower.

resulting in a reduction of the thermal output of the assembly. This behavior was detected by the FFTF instrument tree measuring the assembly coolant outlet temperature. The temperature data suggests the axial growth occurred early in life, starting at ≈0.2 at.% burnup and ending at ≈0.6 at.% burnup (between the 10th and 40th EFPD.) The lack of further changes afterward suggests a possible lockup between the fuel and the cladding.

4-3. Cladding Deformation

The cladding strain profiles for all IFR-1 pins examined were similar. None of them showed anomalous strain peaks that would suggest enhanced localized FCMI from fuel relocation. A representative strain profile, that of the 181193 pin, is shown in Fig. 3. In regions above and below the fuel column, where cladding loading was from fission gas alone, the cladding strain was essentially zero, indicating the plenum volume in the IFR-1 fuel pins was adequate in accommodating the released fission gas from the fuel. The slight "shoulders" (≈0.1-0.2%) at the top and bottom ends of the fuel column reflected the small FCMI contribution to the total strain at those locations. In the fuel region, the strain profile was bell-shaped with a maximum of 3.3% at near the axial midplane. Cladding density data, also displayed in Fig. 3, showed that most of the cladding strain was due to the fast neutron-induced swelling of the D9 cladding material. The creep component, i.e., the difference between the total and swelling strains was small (≈0.6% max.)

4-4. Fission-Gas Release and Fuel Restructuring

Fission gas sampling and metallographic examination was completed for U-19wt.%Pu-10wt.%Zr pin 181193. The gas collected in the pin plenum represented a fission-gas release fraction of 72%, which is consistent with the EBR-II data base (63-78%) for comparable fuels at 10 at.% burnup.[4] The data thus indicates that the longer fuel column did not impede the release of fission gas from the fuel. This result correlates with the cladding strain data which showed no significant FCMI.

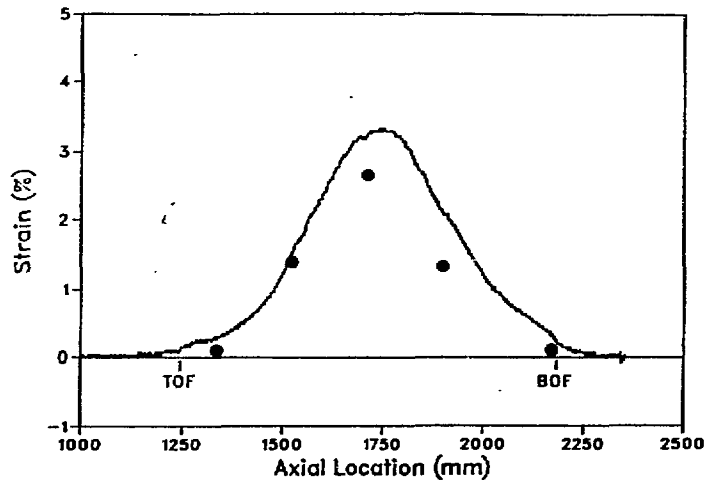


Fig. 3. Cladding strain profile of IFR-1 Pin 181193. The closed circles represent the component of fast neutron-induced swelling of the D9 cladding.

Complex fuel restructuring, i.e., formation of radial zones with different composition and microstructure, resulted from the irradiation. Similar restructuring has been noted in U-Pu-Zr fuels irradiated in EBR-II as well. The extent of restructuring in the 181193 pin varied along the fuel length as affected by the local power and temperature conditions. A transverse metallographic section at X/L = 0.70 and the corresponding alpha and beta-gamma autoradiographs are shown in Fig. 4. Fuel swelling caused the closure of the sodium-filled, 0.76-mm diametral fuel/cladding gap. Eight near-concentric fuel zones, denoted A through H, can be seen in Fig. 4. Distinct redistributions of Pu and fission products are evident from the alpha and beta-gamma radiographs. Results from the electron microprobe analyses revealed that the U and Zr constituents redistributed in the fuel as well. Table 1 shows the approximate compositions of the major constituents in the various zones from the microprobe analysis. Zone A and, to a lesser extent, zone C, are highly porous. These zones probably provided the long-range connectivity in the axial direction that facilitated the release of fission gas to the plenum.

Table 1. Approximate Compositions (in wt.%, normalized) of the Restructured Fuel Zones

Zones*	U	Pu	Zr
A	36	34	30
B	80	19	1
C,D	65	16	19
E	88	11	1
F	66	27	7
G,H	54	24	22

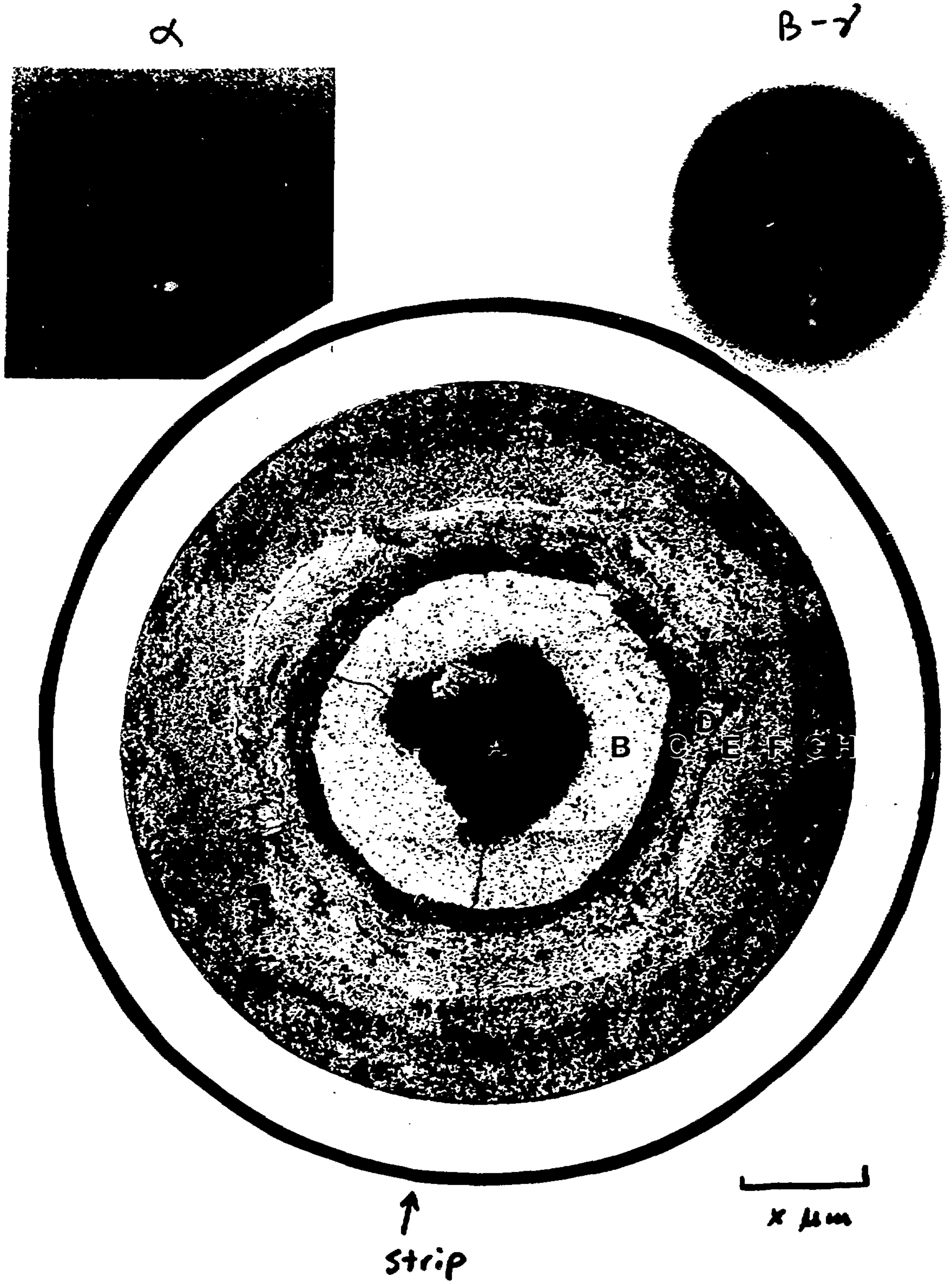
*Refer to Fig. 4.

4-5. Fuel/Cladding Metallurgical Interaction

Metallurgical interaction between the fuel (including fission products) and cladding during the irradiation was minor in the 181193 pin. The depths of cladding reaction ranged from ≈0 μm at both ends of the fuel column to ≈60 μm at the axial location of X/L = 0.70. Compared to the as-built cladding thickness of 560 μm, the amount of cladding wastage was not significant.

The condition of the fuel/cladding interface at X/L = 0.70 where the maximum cladding interaction occurred is shown in Fig. 5. This location is directly opposite a fuel surface crack at approximately two o'clock in Fig. 5. The depth of cladding interaction at other azimuthal locations was considerably less. Two reaction bands in the cladding, denoted A and B in Fig. 5, can be seen. In the microprobe analysis, the

Fig. 4. Transverse metallographic section and α and β-γ radiographs of pin 181193 at $X/L = 0.70$.



α
B-γ
A
B
C
D
E
F
G
H
strip
x μm

presence of lanthanide fission products was detected to the far edge of the outer cladding reaction band (A) adjacent to the unreacted cladding. The fuel crack probably provided a conduit for enhanced vapor-phase transport of lanthanides to the cooler fuel/cladding interface. In the inner reaction band (B), in addition to lanthanides, U and Pu were detected. Both the inner and outer cladding reaction bands were nearly completely depleted in Ni. The loss of Fe, on the other hand, was minor and limited to only the inner band. The Fe and Ni lost from the cladding were detected in the fuel adjacent to the reacted cladding. Chromium, the other major constituent in D9, showed the least mobility in the interdiffusion, being concentrated in a sub-band on the inward side of band A. The source of the Cr concentration was apparently from reaction band B

Fig. 5. Fuel/cladding metallurgical interaction in pin 181193. The arrows denote the extent of the cladding reaction. The bright sub-band in Bond A is encircled in Cr and lanthanide fission products.

As-polished

Etched

X μm

5. DISCUSSION

5.1. Burnup Potential of Full-length IFR Fuel Pins

The IFR-1 test was terminated when the design goal burnup of 10 at.% was reached. Judging from the excellent condition of the discharged fuel pins, e.g., minimal fuel/cladding mechanical and metallurgical interactions, it is reasonable to expect that the lifetime limit of the IFR-1 fuel pins may be substantially beyond the attained 10 at.% burnup. The early concern that longer fuel column may cause localized fuel compaction, enhanced FCMI and premature fuel pin failure was thus unfounded. Six other IFR fuel irradiation experiments have been successfully completed in FFTF with over 800 HT9-clad U-10Zr fuel pins, many with operating conditions substantially more aggressive than those of the IFR-1 pins. All have attained high burnup, with the peak exceeding 16 at.%., and no cladding breaching. The results from these and the IFR-1 tests clearly demonstrated that full-length metallic fuel pins possess excellent high burnup potentials.

5.2. Effects of Pu on Axial Fuel Elongation

The EBR-II irradiation data base shows the magnitude of fuel column elongation is strongly Pu-dependent: the greater the Pu content, the lower the elongation (see Fig. 1). The behavior of the full-length IFR-1 fuel pins was found following the same trend. In U-Pu-Zr fuels, Pu promotes constituent redistribution (zone formation) and radial swelling, resulting in an early contact between the fuel and cladding.[7] In addition, the stresses from the swelling of the hotter inner zones in a high-Pu fuel often cause the colder fuel periphery to crack. (The remnant of one such crack can be seen in Fig. 5, at 2 o'clock.) One effect of the fuel surface cracking is a near instantaneous fuel diameter increase which accelerates fuel/cladding contact. Once fuel contacts the cladding, it is believed that the restraining effect of the much stronger cladding will eliminate essentially any significant further growth of the fuel column. In fuel pins with low or no Pu contents, on the other hand, constituent redistribution and fuel restructuring are usually minor and rarely result in fuel surface cracking. Consequently, fuel/cladding lockup occurs later, allowing a greater fuel swelling in the axial direction.

6. CONCLUSIONS

The IFR-1 test successfully reached the goal burnup of 10 at.%. All pins maintained their cladding integrity. The results from postirradiation examinations indicate that the irradiation behavior of full-length IFR-1 fuel pins is comparable to that of the shorter EBR-II siblings. Fuel column stability was excellent and cladding strain from fission-gas loading and FCMI was insignificant. These positive results support the use of the extensive EBR-II irradiation data base for the development of prototypical ALMRs.

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