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High Density Fuel Development for Research Reactors

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ABSTRACT

An international effort to develop, qualify, and license high and very high density fuels has been underway for several years within the framework of multi-national RERTR programs. The current development status is the result of significant contributions from many laboratories, specifically CNEA in Argentina, AECL in Canada, CEA in France, TUM in Germany, KAERI in Korea, VNIIM, RDIPE, IPPE, NCCP and RIARR in Russia, INL, ANL and Y-12 in USA.

These programs are mainly engaged with UMo dispersion fuels with densities from 6 to 8 gU/cm^3 (high density fuel) and UMo monolithic fuel with density as high as 16 gU/cm^3 (very high density fuel).

This paper, mainly focused on the French & US programs, gives the status of high density UMo fuel development and perspectives on their qualification.

1. INTRODUCTION

The limitations discovered under severe (high power) operating conditions during tests on first generation high density fuel, postponed initial plans and schedules for qualification tests and the subsequent licensing program [1], [2], [3].

These failures, firstly observed in the French FUTURE experiment, and confirmed in RERTR-4 miniplates, IRIS-2 full sized plates and Russian IVV-2M tubes, are clearly attributed to the excessive formation of the interaction product and the inability of this compound to retain fission gas in the form of stable bubbles.

Thus, it was clear that the only way to improve UMo dispersion fuel behavior for medium to high power operating conditions was to drastically reduce the formation of the interaction compound.

To solve the problem, full scientific and experimental programs have been launched to understand the mechanism of UMo/Al interaction, develop and test in-pile solutions to avoid an inappropriate behavior under irradiation.

The article starts by recalling the encouraging results of early experiments (US RERTR-1 through -5 and French IRIS-1), then exposes the limitations demonstrated by French experiments FUTURE, IRIS-2 and Russian IVV-2M fuel assembly irradiation, before coming to the development programs status of the high density UMo fuels, by distinguishing the dispersed case and of monolithic, and finish by the prospects for qualification of the one and the other solution.

This paper is mainly focused on French and US experiments and on in-pile experimental programs. These in-pile irradiation programs are specifically designed to evaluate and optimize the irradiation behavior of UMo dispersion fuels (high density fuel) as well as UMo monolithic (very high density fuel). This experimental work will culminate with irradiation tests on full size plates that will demonstrate the integrated package that includes both manufacturing and irradiation aspects, as has been previously demonstrated by the French UMo Group.

2. DISPERSION FUELS: FIRST GENERATION RESULTS (1997-2003)

2.1. Early Tests Results (1997-2001)

2.1.1. US experiments (RERTR-1 to 5)

RERTR-1 & 2 – The objective of the two first RERTR experiments (RETR-1 &2), performed in ATR at low irradiation temperature (~65°C) to relatively high burnup (71%), was to test various binary or ternary U-Mo alloys regarding their γ -stability and microstructure. Neutron flux, surface heat flux, and fuel temperature were approximately 1.3 x 10^{14} n/cm^2/s, 55 W/cm^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 burnup between 39 and 45% and between 65 and 71%, respectively. Post irradiation examination (PIE), indicated excellent irradiation behavior for U-Mo alloys with greater than 6 wt % Mo content and for the ternary U-Mo-Pt and U-Mo-Ru alloys.

RERTR-3 – The RERTR-1 and -2 irradiation test capsules were followed by the RERTR-3 irradiation test at higher temperature to moderate burnup. The experimental test matrix was composed of U-Mo 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 EFPD 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. On PIE, 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. 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.

RERTR-4 & 5 – As in previous irradiation experiments, the miniplates irradiated in RERTR-4 and 5 contained either atomized fuel particles, supplied by Korea Atomic Energy Research Institute (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. Irradiation began on August 19, 2001, and RERTR-4 and RERTR-5 were irradiated for 204 EFPD and 116 EFPD respectively. The RERTR-5 experiment was removed from the reactor at a peak burnup of ~50% ^{235}U, whereas the RERTR-4 test 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 post irradiation data. Post irradiation 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.

2.1.2. French experiment (IRIS-1)

IRIS-1 – Three Materials Test Reactor (MTR) fuel plates loaded to ~8 g U/cm^3, with ground powder in pure Al matrix, were 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 125-145 W/cm^2 to a peak burnup of 67.5% (4.6 10^{21} fissions/cm^3 UMo), and a peak

cladding temperature of 72°C. The maximum fuel plate thickness increase was 77 µm out of 1.3 mm (Fig. 1a)

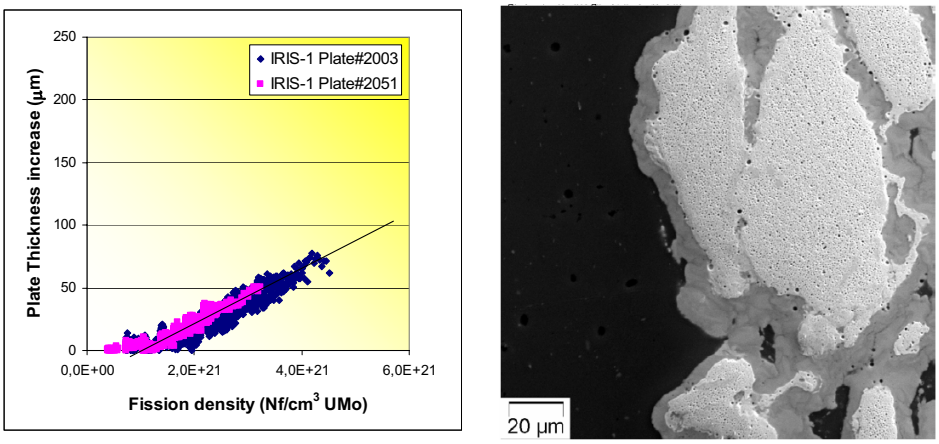

Fig.1a & 1b: IRIS-1 Results (Plates thickness increase vs fission density ; Representative micrograph)

The fuel/aluminum reaction layer composition was found in the range of $(U\text{-}Mo)Al_6$ to $(U\text{-}Mo)Al_8$, and fission products accumulation was detected at the interface between the reaction product and the fuel/aluminum interface. A representative micrograph is shown in Fig.1b showing interaction layer between Al clad or matrix, in black, and UMo ground particles, with uniform density of fission gas bubbles inside.

2.1.3. Status at the end of this early stage

At the end of this early stage, the conclusions on particles swelling and particles / matrix interaction, were as follows ([4] to [8]):

- U-Mo alloys with at least 6 wt.% Mo exhibited low swelling and stable irradiation behavior, with the presence of small uniformly distributed fission gas bubbles.
- 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.
- Lowering of Mo content results in somewhat higher rates of interdiffusion and fission gas swelling.
- 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 been previously identified.
- Mg-matrix fuel specimens showed no interaction between the fuel particles and the Mg-matrix and behaved well under RERTR-3 irradiation test conditions.

2.2. Performances limitations (2002-2003)

2.2.1. FUTURE experiment

The French FUTURE experiment was irradiated in the Belgian Reactor 2 (BR2 Mol Belgium) during the second half of 2002. The fuel plates, atomized powder in pure Al matrix, 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. After 40 EFPD and 29% ^{235}U burnup (fission density 1.25 x 10^{21} f/cm^3), a plate thickness increase of 13% was noted [9]. 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 previous PIE. In the case of the FUTURE experiment, the fuel plate delaminated through the center of the fuel meat (Fig. 2a), with voids at the interface of the (UMo)Al_x interaction phase with matrix and ligaments bridging the void, as shown in Fig. 2b. There were also indications of concentration of xenon gas at the boundary between the remaining matrix aluminum and the (UMo)Al_x layer.

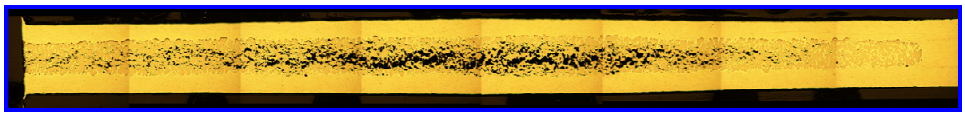

Fig.2a: FUTURE metallography in the pillowed region (plate transverse section)

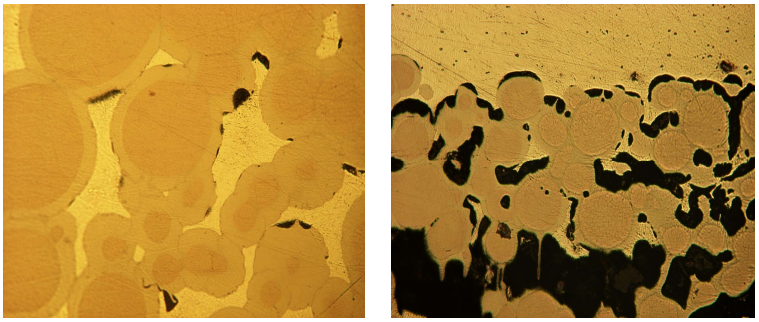

Fig.2b: FUTURE metallographies showing (UMo)Al_x interaction product around the UMo particles, lenticular shape porosities at the interfaces with Al matrix, and final meat decohesion

2.2.2. 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 resulted in a peak beginning of life cladding temperature of 93°C. A plate thickness increase of more than 250 µm was observed after 30 EFPD for one plate, which was withdrawn from the experiment (Fig. 3a). The experiment was terminated after 58 EFPD due to high swelling of the remaining plates [10].

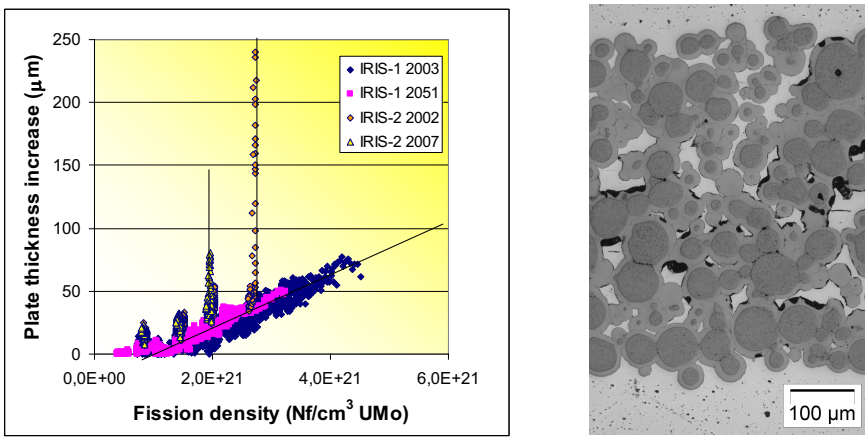

Fig.3a & 3b: IRIS-2 results (Plate thickness increase results with comparison with IRIS-1 results & Micrograph of fuel meat with porosities at the interface between Al matrix and UMo particles)

PIE results revealed the same failure mechanism as found in the FUTURE test 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, but separation between the reaction layer and the aluminum matrix were noted (Fig. 3b).

2.2.3. Russian results on UMo fuel assembly irradiated in IVV-2M reactor

Tubular U-Mo dispersion fuel undergoing irradiation at the IVV-2M reactor at the Institute of Reactor Materials (Zarechny) failed at approximately 60% burnup [11]. 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 Fig.4.

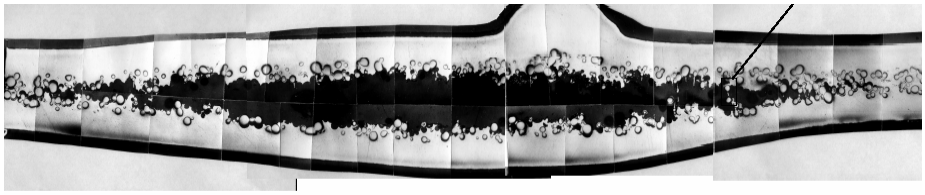

Fig.4: Cross section of failed U-Mo fuel tube from testing in the IVV-2M reactor.

Neutron diffraction studies were performed on samples taken from the fuel elements and did not show the presence of any crystalline UAl_x phases until the samples were annealed at temperatures greater than 450°C [12].

More recently, TEM examination was performed on samples taken from FUTURE experiment, proving the amorphous character of the interaction product [13].

2.2.4. Status at the end of 2003

It became obvious at the end of 2003 that U-Mo/Al dispersion fuels would not perform acceptably under the operating conditions required for high-powered research reactors. It was 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 U-Mo fuel particles (or foils) do not behave well under irradiation. Given these observations, there are three primary 'fixes' available that allow for retention of a conventional aluminum clad fuel design.

These are:

- Chemical modifications to the Al 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,

- Introduction of diffusion barriers between matrix and U-Mo particles to reduce the formation of interaction product,

- 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 first irradiation test results [6]

3. DISPERSION FUELS: RECENT DEVELOPMENTS & PERSPECTIVES

3.1. Improved solutions

3.1.1. Modified matrix dispersion fuels

IRIS-3 – The IRIS-3 test irradiation 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. The plates with 0.3 wt % silicon (Plates 8011 & 8013) showed excessive swelling after five irradiation cycles (Fig 5a) and were removed from the experiment at 2.9 x 10^{21} f/cm^3 (40 at% ^{235}U), while the plate 8021, operating at the highest power with a 2.1wt % matrix silicon content, showed good in-reactor performance to 4.1 x 10^{21} f/cm^3 (60 at% ^{235}U) [14], with a maximum thickness increase of 90 µm (Fig. 5b).

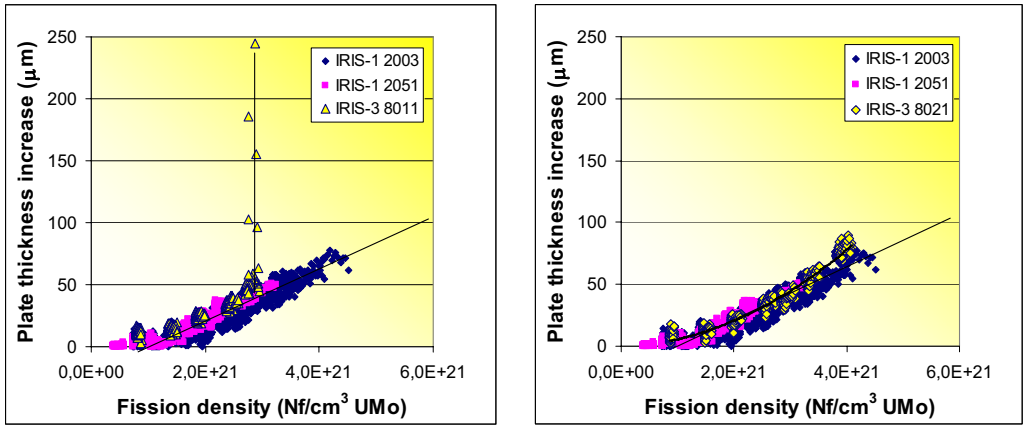

Fig.5a & 5b: IRIS-3 Plates thickness increase (Plate 8011 with 0.3%Si in Al matrix ; Plate 8021 with 2.1%Si) compared with IRIS-1 results

RERTR-6 & RERTR-7A – These two U.S. irradiation tests were designed to provide the initial data required to evaluate the irradiation performance of the modified matrix dispersion fuels. The RERTR-6 experiment was a mini-plate test at moderate power, temperature, and burnup conducted over three long cycles (136 EFPD) in the Advanced Test Reactor (ATR). The RERTR-7A experiment was a follow-up to this experiment and subjected mini-plates to high power, moderate temperature, and high burnup over two ATR cycles (90 EFPD). Several second generation dispersion fuels were tested in both experiments that incorporated modified matrix materials, including Al-6061 (1% Si), Al-0.2 wt% Si alloy, Al-2 wt% Si alloy, and Al-4043 (5% Si). Both tests showed that the incorporation of at least 2% silicon into the fuel meat matrix (Fig. 6a & b) substantially reduced the amount of interaction and showed that the stability of the modified layer apparently exceeds that of the fuel itself [15].

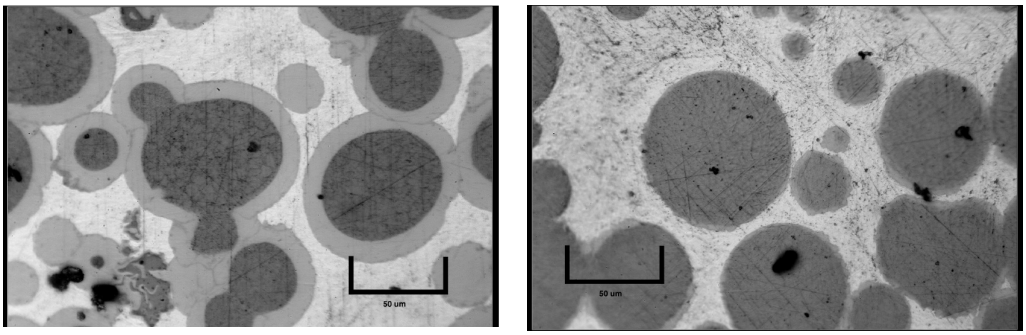

Fig.6a & 6b: Impact of silicon on the interaction layer thickness, (a) shows particles from a plate

(RERTR-6 – C5) with Al-0.2 wt% Si and (b) shows particles from a plate (RERT-6 – C3) with Al-2.0 wt% Si. Both fuel plates were irradiated to approximately 50% burnup.

3.1.2. Ternary fuel alloys

RERTR-7B & RERTR-8 – These two experiments emphasized tests on mini-plates with modified dispersion fuel alloys. Out-of-pile diffusion tests and computational studies suggested that the addition of zirconium or titanium to the fuel alloy could work with the silicon in the matrix to further inhibit interaction layer growth. Mini-plates containing ternary fuel alloys (U-7Mo-2Zr and U-7Mo-1Ti) were fabricated and tested with an Al-4043 matrix [16]. Post irradiation examination of the RERTR-8 experiment is currently underway.

3.2. In-Progress Tests

RERTR-9A & RERTR-9B – The primary objective of these tests was to confirm that the performance of dispersion fuels with silicon enhance matrix material extends from 6 g U/cc to higher loadings. Dispersion mini-plates were fabricated at 8 g U/cc with matrix materials consisting of Al-4043, Al-2 Si alloy, and Al+2 Si mixture for this test. Additional mini-plates were fabricated for experiments to test monolithic mini-plates with a modified interface (which will be discussed in a later section). The RERTR-9A experiment irradiation was completed in December 2007 and the RERTR-9B experiment irradiation will be completed in January 2008. PIE should begin on both in late January.

3.3. Perspectives

Modification to the fuel matrix and particle chemistry used in first generation dispersion fuels was proposed to change the irradiation performance of the $(U,Mo)Al_x$ interaction product. It has been demonstrated in recent irradiation tests that the addition of silicon to the matrix material leads to a dramatic reduction in growth rate and an increase in irradiation stability of the interaction layer [17, 18]. The performance of these second generation fuel meats have been promising enough that qualification of this type of fuel appears viable. Efforts are currently focused on selecting the optimum matrix material and developing high uranium density fabrication techniques for full-size plates. Matrix material candidates include high silicon commercial aluminum alloys (e.g. Al-4043), silicon aluminum binary alloys, and binary mixtures of aluminum and silicon (both with silicon weight fractions between 2 and 10%). Based on observations made in out-of-pile diffusion tests, the use of a ternary fuel alloy is also being examined to determine if it will further reduce the rate of interaction layer growth or allow a decrease in the amount of silicon required in the matrix.

4. MONOLITHIC DEVELOPMENT

4.1. Objectives

The use of a monolithic fuel meat was proposed as alternative to the dispersion fuel form in order to achieve even higher uranium densities (up to 16.5 g U/cc) and because it was believed the reduced interfacial area between aluminum and fuel would lessen the impact of the interaction product. Recent irradiation tests showed generally acceptable behavior of the fuel design and minimal interaction layer growth between the fuel and cladding during irradiation [19]. However, it has been observed that porosity similar to that seen in first generation dispersion fuels may still form in the thin interaction layer between the fuel and cladding. This porosity is likely to degrade the strength of the fuel/clad interfacial bond. Techniques to either incorporate silicon into the interface or to apply a diffusion barrier are currently being developed. Variants of the monolithic fuel type include a zircaloy clad plate version

proposed and under development in Argentina, aluminum clad versions in France and the U.S., and pin type versions in Canada, South Korea, and Russia.

4.2. Fabrication

Monolithic fuel requires significant development to establish techniques for foil and plate fabrication. Multiple paths to production of robust monolithic cores have been recently demonstrated. For example, two techniques for foil fabrication have been developed in the U.S. including 1) immersing as-cast ingots into a salt bath, hot rolling to an intermediate thickness, and cold rolling to final thickness [20] and 2) by cladding the ingots with stainless steel, furnace heating, rolling, and de-cladding the foil [21].

Plate fabrication (fuel to clad bonding) development has focused on three primary approaches including friction bonding (FB), hot isostatic pressing (HIP), and co-rolling the U-Mo ingot with zircaloy clad [21, 22]. Substantial progress has been made in the last year with the FB process. Modifications to the tool head design have greatly improved the tool life, fuel plate surface finish, and bond quality [23]. Ongoing experiments are planned to parametrically evaluate the role of temperature and pressure on the mechanical properties of the fuel/clad interfacial bond. Similarly, fuel plate fabrication using the HIP process was demonstrated at the mini-plate scale and is being evaluated in the RERTR-8, -9A and –9B irradiation tests. Studies are being performed to optimize the temperature, pressure, and hold time parameters to minimize fuel/clad chemical interaction and maximize fuel/clad and clad/clad bonding. Both techniques are being used to fabricate full-size plates for irradiation tests to be conducted in the early 2008.

A novel approach was also proposed and demonstrated by the CNEA to fabricate zircaloy clad fuel plates. The U-Mo ingot was inserted in a zircaloy frame and hot rolled at roughly 650°C to thickness in a fashion similar to aluminum dispersion fuels [22].

Recent irradiation test results have shown that correcting the instability observed in the fuel/clad interaction layer will likely require making changes to the interface. Techniques to incorporate silicon into the interface are being explored to help stabilize the irradiation performance of the interaction layer, as was proven effective in dispersion fuel. Mini-plates have been fabricated in the U.S. by HIP with a thin interlayer of Al-4043 (~5% Si) between the fuel and cladding and were recently inserted in the RERTR-9A experiment. The application of a thin Al-Si layer into the foil pocket by thermal spray is currently under evaluation for use with both FB and HIP. As an alternative, the inclusion of a zirconium diffusion barrier is being explored, partially based on the early results from recently irradiated Argentinean zircaloy clad fuel plates. The application of the barrier to fuel foils by co-rolling and HIP are currently being evaluated by the U.S. RERTR program. Additional testing is also being performed to determine how these interlayers affect the fuel plate fabrication by either FB and HIP processes.

4.3. Recently Completed Tests

RERTR-6 & RERTR-7A – These two experiments included the first tests of the monolithic fuel form. Mini-plates were fabricated with fuel alloy compositions of U-7Mo, U-10Mo, and U-12Mo using FB and Transient Liquid Phase Bonding (TLPB). The tests showed that the fuel phase was stable and that the interaction layer thickness remained thin [19]. It was also observed that fine gas bubbles had a tendency to form within the interaction layer at high burnup (Fig. 7). These bubbles apparently weaken the interfacial bond strength and are believed to be the cause of debonding during metallographic examination and premature blistering during post irradiation annealing tests. However, the interaction layer in plates fabricated using the TLPB process demonstrated much more stable behavior. While the

clad/clad bond obtained with the TLPB process was poor and lead to the breach of one plate in the RERTR-7A experiment, the silicon used in the fabrication process appeared to have stabilized the interaction layer in a fashion similar to that observed in the second generation dispersion fuel meats (Fig. 8a).

Two additional zircaloy clad monolithic mini-plates were supplied by the CNEA for the RERTR-7A experiment. These plates were irradiated under moderate power and temperature to moderate burnup (~35%) and exhibited excellent irradiation performance (Fig. 8b).

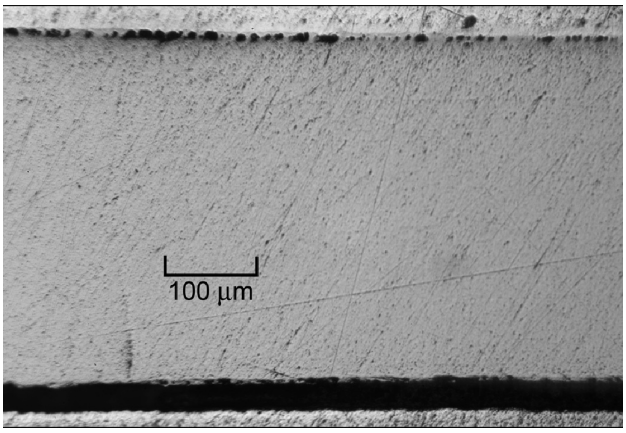

Fig.7: Example of pore formation in the interaction layer at the U-Mo/Al-6061 interface of a monolithic fuel plate (RERTR-7A-L1F140). (The delamination shown on the bottom occurred during sectioning of the plate.)

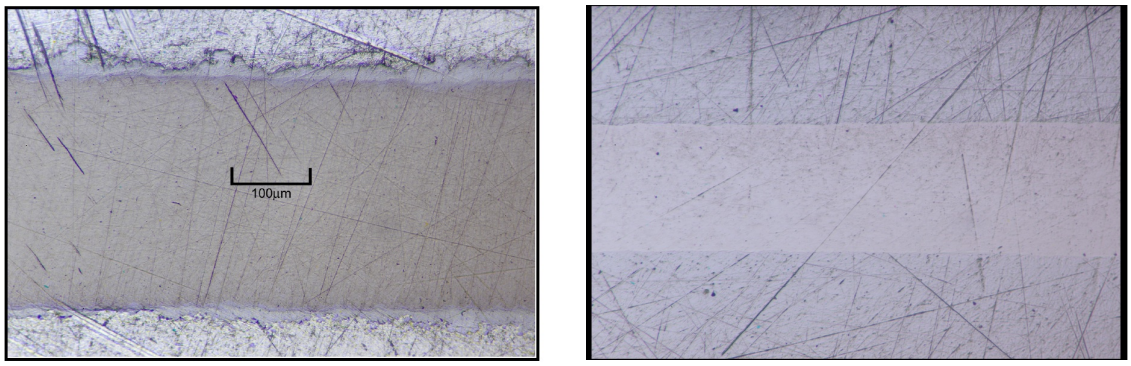

Fig.8a & 8b: Monolithic fuel plates with alternate interface designs after irradiation to ~80% and ~50% LEU equivalent burnup, respectively; (a) shows the fuel/clad interface in a fuel plate fabricated by TLPB with a high silicon interaction layer and (b) a zircaloy clad monolithic fuel plate after irradiation.

RERTR-8 – The first monolithic fuel plates fabricated by HIP were irradiated in this experiment. The original experiment plan was to fabricate a series of monolithic fuel plates by HIP with molybdenum contents between 7 and 12 wt%. However, excessive interaction layer growth occurred in plates with U-7Mo and U-8Mo due to the relatively long period of exposure to high temperature (90 minutes at 580C) that is required for HIP fabrication. FB subjects the plates to a very short period of thermal exposure and was thus used as alternative to fabricate these two plates. Although PIE of this experiment has not been completed, there appears to be very little difference between the performance of FB and HIP plates.

4.4. In-Progress Tests

RERTR-9A & RERTR-9B –In order to improve the 'end-of-life' interfacial bond strength of monolithic fuel plates, fabrication techniques were developed to enable testing of mini-plates with a fuel/clad interlayer. The interlayer compositions were based on the positive performance indicators from the previous tests including the addition of silicon to the interaction product and the stability of a zirconium-UMo interface. For the RERTR-9A experiment mini-plates were fabricated by HIP with a thin layer of high silicon aluminum alloy (Al-4043) inserted between the fuel and cladding. The interface layer is expected to behave similarly to the TLPB plates from the RERTR-7A experiment but with improved clad/clad bonding. Additional mini-plates were fabricated with an improved FB tool that drastically improved the as-fabricated bond strength for testing in the RERTR-9A test.

The RERTR-9B experiment was designed to test similar monolithic mini-plates with alternate interface layers. Mini-plates were fabricated by HIP (and subsequently by FB as well) with a thin zirconium diffusion barrier between the fuel and clad. The diffusion barrier has been applied by both co-rolling the fuel foil with zirconium and by HIPing the layer directly to the foil. Mini-plates were also fabricated by FB and HIP with a thin layer of Al-Si alloy applied by plasma spray techniques to the fuel pocket prior to insertion of the fuel foil such that the foil was encapsulated in a high silicon environment.

The RERTR-9A and RERTR-9B experiments will complete irradiation in December 2007 and January 2008, respectively, and PIE is planned to begin in February.

AFIP-2 & -3 – A primary obstacle to the development of the monolithic fuel form is the demonstration of performance at prototypic scale. Unlike dispersion fuel, the behavior of the fuel phase in the monolithic fuel meat is not buffered by the presence of the aluminum matrix. Deformation (due to fission product induced swelling, thermal expansion, etc.) in one location may impact the mechanical response of the fuel in another location due to the fact that they are physically coupled. A test of full size fuel plates is therefore being designed to explore this behavior. Two plates, one with a zirconium diffusion barrier and one with a silicon enhanced interface, are being fabricated by FB for the AFIP-2 test and two similar plates are being fabricated by HIP for the AFIP-3 test. The AFIP-2 and AFIP-3 tests are planned for insertion in January 2008 and May 2008, respectively, and PIE is expected to begin in late 2008.

5. QUALIFICATION & CONVERSION PROGRAM

As presented in previous sections, in late 2003 it became evident that the U-Mo/Aluminum based dispersion fuel exhibited significant fuel performance problems under the irradiation conditions required for conversion of the highest-powered research reactors. Solutions to this fuel performance issue have been proposed and shown promise in early testing. Based on these results, it is believed that it is possible to allow generic fuel qualification of both the dispersion and monolithic fuel forms in the near future. The US is planning to convert the five US reactors prior to the end of 2014 and will continue to support conversion of additional international reactors under its Global Threat Reduction Initiative (GTRI). To accomplish these missions, a U-Mo fuel will need to be available for use by the end of 2010.

The fuel development program started with tests performed on miniature fuel plates used to screen potential fuel materials and eventually to perform detailed tests on design parameters. Irradiation tests on larger specimens have been performed or are planned using the Advanced Test Reactor (ATR) in the United States, the Belgian Reactor-2 (BR2) reactor in Belgium, and the OSIRIS reactor in France. These irradiation tests provide a large amount of data on the performance of advanced fuel types under irradiation and allow the down selection of technology for larger scale testing during the final stages of fuel qualification. In conjunction with irradiation testing, fabrication processes are being developed and made available to commercial fabricators. The

commercial fabrication infrastructure also being enhanced to ensure that the supply of this new fuel form is reliably available to all users. However, it is important to recognize that final qualification of the dispersion and monolithic fuels designs will likely need to occur independently.

It appears that enough fuel performance information will be available at the completion of currently planned full-size plate tests to select a set of design parameters for U-Mo dispersion fuels that would allow generic approval for use with density less than 8.5 g-U/cm^3 under high power conditions. In order to obtain this approval, a larger scale demonstration of fuel performance and fabrication technology will be necessary. Several representative plate-type fuel assemblies should be irradiated in at least two Materials Test Reactors (MTR). Reactors likely to be used for this purpose include the High Flux Reactor (HFR), the Advanced Test Reactor (ATR), the Belgian Reactor-2 (BR2), and MIR in Russia. Following post irradiation examination of these experiments, a report summarizing the behavior of very-high density dispersion fuel will be assembled to assist fuel designers and regulators in the design and licensing work required for conversion. The required irradiation tests are tentatively planned for 2009 or 2010.

Additional development of the monolithic fuel form is required before qualification tests can be initiated. However, all five US high power research reactors (as well as additional foreign reactors) will require monolithic fuel to convert to LEU. The US fuel development program is making rapid progress with this fuel and plans to conduct a mini-plate test and two full-size fuel plate tests in 2008. After these tests are completed, the performance database necessary to select a design for qualification is expected to be available. Qualification testing is planned for 2009 and will focus on the demonstration of the base U-Mo fuel form (flat and arced fuel plates), which when completed will be summarized in a U-Mo monolithic fuel performance summary report. This report will be submitted to the US Nuclear Regulatory Commission (NRC) for review. It is anticipated that the NRC will issue a report (as a NUREG document) that may be used as the basis for designing and using fuel for NRC regulated reactors. Additional development testing will continue to support the incorporation of burnable poison and graded fuel zones. This testing is expected to be performed in 2010 and 2011 and will be summarized in a report to NRC, which may be released as an addendum to the qualification NUREG.

The final step in the fuel qualification process is the pursuit of a license to use the fuel in specific reactors. The step emphasizes the insertion of lead test elements into the converting reactors. Each reactor that plans to convert using the developed high-density fuels will develop a reactor specific conversion plan based upon the reactor safety basis and operating requirements. For some reactors (FRM-II, High-Flux Isotope Reactor [HFIR], and RHF) conversion will be a one-step process for others the new fuel type will be phased in gradually as spent HEU fuel elements are replaced.

4. CONCLUSIONS

After the limitations in first generation U-Mo/Al fuel performance were identified, an international effort was launched to understand and resolve the issues. After four years it appears that a viable solution to the breakaway swelling problem has been identified and will be carried into full-size tests. In response to the early success of second generation dispersion fuels, the possibility of moving forward with the generic qualification of a high density (~8.5 g U/cc) version of this fuel appears viable. This fuel would be useful to a large number of reactors around the world to either convert to LEU or to enhance performance. This qualification effort will require the fabrication and irradiation testing of full-size plates, transference of fabrication technology to commercial fabricators, and the initiation of element testing over the next two years. It is however, fully recognized that, even if this fuel form is successfully qualified, it will not enable the LEU conversion of all reactors of interest. As a result, substantial emphasis will still be placed on the development of the monolithic fuel form with a very high density dispersion fuel form carried forward as a backup.

For dispersion fuel, the most promising remedies appear to be an addition of silicon to the matrix or the oxidation of the particles without any additive in Al matrix.

For monolithic fuel, tests are in progress to improve cohesion between foil and clad during and after irradiation. Promising solutions include the application of a silicon rich layer to the interface between the cladding and fuel, application of a zirconium diffusion barrier to the interface, or by using Zy cladding co-rolled with the UMo foil.

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