INL/EXT-06-11707

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Experiment Description for AFC-2A and AFC-2B

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EXPERIMENT DESCRIPTION
for
AFC-2A and AFC-2B

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1.0 Introduction

The U.S. Advanced Fuel Cycle Initiative (AFCI), now within the broader context of the Global Nuclear Energy Partnership (GNEP), seeks to develop and demonstrate the technologies needed to transmute the long-lived transuranic actinide isotopes contained in spent nuclear fuel into shorter-lived fission products, thereby dramatically decreasing the volume of material requiring disposal and the long-term radio-toxicity and heat load of high-level waste sent to a geologic repository [1]. One important component of the technology development is actinide-bearing metallic transmutation fuel forms containing plutonium, neptunium, americium (and possibly curium) isotopes. There are limited irradiation performance data available on metallic fuels with high concentrations of Pu, Np and Am, as are envisioned for use as actinide transmutation fuels. Initial scoping level irradiation tests of such metallic fuels are underway in the ATR (AFC-1B, D, F & H). Non-fertile and low-fertile metallic fuels in the AFC-1B & F tests have already been discharged from the reactor at 8% burnup and are nearing the completion of postirradiation examination; performance of all the fuel alloys included in these tests was good [2]. Similar fuel alloys continue to be irradiated in the AFC-1D & 1H tests and are currently over 20% burnup.

The proposed AFC-2A and AFC-2B irradiation experiments are a continuation of the metallic fuel test series in progress in the ATR. These experiments will consist of metallic fuel alloys of U, Pu, Np, Am and Zr, some with minor additions of rare earth elements meant to simulate expected fission product carry-over from pyro-metallurgical reprocessing, to be irradiated to burnup levels of ≥ 10 and ≥25 at.-% burnup. The AFC-2A & B experiments will be irradiated in two of the East Flux Trap positions of the ATR using the same hardware design as for the AFC-1B, D, E & H tests [3, 4, 5].

The AFC-2A & B irradiation experiments are expected to provide important irradiation performance data on metallic transmutation fuel forms, including irradiation growth and swelling, helium production, helium and fission gas release fractions, fission product and fuel constituent migration, fuel phase equilibria, and fuel-cladding chemical interaction. Of particular interest in these tests will be the effect that small additions of rare earth elements, expected to be carried over as a result of pyro-metallurgical reprocessing, might have on irradiation performance parameters.

Experiments AFC-2A & B will have design and test conditions analogous to the AFC-1F & H metallic fuel tests. The fuel test matrix in AFC-2A & B will be identical; only the target discharge burnups will differ for the two tests. An overview of these two experiments is shown in Table 1. The simultaneous insertion of these two irradiation test assemblies is proposed for August 2007 (ATR Cycle 140A).

2.0 Fuel Rodlet Description

The rodlet assembly is designed as a miniature length, fast reactor fuel rod. The rodlet assembly consists of the metallic fuel column, bond sodium, stainless steel Type 421 (HT-9) cladding and an inert gas plenum. A stainless steel capsule assembly will contain a vertical stack of six rodlet assemblies. The capsule and rodlet radial dimensions of the metallic fuel specimens are shown in Figure 1. The annular gap between the fuel column and rodlet inner diameter is initially filled by the sodium bond and is designed to accommodate fuel swelling during irradiation. The annular helium-filled gap between the rodlet outer diameter and capsule inner diameter is designed to provide the thermal resistance necessary to achieve the design irradiation temperature of the fuel specimen.

Figure 2 shows the fuel rodlet assembly axial dimensions. Table 2 shows the materials used in constructing the rodlets along with their radial design dimensions. The design length of the metallic fuel column is 1.5-in.; the metallic fuel column may consist of a maximum of two pins, and the design diameter is 0.168-in. The bond sodium is designed to cover and exceed the fuel column height by between 0.25 and 0.50-in. The cladding for all rodlets is 6.0-in. in length (including welded endplugs) with 0.230-in. outer diameter and 0.194-in. inner diameter.

3.0 Test Matrix

The fuel compositions and positions of the metallic fuel rodlets in AFC-2A & B are shown in Table 3; note that the metallic alloy compositions are expressed as a weight percent for each constituent element.

The fuel compositions in these experiments were selected to build upon the irradiation performance data that have been obtained from the AFC-1B, D, E and H metallic fuel tests in the ATR. Compared to the fuel compositions in the previous tests, the AFC-2A, B uranium contents have been increased and the zirconium contents have been decreased, as the AFCI/GNEP program objectives have evolved to a position where a somewhat higher conversion ratio is deemed as acceptable in future sodium-cooled fast reactors used for actinide transmutation; the lower Zr content results in a denser fuel, though still considerably less dense than the U-20Pu-10Zr fuels irradiated routinely in EBR-II. The transuranic contents in the AFC-2A, B fuels is bounded by the previous ATR tests. The new parameter introduced in the AFC-2A, B fuel test matrix is a minor addition of rare earth elements (La, Nd, Ce) in some of the fuels; this is done to simulate rare earth carryover that may result from pyro-metallurgical reprocessing of metallic fuels.

Uranium enrichment will be varied to achieve target linear heat generation rates for the different fuel compositions. The calculated fuel constituent masses for each rodlet, as well as sum for each irradiation test assembly, are given in Table 4; note that in Table 4 the uranium content has been calculated assuming ‘depleted uranium’, but this will change when the final enrichment is determined for each rodlet.

4.0 Irradiation Test Assembly Description

The irradiation test assembly, capsule assembly and rodlet assembly are shown in Figure 3. The irradiation test assembly consists of the experiment basket and capsule assembly, which contains six vertically-stacked rodlet assemblies. The experiment basket of the test assembly is designed to interface the capsule assembly with the ATR and to act as a thermal neutron flux filter. The current basket design is an aluminum-sheathed cadmium tube. The aluminum sheath accommodates a cadmium tube thickness between 0.021 and 0.045-in. For the AFC-2A & B experiments, it is proposed that the cadmium thickness be the same as in the AFC-1B, D, F & H experiments design, which is 0.045-in. The decrease in the thermal neutron flux will result in a reduction in the linear power in the fuel rodlets, which is necessary to meet the experiment design conditions.

The capsule assembly function is the following: 1) provide a second, robust barrier between the water coolant and the fuel, sodium and fission products, and 2) provide additional free volume for expansion of helium and fission gases should the cladding of any number of rodlets be breached during irradiation (this free volume is sufficient to reduce the gas pressure on the capsule to below 235 psi assuming all six rodlets are breached). The relevant design data for the capsule assembly is summarized in Table 5. The capsule assembly will be fabricated to meet the intent of the ASME, Section III, Class 1 pressure vessel code requirements. The capsule assembly design will be identical in both experiments.

5.0 Irradiation Experiment Conditions

The experiments AFC-2A and 2B are designed for irradiation in the east flux trap. The two tests will be inserted in the ATR east flux trap drop-in positions that, together with the cadmium neutron filter baskets, will achieve the design linear powers.

The fuel experiment design conditions for AFC-2A and 2B are shown in Table 6. The design objective is for each rodlet in these experiments to have a peak linear heat generation rate (LHGR) of 33.0 kW/m. The LHGR for each rodlet will be calculated using the MCNP Coupling With ORIGEN2 (MCWO) analysis methodology. The final uranium enrichment value for each rodlet will be determined based on these results.

The peak cladding temperature should not exceed 525°C during normal operation and 650˚C during off-normal conditions. The expected peak thermal conditions during irradiation for these experiments will be estimated based on the linear powers calculated by MCWO. In addition, the plenum pressures expected in each rodlet at a maximum burnup of 30 at.-% (i.e., depletion of initial Pu-239 + U-235) will be estimated to confirm that the total pressure on the capsule assembly (the safety class boundary) will not exceed 235 psi in the event that all rodlets fail during irradiation.

The AFC-2A experiment should be discharged from the reactor upon reaching a target peak burnup of ≥ 10 at. %, and the AFC-2B experiment should be discharged from the reactor upon reaching a target peak burnup of ≥ 25 at. %

6.0 Post-Irradiation Services

Upon discharge of each experiment from the reactor, cooling in the ATR canal for a minimum of 30 days will be required. The capsule assembly and aluminum sheathed cadmium segment of the basket assembly will be shipped to the Hot Fuels Examination Facility (HFEF) at MFC for postirradiation examinations. Shipment from the Reactor Test Complex (RTC) to HFEF at MFC will be made using the GE Model 2000 shipping cask. After the capsule assemblies and cadmium bearing segments of the basket assemblies are loaded in the cask, the remnants of the basket assemblies should be considered scrap.

7.0 Proposed Schedule

The schedule shown below is proposed to meet the required deadlines prior to insertion of experiments AFC-2A and 2B into the ATR for irradiation beginning in July 2007.

Deliverable	Date
Experiment Description for AFC-2A & B	August 31, 2006
Final Experiment Design and Data Package	April 20, 2007
Fabrication As-Built Data Package	July 3, 2007
AFC-2A&B Delivered for Cycle 140A	July 18, 2007

8.0 Responsibilities

Idaho National Laboratory will fabricate the experiments at the Materials & Fuels Complex and deliver the irradiation test assemblies to the ATR. Steven Hayes is responsible for experiment design and postirradiation examinations. Debbie Utterbeck is responsible for irradiation planning and transportation. Tim Hyde is responsible for fuel and experiment fabrication. Steven Hayes has overall responsibility for these irradiation experiments.

9.0 References

[1] Report to Congress on the Advanced Fuel Cycle Initiative: The Future Path for Advanced Spent Fuel Treatment and Transmutation Research, U.S. Department of Energy Office of Nuclear Science and Technology, January 2003.

[2] B. A. Hilton, D. L. Porter and S. L. Hayes, "AFC-1 Transmutation Fuels Postirradiation Hot Cell Examination 4 to 8 at.% Preliminary Report: Irradiation Experiments AFC-1B, AFC-1F and AFC-1Æ," Idaho National Laboratory Report, INL/EXT-05-00785, September 2005.

[3] S. L. Hayes, "Irradiation of Nitride and Metallic Fuels for Actinide Transmutation in the Advanced Test Reactor: Final Experiment Description and Design & Data Package for AFC-1A, AFC-1B, AFC-1C & AFC-1D," Argonne National Laboratory(West) Document No. W7520-0481-ES-02, February 2003.

[4] B. A. Hilton, "Irradiation of Nitride and Metallic Fuels for Actinide Transmutation in the Advanced Test Reactor: Final Experiment Description and Design & Data Package for AFC-1Æ and AFC-1F," Argonne National Laboratory (West) Document No. W7520-0529-ES-03, November 2003.

[5] B. A. Hilton, "Irradiation of Nitride and Metallic Fuels for Actinide Transmutation in the Advanced Test Reactor: Final Experiment Description and Design & Data Package for AFC-1G and AFC-1H," Argonne National Laboratory (West) Document No.W7520-0678-ES-00, August 2004.

Table 1. Overview of the AFC-2A & AFC-2B Experiments.

ATR Experiment Designation	Fuel Form	ATR Insertion	Target Discharge Burnup*
AFC-2A	Metallic	July 2007	≥ 10%
AFC-2B	Metallic	July 2007	≥ 25%

*Burnup in percent of initial (Pu-239 + U-235).

Table 2. Fuel Rodlet Design Data.

Design Parameter	**AFC-2A & B**
Cladding Material	421SS (HT9)
Cladding O.D.	0.230-in.
Cladding I.D.	0.194-in.
Bond Material	Sodium
Fuel Type	Metallic Alloy
Fuel Smear Density	75%
Fuel Porosity	0%
Fuel O.D.	0.168-in.
Fuel Height	1.50-in.

Table 3. AFC-2A & AFC-2B Fuel Test Matrix.

Rodlet	**Metallic Fuel Alloy †**
1	U-20Pu-3Am-2Np-15Zr
2	U-20Pu-3Am-2Np-0.8RE*-15Zr
3	U-20Pu-3Am-2Np-1.5RE*-15Zr
4	U-30Pu-5Am-3Np-1.5RE*-20Zr
5	U-30Pu-5Am-3Np-0.8RE*-20Zr
6	U-30Pu-5Am-3Np-20Zr

†Alloy composition expressed in weight percent.
*RE designates rare earth alloy (16% La, 53% Nd, 31% Ce).

Table 4. AFC-2A and AFC-2B Rodlet Constituent Masses.

	Fuel Density (g/cm3)	Fuel Column Constituent Masses (g) Np-237	Total U	U-235	U-238	Total Pu	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	RE	Zr	Bond Na
Rodlet 1	13.98	0.152	4.570	0.009	4.561	1.523	0.001	1.257	0.251	0.009	0.005	0.229	0.000	1.143	0.426
Rodlet 2	13.98	0.152	4.510	0.009	4.500	1.523	0.001	1.257	0.251	0.009	0.005	0.229	0.061	1.143	0.426
Rodlet 3	13.98	0.152	4.456	0.009	4.447	1.523	0.001	1.257	0.251	0.009	0.005	0.229	0.114	1.143	0.426
Rodlet 4	12.87	0.210	2.840	0.006	2.834	2.104	0.001	1.736	0.347	0.012	0.007	0.351	0.105	1.403	0.426
Rodlet 5	12.87	0.210	2.889	0.006	2.883	2.104	0.001	1.736	0.347	0.012	0.007	0.351	0.056	1.403	0.426
Rodlet 6	12.87	0.210	2.945	0.006	2.939	2.104	0.001	1.736	0.347	0.012	0.007	0.351	0.000	1.403	0.426
Total		1.088	22.211	0.044	22.166	10.882	0.005	8.982	1.795	0.062	0.037	1.737	0.336	7.635	2.558

Note: RE designates rare earth alloy (16% La, 53% Nd, 31% Ce).

Table 5. Design Data for Capsule Assembly.

Design Parameter	Value
Capsule Material	316SS
Capsule O.D.	0.354-in.
Capsule I.D.	0.234-in.
Capsule Length	52.000-in.
Capsule Free Volume	15.56 cm^3
Capsule-Rodlet Gap	0.0022-in.

Table 6. AFC-2A and AFC-2B Experiment Specifications.

Performance Parameter	**AFC-2A & 2B**
Maximum Burnup	**30 at.-%**
Peak Rodlet Linear Power - Normal Operation - Off-Normal Limit	 **35.0 kW/m** **45.0 kW/m**
Peak Cladding Temperature - Normal Operation - Off-Normal Limit	 **525°C** **650°C**
Fuel Temperature - Normal Operation - Off-Normal Limit	 **900°C** **1100°C**

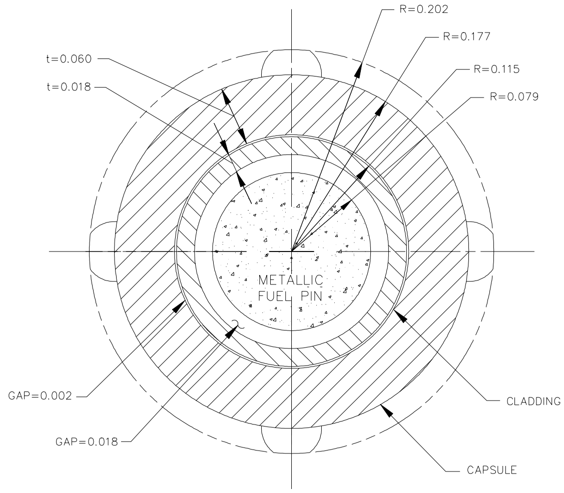

Figure 1. Radial dimensions of capsule and fuel rodlet assemblies for the AFC-2A & B metallic fuel tests.

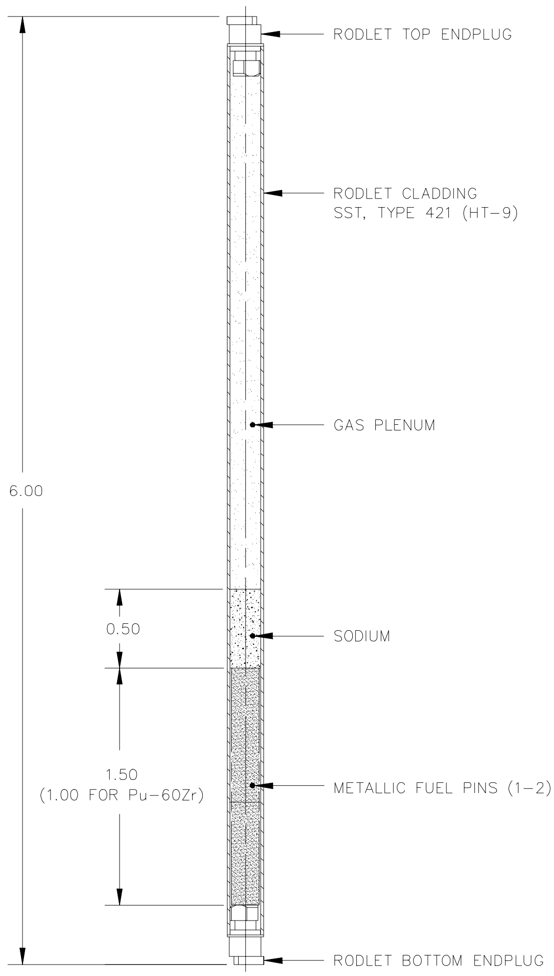

Figure 2. Rodlet assembly axial dimensions for the AFC-2A & B metallic fuel tests.

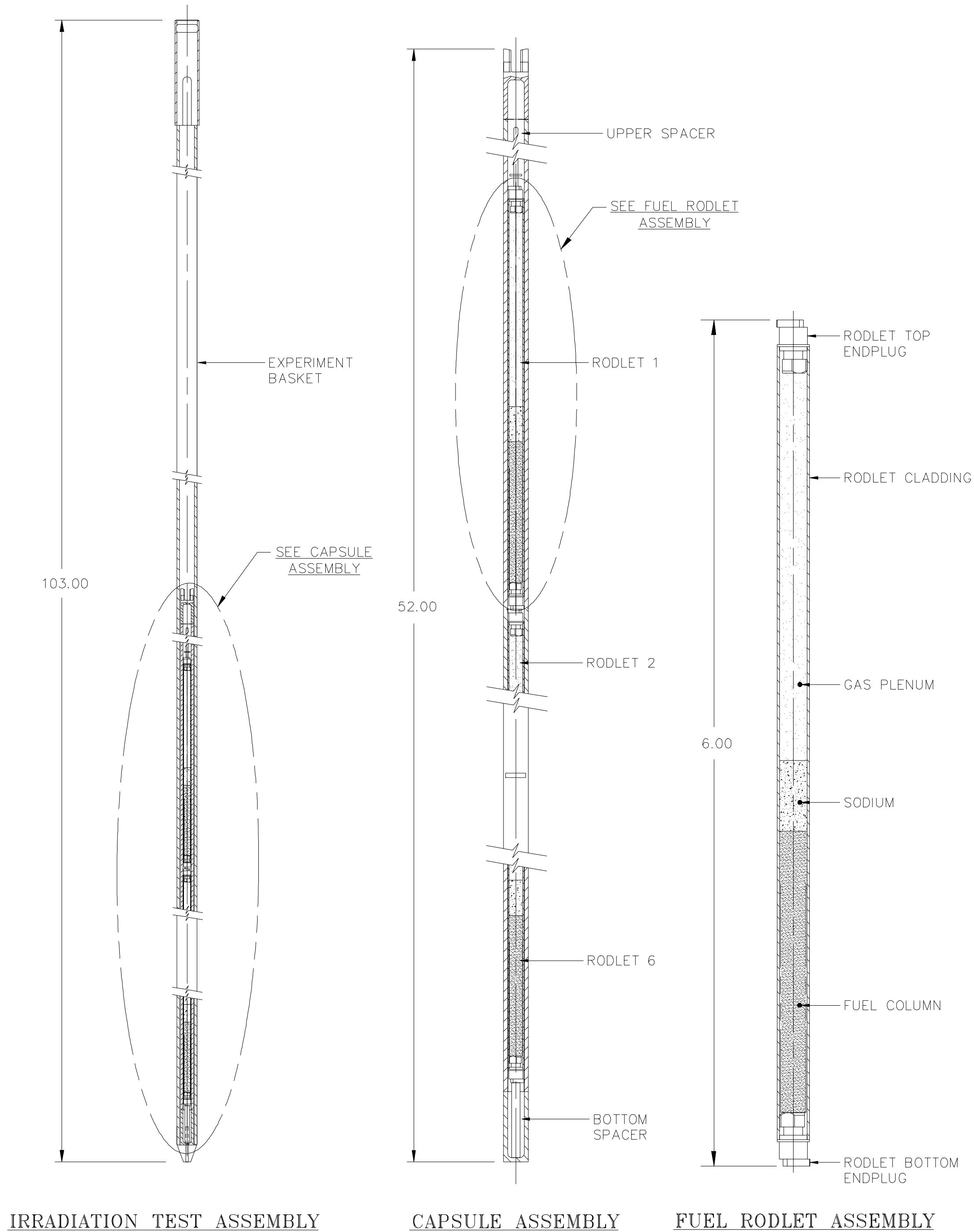

Figure 3. AFC-2 Series Irradiation Test Assembly for ATR East Flux Trap Positions.