NRC FORM 7 (5-2003) 10 CFR 110 — U.S. NUCLEAR REGULATORY COMMISSION

APPLICATION FOR LICENSE TO EXPORT NUCLEAR MATERIAL AND EQUIPMENT

(See Instructions on Reverse)

1. APPLICANT'S USE	a. DATE OF APPLICATION	b. APPLICANT'S REFERENCE	2. NRC USE	a. DOCKET NUMBER	b. LICENSE NUMBER
	07/11/2006	BEA0004		11005636	XSNM03458

3. APPLICANT'S NAME AND ADDRESS

a. NAME: Battelle Energy Allliance LLC. (BEA)

b. STREET ADDRESS *(Facility Site)*: 2525 North Fremont Avenue

c. CITY: Idaho Falls — d. STATE: ID — e. ZIP CODE: 83415-5102

f. TELEPHONE NUMBER: (208) 526-3957 — g. FAX: (208) 526-5432 — h. E-MAIL: vkubiak@inl.gov

4. SUPPLIER'S NAME AND ADDRESS *(Complete if applicant is not supplier)*

a. NAME:

b. STREET ADDRESS:

c. CITY: — d. STATE: — e. ZIP CODE:

5. FIRST SHIPMENT SCHEDULED	6. FINAL SHIPMENT SCHEDULED	7. APPLICANT'S CONTRACTUAL DELIVERY DATE	8. PROPOSED LICENSE EXPIRATION DATE	9. CONTRACT NO.
[redacted]	[redacted]	[redacted]	12/31/2006	None

10. ULTIMATE FOREIGN CONSIGNEE

a. NAME: CEA Cadarache

b. STREET ADDRESS *(Facility Site)*: 13108 Saint-Paul-lez-Durance Cedex

c. CITY: Cedex — d. COUNTRY: France

11. ULTIMATE END USE *(Include plant or facility name)*

Post-irradiation examination will be performed to analyze the UMo alloy fuel plates as part of the U.S. Department of Energy RERTR Program. The irradiated fuel plates will be returned to the U.S. for disposal prior to 2010.

11a. DATE REQUIRED [redacted]

12. INTERMEDIATE FOREIGN CONSIGNEE

a. NAME: CEA-Saclay

b. STREET ADDRESS *(Facility Site)*: 91191 Gif-sur-Yvette cedex

c. CITY: Cedex — d. COUNTRY: France

13. INTERMEDIATE END USE

Up to four LEU (19.75%) monolithic UMo alloy fuel plates will be irradiated in the French OSIRIS reactor as part of the U.S. Department of Energy RERTR Program.

13a. DATE REQUIRED [redacted]

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14. INTERMEDIATE FOREIGN CONSIGNEE

a. NAME:

b. STREET ADDRESS *(Facility Site)*:

c. CITY: — d. COUNTRY:

15. INTERMEDIATE END USE

15a. DATE REQUIRED

16. COM CODE	17. DESCRIPTION *(Include chemical and physical form of nuclear material; give dollar value of nuclear equipment and components)*	18. MAX. ELEMENT WEIGHT	19. MAX. WT. %	20. MAX. ISOTOPE WEIGHT	21. UNIT
	Up to four 19.75%E U235 monolithic UMo Alloy fuel plates clad with either Al 6061 or ALFeNi alloy.	[redacted] per plate	19.75 ± 0.2%	[redacted] U235 per plate	grams

22. FOREIGN OBLIGATIONS BY COUNTRY AND PERCENTAGE *(Use separate sheet if necessary)*

Foreign obligations are 100% U.S. in accordance with Article 9 and Article 10 of the General Agreement.

23. ADDITIONAL INFORMATION ON CONSIGNEES, END USES, AND PRODUCT DESCRIPTION *(Use separate sheet if necessary)*

Reduced Enrichment for Research and Test Reactors (RERTR) Development and Qualification Plan and Test Plan attached.

24. The applicant certifies that this application is prepared in conformity with Title 10, Code of Federal Regulations; and that all information in this application is correct to the best of his/her knowledge.

25. AUTHORIZED OFFICIAL	a. SIGNATURE	b. TITLE
		Export Compliance Coordinator

Idaho National Laboratory

July 11, 2006

CCN 205972

11305636
XSNM03458

Deputy Director, Office of International Programs
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20852

SUBJECT: Request to Export Fuel Plates to the French OSIRIS Reactor in Support of Department of Energy Work

Dear Deputy Director:

Attached please find an application and supporting documentation to export up to four low enriched (19.75%) monolithic UMo alloy fuel plates to France to support irradiation testing in the French OSIRIS reactor as part of the U.S. Department of Energy Reduced Enrichment for Research and Test Reactors (RERTR) Program. The irradiated fuel plates will be returned to the U.S. for disposal prior to 2010.

Based on responses to previous license submittals, since the Battelle Energy Alliance LLC, the managing and operating contractor at the Idaho National Laboratory, is performing this work for the Department of Energy, I have assumed that we are still covered under a fee-exempt status, and no fees have been provided in this application.

If additional information is required, please contact the undersigned at (208) 526-3957.

Sincerely,

Vernon Robert Kubiak, Empowered Official
Export Compliance and Licensing

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Enclosures

1. NRC Form 7, Application for License to Export Nuclear Material and Equipment
2. Letter of Explanation for NRC Form 7, Application for License to Export Nuclear Material and Equipment

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Attachments

1. Reduced Enrichment for Research and Test Reactors Fuel Qualification Plan
2. Monolithic Fuel For Irradiation Testing In The Osiris Reactor
3. Agreement between the Department of Energy of the United States of America and the Commissariat A L'Energie Atomique of France for Cooperation in Advanced Nuclear Reactor Science and Technology

cc: D.E. Coburn, INL, MS 3406 (w/o Att.)
J. J. Grossenbacher, INL, MS 3695 (w/o Att.)
L. A. Sehlke, INL, MS 3810 (w/o Att.)

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ENCLOSURE 1 TO THE BATTELLE ENERGY ALLIANCE REQUEST TO EXPORT FUEL PLATES TO THE FRENCH OSIRIS REACTOR IN SUPPORT OF DEPARTMENT OF ENERGY WORK

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ENCLOSURE 2 TO THE BATTELLE ENERGY ALLIANCE REQUEST TO EXPORT FUEL PLATES TO THE FRENCH OSIRIS REACTOR IN SUPPORT OF DEPARTMENT OF ENERGY WORK

Enclosure 2 to Request to Export Fuel Plates to the French OSIRIS Reactor in Support of Department of Energy Work

Letter of Explanation for NRC Form 7, Application for License to Export Nuclear Material and Equipment

Background

A major element of the "Atoms for Peace" program, announced by Dwight Eisenhower in 1953, was the provision of research reactors and of fuels needed for their operation [1]. The research and test reactors supplied by the "Atoms for Peace" program initially used low-enriched uranium (LEU, <20% ^{235}U) fuel. The demand for higher specific powers and neutron fluxes increased with the accelerating development of nuclear technology, and drove the need for fuels with higher fissile loading. The higher power research and test reactors began to use HEU fuel, and by the late 1970's the U.S. was exporting about 700 kilograms of HEU annually for foreign research and test reactors. In response to increasing congressional and public concern about the potential diversion of HEU for use in nuclear weapons, DOE in August 1978 initiated the Reduced Enrichment for Research and Test Reactors (RERTR) program. The primary objective of the RERTR program is to develop the technology needed to minimize and eventually eliminate use of high-enriched uranium (HEU) for most civilian applications worldwide.

Material Description

Up to four 19.75%E U^{235} monolithic UMo Alloy fuel plats clad with either Al 6061 or ALFeNi alloy.

IRIS-5 Experiment in OSIRIS

The IRIS-5 test in the OSIRIS reactor is a joint test with the CEA that will generate the first fuel performance data on full-size monolithic fuel plates. A drawing of the IRIS-5 Test can be found in the RERTR Fuel Development and Qualification Plan (added as Attachment 2)

IRIS-5 Test Description

The RERTR program will provide two LEU test plates for irradiation, to be complemented by two plates fabricated by CERCA (France). If plates cannot be fabricated by CERCA, the U.S. RERTR program may provide four test plates. Plate thickness profiles will be monitored after each irradiation cycle for excessive swelling. A full post irradiation examination will be conducted at the CEA Cadarache site.

Enclosure 2 to Request to Export Fuel Plates to the French OSIRIS Reactor
in Support of Department of Energy Work

IRIS-5 Objectives

The objectives of the IRIS-5 test are to:

1. Generate the first irradiation performance data for full-size MTR U-Mo monolithic fuel plates through the use of in-canal plate thickness measurements and subsequent PIE.
2. Provide a comparison of the performance of monolithic fuel plates fabricated in France and the U.S. using different processes.
3. Provide diversity in fuel testing with irradiation conditions that differ from those in the Advanced Test Reactor at the Idaho National Laboratory and to establish the framework for continued joint U.S./French testing in OSIRIS and other reactors.

IRIS-5 Schedule

The IRIS-5 test is scheduled for delivery to CEA-Saclay prior to [redacted] Irradiation will begin in OSIRIS in November/December of 2006. Irradiation will require 9 months, and be completed in June/July/August/September of 2007, followed by cooling prior to shipment to CEA Cadarache for PIE in November/December/January/February of 2007 and 2008. It is anticipated that irradiated fuel will be returned to the U.S. for disposal prior to 2010. Preliminary agreement with CEA has been made regarding experiment schedule and cost sharing. The CEA will fund irradiation in OSIRIS and one-half of PIE and disposal costs. The U.S. RERTR program will supply two (or four) monolithic plates for testing and pay one-half of the PIE costs.

ATTACHMENT 1 TO THE BATTELLE ENERGY ALLIANCE REQUEST TO EXPORT FUEL PLATES TO THE FRENCH OSIRIS REACTOR IN SUPPORT OF DEPARTMENT OF ENERGY WORK

INL/EXT-05-01017

RERTR Fuel Development and Qualification Plan

M. K. Meyer

January 2005

INL/EXT-05-01017

Project Execution Plan
for the
RERTR Fuel Development Project

Author:
M. K. Meyer

January 2005

Idaho National Laboratory

Idaho Falls, Idaho 83415

Title: **Reduced Enrichment for Research and Test Reactors Fuel Qualification Plan**
Author: M. K. Meyer
Revision No: 1
Effective Date: 27 January, 2006
Document Number:

Summary:

In late 2003 it became evident that U-Mo aluminum fuels exhibited significant fuel performance problems under the irradiation conditions required for conversion of most high-powered research reactors. An RERTR program strategy has been mapped that will allow generic fuel qualification to occur prior to the end of FY10 through the anticipated issue of an NRC Safety Evaluation Report.

Phase I of fuel qualification will be completed in FY10. The qualification will cover all very-high density fuel variants, including those that require burnable poisons and density zoning. The final phase of qualification will consist of irradiation of MTR plate-type fuel assemblies in two reactors. After this work is completed, reactors that use discrete fuel elements can convert through the process of irradiation of lead test elements.

Phase II fuel qualification will target reactors that utilize more complex annular fuel elements for which performance of an entire fuel element cannot be demonstrated in a test reactor due to space limitations. These reactors include HFIR, RHF, and FRM-II. Because discrete lead test elements cannot be used in these reactors, final qualification of fuel elements for these reactors will have to occur in the actual reactors that are targeted for conversion.

In addition to the U.S. fuel development effort, contracts with Russian Federation institutes in support of fuel development for Russian are in place. A Russian fuel development strategy is under has been developed.

The overall RERTR fuel development program thus currently results in qualification of fuels suitable for conversion of 23 targeted research reactors that consume almost 600 kg of HEU per year.

Several fuel development issues remain to be addressed, principally in relation to ensuring fuel qualification for four Russian high-temperature reactors that are not currently addressed or funded by either the U.S. or Russian RERTR programs. An additional significant cost is that of upgrading the U.S. commercial infrastructure to support LEU fuel supply after qualification.

1.0 Introduction

1.1 Purpose of Plan

This plan is designed to be used as a guiding document, laying out methodology, major milestones and deliverables, decision points, and required inputs to meet GTRI fuel development program goals.

Detailed work plans and budgets to achieve these goals will be developed on a yearly basis, consistent with RERTR Fuel Development Program funding and NNSA guidance. This plan will be revised by June 1 of every year to reflect changes in the status of research and development, budget, and schedule.

1.2 Background

1.2.1 The RERTR Fuel Development Program from 1978 – 1988

A major element of the "Atoms for Peace" program, announced by Dwight Eisenhower in 1953, was the provision of research reactors and of fuels needed for their operation [1]. The research and test reactors supplied by the "Atoms for Peace" program initially used low-enriched uranium (LEU, <20% ^{235}U) fuel. The demand for higher specific powers and neutron fluxes increased with the accelerating development of nuclear technology, and drove the need for fuels with higher fissile loading. The higher power research and test reactors began to use HEU fuel, and by the late 1970's the U.S. was exporting about 700 kilograms of HEU annually for foreign research and test reactors. In response to increasing congressional and public concern about the potential diversion of HEU for use in nuclear weapons, DOE in August 1978 initiated the Reduced Enrichment for Research and Test Reactors (RERTR) program. The primary objective of the RERTR program is to develop the technology needed to minimize and eventually eliminate use of high-enriched uranium (HEU) for most civilian applications worldwide.

The key to making reactor conversions feasible lies in developing fuels with higher uranium density, since in LEU every atom of ^{235}U must be accompanied by approximately four atoms of ^{238}U. In 1978, the main fuels used in western designed reactors were plate-type UAl_x-Al (1.7 g/cm^3) and U_3O_8-Al (1.3 g/cm^3) dispersions enriched to 93% and rod-type TRIGA fuel based on $UZrH_x$ and enriched to 70% with a uranium density of 0.5 g/cm^3.

Initial RERTR fuel development efforts began by increasing the volume loading of fissile particles in these fuels. Miniplates were fabricated by the RERTR program and irradiated in the ORR (Oak Ridge Reactor). The plate-type fuel fabricators CERCA in France, NUKEM in Germany, and Atomics International in the U.S. were informed of test results and began to fabricate and test full-size plates and elements of the more promising materials. The only fabricator of TRIGA fuel (General Atomics) developed LEU TRIGA pins with a uranium density of 3.7 g/cm^3, and these were tested by the RERTR program in the ORR. The outcome of the initial effort was successful demonstration of aluminum-based LEU dispersion fuels with

uranium concentrations of 2.3g-U/cm^3 for UAl_x-Al, 3.2 g-U/cm^3 for U_3O_8- and 3.7 g/cm^3 for $UZrH_x$. The Ford Nuclear Reactor (FNR) at the University of Michigan was the first reactor converted by the RERTR program in December 1981 using an UAl_x-Al fuel with a density of 1.7 g/cm^3.

Also under development were fuels based on U_3Si, U_3Si_2, and U_6Me intermetallics (Me – transition metal). U_3Si miniplates began to fail at high burnup in 1982, and were dropped from the U.S., French, and German programs. In Canada, the AECL continued development of U_3Si-Al fuel in pin form and used it to successfully convert their NRU reactor. (The better behavior of U_3Si-Al in the pin-type NRU fuel is inferred to be due to the constraint that the pin-type fuel imposes on fuel swelling). U_3Si_2-Al dispersion fuel, which has a uranium density of 4.8 g/cm^3, had been tested in parallel with U_3Si_2-Al dispersion fuel and was found to be very stable under irradiation. Irradiation testing of full-size LEU U_3Si_2-Al dispersion fuel elements was successful, and in 1986-1987 a full-core demonstration of this fuel was successfully performed in the ORR. In 1988, LEU U_3Si_2-Al dispersion fuel received generic approval by the NRC in the form of a Safety Evaluation Report (NUREG-1313). Since 1988, U_3Si_2 has been used in the conversion or new startup of more than thirty research reactors. After this program was completed, the RERTR fuel development effort experienced a lull in funding during which no further substantial gains in fuel technology were achieved.

1.2.2 Selection and Early Irradiation Testing of Very-High-Density Alloy Dispersion Fuels

Conversion of the highest-power density reactors requires LEU fuel densities in the range of 8 to 10 g/cm^3. Funding for the RERTR fuel program was restarted in 1996 to develop fuels targeted for these applications. Due to limited program funding, it was deemed preferable to develop this fuel to allow for fabrication using the current commercial infrastructure and methods. This sets a practical upper limit on fuel loading of approximately 55 volume percent, so that fuel phases with uranium densities of at least 14.5 g/cm^3 must be used. Two types of fuel were available that met this density criterion; these are metallic uranium of low alloy content and the U_6Me class of high density intermetallics. In this latter class, U_6Fe and U_6Mn have been shown to have poor irradiation behavior [2,3], and it is likely that other U_6Me compounds will behave in a similar manner. The emphasis for US-RERTR advanced fuel development was therefore placed on metallic uranium alloy fuels. Screening of the potential alloy candidates based on gamma stability and what was known about irradiation behavior eliminated all but U-Nb-Zr and U-Mo alloys from consideration early in the process. Since little irradiation data existed on the performance of metallic fuels in the high burnup, low temperature range, required for research reactors, screening irradiation tests were used to identify alloys with good irradiation performance under these conditions. A chronology of these tests and other irradiations leading to the present is provided below.

1.2.2.1 The RERTR-1 and RERTR-2 tests

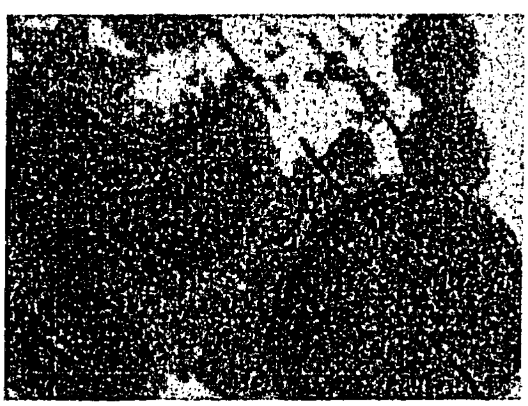
The RERTR-1 and -2 test vehicles contained eleven fuel alloys and two intermetallic compounds irradiated to nominal burnup of 40 and 70 atom percent ^{235}U depletion in the Advanced Test Reactor (ATR). [4,5] The fuel alloys selected for the RERTR-1 and -2 irradiation capsules can be divided into four fuel types based on γ-stability and microstructure, as shown in Table 1.1.

Table 1.1. Fuel alloys chosen for irradiation screening tests (alloying element in wt%)

Most γ stable	Intermediate γ stability	Least γ stable ($\alpha+\gamma'$)	Precipitate Dispersion within Fuel Particles
U-10Mo	U-6Mo	U-5Nb-3Zr	U-10Mo-0.05Sn
U-8Mo	U-6Mo-1Pt	U-4Mo	U-6Mo-0.1Si
U-9Nb-3Zr	U-6Mo-0.6Ru	U-2Mo-1Nb-1Zr	
	U-6Nb-4Zr		

The two test vehicles each consisted of a flow-through "basket" holding eight vertically stacked flow-through capsules. Each capsule held four miniplates in a miniature fuel element configuration. The irradiation vehicles occupied small I-hole positions (I-22 and I-23) in the control drum region of the ATR. Based on calculations, the neutron flux, microplate power, surface heat flux, and fuel centerline temperature were approximately 1.3 x 10^{18} n m^{-2} s^{-1}, 500 kW, 5.5 x 10^{5} W m^{-2}, and 65°C, respectively, at the axial position of highest neutron flux at the start of the irradiation. The RERTR-1 test was irradiated for 94 effective full-power days (EFPD) during the period August 23, 1997, through November 30, 1997, and RERTR-2 was irradiated for 232 EFPD during the period August 23, 1997 through July 6, 1998, achieving (calculated) ^{235}U burnups between 39 and 45% and between 65 and 71%, respectively.

Postirradiation examination, in general, indicated excellent irradiation behavior for U-Mo alloys with greater than 6 wt.% molybdenum content and for the ternary U-Mo-Pt and U-Mo-Ru alloys [6]. The U-Nb-Zr alloys were found to be less suitable due to the propensity for these alloys to exhibit the metallographic precursors to breakaway swelling. Micrographs comparing the behavior of some of the fuels are shown in Figure 1.1.



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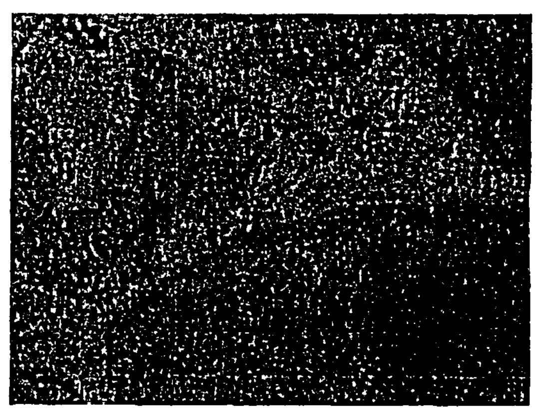

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Figure 1.1. U-10Mo atomized (left) and ground (right) fuels after irradiation to ~ 70 at.% ^{235}U burnup at 65°C.

Conclusions drawn from this experiment at low irradiation temperature (~65°C) to relatively high burnup (71%) were as follows:

- U-Mo alloys with at least 6 wt.% Mo exhibited low swelling and stable irradiation behavior. These alloys are very promising for further development of high-density dispersion fuels.
- Alloys with 4 wt.% Mo and U-Nb-Zr alloys exhibited irradiation behavior that indicated imminent break-away swelling. These alloys did not warrant continued testing.
- The extent of fuel-Al interdiffusion was similar to that seen in U_3Si_2-Al.

1.2.2.2 The RERTR-3 irradiation test

The RERTR-1 and -2 irradiation test capsules were followed by the RERTR-3 irradiation test at higher temperature to moderate burnup. The RERTR-3 experiment consisted of six flow-through capsules stacked vertically inside of an aluminum outer basket. Two columns of four fuel test specimens were arranged within each capsule, for a total of eight specimens per capsule. The experimental test matrix was composed of uranium-molybdenum alloys, and included Mg-matrix fuel test specimens.

The RERTR-3 experiment began irradiation on October 7, 1999 and was exposed for a total of 48 effective full-power days (EFPDs) over two reactor cycles. Peak plate burnup ranged from 24.9 at.% ^{235}U for the plate farthest above the core centerline up to 41.0% burnup for plates near the core centerline. The desire to irradiate at high flux limited the choice of flow-through irradiation positions in the ATR. The B-7 position, a 22.2 mm diameter flow-through hole just outboard of driver fuel element 33 was used for this test. The small diameter of this position necessitated the use of small irradiation test specimens; each test coupon was 10.0 mm x 41.1 mm x 1.52 mm, and contained approximately 0.6 grams of fuel.

Because of the small size of these specimens, analysis of fuel behavior during postirradiation examination was heavily dependant on the use of quantitative metallography. The volume fractions of fuel, unreacted aluminum matrix, and fuel/matrix interaction product were carefully measured. On postirradiation examination, it became immediately evident that the rate of fuel/matrix interaction was faster than anticipated based on extrapolation of the results from the RERTR-1 and RERTR-2 tests. Representative micrographs from the RERTR-3 fuel specimens are shown in Figure 1.2. The consequences of formation of this large amount of fuel/matrix reaction on fuel performance, however, appeared to be benign. Within the range of burnup and temperature examined to this point, the interaction layer formed on U-Mo plates during irradiation exhibited stable irradiation behavior, comparable to that of UAl_x particulate dispersions. These fuels are mixtures of UAl_2, UAl_3, and UAl_4 and exhibit stable irradiation behavior to high burnup. Fuels containing initially stoichiometric UAl_3 and UAl_2 fuel show similar behavior. [7,8]

It also became evident that the rapid growth of the interaction layer was resulting in a positive temperature feedback inside of the fuel meat and an uncertain thermal history. As the fuel reacted with the aluminum matrix, the fuel meat thermal conductivity decreased, resulting in a temperature rise in the fuel and, in turn, a faster reaction rate. The PLATE code was developed, in part, to deconvolute the temperature history of the fuel as a function of irradiation history.

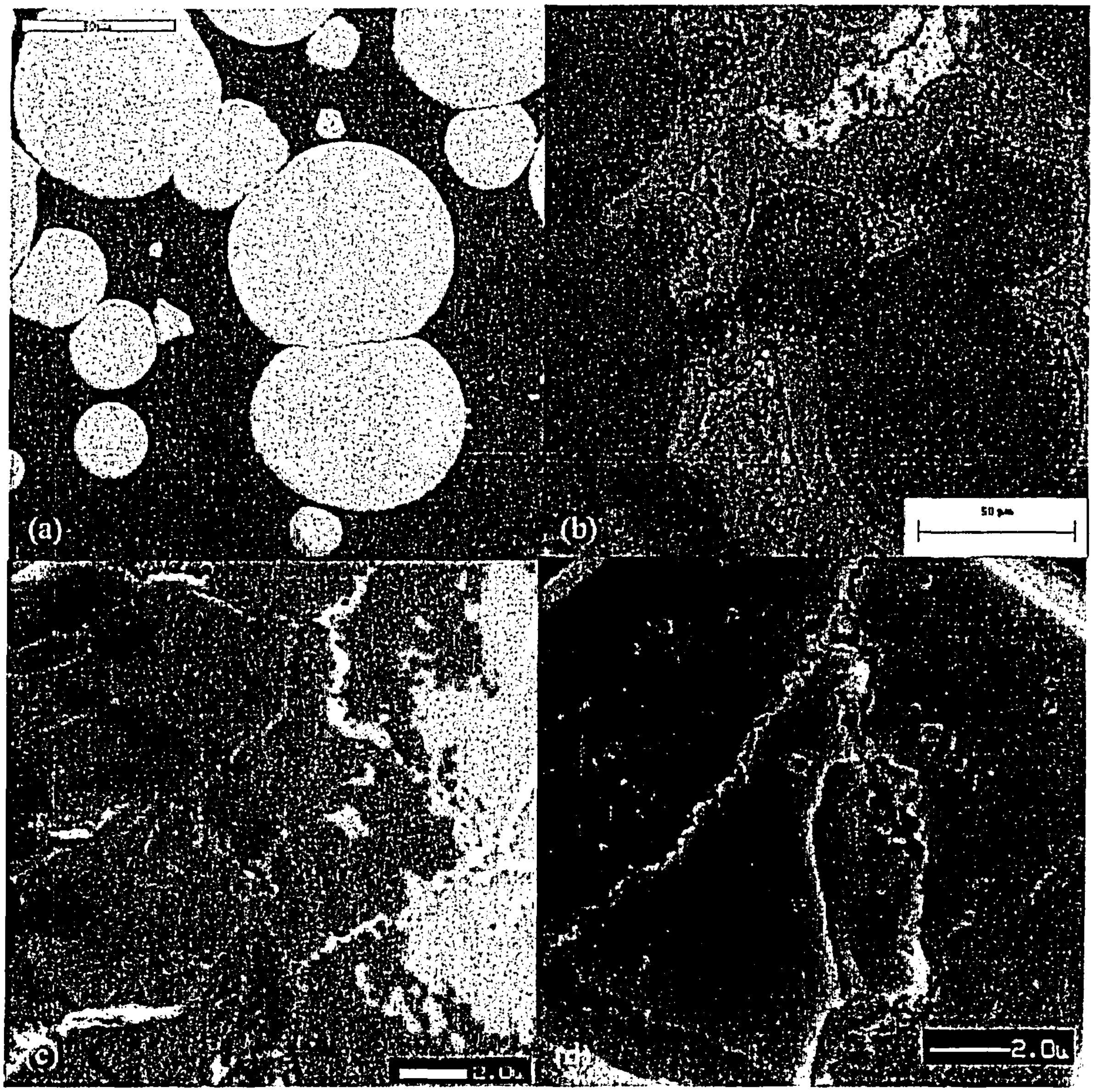


Figure 1.2. Microstructure of U-10Mo atomized fuel from RERTR-3 and comparison with RERTR-1. (a) SEM micrograph of as-fabricated plate. (b) Etched optical microstructure of plate V03 after 37.6% U-235 burnup. Same magnification as (a). (c) SEM micrograph of V03 fracture surface. (d) SEM of RERTR-1 plate V002 after 38.4% burnup.

Conclusions from postirradiation examination of the RERTR-3 test were as follows:

- The extent of fuel swelling was acceptable and stable.
- Swelling was predominantly due to U-Mo/Al interdiffusion up to burnup where the matrix aluminum was essentially consumed by this interdiffusion process.
- The aluminide interaction product appears stable and contained no fission gas bubbles. It has a low thermal conductivity, however, which results in an increased fuel temperature.
- The swelling behavior of the unreacted fuel was athermal in the range of temperature and burnup tested.
- Lowering of Mo content results in somewhat higher rates of interdiffusion and fission gas swelling, as observed in metallographs.
- Mg-matrix fuel specimens showed no interaction between the fuel particles and the Mg-matrix and behaved well under RERTR-3 irradiation test conditions.

1.2.2.3 The RERTR-4 and RERTR-5 irradiation tests

As in previous irradiation experiments, the miniplates irradiated in RERTR-4 and 5 contained either atomized fuel particles, supplied by KAERI, or machined fuel particles, in this case supplied by AECL. The composition of the fuel alloys ranged from, nominally, 6 wt.% Mo to 10 wt.% Mo. The fuel plates in these tests measured 100 mm x 25 mm x 1.40 mm; the meat was in a rectangular zone nominally 0.64 mm thick and contained 6 and 8 g U cm^{-3} in the fuel meat. The RERTR-5 experiment and the RERTR-4 experiment were inserted into positions B-12 and B-11, respectively, of the ATR (Advanced Test Reactor) core for cycle 123B. Irradiation began on August 19, 2001, and RERTR-4 and RERTR-5 were irradiated for 204 EFPD (effective full power days) and 116 EFPD respectively.

In addition to 30 dispersion fuel miniplates, test RERTR-4 also included two miniplates with solid U-Mo alloy cores (monolithic fuels). These plates each contained each thin discs of U-10 Mo of ~12-mm diameter and 0.3-mm thickness. These plates showed stable irradiation behavior.

The RERTR-5 experiment was removed from the reactor at a peak burnup of ~50% U-235, whereas RERTR-4 terminated at ~80% burnup. Unfortunately, the location of the test assemblies in the reactor was switched at some point during the irradiation, so that the irradiation history of the fuel miniplates is not definitively known, complicating quantitative analysis of the postirradiation data. Postirradiation microstructures of the fuel specimens are similar to those of the irradiated RERTR-3 specimens, with the exception of the development of some porosity between the (U-Mo)Al_x reaction layer and the aluminum fuel matrix. Some general conclusions that could be drawn following postirradiation examination of the RERTR-4 and -5 tests were as follows:

- Results from the RERTR-1 through RERTR-5 irradiation experiments continued to show good irradiation behavior of U-Mo/Al dispersion fuel with uranium loadings of up to 8 g-U/cm^3.
- Swelling of the U-Mo alloys was athermal and stable up to ~300°C.
- The interaction between U-Mo and Al affects overall meat swelling to various degrees depending on irradiation conditions, primarily temperature. The main effect of U-Mo/Al interaction is a decrease in meat thermal conductivity.

- Some porosity was noted in the high burnup plates, but this did not appear to be typical of the porosity observed as a precursor to breakaway swelling of fuel that had previously identified.
- The components of the irradiation behavior were largely understood at this point, and allow detailed modeling of the fuel plate swelling and time resolved temperature profile.
- The initial tests of monolithic U-Mo fuel meat were promising, opening the possibility for development of fuel with uranium densities well excess of 8 g-U/cm^3.

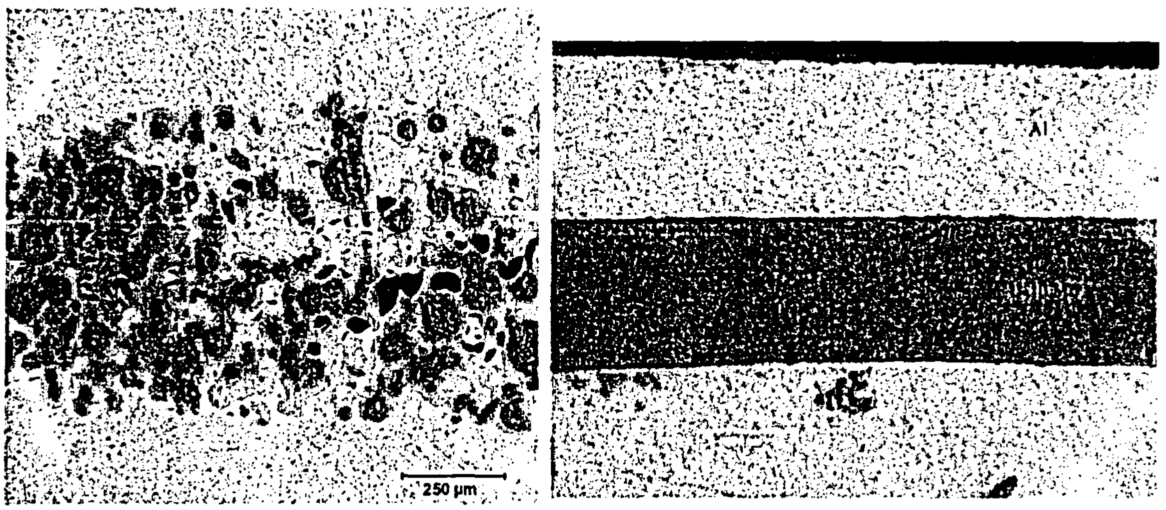


Figure 1.3. Left- Optical micrograph of RERTR-4 U-6Mo fuel plate 623AB after 78 at.% burnup in the RERTR-4 experiment. Note the presence of porosity at the interface between the (u-Mo)Alx interaction product and the aluminum matrix. Right – Micrograph of U-10Mo monolithic fuel specimen 623F.

1.2.3 U-Mo Fuel Failures during U.S. Irradiation Testing

1.2.3.1 Failures in the U.S. Program – There have been no fuel performance related failures in the U.S. miniplate irradiation testing program, however one failure occurred due to a fabrication defect. This failure was useful in determining the extent of reaction of the fuel matrix with reactor coolant water.

Plate Q8003I was fabricated at INL using U-7Mo powder supplied by AECL (Atomic Energy Canada Limited). The plate occupied position A-8 in the RERTR-5 experimental assembly, in the second row from the top of the experiment, with the fuel plate centerline positioned approximately 10 inches (25.4 cm) above the core mid-plane. Reactor startup was on August 19, 2001. A slight increase in coolant and stack activity was noted by ATR operations on August 29, 10 days after irradiation start. Activity decreased to normal levels after approximately two weeks. Irradiation of RERTR-4 and RERTR-5 continued for 106 days. Total irradiation time for the RERTR-5 experiment was 116 EFPD's (Effective Full Power Days).

It appears that the failure of Q8003I was due to a thin area in the cladding that was introduced during fuel plate fabrication. The cladding measured 0.002 – 0.003 inches thick in the failure region. Fuel pile-up in this region increased stresses from thermal expansion and fuel meat swelling (due to reaction and fission products) beyond the ultimate strength of the thin area in the cladding. This is exacerbated by the additional local heat loading from the high-density 'dog-bone' region of the fuel plate. A micrograph of the failed fuel plate is shown in Figure 1.4.

The fuel plate failure was followed by the loss of some of the fuel meat due to water corrosion. An estimate of the total amount of fuel lost by corrosion was made based on metallographic examination, and an upper bound on fuel alloy loss of 0.42 g (including 0.19 g of ^{235}U) was established.

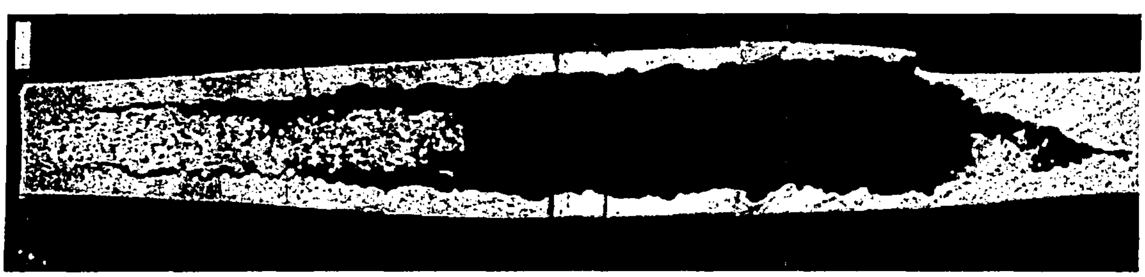


Figure 1.4. RERTR-5 fuel plate Q8003I irradiated for 106 days after cladding failure.

1.2.4 U-Mo Fuel Irradiations in the Other Countries

There have been fuel performance related failures of U-Mo fuel reported in Canada, France, Korea, and Russia. A synopsis of major non-U.S. fuel development activity and fuel failures are summarized below. There were unusual circumstances reported in conjunction with many of these experiments. The poor performance of the FUTURE and IRIS-2 experiments in 2002 and 2003, taken together with new insight from the RERTR-4 and RERTR-5 experiments, definitively showed that U-Mo fuel did have serious irradiation performance issues.

1.2.4.1 Canadian Program

AECL irradiated 12 LEU-Mo pin-type mini-elements in the NRU reactor at Chalk River from September to November 2000. The elements were fabricated using U-10Mo fuel powder produced by a proprietary AECL process. Fuel pins were configured as standard MAPLE reactor elements but loaded to 4.5 g-U/cm^3 and irradiated at a linear power of 165 – 177 kW/m. A day after a short duration increase in local power, at a burnup of approximately 22 at.%, a fuel defect was detected and the experiment was removed from the reactor. Some PIE results that became available in 2002 indicated that 2 pins had breached, and that the cladding failure appeared to be in a tensile mode. The microstructure of the irradiated fuel was similar to that observed in the UMUS experiment (see section below), and indicated that the failure mode was similar and related to overheating of the cladding.

A subsequent experiment was conducted beginning in May 2003 using identical fuel rods but operating at a much lower peak power of 100 kW/m. PIE at 20 at.% burnup reported in 2004 [9] indicated that the swelling rate of U-Mo fuel was higher than that of U_3Si fuel, and that the swelling was consistent with that expected due to extensive formation of the (U-Mo)Al_x interaction phase. Porosity that formed was consistent with that observed in out-of-pile thermal testing. Microchemical analysis of the interaction phase was somewhat inconclusive but indicated that the composition was in the range of (U-Mo)Al_3 to (U-Mo)Al_4 in most regions of the fuel. Some pins continued irradiation under decreasing power conditions to 80 at.% burnup without failure.

1.2.4.2 French Program

IRIS-1 – The IRIS-1 test of 3 MTR fuel plates loaded to ~8 g-U/cm^3 was irradiated in the OSIRIS reactor at CEA-Saclay for 241 EFPD from September 1999 to January 2001. These plates performed well at a peak surface heat flux of 124- 145 W/cm^2 to a peak burnup of 67.5%, and a peak cladding temperature of 72°C. The maximum fuel plate thickness increase was 77 µm (out of 1.3 mm). These plates were fabricated similarly to the UMUS fuel plates (see below). CEA found that the fuel/aluminum reaction layer had a composition in the range of (U-Mo)Al_6 to (U-Mo)Al_8, and that fission products accumulated at the interface between the reaction product and the fuel/aluminum interface. A representative micrograph is shown in Figure 1.5.

UMUS – With indications of good fuel performance in the IRIS-1 test, CEA initiated a second test in the HFR reactor in Petten, The Netherlands. The UMUS experiment tested MTR-size U-Mo fuel plates at 8 g-U/cm^3 at a higher power density than IRIS-1. A peak experiment surface

heat flux of 250 W/cm^2 was achieved, resulting in a BOL clad surface temperature of 110°C. The experiment was composed of four fuel plates, two fueled with LEU and two with fuel at

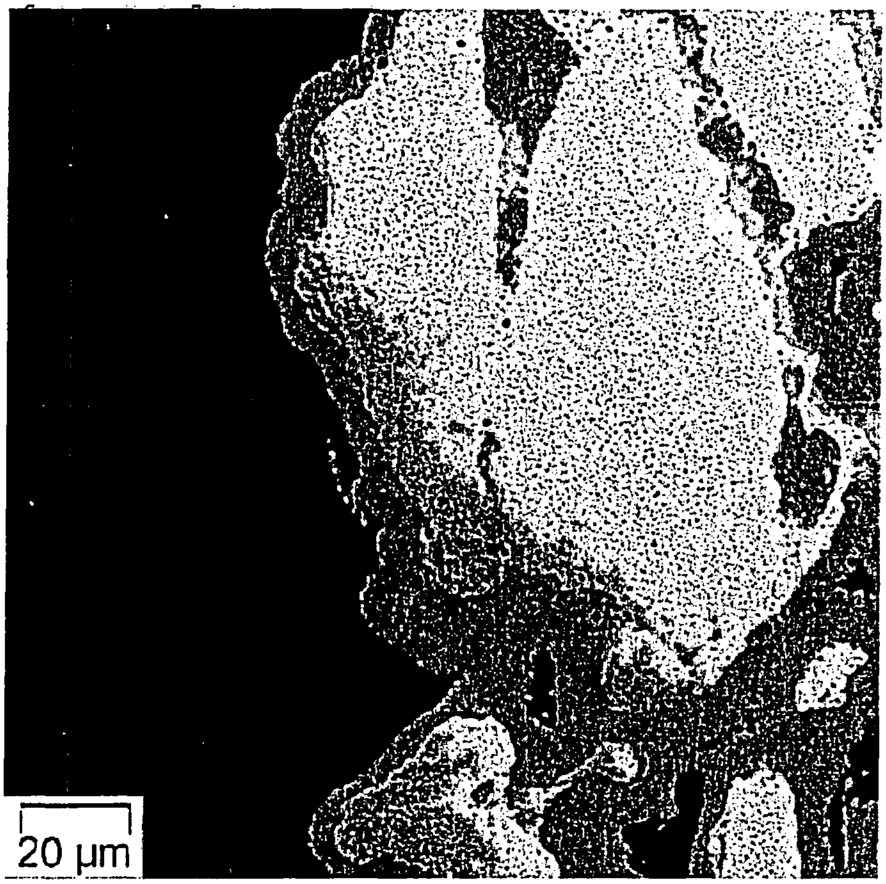


Figure 1.5. Representative micrograph from IRIS-1 fuel plate.

35% enrichment to increase fuel power density. The plates were made by CERCA with irregularly shaped U-Mo powder with a high content of fines (-325 mesh). The experiment was inserted into the HFR (High Flux Reactor) in Petten, The Netherlands in May of 2000 and irradiated for 48 EFPDs over two cycles prior to the detection of a fission product release to the reactor coolant from one of the plates. PIE of a sibling plate beginning in March of 2001 (the failed plate was not examined) showed a low-conductivity surface oxide layer potentially up to 100 µm thick due to the high pH of the HFR coolant water, and high fuel temperature was hypothesized to be the cause of failure; the estimated aluminum cladding temperature was greater than 190°C. The fuel particles reacted extensively with the matrix aluminum, resulting in almost complete aluminum depletion. PIE micrographs of the sibling fuel plate indicated delamination at the interface between the fuel meat and the cladding. There was no indication of excessive swelling or delamination through the fuel meat. A representative optical micrograph is shown in Figure 1.6.

Figure 1.6. Metallographic cross section of sibling to failed plate from the UMUS experiment.

FUTURE – The FUTURE experiment was irradiated by CEA in the in the BR2 reactor (Mol, Belgium) during the second half of 2002. Two U-7Mo fuel plates containing atomized powder loaded at 8 g-U/cm^3 were irradiated at high power to determine whether or not the fuel failure in the UMUS experiment was due to plate overheating (due to the formation of a surface oxide layer) or to an intrinsic fuel performance problem with U-Mo fuel. The plates were irradiated at a power that resulted in a surface heat flux of 340 W/cm^2 and a BOL surface temperature of 130°C. Afer 40 EFPDs and 29% ^{235}U burnup (fission density 1.25 x 10^{21} f/cm^3), a plate thickness increase of 13% was noted. A specimen from the region of the plate with the highest swelling was examined, and showed a much different evolution of the microstructure during irradiation than revealed by the UMUS fuel plate PIE. In the case of the FUTURE experiment, the fuel plate delaminated through the center of the fuel meat, with ligaments of the (U-Mo)Al_x interaction phase bridging the void, as shown in Figure 1.7. The composition of the interaction phase was determined to be in the range of compositions consistent with (U-Mo)Al_3 or (U-Mo)$Al_{4.4}$. There were also indications of concentration of xenon gas at the boundary between the remaining matrix aluminum and the (U-Mo)Al_x layer.

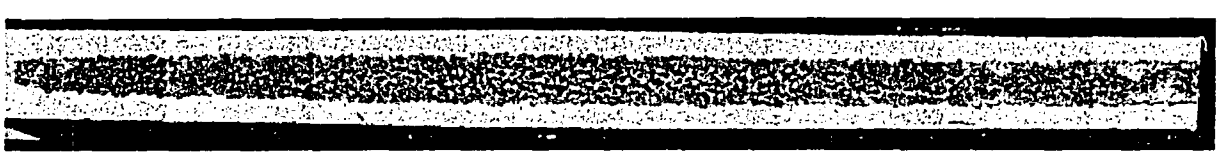


Figure 1.7. Cross section of pillowed fuel plate from the FUTURE experiment.

IRIS-2 – The IRIS-2 test was developed and irradiated by CEA beginning in early 2003 in order to determine whether reasonable limits on operating power could be established that would allow utilization of U-Mo fuel in some reactors. The experiment consisted of four MTR fuel plates fabricated using atomized U-7Mo fuel at 8.3 g-U/cm^3. The plates operating at a peak surface heat flux of 238 W/cm^2 in the OSIRIS reactor, resulting in a peak beginning of life cladding temperature of 93°C. A plate thickness increase of more than 250 μm was observed after 30 EFPDs for one plate, which was withdrawn from the experiment. The experiment was

terminated after 58 EFPDs due to high swelling of the remaining plates. Postirradiation examination results revealed the same failure mechanism as found in the FUTURE test, that is delamination of the fuel plate through the meat center. The U-Mo fuel particles behaved well, with no evidence for excessive swelling in this phase. Separation between the reaction layer and the aluminum matrix were noted, as well as what appears to be viscous flow of the interaction layer in the delaminated region. The composition of the interaction layer was found to vary with location in the fuel plate from (U-Mo)Al_6 in the unaffected regions of the fuel plate to (U-Mo)$Al_{4.5}$ in the failed region. Micrographs of the failed IRIS-2 plate are shown in Figure 1.8.

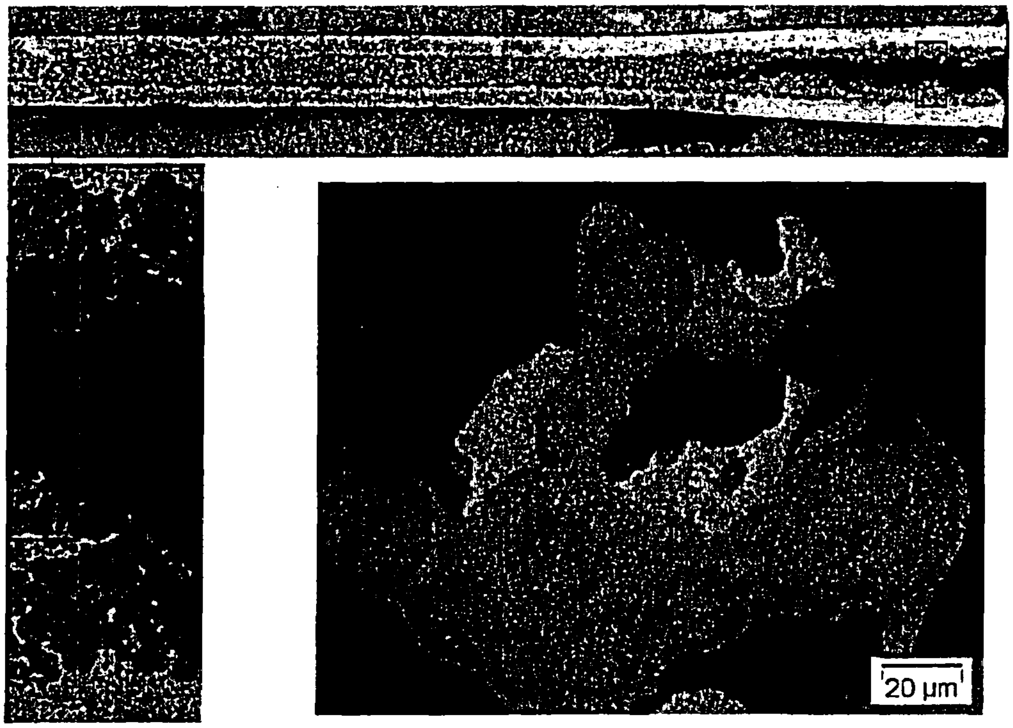


Figure 1.8. Metallographic sections from failed IRIS-2 fuel plate.

IRIS-3 – The IRIS-3 test irradiation [10] began in early 2005 in the OSIRIS reactor to test the efficacy of silicon additions to the fuel matrix in stabilizing the fuel/aluminum interaction layer against breakaway swelling. Fuel plates containing 0.3 wt.% and 2.1 wt.% silicon are under irradiation in the OSIRIS reactor at a BOL power of ~200 W/cm^3. After 4 cycles of irradiation, at fission density and power conditions slightly less than those conditions where plates from IRIS-2 had failed (fission density 2.0 – 2.6 x 10^{21} f/cm^3) both types of plates were exhibiting improved irradiation performance with a maximum thickness increase of 35-40 µm at a peak burnup of 38 at.% ^{235}U. The plates with 0.3 wt.% silicon were reported in December 2005 to have excessive swelling after 5 irradiation cycles, and were removed from the experiment, while a plate operating at the highest power with a 2.1 wt.% matrix silicon content has shown good in-reactor performance and will be irradiated for an additional cycle to a burnup greater than 50 at.% ^{235}U.

1.2.4.3 Korean Program

KOMO-1 – The KOMO-1 irradiation [11] included ten U-Mo pin-type fuel specimens (MAPLE-type geometry) of two diameters under aggressive test conditions in the Korean HANARO reactor. Fuel particles with molybdenum contents of 7 and 9 weight percent were tests at fuel loadings of 3.4 and 6.0 g-U/cm^3. The experiment began irradiation in June of 2001 and was discharged due to fission product release to the coolant after 26.6 EFPDs (62 days) and ~13 at.% ^{235}U burnup. Estimated peak linear power of the failed 6 g-U/cm^3 fuel pin was 107 kW/m. PIE indicated a maximum fuel meat volume swelling of 15.9%, extensive reaction of the fuel particles with the matrix aluminum, and formation of a central void. There was no indication of breakaway swelling behavior, and it was hypothesized that the aluminum cladding had ruptured along a fabrication defect (Figure 1.9). Tests of cladding ductility indicated that elongation to failure ranged from 2.5 to 45.8%, and it was concluded that the lower values were caused by cladding defects formed due to inclusions of die lubricant into the cladding during extrusion. The fuel failure was also attributed to this type of cladding defect. It was estimated that high fuel centerline temperatures might also have contributed to fission gas release, however no evidence was presented.

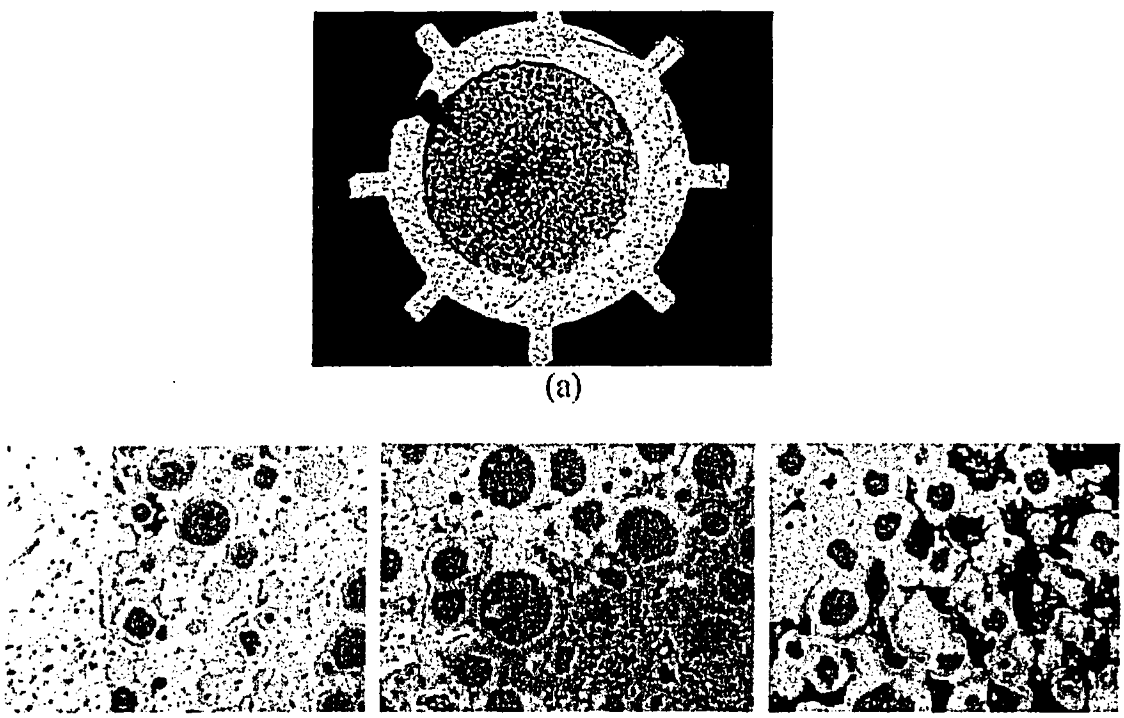


Figure 1.9. Failed KOMO-1 fuel test pin.

KOMO-2 – Following the failure in the KOMO-1 experiment, the KOMO-2 test [12] irradiated similar pin-type U-Mo dispersions with two diameters at lower fuel loading (4.0 – 4.5 g-U/cm^3) in the Korean HANARO reactor. No fuel failures occurred in this test. The experiment was irradiated from January 2003 to January 2004 to a peak burnup of 71 at.% ^{235}U at a peak linear power of 112 kW/m. Fuel meat swelling of the large diameter fuel pins in the range of 8.5-15.6 % was observed for burnup in the range of 62-68 at.% ^{235}U; the upper value being about twice that measured for U_3Si fuel at 4.0 g-U/cm^3. Large fuel particles (were effective in reducing fuel

swelling to the lower end of the observed range. Microstructural examination indicated that porosity that formed in the center of the fuel meat region is similar to that observed in out-of-pile heat treatment and unlike that found in the IRIS-2 and FUTURE tests.

1.2.4.4 Russian Program

Mini-pins – Two test capsules containing 13 variations of U-Mo fuel were inserted into the MIR reactor at the Research Institute of Atomic Reactors (RIAR) in Dimitrovgrad in 2003. Irradiation conditions were moderate with pin surface heat flux of 32 – 90 W/cm^2 and cladding temperatures of 88-117°C. In November of 2004 a fission product release was detected in the capsule containing the highest-burnup pins. The average ^{235}U burnup in the mini-pins ranged between ~60 and ~65%. Subsequent postirradiation inspection determined the cause of the release to be related to an obvious fabrication defect. The irradiation of the second capsule containing mini-pins was resumed and was completed in February 2005 with average fuel burnup of the pins ranging between ~20 and ~60%. The PIE of the pins indicates normal fuel irradiation behavior under the conditions tested, with no indication of breakaway swelling and moderate fuel/matrix interaction.

Full-sized pins - The lifetime test of two full-sized pin-type fuel assemblies are in progress in the IVV-2M reactor. In February 2005, the average burnup in the fuel assembly containing UO_2-Al dispersion fuel (being irradiated as a control) was ~35%, and that in the fuel assembly containing U-Mo dispersion fuel was 12%. The tests are planned to reach an average burnup of 60%.

IVV-2M tubes – Tubular U-Mo dispersion fuel undergoing irradiation at the IVV-2M reactor at the Institute of Reactor Materials (Zarechniiy) failed at approximately 60% burnup. Nominal fuel operating conditions were a surface heat flux of 57-118 W/cm^2 and a cladding temperature of 52-89°C. The failure mechanism appears to be the same seen in the FUTURE and IRIS-2 tests. Gamma scans of the tubes indicate the presence of a high flux gradient across the failed tubes during the latter part of their irradiation, and it is suspected that movement of the experiment to a location near the beryllium reflector resulted in a local power increase that contributed to the failure. The failure cross section is shown in Figure 1.10.

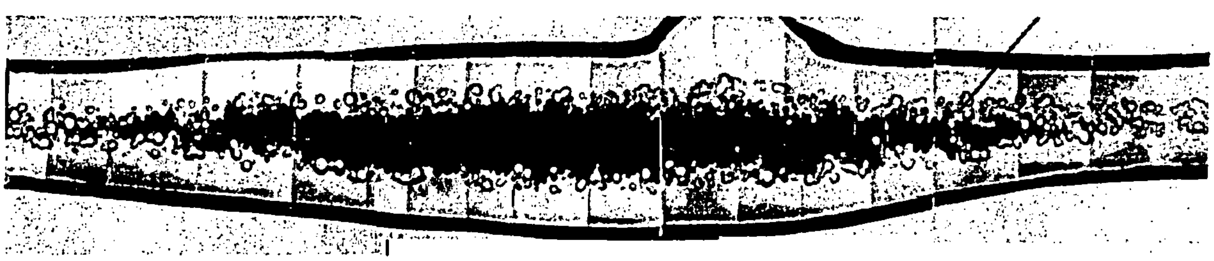


Figure 1.10. Cross section of failed U-Mo fuel tube from testing in the IVV-2M reactor.

1.2.5 Path forward for very-high-density fuel development

It became obvious in 2003 that U-Mo/Al dispersion fuels would not perform acceptably under the conditions required for high-powered research reactors. It is also clear that the fuel performance limitations are related to the unstable irradiation behavior of the (U-Mo)Al_x interaction layer that forms during irradiation under certain conditions. In no case is there evidence that the fuel particles do not behave well under irradiation. Given these observations, there are several potential 'fixes' available that allow for retention of a conventional aluminum clad fuel design. These are:

- Chemical modifications to the aluminum matrix and/or the dispersion fuel particles to stabilize the stoichiometry of the (U-Mo)Al_x interaction layer that forms. The primary candidate for addition to the matrix is silicon in quantities of less than 5 wt.%. The primary candidates for addition as an alloying element to the U-Mo are titanium and zirconium. This tactic also includes the potential for coating particles with materials that tend to stabilize the formation of lower aluminides, such as silicon or multiple layers.
- Replacement of the matrix aluminum with magnesium. Magnesium does not react with uranium molybdenum alloy during irradiation, as shown by RERTR-3 irradiation test results.
- Elimination of the matrix material, resulting in a 'monolithic' instead of a dispersion fuel. The resulting fuel is a laminate of U-Mo alloy clad in aluminum. This fuel has also shown promising irradiation behavior based on RERTR-4 irradiation test results.

In addition to these more conventional aluminum clad fuels, fuels based on zirconium will be considered as an alternative in the case that the fuel performance limitations of the aluminum-based fuels cannot be overcome.

2.0 Fuel development strategy

Fuel development and qualification can only proceed through a program focused on irradiation testing and supported by out-of-pile data and fuel performance modeling. The RERTR program is thus focused on:

- Developing fuel concepts that meet reactor operating requirements,
- Developing fabrication processes for these fuels that are amenable to commercial-scale fuel production,
- Proving performance of fuel concepts through irradiation testing of small specimens,
- Supporting irradiation testing data with fuel performance modeling and generation of out-of-pile data,
- Developing a data set for adequate for NRC approval through large scale demonstration of the performance of successful concepts, and
- Supporting conversions through the provision of fuel performance data and fuel performance models relevant to the reactors being converted.

2.1 Major milestones

GTRI program goals announced by Energy Secretary Abraham on May 26, 2004 in Vienna included the conversion of all civilian reactors that use HEU to LEU by 2014. In order to allow time for reactor conversion, fuels suitable for conversion must be developed well in advance of this date. These considerations lead to the following major milestones for the RERTR Fuel Development task:

Milestone 1

Provide qualified fuels required for conversion of all targeted western-designed research reactors to LEU by the end of 2010.

Milestone 2

Support development of fuel technology sufficient to allow the conversion of Russian designed reactors by 2014.

2.2 Limitations

Due to the complexities associated with the in-reactor behavior of fuel, the novel approaches required for fuel fabrication and the nature of research in general, it is impossible to predict cost and schedule with complete certainty. Proper planning and well-placed expenditures, however, can alleviate some of the risk associated with the RERTR fuel qualification.

Fuel testing cannot be accelerated beyond a certain point due to necessity to maintain prototypic irradiation conditions, the necessity to eliminate excessive fuel failure during testing, and the physical limitations associated with heat transfer from the fuel during testing.

2.3 Targeted reactors

Lists of research reactors targeted by the RERTR program for which a qualified fuel suitable for conversion to LEU is not available are given in Tables 2.1 and 2.2. These reactors have the

potential to consume almost 600 kg of HEU per year. The reactors in Kazakhstan and Poland will be converted by qualifying oxide and silicide fuels, respectively.

Recently the French announced that unless that fuel selection for the planned Jules Horowitz Reactor (JHR) will be made at the end of 2007, and that it is likely that if LEU-Mo fuels do not show significant promise by this point, that the choice for fuel development will be > 30% enriched U_3Si_2. This decision provides additional motivation to deploy a fuel development program that provides early results.

Table 2.1. Western designed and supplied research reactors that require new very-high-density fuels to convert to LEU and estimated HEU consumption.

	Reactor	Country	Power (MW)	HEU (Kg/yr)
1	BR-2	Belgium	80	[redacted]
2	RHF	France	57	[redacted]
3	ORPHEE	France	14	[redacted]
	JHR (planned)	France	100	-
4	FRM-II	Germany	20	[redacted]
5	MITR	USA	5	[redacted]
6	MURR	USA	10	[redacted]
7	NBSR	USA	20	[redacted]
8	HIFR	USA	100	[redacted]
9	ATR	USA	250	[redacted]
10	ATRC	USA	0.005	0
	Approximate total HEU consumption (Kg/yr)			[redacted]

Table 2.2. Russian designed and supplied research reactors that require new very-high-density fuels to convert to LEU and estimated HEU consumption.

	Reactor	Country	Power (MW)	HEU (Kg/yr)
11	LWR-15	Czech R.	10	[redacted]
12	VVR-K*	Kazakhstan	6	[redacted]
13	VVR-K Critical Facility*	Kazakhstan	0	[redacted]
14	MARIA**	Poland	17	[redacted]
15	IRT- MEPI	Russia (MEPI, Moscow)	2.5	[redacted]
16	IR-8	Russia (Kurchatov, Moscow)	8	[redacted]
17	IRT-T	Russia (NIIYaF, Tomsk)	6	[redacted]
18	VVR-TS	Russia (NIFKhI, Obninsk)	15	[redacted]
19	VVR-M	Russia (Gatchina)	18	[redacted]
20	IVV-2M	Russia (Zarechniy)	15	[redacted]
21	MIR-M1	Russia (RIAR, Dmitrovrad)	100	[redacted]
22	CA.MIR-M1	Russia (RIAR, Dmitrovrad)	0	[redacted]
			Approximate total HEU consumption (Kg/yr)	[redacted]

*Conversion of the VVR-K reactor and critical facility will be addressed by qualification of new oxide fuel elements.
**Conversion of the Maria reactor is currently being addressed through qualification of U_3Si_2 fuels elements.

2.4 Fuel Test Sequence

The research reactor fuel development testing sequence has typically included initial testing of miniplates, followed by testing of full-size plates, and finally verification of performance and commercial fabrication through the testing of large numbers of plates in prototypic elements. This testing sequence will be maintained in the current fuel qualification effort in order to maintain a balance between programmatic and technical risk.

2.4.1 Miniplate Testing

Miniplate tests are designed as scoping tests to provide basic fuel behavior feasibility information on new fuel concepts. The consequences of plate failure to reactor operation are low due to the small amount of fuel and diverse pathways for coolant flow incorporated into the design of fuel test rigs. Key factors for postirradiation examination are fuel swelling and characterization of chemical interaction between the fuel particles and the matrix. These tests are often conducted outside of the reactor core to simplify test licensing requirements. Small plate dimensions reduce the cost of testing a large number of variations in fuel design and composition, but do not provide information on the in-pile mechanical performance of full-width plates or the effect of flux and temperature gradients in the axial and transverse directions.

2.4.2 Full-size Plate Testing

These are typically tests of one to ten test plates contained in an MTR subassembly or a special test assembly. Often the plates are demountable; that is they can be removed from the test assembly for inspection in the reactor canal. Test plates are typically visually and dimensionally inspected after each cycle to look for signs of unusual behavior that may be precursors to failure. Plates that show signs of unusual behavior can be pulled from the test assembly. These tests provide assurance of acceptable fuel behavior prior to conducting tests of whole fuel elements, where the consequence of failure of an element is large.

2.4.3 Fuel Qualification Demonstration

Full-size fuel elements provide assurance that the fuel performs as designed in all aspects, and that the fuel can consistently be fabricated in a manner that does not lead to fuel performance issues. Postirradiation examination is nominal, and may consist of dimensional measurement (in-canal and in-cell), visual examination, gamma scanning, and dissection of a plate. Demonstration of acceptable fuel performance in element tests was regarded as the endpoint for fuel qualification for U_3Si_2 fuels, and resulted in the issue of NUREG-1313.

NUREG-1313 (Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors) concluded on the basis of the completed testing of six fuel elements that:

> *Section 4. CONCLUSIONS*
>
> *The tests and examinations summarized in Appendices A and B, including the extensive references to U_3Si_2-Al MTR (plate)-type fuel, provide the necessary bases for the staff's evaluations. On the basis of the currently available information, the staff concludes that the described fuel is acceptable for use in licensed nonpower reactors provided that individual safety analyses do not introduce new safety considerations. The staff will review the report on the full-core irradiation program once it is submitted, and if*

necessary, a supplement to the report will be issued if new safety considerations come to light. This conclusion also is contingent on the use of a uniform set of acceptance specifications consistent with those used for the fabrication of the test specimens.

Other Considerations

Most MTR-type reactors use a flat fuel plate or a fuel plate with a large radius of curvature. High flux reactors such as HFIR, BR2, JHR, and FRM-2 use more complex fuel elements. FRM-2 and HFIR fuel plates use fuel density zoning, burnable poison, and so-called 'involute' curvature. BR2 and JHR use elements composed of rings of three plates bent into a cylinder (each plate comprises a 120° section of the cylinder. Demonstration of the fuel performance of all of these variants will be necessary for fuel qualification for these reactors. These demonstrations are currently planned as part of phase I of RERTR fuel qualification through irradiation of prototypic test plates in the AFIP-3 experiment in ATR. Qualification of fuel elements for BR2 will likely be encompassed by the JHR fuel qualification process if U-Mo fuels are selected for development and qualification by CEA in 2007. If U-Mo fuels are not selected for the JHR startup core, there will likely be an effort to develop these fuels in France for implementation in a conversion core. Phase II fuel qualification will consist of fabricating conversion cores for RHF, HFIR, and FRM-II, inserting the conversion cores in reactor, and reactor startup and verification of fuel performance using the LEU fuel.

2.4.4 Russian Fuel Development

Fuel development in support of the conversion of Russian-designed reactors will proceed by providing consistent technical support and technical exchange with Russian RERTR program counterparts along with adequate and timely financial support through contracts for fuel development.

Fuel development will continue to be funded primarily through the Russian institute VNIIM (Bochvar Institute, Moscow). The U.S. RERTR fuel development program will work with the Russian RERTR program to develop a schedule for qualification of fuel for Russian designed reactors, and fund this work at a level that allows fuel qualification to proceed according to schedule. Support for fuel supply, including fuel specifications and coordination of fuel purchases will be provided according to direction from NA-212.

U.S. contracts will initially focus on the development and demonstration of the feasibility of a universal U-Mo pin-type fuel. This concept, if successful, would provide the base component of fuel that can be configured into element designs suitable for conversion of most Russian designed reactors that use aluminum-based fuel.

It is conceivable that fuel for reactors outside of Russia could be developed, if necessary, by the U.S. RERTR program and that a western supplier could tool up to produce this fuel type and supply it to these reactors. This is not currently under consideration due to the desire to maintain an open and noncompetitive relationship with Russian fuel developers to facilitate technical exchange.

Several issues remain to be addressed to meet GTRI program goals for HEU minimization, principally in relation to ensuring fuel development for four Russian high- temperature reactors (SM-3, PIK, RBT-6, RBT-10/2) and two associated critical facilities that use steel clad fuels with a Cu-Be matrix. Development of LEU fuels for these reactors is not currently funded or addressed by either the U.S. or Russian RERTR programs.

2.4.5 International Fuel Development Collaboration

There are currently six countries with active U-Mo fuel development programs. These are Argentina, Canada, France, the Republic of Korea, Russia, and the United States. In addition, ANSTO has a keen interest in U-Mo fuel development in order to provide a suitable back end solution for the OPAL reactor. Interaction with these countries on fuel development takes place in two ways. First, the RERTR International Fuel Development Working Group includes representatives from these countries and meets approximately on a semiannual basis to exchange information. Second the U.S. RERTR program uses bilateral agreements such as memoranda and contracts in order to accomplish work with several fuel development partners.

2.4.6 Fuel Testing Requirements

Basic Qualification Criteria

The most-important criterion for qualification is that the fuel meat and fuel plate, pin, or tube (or hereinafter fuel element) must behave in a predictable and stable manner during normal operation and during anticipated operational transients up to a U-235 burnup significantly higher than normal discharge burnup. Stable behavior means that:

- The fuel meat and, hence, the fuel element swells in a predictable manner and an acceptably low rate up to the required burnup. Rapidly accelerating, or breakaway, swelling is unacceptable. Swelling is important, of course, because the coolant volume and velocity is reduced when the fuel plate, pin, or tube swells.
- The cladding and meat remain bonded so that the cladding does not blister or "pillow." Therefore, the strength of the cladding/meat bond or the strength of any interaction product formed by interaction of the cladding and the meat must be great enough to resist a rupture during allowed mechanical deformations or during thermal expansion/contraction of the fuel plate. The latter situation must be considered during start-up and shut-down, and especially during scrams.

Predictable behavior means that the fuel behavior can be characterized in terms of the relevant irradiation parameters so that, for example the amount of swelling can be predicted under all operating conditions.

The second important criterion for qualification is the absence of fission-product leaks during irradiation. The manufacturing process must result in cladding without cracks or pin holes. In addition, blisters or pillows must not form during irradiation because stresses during blistering often result in cracks in the cladding through which the fission product gases that formed the blisters can escape.

The third important criterion for qualification is the requirement for a repeatable and robust fuel fabrication process.

Qualification Testing Requirements for Western-Designed Reactors

The RERTR program is collecting data on fuel requirements in the course of conversion studies for the class of high-powered reactors in order to better define testing requirements for fuel under development. As this data is being developed, irradiation test conditions are bracketed by considering the licensed operating envelope for high-powered reactors.

Western-designed reactors that use significant quantities of fuel fall into two broad classes: 1) MTR-type reactors (JMTR, OPAL) and other reactors (e.g., NBSR) with powers of 50 MW or less and moderate surface heat fluxes and 2) MTR-type and other reactors with powers greater

than 50 MW and/or high surface heat fluxes (e.g., ATR, JHR, BR2, HFIR, HFR-Grenoble, FRM-II). This latter class of reactors are the most demanding in terms of fuel operating conditions. In these reactors, high surface heat fluxes result from high volumetric power densities in the fuel meat. Reactors of the second class can have surface heat fluxes (hence, meat volumetric power densities) more than twice that of the first; e.g., BR2 has been licensed to operate with a peak heat flux of 600 W/cm^2. Although HFIR can operate with surface heat fluxes up to 620 W/cm^2, the velocity of the cooling water is greater, resulting in lower fuel-meat temperatures. It appears, then, that BR2 is the limiting case among currently operating reactors.

A heat flux of 600 W/cm^2 in the BR2 reactor results in a temperature rise of 143.4°C from the bulk water to the cladding surface. The temperature rise through the cladding is 24.4°C, so that the temperature at the cladding/meat interface is 206.8°C. Using a thermal conductivity of fresh U-Mo fuel of ~0.15 W/cm-K, the temperature rise from the fuel/meat interface to the center plane of the fuel meat is ~24.4°C. Therefore, the fuel meat at the central plane will be operating at 232°C when the bulk water temperature is 40°C. The peak heat flux likely will not occur at the top of the channel, but somewhere down the channel where the water has been heated above its inlet temperature. However, it is found that the increase in the bulk temperature of the coolant is essentially offset by an increase in the heat transfer coefficient, so that the fuel-meat temperature at the hot spot varies little as the hot spot is moved downward. This simple analysis does not take into account the effect of the surface oxide layer on fuel temperatures.

For purposes of initial design of irradiation tests, the values in the preceding paragraph will be used for the design of irradiation tests that bound fuel operating conditions in all currently operating and planned reactors. As reactor conversion analysis proceeds, a more concrete testing envelope will be developed.

3.0 RERTR Fuel Qualification Plan Summary

A fuel qualification plan is outlined in this section leading to generic qualification of very high density low-enrichment fuel prior to the end of 2010. This strategy relies heavily on the ATR to provide the primary data to support fuel approval by the NRC (through the issuance of a Safety Evaluation Report). The strategy for generating fuel qualification data consists of first identifying fuel testing requirements. Irradiation testing of miniplates and then full-size plates follows under the required conditions to obtain data on fuel performance in the required operating envelope. During this process, technologies are down selected as supported by available data. Finally, a large scale demonstration of the adequacy of the fuel is conducted through testing of a series of fuel elements representative of the fuel geometry and operating conditions in the reactors targeted for conversion.

The RERTR irradiation testing plan is based on the ATR reactor operating schedule as of October 2005. This schedule is subject to change, with corresponding changes in testing dates.

3.1 WBS Elements

The RERTR fuel qualification WBS includes 6 major elements for U.S. fuel development and 1 elements for foreign fuel development collaborations and assistance. These are as follows:

B.0 RERTR Fuel Development (RER200)

- B.1 Fuel Fabrication Development
- B.2 Out-of-Pile Modeling, Testing, and Analysis
- B.3 RERTR Fuel Development NDT Support
- B.4 Foreign Collaborations and Support
- B.5 Irradiation Testing
- B.6 Postirradiation Examination of RERTR Experiments
- B.7 RERTR Fuel Program Technical Integration Support

Detailed descriptions of each of these WBS elements are found in the INL RERTR Project Execution Plan.

3.2 RERTR Fuel Testing Plan

The scope and schedule of the program for developmental fuel irradiation testing is described in this section. Six developmental irradiation tests are planned, beginning with scoping miniplate tests and continuing with full-size plate irradiations in preparation for fuel element testing.

The developmental fuel testing phase of the program began with the RERTR-1 test in 1997 and will end with postirradiation examination of the AFIP-4 full-size plate test in 2009. A schedule for developmental fuel testing in ATR is provided in Table 3.1.

Table 3.1. ATR RERTR experiment schedule

	ATR cycle														
	134AB	135A PALM	135B	135C	136A	136B	137A	137B PALM	138A OUTAGE	138B	139A	139B PALM	140A	140B	141A
Month/Year*	4/05		7/05	9/05	11/05	1/06	3/06	5/06	8/06	10/06	12/06	2/07	3/07	5/07	7/07
RERTR-6 B-12	X		X	X											
RERTR-7A B-11					X	X	X	-	**						
RERTR-7B-C B-12						X	X	***							
RERTR-7B-B B-12, B-11							X	X							
RERTR-8 B-12									X	X	X	X			
RERTR-9 B-12													T	T	T
AFIP-1 Center Flux Trap									C	X	X		T		
AFIP-2 Center Flux Trap													T	X	X
AFIP-3 Center Flux Trap															»
AFIP-4 Center Flux Trap															»

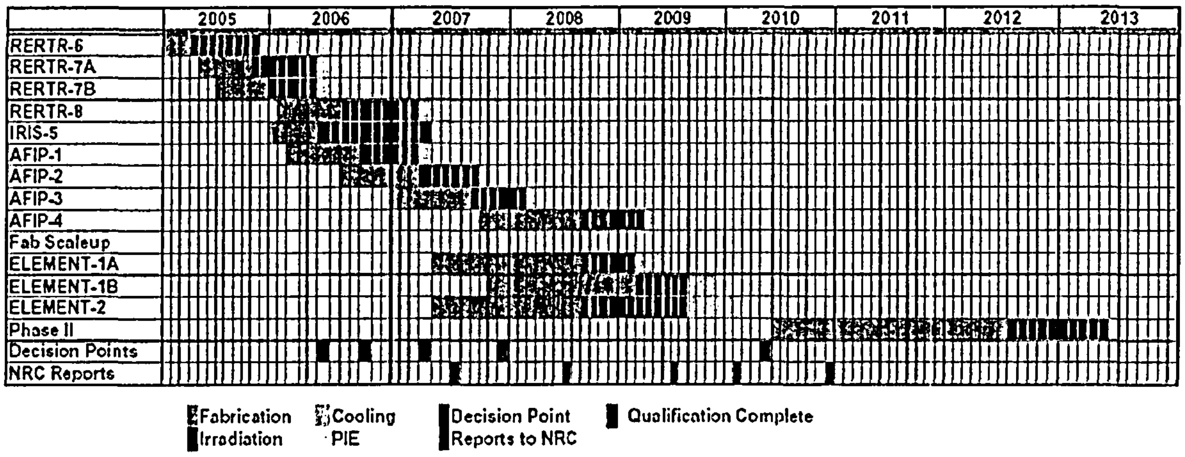
*Reactor operating schedule as of 10/2005, **Additional cycles possible depending on reactor operating history and actual fuel burnup. ***Possibility of irradiation during PALM cycle in low-power position. T -tentative. [C] Insertion into ATR-C critical facility.

Additional irradiation testing of MTR plates will be conducted in the IRIS-5 test in French OSIRIS reactor beginning in the spring of 2006.

Larger scale testing, including testing of fuel elements, is discussed in section 3.3. The overall schedule leading to fuel qualification is shown in Figure 3.1. Major decision points associated with this schedule are:

1) Decision on location of element tests – June 2006
2) Preliminary downselection of primary technology(s) – October 2006
3) Technology downselection for element testing – April 2007
4) JHR fuel downselection (external) – December 2007
5) Decision to fabricate elements for Phase II qualification – May 2010

Figure 3.1. RERTR Fuel qualification timeline.



3.2.1 Miniplate Testing

Four miniplate irradiation testing campaigns (RERTR-6 through RERTR-9) are planned as scoping studies to determine the feasibility of new fuel technologies under consideration. These tests continue to build on the database for U-Mo fuel performance established with the RERTR-1 through RERTR-5 experiments and ends with completion of RERTR-8 in early 2008.

These tests utilize up to 32 small plates per experiment to test a wide range of parameters at relatively low cost. The test specimens are easily fabricated, transported, and examined.

A sketch of the generic miniplate to be used in all tests is shown in Figure 3.2.

The scope and schedule for the planned miniplate tests are given below.

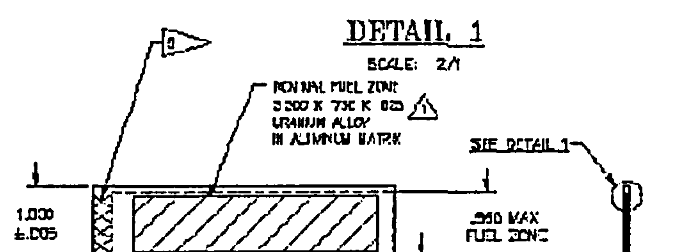


Figure 3.2. Sketch of generic miniplate to be used in the RERTR-6, -7, -8, and -9 experiments.

3.2.1.1 RERTR-6 Experiment

The RERTR-6 miniplate irradiation test, currently undergoing postirradiation examination, is a moderate power test that was irradiated over three long ATR operating cycles to a target peak burnup of 50% U-235. The experiment was fabricated using low-enriched uranium. Surface heat flux for the test plates is in the range of 100-200 W/cm^2. The RERTR-6 experiment includes both monolithic and dispersion fuel types. The monolithic fuel plates are an extension of promising exploratory tests performed in the RERTR-4 experiment. The monolithic fuel

meats consist of U-7Mo and U-10Mo foils with fuel loadings of up to 17.5 g-U/cm^3. Foil thicknesses of 0.01 and 0.02-in. were fabricated and inserted in the ATR. The total fuel loading is similar to that used in previous RERTR experiments. The dispersion fuel plates included in the experiment also leverage previous experience. In an effort to reduce the extent of fuel/matrix interactions observed in previous fuel studies an aluminum alloy (either 6061, 4043, Al-2Si, or Al-0.5Si) matrix replaced the pure aluminum alloy matrix used in earlier experiments. U-10Mo dispersion fuel plates using an unalloyed aluminum matrix were included as a benchmark fuel form.

RERTR-6 Objectives

The objectives of the RERTR-6 miniplate test are to:

1) Provide the first comprehensive set of fuel performance data on U-Mo monolithic fuels operated under moderate power and temperature conditions in a relatively short time period.

2) Provide initial information on the effectiveness of adding Si to the aluminum matrix to stabilize the irradiation performance of the interaction product in a relatively short time period.

RERTR-6 Schedule

Irradiation of the RERTR-6 experiment in the ATR began in cycle 134B on April 12, 2005.

The experiment remained in core through cycles 134B, 135B, and 135C. The experiment was removed from the reactor during short duration PALM cycle 135A. The experiment was discharged from the ATR at the end of cycle 135C in November of 2005 after a total irradiation time of 135.1 EFPDs and a calculated peak U-235 burnup of 50.4%. The RERTR-6 experiment was shipped in the GE-100 cask to the HFEF hot cell in January of 2006, and is currently undergoing postirradiation examination.

The miniplates will be subjected to the following examinations:

1) Visual examination
2) Dimensional measurement (thickness)
3) Immersion density
4) Gamma scanning
5) Metallographic examination
6) SEM/WDS examination
7) Postirradiation blister annealing

PIE is scheduled for completion in October of 2006.

3.2.1.2 RERTR-7A Experiment

The RERTR-7A miniplate irradiation test is a higher power, higher burnup test to be conducted over three long ATR operating cycles to a target fission density representative of a peak LEU burnup of >70% U-235 at peak experiment power sufficient to generate a peak surface heat flux >300 W/cm^2. The test will use the same hardware geometry as was used in the RERTR-4, -5, and -6 tests. Uranium enriched to 58% U-235 will be used to meet the test requirement for higher power density.

The RERTR-7A test was inserted into ATR cycle 136A in November of 2005. The test train contains monolithic and modified matrix dispersion fuels. The fuel plates are similar to those irradiated in the RERTR-6 experiment, but irradiated at higher power and to higher burnup.

The monolithic fuel plates were fabricated by transient phase liquid bonding and friction stir welding using both U-10Mo and U-12Mo alloys. Foils fabricated with thicknesses of 0.010" and 0.020" were included. Dispersion fuel plates include levels of silicon in the matrix that vary from 0 to 5 wt.%. Two zirconium clad LEU-10Mo monolithic fuel plates from CNEA (Argentina) are also included in the RERTR-7A test. These latter plates are fueled with low-enrichment uranium, and operate at lower power than the remainder of the experiments.

U-10Mo dispersion fuel plates using an unalloyed aluminum matrix were included as a benchmark fuel in the RERTR-7A test vehicle.

In-canal inspection of the RERTR-7 experiment capsules will be completed between irradiation cycles using backlit photography to identify gross fuel swelling or other defects.

RERTR-7A Objectives

The objectives of the RERTR-7A experiment are to:

1) Irradiate U-Mo monolithic fuel plates similar to test plates included in RERTR-6 at higher power density (300 w/cm^2) and to higher burnup (>70% LEU equivalent peak) to provide fuel performance data under these conditions.

2) Irradiate silicon modified aluminum matrix dispersions similar to RERTR-6 test plates at higher power density (300 w/cm^2) and to higher burnup (>70% LEU equivalent peak) to provide fuel performance data under these conditions.

3) Generate the first data relative to the irradiation performance of zirconium clad monolithic fuels.

RERTR-7A Schedule

The RERTR-7A experiment was inserted in ATR cycle 136A beginning irradiation in November of 2005. The experiment will remain in reactor for cycles 136A and 136B. Irradiation in cycle 137A is contingent upon meeting burnup goals. The experiment will be withdrawn from the reactor for cooling prior to June of 2006.

The RERTR-7 experiments will be cooled for the minimum time period consistent with ATR and HFEF technical specifications, the cask certificate of compliance, and heat generation from the plates that maintains a fuel temperature below the in-pile operating temperature during shipment.

The experiments will be shipped in the GE-100 cask to the HFEF (Hot Fuel Examination Facility) at INL for PIE (Postirradiation Examination). The miniplates will be subjected to the following examinations:

1) Visual examination
2) Dimensional and/or density measurement
3) Gamma scanning
4) Metallographic examination
5) SEM examination
6) Postirradiation blister testing

PIE is scheduled for completion in March 2007.

3.2.1.3 RERTR-7B Experiment

The RERTR-7B miniplate irradiation test is a high power test of advanced dispersion and monolithic fuels to be conducted over two long ATR operating cycles to a target fission density representative of a peak LEU burnup of >45% U-235 at peak experiment power sufficient to generate a peak surface heat flux >300 W/cm^2. The test includes two experiment capsules, one containing dispersion fuels with ternary element additions to the fuel particles and silicon additions to the matrix (RERTR-7B-C), the second containing monolithic and dispersion fuels to be irradiated in a high power PALM cycle (RERTR-7B-B). Magnesium matrix plates are schedule for inclusion in RERTR-7B-B. The test will use the same hardware geometry as was used in the RERTR-4 through RERTR-7A tests. Uranium enriched to 58% U-235 will be used to meet the test requirement for plate operating power.

RERTR-7B Objectives

The objectives of the RERTR-7B experiment are to:

1) Irradiate dispersion fuels which contain both silicon additions to the matrix and ternary alloy additions to the fuel particles at high power density (peak surface heat flux of 300 w/cm^2) and to moderate burnup (>45% LEU equivalent peak) to provide fuel performance data under these conditions relative to specimens with only silicon additions to the matrix and to U-7Mo/Al dispersions.

2) Provide early data on the irradiation behavior of U-Mo monolithic fuels, Mg-matrix fuels, and U-Mo dispersion fuels with silicon additions to the matrix and ternary element additions to the fuels at very high power (>500 W/cm^2).

RERTR-7B Schedule

The first RERTR-7B test capsule (RERTR-7B-C) will be inserted into ATR cycle 136B in January of 2006. Capsule RERTR-7B-C includes dispersion fuels with silicon additions to the matrix and ternary additions of titanium and zirconium to the fuel particles. U-7Mo dispersions in aluminum and 4043 aluminum matrices are included to differentiate the performance of these fuels. This capsule will be irradiated for 2 normal ATR cycles (136B and 137A).

An additional capsule (RERTR-7B-B) containing U-10Mo monolithic fuel, magnesium matrix fuel, and additional dispersion fuels with silicon additions to the matrix and ternary additions to the fuel particles will be inserted into the test train for ATR cycle 137A. Capsule RERTR-7B-B will remain in the ATR for one normal cycle (137A) to achieve an equivalent LEU burnup of >25% U-235 and then be subjected to high-power operation (>500 W/cm^2 surface heat flux) during ATR PALM cycle 137B.

In canal inspection of the RERTR-7B experiment capsules will be completed between irradiation cycles using backlit photography to identify gross swelling or other defects.

The RERTR-7B experiments will be cooled for the minimum time period consistent with ATR and HFEF technical specifications, the cask certificate of compliance, and heat generation from the plates that maintains a fuel temperature below the in- pile operating temperature during shipment. The target for shipment to HFEF is in July of 2006.

Both experiment capsules RERTR-7B-C and RERTR-7B-B will be shipped in the GE-100 cask to the HFEF (Hot Fuel Examination Facility) at INL for PIE (Postirradiation Examination). The miniplates will be subjected to the following examinations:

1) Visual examination

2) Dimensional and/or density measurement
3) Gamma scanning
4) Metallographic examination
5) SEM examination
6) Postirradiation blister testing

PIE of RERTR-7B is scheduled for completion in May 2007.

3.2.1.4 RERTR-8 Experiment

The RERTR-8 experiment is a high burnup, high power test of selected very-high-density fuels in the ATR. The RERTR-8 test will use the same hardware configuration as used in the RERTR-4 through RERTR-7 experiments. The current concept calls for irradiation of four miniplate capsules (up to 32 miniplates). The test will irradiate the most promising fuel concepts identified to date to high burnup (>75% peak LEU equivalent) at high power density (resulting in surface heat flux > 300 W/cm^2). The test will also include concepts that incorporate burnable poisons and density graded fuel zones into the fuel as proof of concept for these fuels. A subset of the fuel miniplates, including fuels with burnable poisons and density zoning, will be irradiated in high power PALM cycle 139B at a surface heat flux of >500 W/cm^2. The RERTR-8 experiment will benefit from postirradiation examination of the RERTR-6 experiment and preliminary examination of the RERTR-7A and RERTR-7B experiments. In-canal visual examinations of RERTR-8 capsules will be conducted between ATR operating cycles.

RERTR-8 Objectives

1) Test the most promising fuel concepts identified to date to high burnup (>75% LEU equivalent) at high power (power resulting in surface heat flux > 300 W/cm^2).
2) Test the feasibility of using burnable posions and density zoning in U-Mo monolithic and dispersion fuels at nominal and high powers.
3) Provide a further testing opportunity for second tier concepts such as zirconium clad fuels.

RERTR-8 Schedule

The RERTR-8 experiment will be inserted into ATR cycle 138A in August of 2006. RERTR-8 will reside in the ATR for three cycles (138A, 138B, 139A). Three capsules will be withdrawn from the test at the end of cycle 139A in February of 2007[1]. One capsule will remain in core for exposure during PALM cycle 139B.

The experiment will be shipped in the GE-100 cask to the HFEF (Hot Fuel Examination Facility) at INL for PIE (Postirradiation Examination) in April/May of 2007. The miniplates will be subjected to the following examinations:

1) Visual examination
2) Dimensional and/or density measurement
3) Gamma scanning
4) Metallographic examination
5) SEM examination
6) Postirradiation blister testing

[1] The experiment may be left in core for one additional cycle (140A) if burnup goals are not met.

PIE of RERTR-8 is scheduled for completion in December of 2007.

3.2.1.5 RERTR-9 Experiment

The RERTR-9 experiment is tentatively scheduled test of selected very-high-density fuels in the ATR. The RERTR-9 test, if conducted, will use the same basic hardware configuration as used in the RERTR-4 through RERTR-8 experiments. The test will be used to gather additional data on certain fuel types in support of fuel qualification. Irradiation power and burnup remain undefined until more information is available on fuel performance and actual irradiation test conditions that are attained in the preceding tests.

RERTR-9 Objective

1) Obtain additional data as required in support of the qualification of specific fuel types.

RERTR-9 Schedule

The RERTR-9 experiment is tentatively scheduled for insertion into ATR cycle will be inserted into ATR cycle 140A in March of 2007. RERTR-9 will likely reside in the ATR for three cycles.

The experiment would be shipped to the HFEF (Hot Fuel Examination Facility) at INL for PIE (Postirradiation Examination) in October/November of 2007. PIE would be scheduled for completion in August of 2008.

3.2.2 Full-size Plate Testing

Full-size plate tests typically consist of two to six test plates contained in an MTR subassembly or a special test fixture. Often the plates are demountable; that is they can be removed from the test assembly for inspection in the reactor canal. Test plates are typically visually and dimensionally inspected after each cycle for signs of unusual behavior that may be precursors to failure. Plates that show signs of unusual or unacceptable behavior can be pulled from the test assembly and replaced with additional plates. These tests provide assurance of acceptable fuel behavior under the mechanical conditions prior to conducting tests of whole fuel elements, where the consequence of failure of an element is large.

3.2.2.1 IRIS-5 Experiment in OSIRIS

The IRIS-5 test in the OSIRIS reactor is a joint test with the CEA that will generate the first fuel performance data on full-size monolithic fuel plates.

IRIS-5 Test Description

A drawing of the IRIS test fuel plate is shown in Figure 3.3. The RERTR program will provide two LEU test plates for irradiation, to be complemented by two plates fabricated by CERCA (France). If plates cannot be fabricated by CERCA, the U.S. RERTR program may provide four test plates. Plate thickness profiles will be monitored after each irradiation cycle for excessive swelling. A full postirradiation examination will be conducted at the CEA Cadarache site.

IRIS-5 Objectives

The objectives of the IRIS-5 test are to:

1) Generate the first irradiation performance data for full-size MTR U-Mo monolithic fuel plates through the use of in-canal plate thickness measurements and subsequent PIE.

2) Provide a comparison of the performance of monolithic fuel plates fabricated in France and the U.S. using different processes.

3) Provide diversity in fuel testing with irradiation conditions that differ from those in the ATR and to establish the framework for continued joint U.S./French testing in OSIRIS and other reactors.

IRIS-5 Schedule

The IRIS-5 test is scheduled for delivery to CEA-Saclay prior to May 1, 2006. Irradiation will begin in OSIRIS in June of 2006. This will require shipping of the plates to CEA in March of 2006. Irradiation will require 9 months, and be completed in March/April of 2007, followed by cooling prior to shipment to CEA Cadarache for PIE in September of 2007. It is anticipated that irradiated fuel will be returned to the U.S. for disposal prior to 2010.

Preliminary agreement with CEA has been made regarding experiment schedule and cost sharing. The CEA will fund irradiation in OSIRIS and one-half of PIE and disposal costs. The U.S. RERTR program will supply two (or four) monolithic plates for testing and pay one-half of the PIE costs.

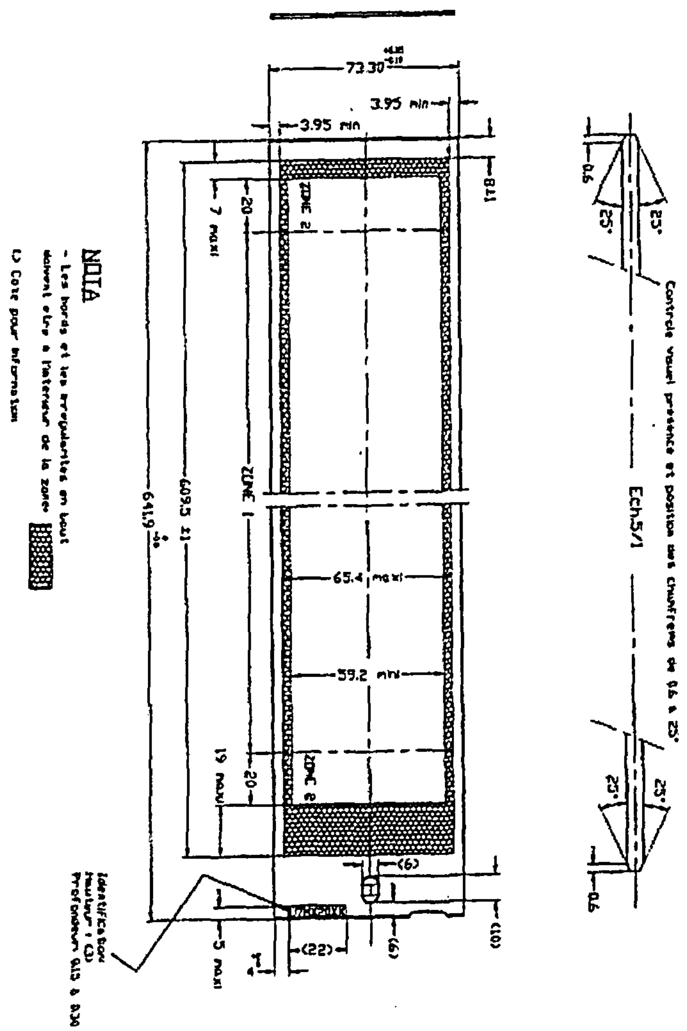


Figure 3.3. IRIS-5 test plate geometry.

3.2.2.2 Full-size Plate Testing in ATR

Four tests of full-size fuel plates are proposed for ATR. Labeled the AFIP (*A*TR *F*ull-size-plate *I*n center flux trap *P*osition) experiments, these tests will be conducted in the ATR center flux trap. Surface heat flux in excess of 350 W/cm^2 can be achieved using this arrangement. The first test (AFIP-1) will focus on monolithic and modified matrix fuel. AFIP-2 will include magnesium matrix fuel and density-zoned plates. AFIP-3 will include plates with burnable poison, density zoning, and involute geometry. Enrichment of 50-60% U-235 will likely be required in order to mimic fission rates typical of LEU plates in cores targeted for conversion. The geometry of the AFIP test hardware is shown in Figure 3.4. AFIP test geometry will be modified as necessary for the AFIP-3 and -4 tests of density zoned and curved plates.

Testing of full-size plates in ATR will benefit from the development of the capability for following plate swelling as a function of burnup through the measurement of plate thickness between cycles. A demountable plate irradiation facility coupled with a thickness measurement system will be used for in-canal measurement of plate swelling for the AFIP series experiments.

Full-size plate testing in the ATR will begin with the insertion of the AFIP-1 test in ATR cycle 138B in October of 2006.

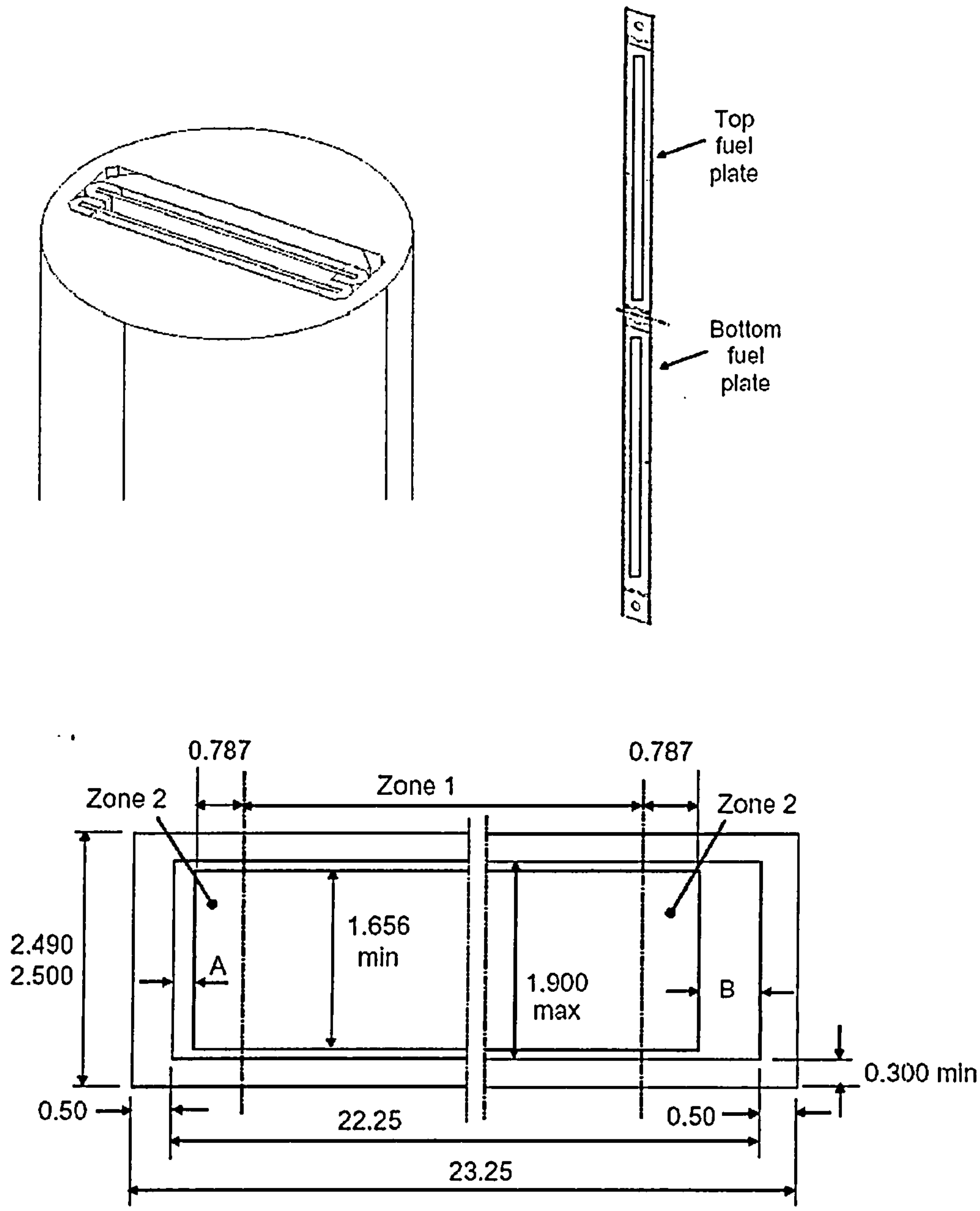


Figure 3.4. Top – AFIP center flux trap position concept. Bottom – Approximate AFIP plate dimensions.

3.2.2.3 AFIP-1 Experiment in the ATR

AFIP-1 Test Description

The AFIP-1 test will irradiate three monolithic and three aluminum matrix dispersion fuel plates to moderate burnup (50-60% LEU equivalent burnup) at moderate to high surface heat flux (~300 W/cm^2). AFIP-1 will be delivered to ATR-C in August 2006. Insertion of AFIP-1 into ATR will be in ATR cycle 138B, scheduled for October 2006 following the long ATR summer outage. AFIP-1 experiment insertion is supported by RERTR-6 PIE data and preliminary RERTR-7 visual examination data. The AFIP test is configured so that plates may be removed and replaced with additional plates pending confirmation of adequate experiment safety performance. AFIP-1 test plates will be subjected to in-canal visual examination and dimensional (thickness) measurement between ATR cycles.

AFIP-1 Objectives

The objectives of the AFIP-1 test are to:

1) Produce data to inform the down-selection of both monolithic and dispersion fuel technologies based on the performance of full-size monolithic and modified matrix dispersion fuel plates at surface heat flux higher than attainable in the IRIS-5 experiment.

2) To provide diversity in test conditions and testing reactors relative to the IRIS-5 test.

3) To provide early data on fuel swelling of advanced very-high-density fuels through the use of an in-canal thickness measurement device.

4) Exercise the U.S. research reactor fuel development infrastructure that will be required for LEU fuel qualification.

AFIP-1 Schedule

The AFIP-1 test will be inserted into the ATR beginning in cycle 138B in October of 2006. The AFIP-1 test will be irradiated during cycles 138B, 139A, and 140A to a peak fission density representing an effective LEU burnup of 50-60%. Fuel swelling data from in-canal measurements will be available at 15-20% burnup in November of 2006. It is important that the in-canal examination equipment be inserted in the ATR canal prior to the long outage associated with cycle 138A so that the equipment and methodology for plate thickness measurement can be refined prior to the need date. The experiment will be removed from the reactor for cooling in February 2007. An additional irradiation cycle (140A) may be added to some or all of the plates pending evaluation of as-run conditions. After cooling, the experiment will be shipped in the GE-2000 cask to the HFEF (Hot Fuel Examination Facility) at INL for PIE (Postirradiation Examination). The fuel plates will be subjected to the following examinations:

1) Visual examination
2) Dimensional and/or density measurement
3) Gamma scanning
4) Metallographic examination
5) SEM examination
6) Postirradiation blister testing

PIE is scheduled in two phases. Phase I will be completed in June 2007 to provide input for monolithic fuel technology down selection. Completion of phase II of the PIE is scheduled for February 2008.

3.2.2.4 AFIP-2 Experiment in the ATR

AFIP-2 Test Description

The AFIP-2 test will continue irradiation of monolithic and dispersion fuel technologies to moderate burnup (>60% LEU equivalent burnup) at high surface heat flux (~350 W/cm^2). Insertion of AFIP-2 will be in ATR cycle 140B, currently scheduled for May 2007. AFIP-2 experiment insertion will be supported by RERTR-6 and RERTR-7 PIE data and AFIP-1 thickness measurement and visual examination data. The AFIP-2 test is configured so that plates may be removed and replaced with additional plates pending confirmation of adequate experiment safety performance.

AFIP-2 Objectives

The objectives of the AFIP-2 irradiation test are:

1) To irradiate monolithic MTR-type fuels to higher burnup at higher power density than achieved in the AFIP-1 test.

2) To irradiate advanced dispersion fuels to higher burnup at higher power density than achieved in the AFIP-1 test.

3) To provide the first fuel performance data for full-size magnesium matrix plates.

4) To irradiate full-size plates containing burnable poisons.

AFIP-2 Schedule

The AFIP-2 test will be inserted in ATR for cycle 140B, currently scheduled for May of 2007. The AFIP-2 test will be irradiated during cycles 140B, 141C, and another unscheduled cycle. Fuel swelling data from in-canal examination will be available at 15-20% burnup in June 2007. The experiment will be removed from the reactor for cooling in July 2007. After cooling, the experiment will be shipped in the GE-2000 cask to the HFEF (Hot Fuel Examination Facility) at INL for PIE

The fuel plates will be subjected to the following examinations:

1) Visual examination
2) Dimensional and/or density measurement
3) Gamma scanning
4) Metallographic examination
5) SEM examination
6) Postirradiation blister testing

Completion of AFIP-2 PIE is scheduled for April of 2008.

3.2.2.5 AFIP-3 and AFIP-4 Experiments in the ATR

AFIP-3 and AFIP-4 Test Description
The AFIP-3 test is targeted at generating fuel performance data for plates with density zoning, involute shape, and burnable poisons. Qualification of this fuel follows qualification of more typical MTR-type plates. AFIP test hardware will be reconfigured as necessary to support irradiation of these fuel plates. Experiment insertion is supported by irradiation data from the RERTR-6, RERTR-7, RERTR-8 miniplate tests and the AFIP-1 and AFIP-2 irradiation tests of full size plates. The AFIP-3 test may also allow the opportunity to bolster test data or repeat plate tests conducted in the AFIP-1 and AFIP-2 experiments. Plates tested in the AFIP-3 experiment will be subjected to in-canal dimensional measurement and visual examination between ATR cycles. The test will be configured to allow for continuation of testing as needed to support development of complex fuels. The AFIP-4 test allows for follow-on irradiations in support of complex fuel testing.

AFIP-3 and AFIP-4 Objectives
The objectives of the AFIP-3 and AFIP-4 irradiation tests are:

1) To provide the first fuel performance data for density-zoned involute plates containing burnable poisons.

2) To continue to test complex fuels as required to demonstrate that very-high-density fuel performance is adequate for high-power reactors that require complex fuels.

3) To provide the opportunity for follow up irradiation testing of other fuel concepts as required by the program.

AFIP-3 and AFIP-4 Schedule
The AFIP-3 test will be inserted into ATR during a currently unscheduled beginning approximately in September/October 2007. The AFIP-3 test will be irradiated for three long ATR cycles to a peak fission density equivalent to 40-60% U-235 LEU burnup at high power density (sufficient to produce a surface heat flux >300 W/cm^2). The AFIP-3 experiment will be removed from the reactor for cooling in (approximately) April 2008. The AFIP-4 experiment will be inserted in the ATR approximately 6 months after removal of AFIP-3, during the last quarter of 2008. AFIP-4 will irradiate until March/April 2009. After cooling, the experiments will be shipped in the GE-2000 cask to the HFEF (Hot Fuel Examination Facility) at INL for PIE (Postirradiation Examination). The fuel plates will be subjected to the following examinations:

1) Visual examination
2) Dimensional and/or density measurement
3) Gamma scanning
4) Metallographic examination
5) SEM examination
6) Postirradiation blister testing

Completion of AFIP -3 PIE is scheduled for December 2008. Completion of AFIP-4 PIE is scheduled for December 2009.

3.3 Qualification Demonstration

The last phase of fuel qualification entails a proof-of-performance demonstration that the developed fuel behaves adequately during all reactor normal (and potentially some off-normal) operating conditions and that it can be fabricated in a manner that guarantees stable fuel performance. This demonstration is normally made by irradiating several fuel elements. The elements can be manufactured by different fabricators using slightly different processes and materials.

Fuel qualification will be completed in two phases. Phase I will target those reactors for which representative elements can be tested adequately in existing test reactors. These reactors use flat or slightly curved MTR- type fuel elements, as well as cylindrical elements for which fuel qualification can be demonstrated by irradiating prototypic elements in a test reactor.

Phase II fuel qualification will target reactors that utilize more complex annular fuel elements for which performance of an entire fuel element cannot be demonstrated in a test reactor due to space limitations. These reactors include HFIR, RHF, and FRM-II. Qualification of fuel elements for these reactors will have to occur in the actual reactors that are targeted for conversion. A recent example of this type of qualification was that of the FRM-II core. The FRM-II fuel element is an annular design composed of plates that feature involute curvature and density zoning. No testing of the full element was conducted prior to installation into FRM-II. Instead, irradiation of test plates was completed in test reactors to demonstrate fuel performance, and the actual 'qualification' of the elements occurred by bootstrapping the reactor to power using the new element design.

Data from the U.S. fuel development program will be supplemented by data from foreign fuel development programs. In particular data from the Jules Horowitz reactor fuel qualification effort (JHR test elements will be irradiated in the BR2 reactor in Belgium), will support qualification of fuels for JHR. Qualification of JHR elements by CEA will also likely envelope fuels required for conversion of BR2.

The last RERTR fuel qualification demonstration began in the early 1980's with fabrication of ORR (Oak Ridge Reactor) nineteen plate fuel assemblies by three vendors; CEA, NUKEM (now defunct), and Babcock & Wilcox (now BWXT). One element from each vendor was irradiated nominally to approximately 50% average burnup, the other to more than 75% burnup. Surface heat flux was 94-140 W/cm^2, much less than the requirements for very high density fuel (peak > 300 W/cm^2).

As a backup or addition to element testing in ATR, loop positions in the MIR test reactor in Dimitrovgrad are under consideration. Final selection of the location for element testing will be made in June of 2006.

3.3.1 Phase I Fuel Qualification

The reference Phase I U.S. qualification demonstration calls for irradiation of four test elements in ATR driver core positions and two test elements in an MTR-type reactor. Between irradiation cycles, channel gap thickness will be measured to indicate swelling or excessive oxide layer growth. Destructive and non-destructive PIE will be conducted after element irradiation.

After successful completion of the ELEMENT-1 and ELEMENT-2 tests a fuel qualification data package will be submitted to NRC for review. Issuance of a NUREG fuel safety evaluation report (SER) by the NRC is anticipated in September 2010, after which the reactors shown in Table 3.1 could be converted to LEU through a program of lead test element irradiations followed by core replacement.

Table 3.1. Reactors that could convert to LEU upon licensing after completion of the Phase I fuel qualification.

	Reactor	Country	Power (MW)	HEU (Kg/yr)
1	ORPHEE	France	14	[redacted]
2	MITR	USA	5	[redacted]
3	MURR	USA	10	[redacted]
4	NBSR	USA	20	[redacted]
5	BR-2	Belgium	80	[redacted]
6	ATR	USA	250	[redacted]
7	ATRC	USA	0.005	-
8	JHR (to be built)	France	100	?
			Total HEU (kg/yr)	[redacted]

3.3.1.2 ELEMENT-1 Fuel Qualification Test

The reference strategy for obtaining primary fuel qualification data in support of licensing will be to generate data by irradiating test elements in ATR driver core positions under conditions enveloping reactors for which the fuel is targeted. Four elements will be tested in the ATR. The reference plan is to first irradiate a lead test element that consists of some or all of the 11 inner plates with LEU plates, leaving the outer burnable poison containing plates intact. When acceptable fuel performance has been demonstrated through in-canal visual examinations after the first irradiation cycle, a second identical element will be inserted. The third and fourth elements inserted in the subsequent cycle will be fuel assemblies incorporating fuel with LEU plates containing burnable poisons.

Element-1 Test Objective

The objective of the ELEMENT-1 irradiation is to demonstrate robust and reliable fuel performance for selected very-high-density fuel types required for conversion of Phase I targeted reactors. This testing will provide data supporting a fuel qualification report to the NRC, after

which issuance of an SER concerning the suitability for use in reactor conversions is anticipated. This SER would serve to 'qualify' VHD fuels for use as LEU replacement fuel suitable for conversion of all reactors listed in Table 3.1. The Element tests will utilize in-canal channel gap width measurements between irradiation cycles as a monitor of fuel swelling.

Element-1 Schedule

ELEMENT-1 fuel fabrication will begin in May of 2007. Planned insertion of the ELEMENT-1 test is in September of 2008. At the time of insertion, PIE data will be available from RERTR-6, RERTR-7, RERTR-8 and the AFIP-1 and AFIP-2 tests. If possible, test elements will be fabricated by commercial vendors (foreign and domestic), or at minimum, using a prototypic fuel fabrication line at a DOE laboratory. The four test elements will reside in in-core for 3-5 irradiation cycles under irradiation test conditions that envelope those in reactors targeted for conversion. The first elements will be withdrawn from the reactor for cooling in approximately May of 2009, and shipped to the HFEF hot cells for postirradiation examination.

PIE of the intact element will consist of visual examination, dimensional inspection (twist, bow, and element dimensions), and gamma scanning. The elements will then be disassembled and all individual fuel plates subjected to visual inspection. Approximately five plates from each element will be selected for thickness measurement and individual plate gamma scanning, and one plate will be selected for destructive metallographic analysis and sampling for burnup measurement. The majority of PIE will be completed in December 2009 and will provide input to the fuel qualification data package submitted to NRC in February 2010.

3.3.1.3 ELEMENT-2 Test

Fuel element testing in either the French OSIRIS or the HFR in Petten, The Netherlands, will add to the quantity and diversity of fuel qualification data generated by the ELEMENT irradiation test. Two U.S. fabricated fuel elements may be supplemented by elements fabricated in France in an irradiation test conducted by replacing standard MTR driver core elements by U-Mo VHD test elements. Testing will be conducted at a lower surface heat flux imposed by the nature of the pool-type MTR reactors under coolant chemistry conditions that differ from that in the ATR. The differing coolant chemistry lead to a potentially thicker surface oxide layer and a different ratio of fission rate to fuel temperature than in ATR testing.

ELEMENT-2 Test Objectives

The objectives of the ELEMENT-2 test are to:

1) Provide supplemental information in addition to the fuel qualification data generated in the ELEMENT-1 test.

2) Provide fuel performance data for fuels fabricated by non- U.S. vendors

3) Provide for diversity in irradiation suppliers and irradiation test conditions.

ELEMENT-2 Test Schedule

ELEMENT-2 test fabrication will begin in May 2007 for a planned insertion in October 2008, lagging slightly the insertion of the ELEMENT-1 test in ATR. At the time of insertion, PIE data will be available from the RERTR-6, -7, and -8 miniplate tests and from the AFIP-1 and AFIP-2 tests. The two test elements will reside in core for approximately one year, and be subject to in-canal visual and channel gap width examinations between irradiation cycles. Elements will be

withdrawn from the reactor and subject to a final examination in canal in October 2009. The elements will be shipped to hot cells either at CEA Cadarache or Petten for postirradiation examination beginning in April 2010. The in-canal examinations will provide support for the phase I fuel qualification package.

PIE will consist of element visual examination, dimensional inspection, and gamma scanning. The elements will then be disassembled and all individual fuel plates subjected to visual inspection. Approximately five plates from each element will be selected for thickness measurement and individual plate gamma scanning, and one plate will be selected for destructive metallographic analysis and sampling for burnup measurement. PIE will be completed in 2010, and will be available to provide supplemental input into the phase I fuel qualification data package submitted to NRC.

3.3.2 Phase II Fuel Qualification

Phase II fuel qualification will target reactors that utilize more complex annular fuel elements for which performance of an entire fuel element cannot be demonstrated in a test reactor due to space limitations. These reactors include HFIR, RHF, and FRM-II. Final qualification of fuel elements for these reactors will have to occur in the actual reactors that are targeted for conversion through the insertion of an entire LEU core. A recent example of this type of qualification was that of the FRM-II core. The FRM-II fuel element is an annular design composed of plates that feature involute curvature and density zoning. No testing of the full element was conducted prior to installation into FRM-II. Instead, irradiation of test plates was completed in test reactors to demonstrate fuel performance, and the actual 'qualification' of the elements occurred by bootstrapping the reactor to power using the new element design.

Elements will be fabricated that are representative of production fuel that is to be used in the targeted reactor. The issuance of the NRC Safety Evaluation Report in late 2010 concerning generic acceptance of new fuels as demonstrated by miniplate, full-sized plate, and element testing will serve as the basis document for supporting the licensing changes required for reactors utilizing new LEU fuels. After installation of the LEU core and successful startup testing, the reactors shown in Table 3.2 would be converted to LEU.

Since each of these reactors has unique licensing requirements, a detailed startup and test plan will be required for each reactor. These specific plans must be generated by the converting facility in conjunction with the RERTR program.

Table 3.2. Reactors that could convert to LEU upon licensing after completion of the Phase II of fuel qualification

	Reactor	Country	Power (MW)	HEU (Kg/yr)
1	RHF	France	57	[redacted]
2	FRM-II	Germany	20	[redacted]
3	HIFR	USA	100	[redacted]
			Total HEU (kg/yr)	>170

Phase II Qualification Strategy

Fabrication of LEU fuel elements for conversion of these three reactors will begin in 2010 and will be completed by 2012. Startup testing requirements will be individually determined by each reactor, including verification of fuel performance, and will be determined prior to delivery of cores in 2012. In some cases, it may be possible to test individual LEU plates assembled into a standard HEU core first, or to design a transition core that allows for removal and inspection of fuel plates during startup testing. Requirements for postirradiation examination will also vary, and may range from in-canal visual and dimensional examination to destructive examination of removable plates in a core, to destructive examination of an entire core to gain access to individual plates.

3.3.3 Fuel Qualification Reports

Three reports are planned for transmittal or submission to NRC; two preliminary reports on U-Mo irradiation behavior for informational purposes and a final fuel qualification report submitted for NRC review as input to the anticipated Safety Evaluation Report (SER) for very-high density fuels.

A preliminary report on U-Mo fuel behavior will be transmitted for information to the NRC in July of 2007, following completion of RERTR-7 PIE. A second report will be transmitted to NRC for information following completion of the RERTR-8, IRIS-5, and AFIP-1 PIE in July 2008.

The final fuel qualification report will be transmitted to NRC as data supporting the issuance of an NRC SER for VHD fuel. Transmittal in February 2010 will follow completion of the majority of PIE on the ELEMENT-1 test in ATR and in-canal examination of the ELEMENT-2 test. It is anticipated that NRC review will require nine months, during which time RERTR staff will be available to support requests for additional information. This schedule will result in issue of a Safety Evaluation Report in September 2010.

4.0 Fuel Fabrication Development

Fuel development hinges on the ability to fabricate representative fuel specimens for testing. Fuel fabrication methods and the details of particular processes can impact fuel performance. It is important to realize that the RERTR fuel development task be front loaded in this area to allow for the development of fabrication processes and the production of specimens for testing. Technologies to be developed are discussed in the following sections.

4.1 Monolithic Fuel

Monolithic fuel features the highest possible uranium density, thus allowing the most flexibility in reactor conversions. Monolithic fuel is based on elimination of matrix aluminum and replacement with a solid U-Mo alloy foil. This concept eliminates the fuel performance problems related to reaction of fuel and aluminum during irradiation. PIE data from two miniplate specimens have been irradiated to a peak burnup of >70% at relatively low power, in the RERTR-4 experiment and show acceptable irradiation behavior. Principal obstacles to development of this fuel include unknown irradiation behavior of full-size plates and fabrication scale up.

4.1.1 Monolithic Fuel Fabrication Development

Fabrication requires a significant departure from the conventional dispersion fuel fabrication process. Three methods are currently under consideration; these are FSW (friction stir welding),

TLPB (Transient Liquid phase Bonding), and HIP (Hot Isostatic Pressing). All of these processes require demonstration of fuel performance through irradiation testing.

4.1.2 Uranium Alloy Foil Fabrication

Integral to the deployment of monolithic fuels is the development of a simple and economic process for foil production. Three lines of R&D are being followed. These are cold rolling and continuous casting. The continuous casting process is being developed using custom designed process equipment. The cold rolling process will require the acquisition of a small rolling mill. This equipment will also be used to finish foils made by continuous casting to final thickness. DOE's Y-12 complex is also developing methods for commercial-scale foil fabrication beginning in FY06.

4.2 Mg-matrix Fuels

Magnesium matrix fuels are the second line of fuels that require fabrication process development. The rolling properties of magnesium are much different than those of aluminum. The formation of an aluminum/magnesium eutectic at 437°C limits fuel rolling temperatures to less than that temperature, which makes aluminum to aluminum bonding more difficult.

Process development includes fine tuning of the traditional dispersion fuel fabrication techniques for use with magnesium matrix fuel and the development of hybrid fabrication processes that ensure aluminum-to-aluminum bonding prior to rolling.

4.3 Modified matrix fuels

Development of novel techniques for fabrication of aluminum matrix fuels modified with silicon is not foreseen. Indications from fabrication of Al/Si matrix miniplates indicates that some process development will be required to produce test plates for irradiation.

4.4 Other Fuel Types

If the performance of monolithic, Mg-matrix, and modified matrix fuels should prove to be inadequate, other possibilities can be explored, although it will be more difficult to meet the FY10 milestone for fuel qualification using these fuels. Some alternate possibilities are listed here.

Zr-matrix Dispersion Fuels

If the potential for the reaction of magnesium with reactor coolant during operation or canal water during long-term storage becomes an issue, the potential for use of a zirconium matrix exists.

Zr Clad Monolithic Fuels

If reaction of U-Mo with aluminum remains an issue with monolithic fuels, it may be possible to substitute zirconium cladding. The Argentine fuel development program has conducted some preliminary experiments in rolling Zr-clad monolithic fuel plates. Two zirconium clad U-Mo monolithic miniplates are currently under irradiation in the RERTR-7A experiment. U-Nb-Zr monolithic fuels may also be pursued as an alternative to U-Mo monolithic fuels.

Pin-type Fuels

A larger departure from current core design but still within the realm of possibility would be conversion of some reactors to pin-type fuels. It has been shown that pin- type fuels offer more

restraint to fuel swelling than plate fuels. Due to the smaller surface area for heat transfer, pin-type fuels operate at higher fuel centerline temperatures than plate fuels for the same power density, and may require the use of matrix materials other than aluminum. An analysis is necessary before firm fuel concepts can be established. This fuel development activity would parallel RERTR funded development of pin-type fuels in the Russian program.

5.0 Out-of-Pile Testing & Analysis

Out-of-pile analysis activities are conducted in support of the core irradiation testing. This effort falls into three areas: (1) fuel performance code development, (2) fuel/matrix interaction, and (3) ion beam irradiation.

5.1 Fuel Performance Codes

Two fuel performance codes are currently developed and used by the RERTR program. These codes are the DART code and the PLATE code. The PLATE code is based on a thermohydraulic backbone and is used to analyze fuel performance through the understanding of fuel operating conditions. DART is a more complex code that attempts to use mechanistic models to predict fuel behavior on a more fundamental level. Although both codes are useful to some degree, development of a single code provides for the best use of program resources. This code will be validated against fuel performance data from U.S. and foreign experiments and benchmarked against foreign fuel performance codes. The code will eventually support reactor conversion through analysis of fuel operating conditions in targeted reactors. It is not anticipated that the code will support NRC licensing of developed fuels.

5.2 Fuel Matrix Interaction

Studies of fuel matrix interaction support the understanding of in-reactor fuel behavior through the identification and analysis of reaction products that may potentially form in reactor. Measurement of U-Mo reaction enthalpy is also important in order to understand fuel behavior during off-normal conditions.

5.3 Ion Irradiation

Ion irradiation allows out-of-pile observation of the effects of irradiation damage and the combined processes of fuel/aluminum reaction and irradiation. This task will continue through FY06.

6.0 Capability and Infrastructure

Development of infrastructure to support RERTR fuel development is required in four areas. These are (1) fuel fabrication, (2) minor upgrades to HFEF equipment to allow for measurements on flat plates, (3) the qualification of additional casks for irradiated fuel transport, and (4) design and installation of ATR in-canal fuel examination facilities for plates and elements.

6.1 PIE Capability

The U.S. infrastructure for postirradiation examination is adequate for basic miniplate examinations, however some minor upgrades are required for full-size plate and fuel element examinations.

6.1.1 Hot Cell Engineering

Hot cell upgrades are conducted in three phases as needed to support handling and examination of fuel specimens. Phase I designs and implements necessary handling fixtures in HFEF, and includes modifications to the gamma scanner to allow for miniplate scanning, provisions for measuring fuel thickness and density, and postirradiation blister testing. Phase II makes similar modifications for full-size plates. Phase III allows for the handling and examination of fuel elements.

6.2 In-canal Examination

Facilities for in-canal measurement of the thickness changes in fuel due to irradiation do not exist in the U.S. Such a facility will need to be developed to support U.S. VHD fuel qualification. This capability allows monitoring of fuel swelling between irradiation cycles, which gives early indications of fuel performance problems.

6.3 Transportation of Irradiated Fuel

Transport of irradiated fuels from reactor to hot cell may be a significant issue due to the projected retirement of the GE- 100 and T-2 casks in 2008. Retirement of theses casks affects not only the ability to ship RERTR fuels using the GE- 100 and T-2, but also impacts the availability of other casks in service that must make up for shipments that would have been made using the GE-100. This situation can cause long delays in shipment of fuel due to cask unavailability.

The GE-100 cask will be used for miniplate transport through 2007, with the more expensive GE-2000 as a backup option. As the GE-100 is retired from service and the emphasis of the program shifts to larger plates, another cask will be qualified for transport of fuel specimens between ATR and HFEF.

6.4 Fuel Fabrication Facilities

RERTR fuel fabrication laboratories at ANL and INL will be upgraded as required including equipment for friction stir welding, fuel plate cleaning, and a clean test assembly and inspection area.

7.0 Summary

In late 2003 it became evident that U-Mo aluminum fuels exhibited significant fuel performance problems under the irradiation conditions required for conversion of most high-powered research reactors. An RERTR program strategy has been mapped that will allow generic fuel qualification to occur prior to the end of FY10 through the anticipated issue of an NRC Safety Evaluation Report.

Phase I of fuel qualification will be completed in FY10. The qualification will cover all very-high density fuel variants, including those that require burnable poisons and density zoning. The final phase of qualification will consist of irradiation of MTR plate-type fuel assemblies in two reactors. After this work is completed, reactors that use discrete fuel elements can convert through the process of irradiation of lead test elements.

Phase II fuel qualification will target reactors that utilize more complex annular fuel elements for which performance of an entire fuel element cannot be demonstrated in a test reactor due to space limitations. These reactors include HFIR, RHF, and FRM-II. Because discrete lead test elements cannot be used in these reactors, final qualification of fuel elements for these reactors will have to occur in the actual reactors that are targeted for conversion.

In addition to the U.S. fuel development effort, contracts with Russian Federation institutes in support of fuel development for Russian are in place. A Russian fuel development strategy is under has been developed.

The overall RERTR fuel development program thus currently results in qualification of fuels suitable for conversion of 23 targeted research reactors that consume almost 600 kg of HEU per year.

Several fuel development issues remain to be addressed, principally in relation to ensuring fuel qualification for four Russian high-temperature reactors that are not currently addressed or funded by either the U.S. or Russian RERTR programs. An additional significant cost is that of upgrading the U.S. commercial infrastructure to support LEU fuel supply after qualification.

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ATTACHMENT 2 TO THE BATTELLE ENERGY ALLIANCE REQUEST TO EXPORT FUEL PLATES TO THE FRENCH OSIRIS REACTOR IN SUPPORT OF DEPARTMENT OF ENERGY WORK

	DIRECTION DE L'ENERGIE NUCLEAIRE – Centre de Saclay DÉPARTEMENT DES RÉACTEURS ET SERVICES NUCLEAIRES **Service d'Irradiations en Réacteur et d'Études Nucléaires**	Nature du document : Cahier des Spécifications et des Charges Techniques
		Référence : DRSN/SIREN/LECSI/CSCT/XXX
		Affaire : A-MACOB-02-09

Attachment 2 to Request to Export Fuel Plates to the French OSIRIS Reactor
in Support of Department of Energy Work

MONOLITHIC FUEL FOR IRRADIATION TESTING IN THE **OSIRIS** *REACTOR*

A	22/07/2005		12	J.M. CHAUSSY	P. BOULCOURT	J.M. CHAUSSY	S. LOUBIERE
Indice	Date	Etat du document	Nbre de pages	Rédacteur	Vérificateur	Approbateur	Chef de Service

Mots clés	IRIS		
Nom du fichier	: DRSN-SIREN-Second Attempt for RERTR Export CSCT specif UMo monolithique.doc	Logiciel/modèle utilisé	: Word 2000/modèle SIREN c A.dot

Liste de diffusion		Diffusion contrôlée	Diffusion non contrôlée
DRSN/SIREN	S. Loubière, Ch Pech		+
DRSN/SIREN/LECSI	C Blandin, J-M Chaussy, P. Boulcourt		+
DRSN/SIREN/LASPI	S Naury		+
DSOE	P-M Lemoine, A Chabre		+
DEC/SESC/LIPA	S. Dubois		+

INDICE	DATE	NATURE DE L'ÉVOLUTION	PAGES CHAPITRES
A	22/07/2005	Initial edition	

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1. INTRODUCTION

This technical specification is related to the fabrication and inspection of two monolithic UMo fuel plates dedicated to the study of new MTR fuel.

The plates will be irradiated in OSIRIS reactor in the specific device called IRIS.
The fuel plates will contain 20 % enriched Uranium as monolithic U-Mo alloy.

2. APPLICABLE DOCUMENTS

2.1 Drawings

• UMo Fuel Plate as Drawing

- 9300-AH-29-00-012 : Plate Monolithic for Irradiation Test in IRIS Device

2.2 Documents

• Technical specifications as described in the present document

• Manufacturer manual

2.3 Precautions and general cleanness

The manufacturer will respect the state of the art and will watch to ensure all the required conditions of cleanness.

3. MATERIALS

3.1 Fuel core

The fuel core contains a foil of monolithic Uranium / 7-11 % Molybdenum

The U is provided with enrichment of ^{235}U 19.75% ± 0.2 %

3.2 Frame and cladding

The frame and the cladding are made of either Al 6061 or AlFeNi alloy, whose chemical compositions are as follows:

Alloy materials & Impurities

Sample	Inspection method	Inspection Criteria
1 / batch	Chemical analysis (Weight %)	Al6061: 0.40 ≤ Si ≤0.80 Fe ≤0.70 0.40 ≤ Cu ≤0.80 Mn ≤0.70 0.80 ≤ Mg ≤1.20 0.04 ≤ Cr ≤3.5 Zn ≤0.25 Ti ≤0.15 Other elements ≤0.15 Al = balance
1 / batch	Chemical analysis (Weight %)	AlFeNi: 0.8 ≤ Fe % ≤ 1,2 0.8 ≤ Ni % ≤ 1,2

		0.02 ≤ Ti% ≤ 0.08
		0.2 ≤ Mn % ≤ 0.6
		0.2 ≤ Cr % ≤ 0.5
		0.06 ≤ Zr % ≤ 0.14
		Si % ≤ 0.30
		0,8 ≤ Mg % ≤ 1,2
		Cu % ≤ 0.008
		Zn % ≤ 0.03
		B % ≤ 0.001
		Li % ≤ 0.001
		Cd % ≤ 0.001
		Others ≤ 0.03
		Sum other ≤ 0.5
		Al reminding

Any variations will be discussed with the reactor prior shipment

The manufacturer will give the room temperature mechanical properties of the aluminum cladding alloy (Al 6061 or AlFeNi) after friction stir welding

4. PLATE FABRICATION

4.1 General description

The fuel plates are made of a fissile core clad in an aluminum alloy.

The core of the fuel plate is made with a uranium molybdenum alloy as per section 4.2. The nominal dimensions of the U-Mo alloy fuel foil are 5.4W x 59.65L x 0.025T cm.

The U-Mo monolithic fuel is assembled in a sandwich using two aluminum plates with a recess for the foil machined into one plate.

The plate is then bonded by friction stir welding to obtain a leak tight cladding with good thermal contact between the fuel and the clad.

The plate is then surface finished, cut to final size, degreased and cleaned as required.

4.2 Fissile core fabrication

U-Mo alloy composition

Sample	Inspection method	Inspection Criteria
100 %	Weighing	7.0 ≤ Mo (wt %) ≤ 11.0 U % = Remaining

UMo Impurities

Sample	Inspection method	Inspection Criteria
1 / batch	Chemical analysis	The analysis certificate for UMo foil will be included in the fabrication document package.

Core Uranium Content

Sample	Inspection method	Inspection Criteria
100 %	Weighing (see section 6.2.1 for foil dimentions)	Total Uranium: [redacted] per plate Total ^{235}U: [redacted]er plate ^{235}U areal density 0.084 ± 0.008 gU/cm^2 per plate

UMo foils state

Sample	Inspection method	Inspection Criteria
100 %	Metalographic inspection	For information : at least grain size on one representative UMo foils (Transversal and longitudinal direction)
100%	Visual inspection	For information : Default Location if any (Holes,tears, cracks will be located)
100%	Thickness	Representative measurment (At least 5 area by means of micrometer). Thickness variation +/- 20 %

5. TRACEABILITY

During fabrication, the traceability of the products is ensured by issued quality documentation, and by using mark-up as required.

Each plate is assigned a serial number that will be clearly marked on the surface of the plate. The components of the fuel plate will be traceable back to the raw materials.

After friction stir welding, the plate reference is engraved or stamped on a non fissile area according to the applicable plate drawing (see section 2.1)

The plate reference shall be in conformance with the customer requirements, if any.

The engraving height shall be approximately 0,3 mm.

Each plate will be identified using specific tooling mark in its top side in order to identify easily the plate during handling (See drawing).

6. FUEL PLATE INSPECTION

6.1 Core ^{235}U content

Each fuel core not in conformance with criterion given in section 4.2 shall be rejected.

6.2 Fuel plate fissile area

6.2.1 Fissile area outer dimension

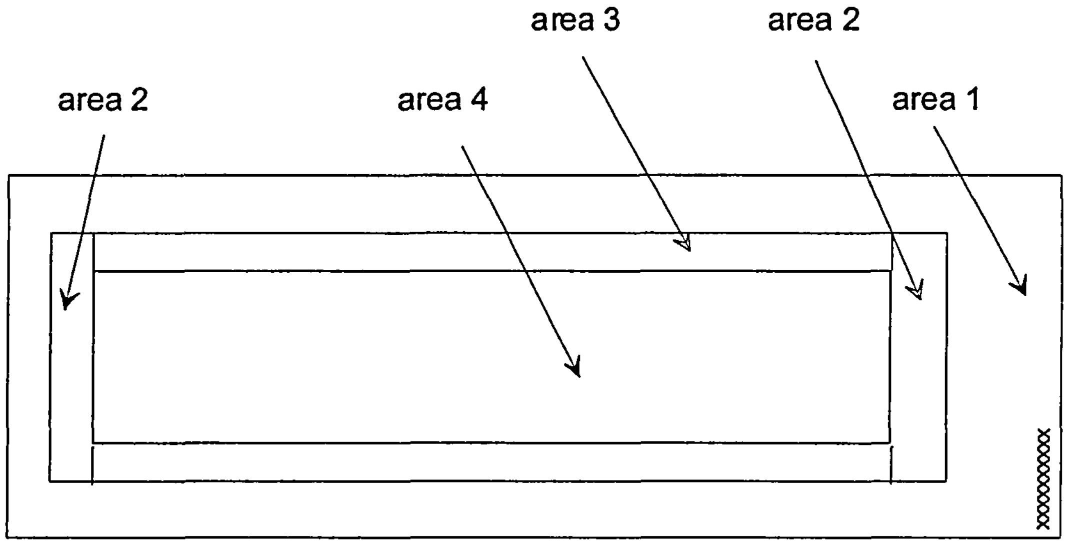
Fissile area dimensions

Sample	Inspection method	Inspection Criteria
100 %	Each plate shall be inspected using x-ray transmission radiography at 1x magnification so that both fissile and non fissile areas as well as the plate reference number is clearly visible on the film	The fissile areas outer dimension shall be conforming to dimensions of applicable drawing referred in section 2.1. Plates that present segregations type defects or lack of material type defects shall be confirmed through X-Ray homogeneity inspection (see section 6.2.2.) Some defects, such as lateral or longitudinal lack of material, outlying fissile areas, tears, cracks, and holes in foils may be accepted on an individual basis upon agreement with the reactor.
	U-Mo foil thickness (measured prior to assembly)	Foil thickness is: 0.025 ± 0.005 mm

6.2.2 *Homogeneity*

U distribution

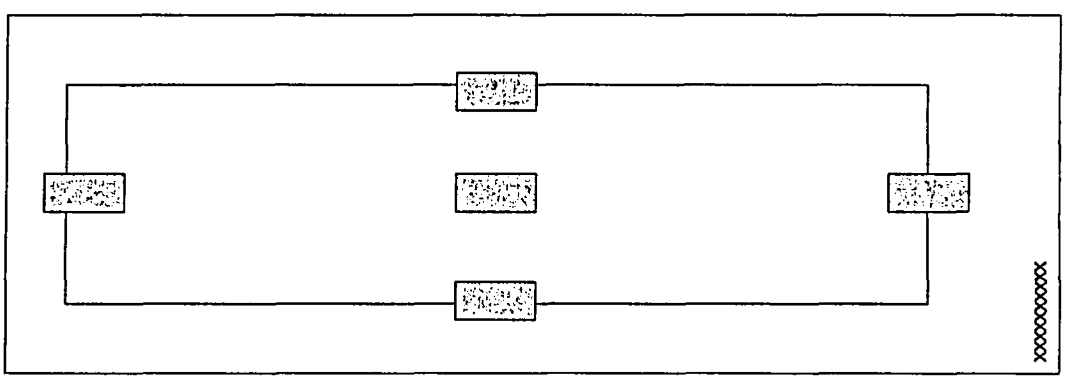
Sample	Inspection method	Inspection Criteria
100 % of the plates shall be inspected.	Uranium homogeneity will be verified by transmission x-ray radiography in reference to a thickness standard prepared from an applicable U-Mo alloy (see section 4.2).	**Area 1** Absence of Uranium (except deviations see 6.2.1). **Area 2 and 3** The U distribution per cm^2 is lower than the nominal value(section 4.2) + 20% whatever the position, measured over a 1 cm^2 area. **Area 4** The U distribution per cm^2 is within the nominal value ± 20 % whatever the position except (accepted defects as per section 6.2.1).


area 3
area 2
area 2
area 4
area 1
XXXXXXXXX

6.3 Fuel core and cladding thickness

Destructive test (cladding thickness)

Sample	Inspection method	Inspection Criteria
One sample per plate type	Five test specimens shall be taken from one selected fuel plate from a production batch (this plate can be depleted uranium of the same alloy composition). Sampling locations will be selected per the sketch below. Photographs of each test specimen magnified equal to or more than 20 times shall be taken Cladding and fuel core thickness shall be measured as follows: **Core center area** Minimum cladding thickness average of cladding thickness (10 measurements every 1 mm)	Nominal cladding thickness is: - 0.51 mm Minimium cladding thickness is: - 0.20 mm



Ultrasonic testing

Sample	Inspection method	Inspection Criteria
100 %	UT procedure Indications are recorded	For information on plate characteristics

6.4 Cohesion and fuel core/cladding bond

The manufacturer will propose a method to test the cohesion and fuel core/cladding bond of the monolithic fuel plate. This method will be discussed with CEA prior to shipment.

For information, the typical roll-bonded dispersion fuel plate blister test technique is as follows:

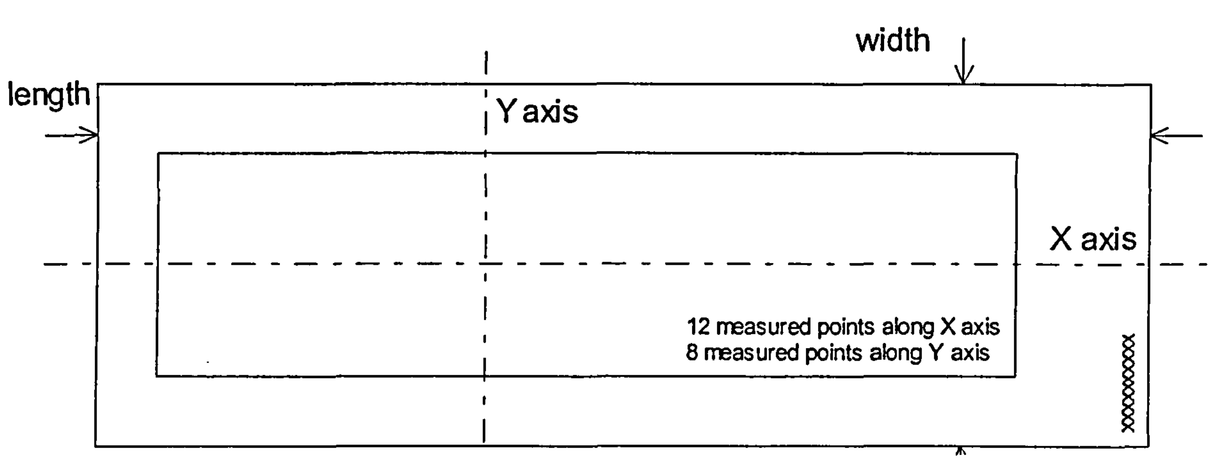
Sample	Inspection method	Inspection Criteria
100 %	Each plate shall be blister tested after annealing at a temperature of 425°C ± 25°C atmosphere for 30 min or more. Indications of blisters and the location of these indications shall be recorded The acceptability of blister indications are evaluated according to the listed inspection criteria.	On the fissile areas including a 2.5 mm wide periphery, blisters with area > 2 mm^2, shall be cause for rejection. (needs clarification of this zone) On the non fissile ares, the following blister indications are acceptable: d ≤ 5 mm & s ≤ 7mm^2 d > 5mm & s ≤15 mm^2 With d = shorter distance between the blister edge and the fissile area edge s = blister surface area

6.5 Fuel plate dimensions

The fuel plates shall have the dimensions & geometry defined in the applicable drawing: "Fuel Plate".

Dimensional inspection

Sample	Inspection method	Inspection Criteria
100 %	Measuring point as per sketch below.	Length: 641.9 + 0.8 / 0 mm Width: 73.3 + 0.05 / - 0.15 mm Thickness: 1.27 ± 0.08 mm Transverse bowing: ≤ 0.10 mm



6.6 Surface cleanness

The plates shall surface treated (degreasing, cleaning, rinsing, drying, etc.) and shall be carefully packed into a polyethylene sheath.

Any cleaning of the plate with a chlorinated product is prohibited.

Cleaned plates shall be manipulated using clean cotton or lint free gloves.

6.7 Surface finish - contamination

6.7.1 Surface finish

Visual inspection

Sample	Inspection method	Inspection Criteria
100 %	Visual test	The plate shall be sound with a chemicaly etched surface. **On the fissile area including a 2 mm wide periphery** Surface defects such as scratch, marks, grooves, pores and other surface flaws deeper than 110 µm shall be cause for rejection. **On the non fissile area** Defects more than 200 µm deep shall be cause for rejection.

6.7.2 Contamination

Surface contamination inspection

Sample	Inspection method	Inspection Criteria
100 %	Smear test or direct a counting	The surface contamination shall be ≤ 1.7 Bq/dm^2

6.8 Fabrication and inspection documents

The following documents shall be provided to the customer for products inspection and included in the EOMR :

- Uranium chemical and isotopic analysis reports
- Cladding material and U-Mo alloy chemical analysis reports
- ^{235}U and total U content in core
- General UT inspection report (for information only)
- Uranium homogeneity inspection report
- Core and cladding thickness inspection report
- As-fabricated cladding mechanical properties report
- Surface finish inspection report
- Surface contamination inspection report
- Dimensional inspection report
- Plate summary data sheet summarizing the inspection results (dimension...)

ATTACHMENT 3 TO THE BATTELLE ENERGY ALLIANCE REQUEST TO EXPORT FUEL PLATES TO THE FRENCH OSIRIS REACTOR IN SUPPORT OF DEPARTMENT OF ENERGY WORK

AGREEMENT

between

THE DEPARTMENT OF ENERGY
OF THE UNITED STATES OF AMERICA

and

THE COMMISSARIAT A L'ENERGIE ATOMIQUE
OF FRANCE

FOR COOPERATION IN

ADVANCED NUCLEAR REACTOR SCIENCE AND TECHNOLOGY

Preamble

The Department of Energy of the United States of America (DOE) and the Commissariat à l'Energie Atomique of France (CEA), herein referred to as the Parties,

Sharing a mutual interest in fostering advanced nuclear engineering and pursuing scientific research and development (R&D) in the nuclear field,

Believing that cooperation based on equitable sharing of their respective R&D data, technology and experience in the nuclear domain, focusing on the field of advanced nuclear reactors would be of mutual benefit, and

Recognizing the contribution such R&D in the field of nuclear energy applications can make to further the safe and economic application of nuclear energy,

It is agreed as follows :

Article 1: OBJECTIVE

1.1 The objective of this Agreement is to establish the basis for cooperation between the Parties in the field of advanced nuclear reactor engineering and scientific R&D.

1.2 Cooperation between the Parties shall be on the basis of mutual benefit, equality, and reciprocity.

Article 2: AREAS OF COOPERATION

2.1 DOE and CEA shall cooperate in joint planning to utilize their existing R&D capabilities in the field of advanced reactor engineering and scientific research, including such test reactors as the Fast Flux Test Facility, if the facility is restarted for civilian use, and the future Jules Horowitz reactor, if a decision is taken with regard to its construction.

2.2 Areas of advanced nuclear reactor engineering and scientific R&D cooperation may include:

- -- Advanced reactor materials irradiation development and testing
- -- Advanced reactor fuel development for next generation reactors
- -- Medical and industrial applications of isotopes and the related research
- -- R&D on transmutation as applied to nuclear waste disposition (not covered in the Agreement between the United States Department of Energy and the French Commissariat à l'Energie Atomique in the Field of Radioactive Waste Management of October 8, 1995)

-- Other areas of mutual R&D interest.

Article 3: FORMS OF COOPERATION

Cooperation under this Agreement may include the following forms:

3.1 Exchange on a current basis of scientific and engineering information and results and methods of research and development.

3.2 Organization of and participation in seminars or other meetings on specific agreed topics in the areas listed in Article 2.

3.3 Short visits by specialist teams or individuals to the facilities of the other Party, subject to the prior written agreement of that Party.

3.4 Assignment of the staff of one Party, its contractors or subsidiaries to the facilities of the other Party, its contractors or subsidiaries for participation in agreed research, development, design, analysis or other experimental activities.

3.5 Exchange of materials and equipment for testing.

3.6 Exchange of technology and engineering drawings (including specifications of components and of industrial plants) as appropriate to the areas of cooperation and as agreed to by the Parties.

3.7 Joint projects in which the Parties agree to share the work and/or costs.

3.8 Such other specific forms of cooperation as the Parties may agree.

Article 4: IMPLEMENTING ARRANGEMENTS

When the Parties agree to undertake a form of cooperation set forth Article 3, they shall conclude an Implementing Arrangement, which shall be subject to this Agreement. Each Implementing Arrangement shall include detailed provisions for carrying out the activity and shall cover such matters as technical scope, total costs, cost sharing between the Parties, project schedule, management of the cooperation, exchange of equipment, and information disclosure specific to the particular project. Activities under Implementing Arrangements may involve, as appropriate, associated firms or laboratories of the Parties or their contractors or subsidiaries.

Article 5: MANAGEMENT

5.1 The Parties shall establish a DOE/CEA advanced nuclear reactor engineering and scientific R&D Steering Committee under which expert groups will be established in areas such a those listed in Article 2 of this Agreement.

5.2 To supervise the execution of this Agreement, each Party shall name a Principal Coordinator. The Principal Coordinators, who shall lead the Steering Committee noted in Article 5.1, shall meet each year, alternately in the United States and in France, or at such other times and places as agreed.

5.3 At their meetings, the Principal Coordinators shall evaluate the status of cooperation under this Agreement. This evaluation may include a review of the past year's activities and accomplishments under this Agreement, a review of the activities planned for the coming year within each of the various areas of cooperation listed in Article 2, an assessment of the balances of exchanges under this Agreement within each of the areas of cooperation listed in Article 2, and a consideration of measures required to correct any imbalances. In addition, the Principal Coordinators shall consider and act on any major new proposals for cooperation.

5.4 Day-to-day management of the cooperation under this Agreement shall be carried out by Technical Coordinators designated by the Principal Coordinators. The Technical Coordinators shall agree on specific details of cooperation in the technical areas listed in Article 2 within policy guidelines established by the Principal Coordinators. The Technical Coordinators shall be responsible for working contacts between the Parties in their respective areas of cooperation.

Article 6: INTELLECTUAL PROPERTY RIGHTS

The treatment of intellectual property created or furnished in the course of cooperative activities under this Agreement is provided for in Annex I, which is an integral part of this Agreement and shall apply to all activities conducted under the auspices of this Agreement.

Article 7: DISCLAIMER

Information transmitted by one Party to the other Party under this Agreement shall be accurate to the best knowledge and belief of the transmitting Party, but the transmitting Party does not warrant the suitability of the information transmitted for any particular use or application by the receiving Party or by any third party.

Article 8: LIABILITIES

A Party sending information, materials, or supplies to the other Party under this Agreement shall not be liable for damages of any nature, either direct or indirect, to property or personnel of the Party receiving the information, material, or supplies or to any third party resulting from the use by the Party receiving such information.

Article 9: LEGAL PROVISIONS

Each Party's activities under this Agreement shall be in accordance with its national laws and regulations. All questions related to the Agreement arising during its term shall be settled by the Parties by mutual agreement.

Article 10: SECURITY OBLIGATIONS

If either Party believes that information or equipment proposed to be provided or exchanged under this Agreement requires protection in the interest of that Party's national defense or foreign relations, that Party shall so notify the other Party, and the Parties shall promptly consult to identify and agree upon appropriate measures for the protection of the information or equipment.

Article 11: FINANCIAL OBLIGATIONS

Except when otherwise specifically agreed in writing, all costs resulting from cooperation under this Agreement shall be borne by the Party that incurs them. It is understood that the responsibilities of each Party to carry out its obligations under this Agreement are subject to the availability of personnel and appropriated funds.

Article 12: DURATION, AMENDMENT, AND TERMINATION

12.1 This Agreement shall enter into force upon the latter date of signature and shall remain in force for five (5) years.

12.2 This Agreement may be amended or extended by written agreement of the Parties.

12.3 This Agreement may be terminated at any time at the discretion of either Party, upon six (6) months advance notification in writing by the Party seeking to terminate the Agreement. Such termination shall be without prejudice to the rights that may have accrued under this Agreement to either Party up to the date of such termination.

12.4 Joint efforts and experiments not completed at the expiration or termination of this Agreement may, on agreement of the Parties, be continued until their completion under the terms of this Agreement.

12.5 The rights and obligations set forth in Article 8 shall survive termination of this agreement.

Done, in duplicate, at Vienna, this 18th day of September 2000, in the English and French languages, each text being equally authentic.

FOR THE DEPARTMENT OF ENERGY OF THE UNITED STATES OF AMERICA

Bill Richardson

FOR THE COMMISSARIAT A L'ENERGIE ATOMIQUE OF FRANCE

ANNEX I: INTELLECTUAL PROPERTY

PREAMBLE

PURSUANT TO ARTICLE 6 OF THIS AGREEMENT:

The Parties shall ensure adequate and effective protection of intellectual property created or furnished under this Agreement and relevant Implementing Arrangements. The Parties agree to notify one another in a timely fashion of any inventions or copyrighted works arising under this Agreement and to seek protection for such intellectual property in a timely fashion. Rights to such intellectual property shall be allocated as provided in this Annex.

I SCOPE

I-A. This Annex is applicable to all cooperative activities undertaken by the Parties or by the relevant entities (hereafter "cooperative entities") pursuant to this Agreement, except as otherwise specifically agreed by the Parties or their cooperative entities.

I-B. For purposes of this Agreement, "intellectual property" shall have the meaning found in Article 2 of the convention establishing the World Intellectual Property Organization, done at Stockholm, July 14, 1967.

I-C. This Annex addresses the allocation of rights, interests, and royalties between the Parties. Each Party shall ensure that the other Party or cooperative entities can obtain the rights to intellectual property allocated in accordance with the Annex. The allocation between a Party and participants on behalf of this Party in the cooperative activities, which shall be determined by the Party's laws and practices, shall not be altered or prejudiced by application of this Annex.

I-D. Disputes concerning intellectual property arising under this Agreement should be resolved through discussions between the concerned participating institutions or, if necessary, the Parties or their designees. Upon mutual agreement of the Parties, a dispute shall be submitted to an arbitral tribunal for binding arbitration in accordance with the applicable rules of international law. Unless the Parties or their designees agree otherwise in writing, the arbitration rules of UNCITRAL shall govern.

I-E. Termination or expiration of this Agreement shall not affect the rights or obligations under this Annex.

II ALLOCATION OF RIGHTS

II-A. Each Party, subject to the restrictions of Article III of this Annex, shall be entitled to a nonexclusive, irrevocable, royalty-free license in all countries to translate, reproduce, and publicly distribute scientific and technical journal articles and publicly available reports directly arising under this Agreement. All publicly distributed copies of a copyrighted work prepared under this provision shall indicate the names of the authors of the work unless an author explicitly declines to be named. Each Party or its cooperative entities shall have the right to review a translation prior to public distribution.

II-B. Rights to all forms of intellectual property, other than those rights described in section II-A above, shall be allocated as follows:

II-B-1. Visiting researchers, for example, scientists visiting primarily in furtherance of their education, shall receive intellectual property rights under the policies of the host institution unless a specific agreement is or has been signed between the host and forwarding institutions. In addition, each visiting researcher named as an inventor shall be entitled to treatment as a national of the host country with regard to awards, bonuses, benefits, or any other rewards in accordance with the policies of the host institution.

II-B-2(A). For intellectual property created during joint research, the Parties or their cooperative entities shall jointly develop a technology management plan either prior to the start of their cooperation, for example in research areas likely to lead rapidly to industrial applications, or within a reasonable time from the time a Party becomes aware of the creation of intellectual property. The technology management plan shall consider the relative contributions of the Parties and their cooperative entities, the benefits of exclusive or nonexclusive licensing by territory or for field of use, requirements imposed by the Parties' domestic laws, and other factors deemed appropriate. If needed, the technology management plan shall be jointly modified or completed in a timely fashion subject to the approval of both Parties or their cooperative entities.

II-B-2(B). If the Parties or their cooperative entities cannot reach agreement on a joint technology management plan within a reasonable time not to exceed six months from the time a Party becomes aware of the creation of the intellectual property in question, each Party may designate one co-exclusive licensee to have world-wide rights to said intellectual property. Each Party shall notify the other two months prior to making a designation under this paragraph. When both Parties (or their licensees) exploit the intellectual property in a country, they shall share equally the reasonable cost of intellectual property protection in that country.

II-B-2(C). A specific program of research will be regarded as joint research for purposes of allocating rights to intellectual property only when it is designated as such in the relevant Implementing Arrangement, otherwise the allocation of rights to intellectual

property will be in accordance with paragraph II-B-1.

II-B-2(D). In the event that either Party believes that a particular joint research project under this agreement will lead, or has led, to the creation or furnishing of intellectual property of a type not protected by the applicable laws of one of the Parties, the Parties shall immediately hold discussions to determine the allocation of the rights to the said intellectual property; the joint activities in question will be suspended during the discussions unless otherwise agreed by the Parties thereto. If no agreement can be reached within a three-month period from the date of the request for discussions, the Parties shall cease the cooperation in the project in question. Notwithstanding paragraphs II-B-2(A) and (B), rights to any intellectual property that have been created will be resolved in accordance with the provision of Article I-D.

III BUSINESS-CONFIDENTIAL INFORMATION

In the event that information identified in a timely fashion as business-confidential is furnished or created under the Agreement, each Party and its cooperative entities shall protect such information in accordance with applicable laws, regulations, and administrative practice. Information may be identified as business confidential information if a person having the information may derive an economic benefit from it or may obtain a competitive advantage over those who do not have it, the information is not generally known or publicly available from other sources, and the owner has not previously made the information available without imposing in a timely manner an obligation to keep it confidential. Without prior written consent, neither of the Parties shall disclose any business-confidential information provided by the other Party except to appropriate employees and government personnel. If expressly agreed between the Parties, business-confidential information may be disclosed to prime and subcontractors. Such disclosure shall be for use only within the scope of their contracts with the Parties relating to cooperation under the Agreement. The Parties shall impose, or shall have imposed, an obligation on those receiving such information to keep it confidential. If one of the Parties becomes aware that, under its laws or regulations, it will be, or may reasonably expected to become, unable to meet the nondisclosure provision, it shall immediately inform the other Party. The Parties shall thereafter consult to define an appropriate course of action.