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ABSTRACT

Two irradiation test vehicles have been designed, fabricated, and inserted into the Advanced Test Reactor in Idaho. Irradiation of these experiments began in August 2000. These irradiation tests were designed to obtain high-burnup irradiation performance data on a series of intermediate and high-density U-Mo alloy dispersion fuels at nominal research reactor thermal conditions. The two irradiation experiments each contain 32 miniature fuel plates. The test matrix of fuel compositions include U-10Mo, U-7Mo and U-6Mo fuel plates fabricated at densities of both 6 and 8 g-U/cm^3, and U-6Mo-1.7Os and U_3Si_2 (included as a standard) fuel plates fabricated at 6 g-U/cm^3. Fabrication techniques used to produce the fuel powders included grinding and atomization. The matrix phase of each fuel plate is aluminum, with the exception of two fuel plates which are aluminum-clad U-10Mo foils (i.e., no matrix phase). The peak fuel temperatures are approximately 180°C at beginning-of-life. These experiments, designated RERTR-4 and -5, will be discharged at peak fuel burnups of approximately 50 and 80 at.% U^{235}. Of primary interest are the collection of quantitative data on the extent of interaction between fuel and matrix phases and the fission gas retention/swelling characteristics of these fuels up to very high burnup. This paper presents the design of the irradiation tests and the irradiation conditions.

INTRODUCTION

Three prior irradiation experiments for the U. S. Reduced Enrichment for Research and Test Reactors (RERTR) Program, designated RERTR-1, -2 and -3 [1,2], have been conducted to investigate the potential of metallic uranium alloy dispersion fuels to meet the high density requirements for conversion of research reactors that employ the highest enrichments and operate at the highest powers. The results of these irradiation experiments have been reported elsewhere

[3-5]. In general, the irradiation performance of the U-Mo (with 6 to 10 wt.% Mo) alloy dispersion fuels tested to date appears promising. While the RERTR-1 and -2 tests took these alloy fuels to high exposures (70 at.% U^{235} burnup, $5 \cdot 10^{21}$ fiss./cm^3 fuel fission density), the peak fuel temperatures of <100°C and fuel loadings of ~4 g-U/cm^3 in these tests were below the values needed for use in high power research reactors. The third irradiation test, RERTR-3, irradiated these alloys at conditions of U^{235} density (8 to 9 g-U/cm^3) and fuel temperature (>200°C) that are prototypic of the highest-power research reactors that currently use high-enrichment fuels, but reached only moderate exposures (40 at.% U^{235} burnup, $3 \cdot 10^{21}$ fiss./cm^3 fuel fission density).

The RERTR-4 and -5 experiments currently under irradiation in the Advanced Test Reactor (ATR) in Idaho will irradiate these U-Mo alloy dispersion fuels at nominal research reactor thermal conditions (180°C peak fuel temperatures) and at both intermediate and high uranium loadings (6 and 8 g-U/cm^3); RERTR-4 will be discharged at a peak fuel burnup of 50 at.% U^{235}, while RERTR-5 will continue irradiation to a terminal burnup of 80 at.% U^{235}.

IRRADIATION TEST VEHICLES

The RERTR-4 and -5 irradiation test vehicles are currently undergoing irradiation in large "B" positions (B-11 and B-12) of the ATR. These positions are vertical, 1.5-in. (3.8 cm) diameter holes located near reflector control drums adjacent to the south and west reactor lobes.

In these experiments, the irradiation vehicle consists of a flow-through "basket" holding four vertically stacked, flow-through capsules which contain the miniature fuel plates. All fuel plates are exposed to and cooled by the reactor's primary coolant. The eight fuel plates within each fueled capsule are positioned as two sets of four parallel fuel plates stacked axially on top of each other; a cross-section of a fueled capsule is shown in Figure 1. Flow-through spacers are included at both the top and bottom of the stack of fueled capsules to center the stack about the axial midplane of the 4.0-ft. (1.2 m) high core. A flow orifice was incorporated into the bottom of the basket to achieve the desired fuel temperatures.

The flow-through capsules are designated from top-to-bottom as "4A" through "4D" in RERTR-4 and "5A" through "5D" in RERTR-5. Each capsule holds eight fuel plates in a configuration such that the long dimension of the fuel plate is parallel to the coolant flow. The large B positions receive primary reactor coolant which flows from top to bottom; thus, coolant flow enters the experiments at capsule "A" and exits at capsule "D". Table 1 shows the configuration of the fuel plates within each irradiation vehicle and within each capsule. In this table, fuel plate positions 1-4 in each capsule are located vertically above positions 5-8.

FUEL PLATES

The miniature dispersion fuel plates fabricated for use in this test have external dimensions of 3.995-in. (10.15 cm) in length, 1.000-in. (2.54 cm) in width, and 0.055-in. (1.4 mm) in thickness. Whereas a circular die was used to produce cylindrical fuel-matrix powder compacts for the fuel

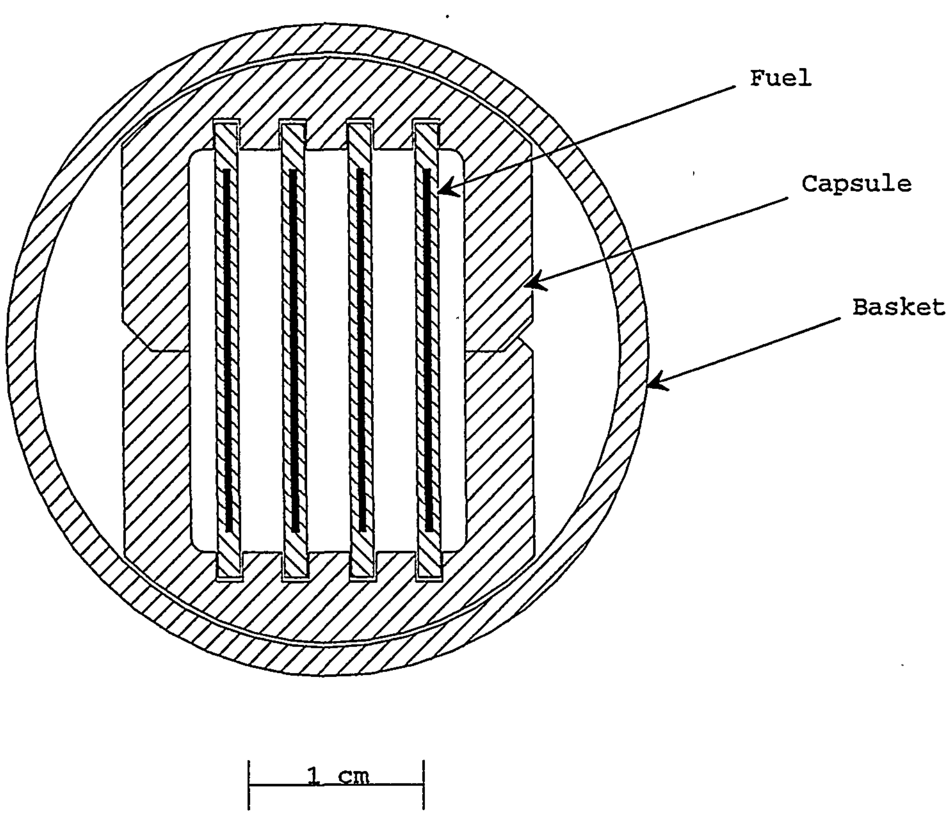


Figure 1. Horizontal cross-section of ATR basket, flow-through capsule, and four fuel plates.

plates fabricated for RERTR-1, -2 and -3, the fuel plates used in RERTR-4 and -5 were fabricated using a square die (with rounded corners); thus, the fuel zones in the finished fuel plates are nominally rectangular in shape as opposed to the elliptical fuel zones produced previously. The fuel zone dimensions in the fuel plates are nominally 3.200-in. (8.13 cm) in length, 0.730-in. (1.85 cm) in width, and 0.025-in. (0.6 mm) in thickness. All the dispersion fuel plates are clad in 0.015-in. (0.38 mm) thick Al-6061. The size of these fuel plates is a significant increase over those tested in RERTR-3 and were designed to be as large as the irradiation position would accommodate to allow meaningful quantitative swelling measurements to be performed on each fuel plate during postirradiation examination.

Fuel powder for use in fuel plate fabrication was produced using three techniques. U-10Mo and U-7Mo metallic powders were produced at the Atomic Energy of Canada, Ltd. (AECL) using a grinding technique (ground fuel alloys appear without a superscript in Table 1); U-10Mo, U-7Mo, U-6Mo and U-6Mo-1.7Os metallic powders were produced at the Korea Atomic Energy Research Institute (KAERI) by atomization (atomized fuel alloys are denoted by a superscript "a" in Table 1); and U_3Si_2 powder was produced at Argonne National Laboratory by communition. The three uranium silicide fuel plates have been included in the test matrix as control fuel plates having well-known irradiation performance characteristics. The AECL and KAERI powders were provided without cost to the U.S. RERTR program.

Fuel powder was blended with aluminum powder and pressed into compacts. Compacts were placed in Al-6061 picture frames, sandwiched between Al-6061 coverplates, roll bonded to final thickness, and sheared to final fuel plate length and width. Finished fuel plates were cleaned and autoclaved in saturated water at 180°C for four to six hours to pre-film the cladding surface with a 0.1 mil (3 μm) thick corrosion-resistant boehmite layer.

All fuel plates were fabricated using low-enriched uranium (~19.5% U^{235}). Confirmatory x-ray density measurements indicated that the target 6 and 8 g-U/cm^3 fuel loadings were achieved in the finished fuel plates. Metallographic cross-sections of typical fuel plates fabricated to these densities are shown in Fig. 2.

Two fuel plates having no aluminum matrix were fabricated and included in RERTR-4, the lower-burnup test. The fabrication of these fuel plates is detailed in another paper in these proceedings [6]. These fuel plates are being tested as the "ultimate" in density achievable using an aluminum-clad U-Mo fuel plate. These fuels appear in Table 1 with the superscript "foil."

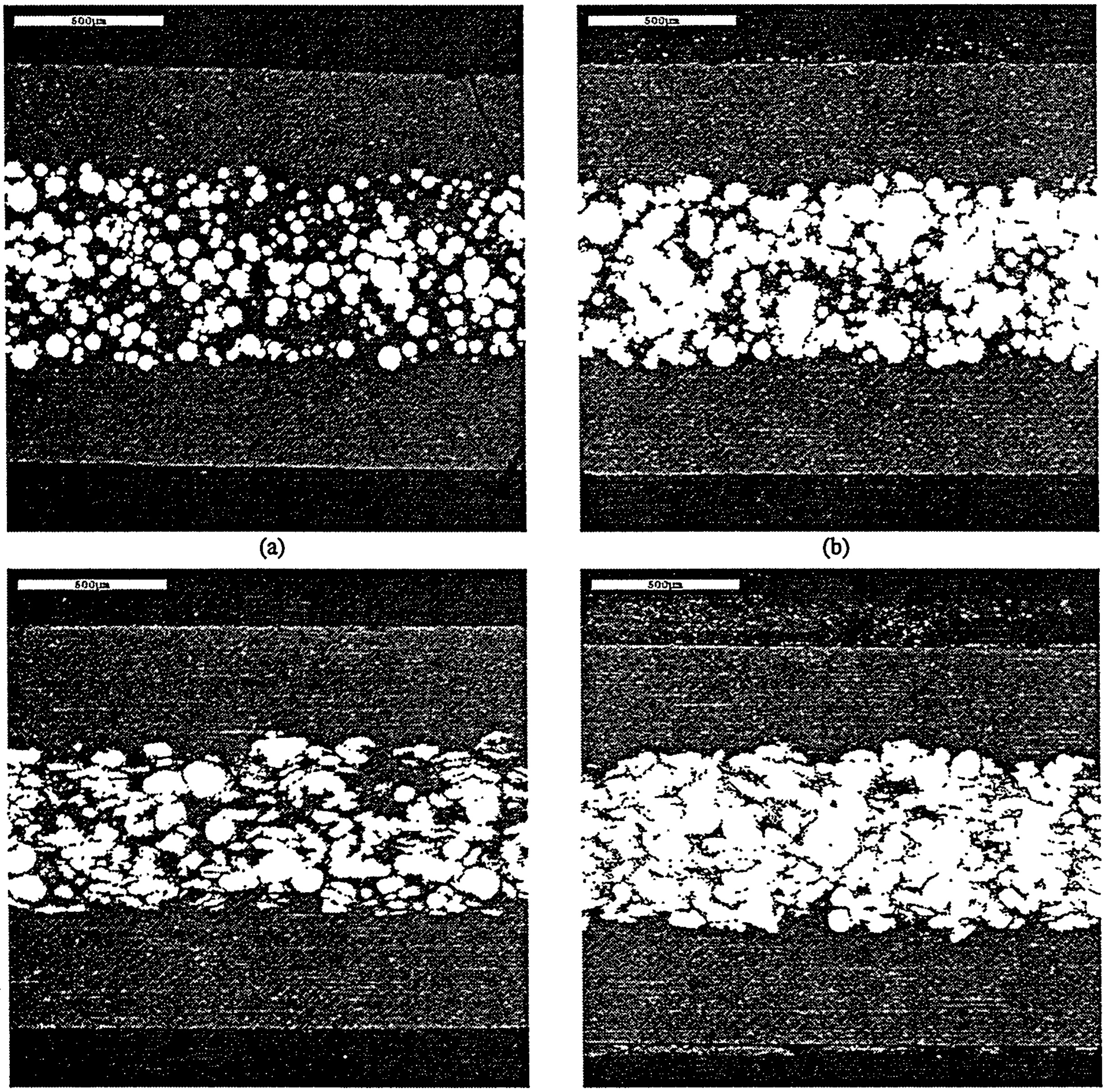
IRRADIATION TEST CONDITIONS

Irradiation of these experiments in the large "B" positions of the ATR provides the highest thermal flux, thus highest fuel plate powers, for experimental fuel plates of this size in a flow-through channel. Still, fuel temperatures were projected to be below the desired values using the basket design of RERTR-1 and -2, which had been designed to be low-temperature tests. A flow orifice was incorporated into the bottom of the baskets for both experiments; the orifice was designed to reduce the flow velocity to 400 cm/sec in the region of the fuel plates.

The 32 fuel plates in RERTR-4 generate a total of 179.3 kW at beginning-of-life, while those in RERTR-5 generate 143.4 kW. The fuel plates in RERTR-5 generate less power, and consequently run somewhat cooler, than those in RERTR-4 due to the lower ATR power in the quadrant where the RERTR-5 test resides. The flow-through capsules allow cooling of the experiment by the primary reactor coolant which enters capsule A at 52°C and exits capsule D at 103°C and 93°C in RERTR-4 and -5, respectively. Thermal calculations indicate that a peak cladding temperature of 171°C and a fuel temperature of 182°C at beginning-of-life are expected in the fuel plate located in capsule position 4C-4. Over half of the fuel plates in RERTR-4 have peak fuel temperatures above 160°C at beginning-of-life. The peak fuel temperature in RERTR-5 at beginning-of-life is calculated to be 156°C for the fuel plate located in capsule position 5C-4.

SUMMARY

Irradiation of RERTR-4 and -5 began in August 2000. RERTR-4 is scheduled to be discharged from the ATR in January 2001 following three irradiation cycles (115 days of irradiation) having peak fuel burnups near 50 at.% U^{235} ($3.7 \cdot 10^{21}$ fiss./cm^3). RERTR-5 will continue irradiation into the summer of 2001 to a terminal burnup of 80 at.% U^{235} ($6.0 \cdot 10^{21}$ fiss./cm^3). Postirradiation



(a) (b) (c) (d)

Figure 2. U-10Mo fuel plates fabricated using KAERI atomized fuel powder at (a) 6 g-U/cm^3, and (b) 8 g-U cm^3; and AECL ground fuel powder at (c) 6 g-U/cm^3, and (d) 8 g-U cm^3.

examination of the discharged experiments will be performed at the Alpha-Gamma Hot Cell Facility at ANL in Chicago beginning in the spring of 2001 for RERTR-4 and the fall of 2001 for RERTR-5. These experiments will provide irradiation performance data on U-Mo alloy dispersion fuels at thermal conditions typical of high power research reactors, at both high uranium loadings and high burnup.

REFERENCES

[1] S. L. Hayes, C. L. Trybus and M. K. Meyer, "Irradiation Testing of High-Density Uranium Ally Dispersion Fuels," Proc. of the 1997 International Meeting on Reduced Enrichment for Research and Test Reactors, 5-10 October 1997, Jackson Hole, Wyoming, USA.

[2] S. L. Hayes, M. K. Meyer, C. R. Clark, J. R. Stuart, I. G. Prokofiev and T. C. Wiencek, "Prototypic Irradiation Testing of High-Density U-Mo Alloy Dispersion Fuels," Proc. of the 1999 International Meeting on Reduced Enrichment for Research and Test Reactors, 3-8 October 1999, Budapest, Hungary.

[3] M. K. Meyer, G. L. Hofman, R. V. Strain, C. R. Clark and J. R. Stuart, "Metallographic Analysis of Irradiated RERTR-3 Fuel Test Specimens," Proc. of the 2000 International Meeting on Reduced Enrichment for Research and Test Reactors, 1-6 October 2000, Las Vegas, Nevada, USA.

[4] M. K. Meyer, G. L. Hofman, J. L. Snelgrove, C. R. Clark, S. L. Hayes, R. V. Strain, J. M. Park and K. H. Kim, "Irradiation Behavior of Uranium-Molybdenum Fuel: Quantitative Data from RERTR-1 and -2," Proc. of the 1999 International Meeting on Reduced Enrichment for Research and Test Reactors, 3-8 October 1999, Budapest, Hungary.

[5] S. L. Hayes, M. K. Meyer, G. L. Hofman and R. V. Strain, "Postirradiation Examination of High-Density Uranium Alloy Dispersion Fuels," Proc. of the 1998 International Meeting on Reduced Enrichment for Research and Test Reactors, 18-23 October 1998, São Paulo, Brazil.

[6] T. C. Wiencek and I. G. Prokofiev "Low-Enriched Uranium-Molybdenum Fuel Plate Development," Proc. of the 2000 International Meeting on Reduced Enrichment for Research and Test Reactors, 1-6 October 2000, Las Vegas, Nevada, USA.

Table 1. Fuel Plate Configuration in RERTR-4 and -5

Capsule Designation	Fuel Plate Designation	Cladding Alloy	Matrix Phase Material	Uranium Density (g-U/cm^3)	Fuel Phase Composition†
4A/5A	4A-1/5A-1	Al-6061	Al	6.0	U-10Mo[a] / U-7Mo
	4A-2/5A-2	Al-6061	Al	6.0	U-6Mo[a]
	4A-3/5A-3	Al-6061	Al	6.0	U-10Mo
	4A-4/5A-4	Al-6061	Al	6.0	U-10Mo[a]
	4A-5/5A-5	Al-6061	Al	6.0	U_3Si_2
	4A-6/5A-6	Al-6061	--- / Al	15.2 / 8.0	U-10Mo[foil] / U-6Mo-1.7Os[a]
	4A-7/5A-7	Al-6061	Al	8.0	U-10Mo
	4A-8/5A-8	Al-6061	Al	8.0	U-7Mo
4B/5B	4B-1/5B-1	Al-6061	Al	6.0	U-7Mo
	4B-2/5B-2	Al-6061	Al	6.0	U-6Mo[a]
	4B-3/5B-3	Al-6061	Al	6.0	U-10Mo
	4B-4/5B-4	Al-6061	Al	6.0	U-10Mo[a]
	4B-5/5B-5	Al-6061	Al	6.0	U-7Mo[a]
	4B-6/5B-6	Al-6061	Al	8.0	U-6Mo[a]
	4B-7/5B-7	Al-6061	Al	8.0	U-7Mo[a]
	4B-8/5B-8	Al-6061	Al	8.0	U-7Mo
4C/5C	4C-1/5C-1	Al-6061	Al	6.0	U_3Si_2
	4C-2/5C-2	Al-6061	Al	8.0	U-10Mo[a]
	4C-3/5C-3	Al-6061	--- / Al	15.2 / 8.0	U-10Mo[foil] / U-6Mo-1.7Os[a]
	4C-4/5C-4	Al-6061	Al	8.0	U-10Mo
	4C-5/5C-5	Al-6061	Al	6.0	U-7Mo
	4C-6/5C-6	Al-6061	Al	6.0	U-6Mo[a]
	4C-7/5C-7	Al-6061	Al	6.0	U-10Mo
	4C-8/5C-8	Al-6061	Al	6.0	U-10Mo[a]
4D/5D	4D-1/5D-1	Al-6061	Al	6.0	U-7Mo
	4D-2/5D-2	Al-6061	Al	6.0	U-6Mo[a]
	4D-3/5D-3	Al-6061	Al	6.0	U-10Mo
	4D-4/5D-4	Al-6061	Al	6.0	U-10Mo[a]
	4D-5/5D-5	Al-6061	Al	6.0	U_3Si_2
	4D-6/5D-6	Al-6061	Al	8.0	U-6Mo[a]
	4D-7/5D-7	Al-6061	Al	6.0	U-7Mo[a]
	4D-8/5D-8	Al-6061	Al	8.0	U-10Mo[a]

†Alloy compositions given in wt.%. [a]Atomized alloy powder.
[foil]Al-6061 clad U-10Mo foil.