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IRRADIATION TESTING OF MINIATURE FUEL PLATES
FOR THE RERTR PROGRAM

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IRRADIATION TESTING OF MINIATURE FUEL PLATES FOR THE RERTR PROGRAM

R. L. Senn M. M. Martin*

ABSTRACT

An irradiation test facility, which provides a test bed for irradiating a variety of miniature fuel plates (miniplates) for the Reduced Enrichment Research and Test Reactors program, has been placed into operation. These tests screen various candidate fuel materials on their suitability for replacing the highly enriched uranium fuel materials currently used by the world's test and research reactors with a lower enrichment fuel material, without significantly degrading reactor operating characteristics and power levels. The use of low uranium enrichment of about 20% ^{235}U in place of highly enriched fuel for these reactors would reduce the potential for ^{235}U diversion.

Fuel materials currently being evaluated in this first phase of these screening tests include aluminum-base dispersion-type fuel plates with fuel cores of (1) high uranium content $\text{U}_3\text{O}_8\text{-Al}$ being developed by Oak Ridge National Laboratory, (2) high uranium content $\text{UAl}_x\text{-Al}$ being developed by EG&G Idaho, Inc., and (3) very high uranium content $\text{U}_3\text{Si-Al}$ being developed by Argonne National Laboratory.

The miniplates are 115 mm long by 50 mm wide with overall plate thicknesses of 1.27 or 1.52 mm. The fuel core dimensions vary according to overall plate thicknesses with a minimal clad thickness requirement of 0.20 mm. Sixty such miniplates (thirty of each thickness) can be irradiated in one test facility.

The irradiation test facility, designated as HFED, is operating in core position E-7 in the Oak Ridge Research Reactor (ORR), a 30-MW water-moderated reactor. The peak neutron flux measured for this experiment is 1.96×10^{18} neutrons $\text{m}^{-2} \text{s}^{-1}$. The various types of miniplates will achieve burnups of up to $\sim 2.2 \times 10^{27}$ fissions/ m^3 of fuel, which will require approximately eight full-power months of irradiation.

During reactor shutdown periods, the experiment is removed from the reactor, moved to a special poolside station, disassembled, and inspected for dimensional changes, blisters, or any other visible defect. After inspection the experiment is reassembled and reinserted into the reactor for further irradiation. At completion of the specified in-reactor exposure of the miniplates, the experiment will be disassembled, and the irradiated samples will be returned to the respective fuel-plate fabricators for detailed postirradiation examination. Ultimately, fuel-plate types with suitable characteristics will be manufactured into full-sized plate-type fuel elements suitable for testing in the ORR.

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1. INTRODUCTION

The Reduced Enrichment Research and Test Reactors (RERTR) program is directed toward the replacement of fully enriched uranium fuel materials currently used by the world's research and test reactors with a lower enrichment fuel material. Implementation of this change is necessary, however, without significantly degrading reactor operating characteristics and power levels. The use of low-uranium enrichment of about 20% ^{235}U in place of highly enriched fuel for these reactors would reduce the potential for ^{235}U diversion.

As part of the overall program designed to implement this goal, an irradiation test facility has been designed, built, and put into operation. This facility provides a test bed for irradiating a variety of miniature fuel plates (miniplates). The experiment is in direct support of the Argonne National Laboratory (ANL) program to develop high uranium density research and test reactor fuel to accommodate the use of low-uranium enrichment. The ANL coordinates and manages the national program for the Department of Energy (DOE). Through DOE, ANL has delegated the responsibility for irradiation testing of the various candidate fuels to the Oak Ridge National Laboratory (ORNL).

2. OBJECTIVE

The objective of these irradiation tests is to screen various candidate fuel materials as to their suitability for replacing the fully enriched uranium fuel materials currently in use. Fuel materials to be evaluated in the first phase of these screening tests include aluminum-base dispersion-type fuel plates with fuel cores of (1) high uranium content U_3O_8 -Al being developed by ORNL, (2) high uranium content UAl_x -Al being developed by EG&G Idaho, Inc., and (3) very high uranium content U_3Si -Al being developed by ANL. These plates are being irradiated in a core position in the Oak Ridge Research Reactor (ORR) with a peak neutron flux of $\sim 1.96 \times 10^{18}$ neutrons $m^{-2} s^{-1}$ to burnups of up to 2.2×10^{27} fissions/ m^3 of fuel core volume. Ultimately, fuel plates with suitable characteristics will be incorporated into full-sized plate-type fuel elements suitable for testing in the ORR.

3. DESCRIPTION OF TEST FACILITY

The irradiation test facility, designated HFED-1 (High-Uranium-Loaded Fuel Element Development test 1) is designed to fit into a core position in the ORR, a 30-MW water-moderated reactor. The HFED-1 experiment is currently operating in core position E-7 of the ORR.

A rectangular miniplate, 115 mm long by 50 mm wide, was chosen as the standard test size. The facility was designed to accommodate over-all plate thicknesses of 1.27 and 1.52 mm; the fuel core dimensions vary according to overall plate thickness, with a minimal clad thickness requirement of 0.20 mm. Sixty such miniplates (thirty of each thickness) can be irradiated in one facility.

Five separate modules, each containing 12 miniplates, are stacked one above another inside a split aluminum holder. The holder, when closed, fits inside a standard ORR aluminum core piece. This assembly, with the same outside dimensions as an ORR fuel element, can be placed in any of the fueled positions of the ORR core. A photograph showing an exploded view of the HFED-1 experiment and its various parts is presented in Fig. 1.

Each aluminum module is designed to hold 12 miniplates: 6 of each thickness alternated one by one. Figure 2 shows a detailed view of a typical module with dummy miniplates installed. The miniplates are held in position in their respective slots by a hafnium (Hf) grid located in the bottom of each module. The Hf grids not only retain the miniplates but also serve as neutron absorbers to depress the power peaks that might occur at the ends of the plate because of (1) higher fuel loading caused by dogboning* and/or (2) neutron flux peaking that may occur in the water gap between the ends of each module assembly. The modules are designed to provide a 2.54-mm-wide water channel between each plate for cooling. The reactor coolant water flows down through all five modules in series. Plates of the same thickness are all in the same vertical plane through all five modules, thus ensuring that each 2.54-mm coolant channel is a constant dimension from the top to the bottom of the experimental section.

*Dogboning — A term used to describe fuel core thickening that may occur near the ends of a fuel plate as a consequence of fabrication technique.

Fig. 1. Exploded view of the RERTR HFED-1 irradiation experiment showing all pertinent components.

Fig. 2. Detailed view of module 1 with dummy miniplates installed showing typical arrangement of miniplates. Note electron-beam welds along edges. Slots are widened at top edge to aid in miniplate installation.

The modules are stacked into the module holder one above the other, with module 1 uppermost, through module 5 at the bottom of the assembly. Each of the 12 slots in each module is numbered so that a specific location can be assigned to each miniplate being tested.

When the five modules are placed in their respective positions within the module holder, the holder can be closed and fastened with a screw at

the top. The entire assembly can then be inserted into the aluminum core piece, which is then ready for insertion into the reactor. Conversely, the experiment may be removed from the reactor during reactor down periods and disassembled underwater for inspection, removal, or replacement of modules or individual fuel plates; the experiment is then reassembled and reinserted into the reactor.

4. EXPERIMENT DESIGN PARAMETERS

Extensive thermal-hydraulic calculations were required to define acceptable operating limits for the various fuel materials, enrichments, and uranium loadings that were proposed for irradiation. A major criterion was that no boiling is permitted within the reactor. Because of the reactor flux pattern, module 4 (fourth in line from the top) sees the peak flux. Therefore, maximal fuel loadings for this module are restricted to a level that will provide for a peak surface temperature of the fuel plate that is $\sim 17^{\circ}\text{C}$ ($\sim 30^{\circ}\text{F}$) below the saturation temperature, $\sim 118^{\circ}\text{C}$ ($\sim 245^{\circ}\text{F}$), of the reactor coolant water at that position. Similar determinations were made for the maximal fuel loadings permitted for each of the five modules. Typical results from these calculations are shown in Fig. 3.

The hydraulic calculations were confirmed by performing an out-of-reactor flow test in a facility designed to provide flow rates and pressure drops equivalent to those obtained in the reactor core. Although the flow patterns through the experiment are complicated because of the flow (1) through the channels between the plates, (2) around the module holder

Fig. 3. Temperature profile of miniplate surface temperature vs elevation within the reactor core — middle channel.

and the modules, and (3) between the module holder and the core piece, the calculations and test results were in good agreement (Fig. 4).

Further confirmation of the thermal calculations was achieved by performing fission rate tests with the HFED-1 assembly in the ORNL Pool Critical Assembly (PCA). Three tests were performed to determine fission rates at various elevations in the experiment as compared with a standard 240-g ^{235}U -bearing ORR fuel element at the same elevations. A small fission chamber mounted on the end of a long probe was moved up and down to known elevations within the assemblies. The fission chamber was attached to electronic counting equipment that provided a digital printout of the ^{235}U fission count rate at predetermined time intervals.

The first of these tests was conducted with the HFED-1 experiment loaded with the test miniplates in their design locations. Module 1 was replaced with trial module 1S, which has a stainless steel grid in place

Fig. 4. Full-scale flow test results for HFED-1 irradiation test.

of the Hf grid located in the bottom of each module. The fission chamber was inserted into the center channel of the miniplate array, and these results were compared with similar measurements made on the standard ORR element. The PCA was operated at 50 W during all tests. The results from this first test (Fig. 5) showed that a sharp flux depression occurred in the vicinity of the Hf grids, and a sharp flux peaking occurred in the vicinity of the stainless steel grid. The general shape of the fission rate profile for the HFED-1 assembly compared quite well with the shape of the profile on the standard element, except in the grid areas. The effects of the Hf and stainless steel did not seem to influence the center regions of the miniplates, however. There was a general flux depression of ~50% as shown by a count ratio of ~0.5, compared with a test of an ORR fuel element in the same position.

A second test was performed to assess the value of removing some of the Hf from the grids (without remaking the module assembly). The lower bars and the comb were cut away from the Hf grid in module 6, one of the spare modules. This modification removed about 35% of the Hf that was originally in the grid and left an ~5-mm water gap between the bottom of this module and the top of the next. The miniplates previously installed in module 3 were placed in module 6, which was installed in the third position in the HFED-1 assembly. The miniplates originally installed in

Fig. 5. Fission-rate profile for HFED-1 irradiation test in the PCA.

module 1S were removed and placed in module 1, and this module was installed in the first (top) position in the assembly. Thus, the configuration for this second experiment was as follows: module 1 with Hf grid, module 2 with Hf grid, module 6 with 2/3 Hf grid, module 4 with Hf grid, and module 5 with Hf grid. The fission-rate scan was then conducted by inserting the fission chamber down the center channel and measuring the fission rates at various elevations. Results from this experiment showed that, as expected, there was a sharp increase in the fission rate where the water gap existed at the bottom of the third module (No. 6) and a smaller depression where the remaining Hf was located than was seen with the complete Hf grid. Again, these localized fission-rate changes did not seem to influence the adjacent central areas of the miniplates, and the fission rates in the center of the miniplates were virtually the same as in the previous test.

A third test was conducted using the same configurations described previously. The fission chamber was inserted into the first full-sized channel at the edge of the module (between the first and second miniplates in the module), and the fission rates were again measured at various elevations. This test was an attempt to determine the effects of self-shielding and to compare the fission rates of the center plates with the edge plates under the same conditions. The shape of the curve of the count ratio between the experiment and the standard element was essentially the same as in previous tests, except that the ratios were about 35% higher in the edge channel. Plates along the edge of the assembly will still operate within the required temperature envelope, even at these higher fission rates, but probably would not operate without a flux-depressing grid of some sort between the modules.

Based on the results from these three tests, it was determined that the experiment would operate within the specified limits using the facility as designed with the Hf grids intact. A typical comparison of the fission-rate profile of the HFED-1 assembly and an ORR fuel element is shown in Fig. 5. The Hf grids served to depress the flux locally and prevent excessive temperatures from dogboning and flux-peaking in the vicinity of the water gap between modules.

Based on both the calculations and the experimental results discussed previously, the maximal fuel loadings that could be accommodated within each module were determined. As additional safety factors, a 15% flux uncertainty was imposed, and each miniplate was assumed to contain the maximal loading uniformly distributed over the heat transfer area. The latter assumption accounts for any localized region of excess uranium content that may occur during plate fabrication. The resulting maximal fissile loadings are 500 g/m² for fuel in modules 1 and 2 and 290 g/m² for fuel in modules 3, 4, and 5. Under these conditions, the fuel-plate surface temperatures should remain below the coolant saturation temperature during normal reactor operation. Also, the reactor upper-limiting setpoints are adequate to prevent fuel burnout should an excursion occur in the ORR.

Some typical results from the thermal-hydraulic calculations are shown in Tables 1 and 2; coolant flow rates and peak surface temperatures are listed for a middle channel and an outer channel, respectively. A computer program was written that provides this information for all coolant channels and for each module.

Table 1. Results from the HFED-1 experiment
thermal-hydraulic calculations

(Middle channel)

Parameters	Normal	Accident
Reactor power level, MW	30	33
ΔP across experiment, kPa	132	72
One main-channel coolant flow, m ³ /h	3.7	2.7
One end-channel coolant flow, m ³ /h	2.0	1.4
Stalk-channel coolant flow, m ³ /h	8.3	6.0
Total coolant flow, m ³ /h	52.6	38.0
Maximum fuel surface temperature, °C	104	124
Minimum ΔT , saturation to surface, °C	20.4	-4.2

Table 2. Results from the HFED-1 experiment
thermal-hydraulic calculations

(Edge channel)

Parameters	Normal	Accident
Reactor power level, MW	30	33
ΔP across experiment, kPa	132	72
Main-channel coolant flow, m ³ /h	3.7	2.7
End-channel coolant flow, m ³ /h	2.0	1.4
Stalk-channel flow, m ³ /h	8.4	6.0
Total coolant flow, m ³ /h	52.8	38.1
Maximum fuel surface temperature, °C	114	138
Minimum ΔT , saturation to surface, °C	8.9	-17.8

5. FUEL-PLATE SPECIFICATION

A fuel-plate specification was prepared to provide guidance to the fabricators and to ensure uniform clad properties of the miniplates. The latter condition confines all testing variables to the uranium-bearing regions of the miniplates. Appendix A presents document Q-11821-ET-002-S-0, "Fuel-Plate Specification for RERTR Irradiation Tests."

6. POOLSIDE STATION AND METROLOGY

As part of the overall irradiation experiment plan, a facility was necessary to permit underwater examination of the HFED-1 experiment during the course of the test period. Accordingly, an underwater table was designed, built, and installed in the ORR pool. Fixtures designed to hold the experiment and the disassembled components of the experiment were fastened to the table. Special handling tools were built that permit opening the module holder and removing the modules, removing single plates, and reassembling the experiment. Figures 6 and 7 are photographs of the poolside station with the HFED-1 experiment partially disassembled after ~2 months of irradiation.

Another technique used for intermittent (during reactor downtimes) examination of the experiment is to measure the channel-gap spacing. A probe, supplied by EG&G Idaho, Inc., incorporating two ultrasonic transducers and the necessary electronics, may be inserted down through each of the channels. The elevation of the probe and the width of the channel can be measured over the entire length of the experiment. The channels are nominally 2.54 mm wide. The probe has a range of ~1.52 to 3.18 mm. The channel-gap measuring device has been used successfully for five separate runs to date.

Fig. 6. Poolside station and fixtures in the ORR pool with the HFED-1 experiment disassembled.

Fig. 7. Detailed view of HFED-1 components in the poolside station.

7. IRRADIATION TESTING

The HFED-1 experiment was installed in core position E-7 of the ORR on July 17, 1980. The reactor subsequently reached full-power operation at 2250 h on July 18, 1980. The experiment continues to operate satisfactorily.

A description of the various miniplates and their positions within the assembly is shown on drawing X3E-11821-0105-Rev. B, attached as Appendix B. Note that the actual maximal fuel loadings are 472 g/m² for plate 0-48-2 in module 1 (of the high-loaded group, modules 1 and 2) and 284 g/m² for plate A-16 in module 3 (of the low-loaded group, modules 3, 4, and 5). A detailed drawing of the miniplate and notations regarding the numbering system are also shown on drawing -0105 in Appendix B. The irradiation of a few miniplates with uranium enrichments of 40 and 45% ²³⁵U was required to achieve the burnup goal of 2.2×10^{27} fissions/m³. The one miniplate with 93% enriched fuel permits a comparison of performance with existing ORR fuel.

Cobalt flux monitor wires were inserted in the middle of the HFED-1 miniplate array and at the outer edge of the array just prior to the initial start-up of the reactor with the experiment in place. These wires were irradiated with the reactor operating at 0.265 MW, subsequently cut into measured segments, and analyzed to determine the neutron flux and flux profile for the experiment.

The peak-measured neutron flux extrapolated to 30-MW reactor power level, based on the 2200-m cross section for cobalt (37 b), was 1.96×10^{18} neutrons m⁻² s⁻¹. This peak was located at the midplane of module 4. A minimum flux of $\sim 0.5 \times 10^{18}$ neutrons m⁻² s⁻¹ was measured near the top of module 1.

The flux profile was as expected from earlier tests in the PCA, showing an increasing flux from the top of module 1 to the midplane of module 4 and then decreasing through module 5 (Fig. 8). Strong local depressions were seen where the Hf grids are located. Note the shift in flux gradient that occurred near the midplane of module 4, where the flux measured for the inner plates is higher than that measured for the outer plates. This shift probably is caused by control rod effects near the experiment.

Fig. 8. Results from flux wire measurements made at initial start-up of the HFED-1 irradiation experiment.

8. RESULTS TO DATE

During some of the reactor refueling shutdowns, which occur approximately every 10 d, the experiment has been removed from the reactor and channel-gap spacing measurements have been made. The measurements to date have included measurements made initially (preirradiation) and during shutdowns on July 31, August 14, October 5, and prior to installation on October 20, 1980 (with new plates in module position 3). Results from these measurements have been compared. Some uncertainty was experienced during the second measurement as to the channel in which the probe was actually inserted. Templates were prepared for subsequent measurements to ensure that the probe was in the desired channel and that the measurements could be repeated.

The probe revealed minor changes in spacing of a few channels, but no measurements have been detected that show any blisters in the cladding or severe narrowing of the channel. A minimum gap spacing of ~ 2 mm (gap is nominally 2.54 mm) has been measured but appears to be the result of bowing rather than a fuel plate defect. The assumed bowing has not changed during the last three measurements. A typical channel-gap spacing measurement is shown in Fig. 9. Twenty-four such scans are made with each channel-gap measuring run: one on each side of the center comb (in the Hf grid) for each of the twelve inner channels within a module and through the experiment.

The experiment was removed from the reactor for an extended period after the October 5, 1980, shutdown. After the fourth channel-gap spacing measurement, the experiment was disassembled to remove module 3 for replacement with a new set of ANL miniplates. This disassemblage afforded the first opportunity for a detailed visual examination of the modules and plates. No detectable differences were evident between the pre- and post-irradiated modules. A 4 $\times$ underwater periscope was used to examine the fuel plates without removing them from the modules. Again, no detectable changes were noted. Even the bowing indicated by the channel-gap spacing measurements was not apparent.

In accordance with the fuel module management plan (Fig. 10), module 3, containing 12 ANL uranium-silicide-fueled miniplates, was scheduled for

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Fig. 9. Typical channel-gap spacing measurements.

Fig. 10. Fuel module management plan.

removal and replacement with new miniplates after approximately two full-power months of operation. Accordingly, at the October 5 shutdown, module 3 was removed in preparation for shipping back to ANL for postirradiation examination, and 12 new miniplates were inserted into a spare module and returned to position 3. At the time of the removal of module 3, the experiment had been irradiated for 2241 MWd, corresponding to ~ 75 full-power days. The estimated burnup for these plates is $\sim 45\%$ fissions per initial fissile atom (FIFA) or $\sim 5 \times 10^{26}$ fissions/ m^3 .

The fuel module management scheme (Fig. 10) is designed to equalize, to some degree, the burnups for plates in the various modules. By exchanging modules in the high-flux regions (modules 4 and 5) with those in the low-flux regions (modules 1 and 2), the desired burnups can be approached for all test miniplates.

Detailed descriptions of the 12 new miniplates received from ANL are presented in Table 3. These plates have been examined in accordance with our established quality assurance (QA) procedures to ensure that they meet all required specifications. The miniplates were accepted as received and

Table 3. Replacement fuel plates for module position 3 in HFED-1 experiment.

Slot No.	Nominal fuel-plate thickness [μm (mils)]	Fuel- plate No.	Meat attributes						
			Fissile- bearing compound	Thickness [μm (mils)]	Uranium density (kg/m^3)	^{235}U enrichment (wt %)	Total ^{235}U loading (mg)	Maximal areal ^{235}U loading	
								g/m^2	% ^a
1	1270 (50)	A-26	U_3SiAl	510 (20.1)	4639	19.88	2088	273	16
2	1524 (60)	A-22	U_3SiAl	503 (19.8)	4592	19.88	2088	258	12
3	1270 (50)	A-42	U_3SiAl	508 (20.0)	4513	19.88	2089	252	11
4	1524 (60)	A-30	U_3SiAl	488 (19.2)	4619	19.88	2090	238	6
5	1270 (50)	A-32	U_3Si_2	480 (18.9)	3754	19.87	1591	197	10
6	1524 (60)	A-36	U_3Si_2	496 (19.5)	3751	19.87	1591	200	8
7	1270 (50)	A-34	U_3Si_2	442 (17.4)	3761	19.87	1591	186	13
8	1524 (60)	A-46	U_3Si_2	498 (19.6)	3723	19.87	1590	195	6
9	1270 (50)	A-24	U_3Si	494 (19.4)	4798	19.88	2079	261	11
10	1524 (60)	A-23	U_3Si	465 (18.3)	4773	19.88	2078	284	28
11	1270 (50)	A-38	U_3Si	468 (18.4)	4813	19.88	2080	281	25
12	1524 (60)	A-28	U_3Si	483 (19.0)	4812	19.88	2079	269	16

^aPercentage maximum above average areal loadings.

have been installed in a spare module (labeled No. 7) and placed in the No. 3 position in the HFED-1 assembly. The experiment was replaced in the reactor for further irradiation on October 21, 1980. Design burnup will be achieved for some of the test miniplates after about 8 months of irradiation. Miniplates from other plate fabricators may be installed in the facility for irradiation testing. When a group of miniplates has reached design burnup, or completed a scheduled irradiation period, the plates will be removed from the assembly and shipped to the respective fuel-plate fabricators for detailed postirradiation examination.

9. FUTURE PLANS AND CAPABILITIES

To ensure that adequate parts would be available for the first HFED-1 experiment, a complete spare set of modules and a spare module holder were built. One of the modules was modified by the removal of part of the Hf grid during the PCA tests but could still be used in position 5 at the lower end of the experiment. Module 7 was used to replace module 3, which was recently removed. Module 3 can probably be reused after removal of the miniplates. If this is the case, a full set of modules is ready for another experiment.

Discussions are under way within the RERTR program with various fuel-plate fabricators regarding future tests in the HFED facility. These possibilities, of course, depend on the need for future testing, which may be deduced from the results of these present tests, the requirements of other fabricators, and the necessary funding required for irradiation tests such as these. In any case, a facility is available in which fuels of many different characteristics may be tested, assuming that the maximal fissile loadings and fuel-plate fabrication techniques meet the specifications determined for the RERTR irradiation test series.

10. CONCLUSIONS

Operation of the HFED-1 irradiation test facility to date has established the capability for testing miniature fuel plates of various (1) fuel matrix compositions, (2) fissionable material densities, and (3) enrichments in a core position in the ORR. The maximum total flux in this position has been measured at 1.96×10^{18} neutrons $m^{-2} s^{-1}$. Burnups of about 2.2×10^{27} fissions/ m^3 of fuel can be achieved in approximately eight full-power months.

The miniplates have been shown to operate under reactor conditions without evidence of cladding failure for more than 75 full-power days to burnups greater than 45% FIFA ($>5 \times 10^{26}$ fissions/ m^3).

The capability has been demonstrated of removing the experiment from the core, making channel-gap spacing measurements, disassembling and exchanging modules, and reassembling and reinstalling the experiment in the reactor to continue the irradiation. To date, remote removal of an individual miniplate from a module has not been demonstrated nor has reinsertion or replacement of a miniplate, although a special tool exists for this task.

In summary, a test facility is in operation in the ORR that serves as a test bed for the RERTR program's miniature fuel plates. The experiment is currently operating with miniplates made with U_3O_8 -Al, UAl_x -Al, and U_3Si -Al fuel cores at a peak flux of 1.96×10^{18} neutrons $m^{-2} s^{-1}$ in an environment equivalent to that seen by the standard ORR fuel elements.

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The experiment could not have succeeded, of course, without the close cooperation of fuel-plate fabricators, their constructive criticism of the fuel-plate specification, and their timely delivery of the various fuel plates manufactured under close quality control with regard to test specifications and requirements. The plate manufacturers, represented by G. L. Copeland (ORNL), R. F. Domagala (ANL), and D. G. Newton (EG&G Idaho, Inc.), provided complete descriptions and inspection information for the fuel plates, which greatly assisted the completion of necessary acceptance inspections. George Copeland also helped in the detailed analysis and inspection of the final fuel plate documentation and assisted in handling and loading the miniplates into their respective modules.

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Appendix A

FUEL PLATE SPECIFICATION FOR RERTR IRRADIATION TESTS

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FUEL PLATE SPECIFICATION

for

RERTR IRRADIATION TESTS

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OAK RIDGE RESEARCH REACTOR

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OAK RIDGE NATIONAL LABORATORY

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FUEL PLATES FOR RERTR IRRADIATION TEST

1.0 Scope

This specification applies to Reduced-Enrichment Research and Test Reactor (RERTR) fuel plates suitable for irradiation in the Oak Ridge Research Reactor (ORR). These aluminum-base dispersion-type plates presently include (1) the high uranium content U_3O_8 -bearing miniplates being developed and fabricated at ORNL, (2) the high uranium content UAl_x -bearing miniplates being developed and fabricated at EG&G Idaho, Inc., and (3) the very high uranium content U_3Si - and $(U_3Si + Al)$ -bearing miniplates being developed and fabricated at ANL.

2.0 UCC-ND Drawings

ORR-HFED Experiment Plate Details	X3E-11821-0105-B
ORR-HFED Experiment Assembly	X3E-11821-0101-A

3.0 Materials

3.1 All materials that compose the cored region of the miniplates shall conform to the requirements of this document and the specifications of the cognizant organization (e.g., ORNL will specify the core materials within the U_3O_8 -bearing miniplates).

3.2 Cover plate and frame cladding shall conform to Standard Specification for Aluminum-Alloy Sheet and Plate, ANSI/ASTM Designation B209-77, for alloy 6061 and Alclad 6061.

4.0 Fuel Plate Requirements

4.1 The fuel plates shall be rolled by a combination of, first, hot rolling, and second, cold rolling at room temperature to a reduction in thickness of at least 10% to insure uniform cladding properties among fabricators.

4.2 After completion of hot rolling and before cold rolling, the fuel plates shall be annealed for 60 ± 10 min at 485 $\pm 20^\circ\text{C}$. After cooling to room temperature, both sides of every fuel plate shall be visually examined for blisters. Visual detection of one or more blisters that would remain in the finally-sized fuel plate shall be cause for rejection.

4.3 After completion of cold rolling, the fuel plates shall be heat-treated at 485 $\pm 20^\circ\text{C}$ for 60 ± 10 min followed by a cooling at a rate of $\leq 28^\circ\text{C/hr}$ to $\leq 260^\circ\text{C}$. After further cooling to room temperature, both sides of every fuel plate shall be visually examined for blisters. Visual detection of one or more blisters that would remain in the finally-sized fuel plates shall be cause for rejection.

4.4 Fuel plate, core, and cladding dimensions shall conform to UCC-ND Drawing X3E11821-0105-A.

4.5 Within the maximum core outline of UCC-ND Drawing X3E11821-0105-A and for any 4-mm to 2-mm-diam spot, the U^{235} surface density, in g/m^2 of core surface area, shall be ≤ 290 for fuel plates to be irradiated in Modules 3, 4, and 5 (see UCC-ND Drawing X3E11821-0101-A for module numbering scheme); and ≤ 500 for fuel plates to be irradiated in Modules 1 and 2.

4.6 The interfaces between the cover plates and frames of the fuel plates shall be metallurgically bonded. Grains that have grown across these interfaces for a minimum of 50% of the interfacial length constitute acceptable bonding.

4.7 The surfaces of the fuel plates shall be free from all foreign matter including moisture, dirt, oil, organic compounds, scale, paint, ink, graphite, all solders, silver, lead, mercury, thorium, chlorine, and fluorine.

4.8 The exterior surfaces of the fuel plates shall be free from uranium after fabrication. A final pickling treatment, which shall be performed before packaging for shipment to ORNL, of submerging the fuel plates in 45 vol% HNO_3 -55 vol% H_2O solution for 2 to 4 minutes at room

temperature followed by rinsing in either demineralized or distilled water meets the intent of this requirement.

4.9 The fuel plates shall be identified as shown on UCC-ND Drawing X3E-11821-0105-A. Serial numbers of 0-001 through 0-999, E-001 through E-999, and A-001 through A-999 are permitted, respectively, for fuel plates supplied by ORNL, EG&G Idaho, Inc., and ANL.

5.0 Inspection

The fuel plate fabricator shall be responsible for inspection and certification of the achievement of requirements set forth in paragraphs 3 and 4 above and in paragraph 6 below.

6.0 Quality Assurance Documentation

6.1 The preirradiation data package for the fuel plates shall include the following items:

- (1) dimensional inspection reports (including an estimation of minimum clad thickness) for each plate;
- (2) radiographs of each plate; and
- (3) results from determination of the meat attributes including a tabulation of (a) plate number, (b) weight of plate, (c) weight of meat, (d) elemental composition of the meat, (e) ^{235}U enrichment (wt %), and (f) total ^{235}U loading.

6.2 Certified copies of inspection and test records to assure compliance with this specification and the preirradiation data package shall be supplied to R. L. Senn, Bldg. 9201-3, MS-5, P. O. Box "Y", Oak Ridge, Tennessee 37830.

7.0 Shipping

7.1 Storage of the fuel plates and final assembly of the HFED experiment will be conducted in Building 3012, ORNL. The fuel plate

fabricators shall provide shipping containers and shall have full responsibility for the fuel plates while in transit to the site designated below. A duplicate copy of the tabulation of 6.1 (3) shall be included in the shipping container with the plates.

7.2 Fuel plate fabricators shall ship the fuel plates by acceptable means to the Oak Ridge National Laboratory, Building 3012, Oak Ridge, Tennessee 37830, Attention: H. C. Austin, Bldg. 3037 for M. M. Martin, Bldg. 4508, ORNL.

7.3 Accountability shall be transferred to facility F.Z.C., H. C. Austin, Bldg. 3037, ORNL.

Appendix B

ORR-HFED EXPERIMENT AND FUEL-PLATE DETAILS
FROM DRAWING X3E-11821-0105-B

- 1. THERE SHALL BE NO SCRATCHES OR FITE MARKS THAN 1/8" DEEP, LONGER THAN 1/8" OR ONE LOCATED WITHIN THE MAXIMUM CORE OUTLINE.
- 2. WEARINGS SHALL NOT EXTEND BEYOND THE MAXIMUM CORE OUTLINE.
- 3. NO FUEL PLATE SHALL HAVE VISUAL EVIDENCE OF A CRACK.
- 4. FUEL PLATES SHALL BE FREE OF IMPURED FOREIGN MATERIAL.
- 5. EACH FUEL PLATE SHALL BE FLAT WITHIN .002 T.I.R.
- 6. MINIMUM CLADDING THICKNESS SHALL BE .003".

MATL: AS SHOWN
REQD: 60 TOTAL (SEE TABLE)
SCALE: 1" = 1' AND AS NOTED
ITEM # 6, PWS-0101

Fig. B.1. Fuel-plate details.

Table B.1. Irradiation test matrix for HFED-1-D2 facility

Module No.	Slot No.	Nominal fuel-plate thickness [μm (mils)]	Fuel-plate No.	Meat attributes						
				Fissile-bearing compound	Thickness [μm (mils)]	Uranium density (kg/m ³)	235U enrichment (wt %)	Total 235U loading (mg)	Maximal areal 235U loading	
									g/m ²	% ^a
1	1	1270 (50)	E-101	UAl _x	697 (27.4)	1937	40.24	2201	282	4
	2	1524 (60)	E-114	UAl _x	846 (33.3)	1953	40.24	2649	374	12
	3	1270 (50)	E-141	UAl _x	536 (21.1)	2217	40.24	1901	272	14
	4	1524 (60)	E-116	UAl _x	864 (34.0)	2306	40.24	3173	446	11
	5	1270 (50)	E-137	UAl _x	561 (22.1)	1907	40.24	1578	236	10
	6	1524 (60)	0-54-1	U ₃ O ₈	859 (33.8)	2794	19.47	2085	267	14
	7	1270 (50)	E-139	UAl _x	569 (22.4)	1879	40.24	1576	232	8
	8	1524 (60)	0-57-2	U ₃ O ₈	632 (24.9)	2766	19.47	1498	202	18
	9	1270 (50)	0-48-2	U ₃ O ₈	516 (20.3)	3100	44.95	3033	472	31
	10	1524 (60)	0-57-1	U ₃ O ₈	638 (25.1)	2768	19.47	1498	196	14
	11	1270 (50)	E-142	UAl _x	555 (21.9)	2202	40.24	1899	272	11
	12	1524 (60)	E-117	UAl _x	875 (34.4)	2304	40.24	3169	451	11
2	1	1270 (50)	A-01	U ₃ SiAl	503 (19.8)	5885	19.88	2786	385	31
	2	1524 (60)	A-03	U ₃ SiAl	545 (21.5)	5614	19.88	2787	424	40
	3	1270 (50)	A-25	U ₃ SiAl	508 (20.0)	4660	19.88	2088	271	15
	4	1524 (60)	0-54-6	U ₃ O ₈	859 (33.8)	2785	19.47	2085	272	17
	5	1270 (50)	0-48-1	U ₃ O ₈	513 (20.2)	3095	44.95	3034	444	24
	6	1524 (60)	0-55-4	U ₃ O ₈	877 (34.5)	3134	19.47	2316	313	17
	7	1270 (50)	0-47-2	U ₃ O ₈	494 (19.4)	2771	44.95	2666	372	21
	8	1524 (60)	0-53-4	U ₃ O ₈	848 (33.4)	2485	19.47	1835	224	9
	9	1270 (50)	0-46-6	U ₃ O ₈	498 (19.6)	2461	44.95	2381	314	14
	10	1524 (60)	0-55-3	U ₃ O ₈	877 (34.5)	3115	19.47	2316	323	22
	11	1270 (50)	0-48-6	U ₃ O ₈	508 (20.0)	3098	44.95	3033	441	25
	12	1524 (60)	0-54-2	U ₃ O ₈	861 (33.9)	2791	19.47	2085	260	11
3	1	1270 (50)	A-05	U ₃ SiAl	500 (19.7)	4673	19.88	2091	259	12
	2	1524 (60)	A-08	U ₃ SiAl	496 (19.5)	4658	19.88	2090	245	7
	3	1270 (50)	A-06	U ₃ SiAl	489 (19.3)	4683	19.88	2091	242	6
	4	1524 (60)	A-09	U ₃ SiAl	489 (19.3)	4658	19.88	2090	247	9
	5	1270 (50)	A-07	U ₃ SiAl	501 (19.7)	4683	19.88	2091	258	11
	6	1524 (60)	A-10	U ₃ SiAl	490 (19.3)	4653	19.88	2089	244	8
	7	1270 (50)	A-11	U ₃ Si	462 (18.2)	4834	19.88	2077	238	7
	8	1524 (60)	A-21	U ₃ SiAl	499 (19.6)	4681	19.88	2091	276	19
	9	1270 (50)	A-12	U ₃ Si	468 (18.4)	4829	19.88	2078	240	7
	10	1524 (60)	A-15	U ₃ Si	480 (18.9)	4814	19.88	2079	265	15
	11	1270 (50)	A-13	U ₃ Si	498 (19.6)	4789	19.88	2079	265	12
	12	1524 (60)	A-16	U ₃ Si	502 (19.8)	4804	19.88	2078	284	19
4	1	1270 (50)	A-17	U ₃ SiAl	519 (20.4)	4618	19.88	2093	263	10
	2	1524 (60)	A-18	U ₃ SiAl	486 (19.1)	4658	19.88	2090	246	9
	3	1270 (50)	A-19	U ₃ Si	462 (18.2)	4814	19.88	2081	257	16
	4	1524 (60)	A-20	U ₃ Si	470 (18.5)	4789	19.88	2080	241	8
	5	1270 (50)	0-50-1	U ₃ O ₈	502 (19.8)	2759	19.47	1154	174	29
	6	1524 (60)	0-58-7	U ₃ O ₈	642 (25.3)	3102	19.47	1663	243	25
	7	1270 (50)	0-50-4	U ₃ O ₈	501 (19.7)	2771	19.47	1154	174	29
	8	1524 (60)	0-58-8	U ₃ O ₈	643 (25.3)	3102	19.47	1663	248	28
	9	1270 (50)	E-061	UAl _x	701 (27.6)	1878	19.92	1087	141	8
	10	1524 (60)	E-094	UAl _x	907 (35.7)	1967	19.92	1525	198	12
	11	1270 (50)	E-127	UAl _x	669 (26.3)	2285	19.92	1202	166	9
	12	1524 (60)	E-093	UAl _x	965 (38.0)	1989	19.92	1525	212	11
5	1	1270 (50)	0-51-1	U ₃ O ₈	517 (20.4)	3103	19.47	1312	215	38
	2	1524 (60)	0-58-2	U ₃ O ₈	638 (25.1)	3093	19.47	1663	240	25
	3	1270 (50)	0-49-4	U ₃ O ₈	496 (19.5)	2456	19.47	1031	137	15
	4	1524 (60)	0-56-4	U ₃ O ₈	622 (24.5)	2470	19.47	1319	174	16
	5	1270 (50)	0-50-2	U ₃ O ₈	501 (19.7)	2767	19.47	1154	165	22
	6	1524 (60)	0-57-4	U ₃ O ₈	639 (25.2)	2778	19.47	1497	207	20
	7	1270 (50)	0-51-5	U ₃ O ₈	519 (20.4)	3111	19.47	1313	190	27
	8	1524 (60)	0-58-4	U ₃ O ₈	640 (25.2)	3098	19.47	1664	255	32
	9	1270 (50)	0-52-2	U ₃ O ₈	734 (28.9)	2442	19.47	1577	188	8
	10	1524 (60)	0-59-3	U ₃ O ₈	475 (18.7)	713	93.21	1448	171	8
	11	1270 (50)	0-52-1	U ₃ O ₈	738 (29.1)	2442	19.47	1577	191	9
	12	1524 (60)	0-58-6	U ₃ O ₈	642 (25.3)	3099	19.47	1663	250	29

^aPercentage maximum above average areal loadings.

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