EXTENDED LIFE ALUMINIDE FUEL FOR
UNIVERSITY RESEARCH REACTORS

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ABSTRACT

EG&G Idaho/Department of Energy, University of Missouri, and Massachusetts Institute of Technology are conducting a test program to determine if the fuel loading and burnup limits for fuel elements in university research reactors can be safely increased beyond the limits presently allowed by reactor licensing restrictions. For the tests, 30 fuel plates were constructed to a maximum fuel loading which could be produced on a commercial basis and to contain a maximum boron content as used in the Advanced Test Reactor at the Idaho National Engineering Laboratory to reduce initial reactor reactivity. A $UA1_2$ fuel matrix was used to gain higher uranium content. The test program planned for the fuel plates to be irradiated to a 3.3×10^{21} fissions/cm³ average burnup (45% of U-235 for the 50 vol% fuel plate cores). This would be twice the burnup presently allowed in the university reactors. Irradiation performance of the heavy loaded fuel plates has been good at burnups exceeding 2.3×10^{21} fissions/cm³, with one fuel plate reaching a peak burnup of about 3×10^{21} fissions/cm³. Three fuel plates failed, however, during the irradiation, and are undergoing destructive analysis. Corrosion pitting occurred in cladding of both $UA1_2$ and $UA1_3$ fuel plates. Some plates appear to be more resistant to corrosion pitting than others. Localized swelling in high fuel loaded plates also is being investigated as a possible failure mode.

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BACKGROUND

The Reduced Enrichment Research Test Reactor (RERTR) Program seeks to reduce enrichment levels in most research reactors by coupling an enrichment reduction with increased uranium weight over present loadings. Research reactors at several universities, however, cannot continue present performance levels with reduced fissionable atom concentration. Fuel elements for these reactors are provided under the university fuel procurement Test, Research, and Training Reactor (TRTR) Program by the Idaho National Engineering Laboratory (INEL) as directed by the United States Department of Energy--Idaho Office (DOE-ID).

As the prices of fuel fabricating, shipping, and reprocessing continue to rise at rapid rates, researchers look for alternate methods to keep their reactor fuel costs within limited funding. Extending fuel element lifetimes without jeopardizing reactor safety can reduce fuel costs by up to a factor of two. But to gain this factor, fuel plates must be tested to the higher burnup limits to verify the fuel plate burnup performance.

Early in 1980, support for a heavily loaded, high-burnup fuel plate irradiation test program was received from the University of Missouri and the Massachusetts Institute of Technology (MIT). This support is a result of the fuel cost savings to be realized from high burnup of high-fuel-loaded plates in their research reactors. A program plan was completed, and 30 fuel plates containing 40, 45, and 50 vol% $UA1_2$ and 50 vol% $UA1_3$ core materials were fabricated. The $UA1_3$ plates were included as a baseline case.

The test program was started in July 1981. When the test program is completed, results are expected to provide the University of Missouri and the MIT with technical justification for using high-burnup, high-fuel-loaded long-life aluminide fuel elements. Fuel cost savings at Missouri University Research Reactor (MURR) alone would be about \$250,000 per year. Results of

these tests could also be used by the RERTR Program to justify the higher uranium loadings in fuel plates in the form of UAl_2 .

UAl_2 VERSUS UAl_3 FUEL CORE MATRIX

Considerable radiation testing has been done with UAl_x powder in which UAl_3 is the principal crystalline constituent; and several years of operational experience in the Advanced Test Reactor (ATR) at the INEL and other reactors has justified the MIT reactor (MITR) and MURR maximum burnup of 1.8×10^{21} fissions/cm³. The ATR burnup limit has been extended in steps to a fission density of 2.3×10^{21} fissions/cm³ in a UAl_3 core. A report by Beeston et al.¹ indicates that an extension to 2.7×10^{21} fissions/cm³ of core burnup is possible.

The fuel powder composition which is predominantly UAl_3 was originally chosen since the powder can be prepared in air. UAl_2 powder is more pyrophoric than UAl_3 . Some work has been done by Dienst et al.² at Karlsruhe on irradiation behavior of UAl_2 -Al and UAl_3 -Al dispersion fuels for thermal high-flux reactors. Although they did not study the range of our required fuel loading, data strongly suggest that for high-loaded, high-burnup fuel plates, UAl_2 -Al dispersion performs considerably better than UAl_3 -Al.

It is shown in Figure 1 that higher total uranium concentration can be achieved with UAl_2 than with UAl_3 for the same weight or the same volume percent. However, UAl_x is a brittle constituent so that some limit on the core uranium content is required for rolling fabrication. The uranium content of UAl_2 is 81.0 wt%, and that of UAl_3 is 74 wt%.

If 50 vol% of UAl_x were taken as a limit of brittle constituent UAl_x , then from Figure 1, the uranium concentration using UAl_3 would be 2.53 g/cm³ of core, and using UAl_2 would be 3.31 g/cm³ of core, and the corresponding core weight percent of UAl_3 and UAl_2 would be 72 and 75, respectively. Thus, using UAl_3 , the core would be mostly brittle constituent (some of the volume is voids), while using UAl_2 , more ductile aluminum can be present. A rolling test program at Atomics International

Figure 1. Uranium content versus weight or volume percent UAl_x .

(AI)³ indicated current technology can be used to produce quality fuel plates on a production line basis with fuel cores containing up to 50 vol% UAl_x.

The MURR and MITR fuel plates, however, are currently being made with a uranium content of $\sim 1.6 \text{ g/cm}^3$, or about half the possible fuel content limit. Therefore, by using the recent data from the AI rolling tests and UAl₂ powder, the uranium content in the plates can be increased from the present value by about 100%, depending on void content. This increase will be important where long-life elements are desired, although some reactors at present may not be able to take full advantage of the full 100% increase due to excess reactivity limitations even with a high boron loading.

Although UAl₂ is more pyrophoric than UAl₃, current fabrication techniques are compatible with either dispersion. The glove boxes use an argon atmosphere with a maximum oxygen content of 4%.

FUEL PLATE DESIGN

Fuel plate dimensions were selected to fit the ATR I-hole configuration and to provide the plate area required for testing. Thickness of plates and cores and plate construction methods were selected to match the MURR and ATR fuel. Extrapolation of the test data to a 1.524-mm (0.060-in.) plate will provide MIT with the required supporting data for extended fuel burnup in the MITR. Table 1 indicates the test specifications.

The finished plates measure $25.4 \pm 0.127 \times 317.5 \pm 0.762 \times 1.27 \pm 0.025 \text{ mm}$ ($1.000 \pm 0.005 \times 12.50 \pm 0.030 \times 0.050 \pm 0.001 \text{ in.}$). The fuel core dimensions are $\sim 20.32 \times 266.7 \times 0.508 \text{ mm}$ ($0.8 \times 10.5 \times 0.02 \text{ in.}$). A 9.525-mm (3/8-in.) hole centered in the top end of each plate provides a means for individual plate removal in the canal or hot cell.

The UAl powder was prepared, and the fuel plates were fabricated by AI. The U-235 enrichment was $93.0 \pm 1.0\%$ for all batches. The metal impurities were less than 0.3%, with no individual impurity exceeding 600 ppm. No free metallic uranium was present in any powder sample.

TABLE 1. TEST SPECIFICATIONS

<u>Compound</u>	<u>U (g/cm³)^a</u>	<u>UA1_x (core vol%)</u>	<u>Estimated Maximum Burnup (fission/cm³)</u>
UA1 ₂	2.51	40	2.9×10^{21}
UA1 ₂	2.82	45	3.0×10^{21}
UA1 ₂	3.13	50	3.3×10^{21}
UA1 ₃	2.22	50	2.7×10^{21}

a. These are theoretical values. Expected values may differ slightly due to other crystalline forms of UA1_x in the core material.

Nondestructive testing (NDT) inspections for nonbond and minimum cladding thickness met the acceptance criteria of the ATR Fuel Element Specification IN-F-9-ATR, Rev. 5.

IRRADIATION

Irradiation was begun in the ATR I-9 facility in July 1981, and has continued intermittently since then. The thermal flux in the facility varies between 3 and 7×10^{13} n/cm²s. The peak gamma heat in the facility of 1.55 W/g was used with corrosion film estimates in the MACABRE computer code to calculate maximum fuel plate temperatures versus operating time. The maximum nominal and 2σ plate temperatures were 395 and 407 K, respectively, and decreased with operating time.

Neutron flux and burnup calculations were made with the PDQ neutron diffusion-depletion program through the irradiation history of the fuel plates. Two-dimensional XY and RZ-4 energy group PDQ problems were developed to model the tests. In the burnup calculation for each test cycle, a correction factor was applied so that the calculated thermal neutron flux matched the measured value obtained from the flux wire monitors, which were placed in the test and removed after each test cycle.

RESULTS

Irradiation of 19 plates to burnups indicated in Figure 2 have been conducted. The cumulative average and peak fission densities are given in Table 2. In general, the performance has been good. One plate in the 50 vol% UA1₂ composition has reached a burnup of about 3×10^{21} fissions/cm³ (57.8% of U-235 burned up). The reference 50 vol% UA1₃ plates are also performing well, but at lower burnup because of their location in the irradiation facility. It is a reference fuel only in the sense of proven fabrication techniques, but not in the sense of proven irradiation burnup behavior. Most ATR U-Al dispersion fuel plates have been fabricated at about 32 vol% UA1_x. From Figure 2, it can be seen that the 40 vol% UA1₂ is performing well with a peak burnup of about 2.5×10^{21} fissions/cm³ of core which compares with most ATR UA1_x

Fuel plate

- Failed plate
 - Removed because of pitting, etc.
 - ▲ Continuing in test
- * Average plate burnup about 75% of peak.
- INEL 3 1293

Figure 2. Fuel plate peak burnup levels.

TABLE 2. RANGES OF FUEL PLATE BURNUP

Composition	Cumulative Average Fission Density (x10 ²¹ fissions/cm ³)	Cumulative Peak Fission Density (x10 ²¹ fissions/cm ³)
40 vol% UAl ₂	1.01 to 1.89	1.39 to 2.49
45 vol% UAl ₂	0.95 to 1.25	1.31 to 1.75
50 vol% UAl ₂	1.12 to 2.23	1.45 to 2.95
50 vol% UAl ₃	0.73 to 1.43	1.02 to 1.61

dispersion fuel plates with a burnup limit of 2.3×10^{21} fissions/cm³. The 40 vol% UAl₂-Al dispersion has achieved an irradiation behavior significant in two respects: (a) it has surpassed the burnup limit of 2.3×10^{21} fissions/cm³ and is approaching the experimentally achieved fission density¹ of 2.7×10^{21} fissions/cm³ (in ATR), but is below the experimentally achieved fission density of 3.4×10^{21} fissions/cm³ (reported by Dienst²) and the test program target fission density⁴ of 3.3×10^{21} fissions/cm³; and (b) the 40 vol% composition (2.5 g/cm³) has a higher uranium fuel loading than ATR U-Al dispersion fuel plates.

The irradiation achievement of the 40 vol% UAl₂ can also be applied to the 50 vol% UAl₂, and at lower fission density for the 45 vol% UAl₂, except that two plate failures occurred in the 50 vol% UAl₂ and one in the 45 vol% UAl₂ (see Figure 2). The cause of these plate failures is being investigated, but appears to be attributable to one of two possible failure modes: (a) swelling of nonbonded or large void regions and (b) corrosion pit perforation of the fuel cladding. Two other unlikely failure modes are (a) swelling of UAl_x fuel from fission gas production and (b) restricted cooling to the plate surface with overheating. Swelling of UAl_x fuel is unlikely because:

1. The blister temperature is probably still high¹ at >720 K. Measurements on the three failed plates (Figure 2) also gave high blister temperatures, >720 K, which indicates accommodation of the fission gas in small nuclei or within the UAl_x fuel structure.
2. UAl_x fuel has been irradiated to higher fission density² without indication of general fission gas swelling.

Restricted cooling of the plate surface is unlikely because: (a) no evidence was seen of restricted channels except for corrosion products, and (b) the maximum 2 σ plate temperature of 407 K gives such a high margin for this failure mode, as compared with the blister temperature of 720 K.

Evidence for the first failure mode has not been observed. Swelling of nonbonded or large void regions should produce localized swelling of the

fuel plate. Localized swelled regions have been observed, but these were associated with perforation by pitting. Thickness measurements were made on two plates after reactor Cycle 58D-1. Nine measurements of thickness over the length of the surface of one plate resulted in a thickness increase of 5%, based on a core thickness of 0.508 mm (0.020 in.) (a volume swelling of 15%), and on the second plate the result was a thickness increase of 4.6% (14 vol%). This volume swelling for the peak fission density along the edge of the fuel region is about the same as observed in Reference 2 at about the same fission density and plate temperature of 453 K.

Corrosion pitting in the surface of the fuel cladding has also been observed. After reactor Cycle 58D-1, an oxide prefilm was applied to the fuel plate cladding surface to reduce the incidence of pitting. However, the effect was minor. After further irradiation, the old plates had fewer corrosion spots (pits) than the prefilmed replacement plates. By comparison of the number of corrosion pits on the prefilmed replacement plates with the old plates, the prefilming has not reduced the corrosion pitting problem. As indicated in Figure 2, five plates have not developed corrosion pits and have continued for all test periods; hence, some plates are more resistant to formation of corrosion pits than others.

CONCLUSIONS

1. The irradiation performance of UA1_2 fuel plates with higher volume percent of UA1_2 than conventional ATR fuel plates (32 vol% UA1_x) has been good.
2. An increase in the burnup limit beyond 2.3×10^{21} fission/cm³ of core for higher fuel loaded plates than conventional appears to be feasible without major risk.
3. Corrosion pitting of high-fuel-loaded plates as well as conventional plates is an important failure mode.
4. Some plates seem to be more resistant to corrosion pitting than others.

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