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Postirradiation examination results of several metallic fuel alloys and forms from low burnup AFC irradiations

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HIGHLIGHTS

- Observations from recent AFC-3 irradiation experiments are compared to the historical behavior of metallic fast reactor fuel.
- Significant FCCI was observed in U-10Mo alloy fuel.
- A palladium additive to control FCCI at high burnup was tested and its impact on fuel performance is discussed.

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ABSTRACT

Several different U based metallic nuclear fuel alloys and forms were irradiated as part of the Advanced Fuels Campaign (AFC) series of irradiation experiments. Results from postirradiation examinations of the AFC-3A and AFC-3B are presented here. In this irradiation, U alloyed with Mo, Zr, and Zr with Pd were investigated. Different geometric forms (solid, annular), smear densities and bonding media (sodium, helium) were also irradiated and evaluated. All fuel forms were irradiated in ferritic-martensitic HT-9 cladding. Examination of the irradiated fuel included neutron radiography, evaluation of the distribution of gamma-ray emitting radionuclides, dimensional inspection, fission gas release analysis, optical microscopy, and chemical analysis. No in-pile failure was observed with this fuel. Neutron radiography, gamma spectrometry, and dimensional inspections indicated the fuel largely performed acceptably. Optical microscopy revealed significant fuel-cladding interaction between the Mo based alloys and the cladding. The Zr based alloys generally performed better although some forms revealed behaviors that present specific fabrication challenges such as tighter fabrication tolerances that will need to be better controlled to match historical U-Zr fuel performance. The addition of Pd as an additive to prevent fuel cladding chemical interaction (FCCI) also appears to present some challenges to fuel performance.

1. Introduction

A long term research focus of the US Department of Energy Advanced Fuels Campaign (AFC) is to investigate technologies that can allow for increased actinide utilization in nuclear fuel for fast neutron spectrum reactors. This includes both the incorporation of minor actinides into nuclear fuel for transmutation and high utilization of actinides through “ultra high” burnup fuel. This goal is defined as fuel that can reach a burnup of 30–40% fission per initial metal atom (%FIMA) which is higher than what has historically been achieved [1,2]. Historically, EBR-II Mark III/IIIA/IV driver fuel (U-10Zr fuel with stainless steel 316 or ferritic martensitic steel HT-9 cladding) was qualified to 10%FIMA, but many experimental assemblies and fuel pins achieved higher peak burnup up to near 20% FIMA without failure [3]. The purpose of the irradiation tests explored in this work was to screen candidate alloys and forms that could tolerate very high burnup irradiations of up to 30% FIMA. In these irradiations, termed AFC-3A and AFC-3B, uranium based alloys and forms were irradiated in a prototypic fast reactor spectrum to investigate low burnup fuel performance. Irradiations were performed in the Idaho National Laboratory (INL) Advanced Test Reactor (ATR). The performance of these fuels is compared to the historic performance of U-10Zr [4]. Metallic fuel for fast reactors has a long history that has been reviewed several time in the literature [3–8].

Uranium alloyed with both Mo and Zr were investigated and some alloys were tested with small amounts of Pd additive meant

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to mitigate fuel cladding chemical interaction (FCCI). Additionally, different fuel forms or fuel geometries were also tested. Solid fuel slugs of different diameters were bonded to the cladding with sodium, and annular fuel slugs were bonded to the cladding with helium. There is interest in removing sodium bonding from fuel to avoid mixed hazardous waste treatment on the back end of the fuel cycle [9]. In a once through fuel cycle, sodium bonded spent nuclear fuel is treated as a “mixed hazardous waste” because it is both irradiated nuclear fuel (high level waste) and contains sodium (hazardous waste). This creates a disposal problem because there is no ultimate repository for “mixed hazardous waste.” Therefore sodium bonded spent nuclear fuel must be treated to remove the sodium prior to disposal. This process is very costly which creates an incentive to research fuels that can perform well without a sodium bond. All fuels tested in the irradiations reported in this work were clad with the ferritic-martensitic steel HT-9.

Results of the AFC-3A and AFC-3B irradiation are the focus of this paper additional details on the collection of data are documented in Ref. [10]. The AFC-3 experiment is a continuation of the AFC experiments that included the AFC-1 and AFC-2 series of experiments [11–13].

2. Fuel fabrication and irradiation conditions

The fabrication of the AFC-3A is documented in Ref. [14] and the pre-irradiation characterization is documented in Ref. [15]. All alloys were formed by mixing appropriate quantities of metals in an arc melter. The alloys were homogenized through a series of several melts. The alloys were then cast into quartz molds using a counter gravity injection casting system where a vacuum is used to draw the alloy melt into the mold. The annular fuel molds contained an inner quartz core. After casting, the molds are removed by breaking the quartz off the outside of the cast pins. For the annular pins the inner quartz core was removed from the cast pin by drilling. The fuel slugs were then cut to length. Samples for microscopy, chemical analysis and archive were also collected. The compositions and as-built dimensions of the 4 fueled rodlets from AFC-3A and the 4 fueled rodlets from AFC-3B are summarized in Table 1. AFC-3A R5 contained two fuel slugs with different amounts of Pd additive. Slug A the upper slug (AFC-3A R5A) contained 1% Pd while slug B (AFC-3A R5B) contained 2% Pd. The geometry of an AFC-3 experiment has 3 major parts: fuel, rodlet and capsule. The fuel is either a solid or annular cylinder of uranium alloy. The rodlet is the fuel sealed inside a cladding material. The cladding material is HT-9 with an outer diameter of 5.842 mm (0.230 inch) and a wall thickness of 0.4445 (0.0175 inch). The rodlets with solid fuel contain sodium to thermally bond the fuel to the cladding and are back filled with helium and Ar. The rodlets with annular fuel only contain the helium back fill gas. The rodlets are sealed inside another cylindrical container called the capsule which acts as the primary safety boundary between the experiment and the ATR coolant. The capsules are back filled with helium, and the gap between the rodlet and the capsule along with enrichment is used to control the temperature in the rodlets to the desired level. The AFC capsules are irradiated in a cadmium shrouded position that hardens the ATR neutron spectrum to create a radial power profile inside the fuel that is appropriate for fast spectrum fuel testing [16–19]. A cross section showing the typical dimensions of the AFC-3 irradiation device is shown in Fig. 1a and the typical axial configuration is shown in Fig. 1b.

The irradiation conditions in the rodlets were simulated using a coupled monte carlo-depletion methodology to calculate the linear heat generation rate and burnup at several points during each reactor cycle [20–23]. The ATR reactor has 4 lobes that can be operated at different powers depending on the needs of the irradiation tests in the reactor in any given cycle [24]. AFC-3A was in the southeast outer A position in ATR which is dependent on the southeast lobe power of ATR, and AFC-3B was in the southwest outer A position in ATR which is dependent on the southwest lobe power of ATR. The hourly variation in LHGR for each effective full power day (EFPD) of the irradiation is shown in Fig. 2 for AFC-3A and Fig. 3 for AFC-3B. The hourly variation in peak inner cladding temperature (PICT) for each rodlet, which is a key metric for evaluating metallic fuel performance, is shown for each EFPD in Fig. 4 for AFC-3A and Fig. 5 for AFC-3B. The hourly peak inner cladding temperature is correlated to LHGR and calculated from the as-built dimensions of the rodlets and capsules [25]. In these thermal calculations, the Kennard model [26] was used to properly model heat conduction across the gap between the rodlet and the capsule.

Although the LHGR's and PICT's shown in Fig. 2 through 5 indicate that the fuel should generally be well behaved these simulation results assume an idealized geometry that was not present in the as-fabricated capsules. After insertion it was discovered that the capsules used in AFC-3A and AFC-3B had poor cylindricity in the inner bore. Cylindricity is a measure of how close an object conforms to a true cylinder. Because of this the gap between the rodlet and the capsule was asymmetric and may have resulted in local variations in temperature beyond what is shown in the above. This also resulted in the early termination of the irradiation after about only 120 days of irradiation. This complicates the interpretation of this irradiation as there is considerable uncertainty in the irradiation conditions due to these fabrication defects.

3. Results from PIE

Several different postirradiation exams were performed on all the AFC-3A and AFC-3B rodlets including: visual examination, neutron radiography, gamma spectrometry, dimensional inspection, fission gas release measurement, chemical burnup analysis and optical microscopy. The majority of PIE is performed in the INL Hot Fuel Examination Facility (HFEF). Cursory non-destructive

Table 1
AFC-3a and 3B test matrix.

Rodlet ID	Nominal Composition	Fuel Form	Bond Material	Nominal Smear Density	Outer Diameter (mm)	Inner Diameter (mm)	Total Height (mm)	As-built Smear Density
3A-R1	U-10Mo	Solid	Sodium	75%	4.28	N/A	35.2	74.5%
3A-R2	U-10Mo	Annular	Helium	55%	4.86	3.25	37.6	53.1%
3A-R4	U-10Zr	Annular	Helium	55%	4.86	3.25	38.0	71.7%
3A-R5A	U-1Pd-10Zr	Solid	Sodium	75%	4.19	N/A	18.8	73.1%
3A-R5B	U-2Pd-10Zr	Solid	Sodium	75%	4.23	N/A	19.1	54.0%
3B-R1	U-4Pd-10Zr	Solid	Sodium	55%	3.64	N/A	38.0	52.9%
3B-R2	U-4Pd-10Zr	Annular	Helium	55%	4.85	3.25	37.8	55.7%
3B-R4	U-10Mo	Solid	Sodium	55%	3.70	N/A	37.9	55.4%
3B-R5	U-10Mo	Solid	Sodium	55%	3.69	N/A	38.1	53.1%