INL/EXT-09-16807
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Irradiation Performance of U–Mo Alloy Based 'Monolithic' Plate-Type—Design Selection Update

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Prepared for the
U.S. Department of Energy
Office of National Nuclear Security
Administration
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

SUMMARY

Based on data available in 2009, a decision was made to focus U–Mo monolithic fuel development and qualification efforts on a single fuel design. This fuel design consists of a U–10Mo (wt.%) monolithic fuel foil, a zirconium barrier layer applied to the faces of the foil, and 6061 aluminum cladding. This document updates the basis for the selection of that fuel system, drawing on additional fuel-performance data available as of May 2013. This update also applies recently developed GTRI (Global Threat Reduction Initiative) requirements for fuel qualification to the selection process.

Because of its inherent gamma-phase stability and qualitatively better rolling behavior, U–10Mo was selected as the fuel alloy of choice, exhibiting reasonable uranium density and good irradiation performance. A zirconium barrier layer between the fuel and cladding was chosen to provide a predictable, well-bonded, fuel-cladding interface, greatly reducing fuel-cladding chemical interaction. The fuel plate testing conducted to inform this selection was based on the use of U–10Mo foils fabricated by hot- and cold co-rolling with a Zr foil. The foils were subsequently bonded to the Al–6061 cladding by hot isostatic pressing or friction stir welding.

Since the original down-select decision was made, additional information has become available on fuel performance and on the cost of the fuel system. Preliminary cost estimates indicate that U–Mo-Zr waste generated during fabrication is difficult to recycle and adds substantially to the production cost.

It is the purpose of this report to review fuel-performance data available as of May 2013, outline data and observations that are significant to informing the down-selection process, review the data against down-select requirements, and update the recommendation on a fuel system suitable for fuel qualification. In addition, this report assesses the potential of a fuel system without a zirconium barrier layer for use in USNRC-licensed research reactors.

This revised report is issued as a working draft for discussion by the United States High Performance Research Reactor (USHPRR) program. The document will continue to be revised as additional information becomes available. This may include updates to information about normal and anticipated transient fuel operating conditions as conversion fuel-element designs evolve or additional information on fuel performance gained from U.S. or international testing programs. In particular, information from the RERTR–12 experiment will be included when available.

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ACRONYMS

AFIP	ATR Full-size-plate In center flux trap Position
ATR	Advanced Test reactor
DOE	Department of Energy
FB	Friction Bonding
FD	Fuel Development
FSW	Friction Stir Welding
GTRI	Global Threat Reduction Initiative
HEU	Highly Enriched Uranium
HIP	Hot Isostatic Press
HPRR	High Performance Research Reactor
IL	Interaction Layer (fuel/clad chemical interaction)
LTA	Lead Test Assembly
NNSA	National Nuclear Security Administration
RERTR	Reduced Enrichment for Research and Test Reactors
TLPB	Transient Liquid Phase Bonding
USHPR	United State High Performance Research Reactor

Irradiation Performance of U–Mo Alloy Based 'Monolithic' Plate-Type—Design Selection Update

1. INTRODUCTION

For over 30 years the Reduced Enrichment for Research and Test Reactors (RERTR) program has worked to provide the fuel technology and analytical support required to convert research and test reactors from the use of highly enriched uranium (HEU) fuels to fuels based on low-enriched uranium (LEU) (defined as <20% U–235/U). While many of these reactors can be converted using LEU nuclear fuels that are based on traditional dispersion-type fuel designs—including U_3Si_2 or U–Mo alloy fuel particles in an aluminum based matrix—some reactors have been identified that can only be converted to LEU fuel by using a fuel design that provides higher uranium density than can be achieved in dispersion fuels.[1]

The basic design selected for development was a very high-density plate-type fuel based on a U–Mo-alloy fuel foil, encapsulated in currently used aluminum cladding alloys.[2] This design was identified as the "monolithic" fuel design, referring to the solid U–Mo alloy fuel meat. This general type of fuel meat configuration had already been tested with rod-type fuel.[3]

The Global Threat Reduction Initiative's (GTRI's) High Performance Research Reactor Fuel Development Program (HPRR-FD) has been tasked with the development of this fuel.

The program has performed a series of fuel tests since 2005 that have established the viability of monolithic research-reactor fuels based on uranium-molybdenum (U–Mo) alloys. The monolithic fuel design represents a significant departure from existing dispersion fuel technology because the entire fuel zone is replaced by a U–Mo alloy.

This modification poses unique fabrication challenges that must be resolved as part of the performance evaluation. Providing an economical fabrication process that provides repeatable results and is applicable to all fuel-plate geometries is required for the conversion of USHPRRs. Data in this report and the recommendation for selection of a base fuel design are based on a bench-scale process developed for the production of test fuel plates.[4] The history of development of fabrication methods in the U.S. and abroad is found in a number of documents.[5,6,7,8,9,10,11,12,13,14]

Since the original down-select decision was made in 2009, additional information has become available on fuel performance and on the cost of the fuel system. Preliminary cost estimates indicate that U–Mo–Zr waste generated during fabrication is difficult to recycle and adds substantially to the production cost.

The development of a commercial-scale fabrication line and associated fabrication processes is ongoing, with the goal of selecting an optimized process that provides a product that meets fuel performance requirements at an acceptable cost. A description of the research and development being conducted is found in the Fuel Fabrication Capability Research and Development Plan.[15] The results of this research and development will be used to establish an improved baseline process that is designed to accommodate full-scale production requirements. Fabrication development work has been identified that may result in significant process changes targeted at reducing fuel cost, for example, by reducing the number of process steps required, increasing yield, and eliminating or reducing difficult-to-recover scrap material.

New variables introduced by fabrication-process development require an experimental irradiation program that provides quantitative data allowing down-select of the fabrication process. These concepts will be included in the MP–1 irradiation test to determine whether fuel performance associated with optimized or new fabrication processes is adequate to meet USHPRR reactor requirements. A final down-selection of the fuel design and fabrication technology will be made as a result of the MPl–1 test.

Some USHPRR conversion fuel elements may include the need to incorporate a burnable poison and complex fuel-meat shapes into the fuel design.[16] The fuel testing performed to date has not been focused on the issues of so called "complex fuel" fabrication or irradiation performance. Consideration has been given to the potential for extrapolation of the selected "base fuel" designs and fabrication processes to complex fuels. Some potential complex fuel designs have been irradiation–tested, but have not yet been examined post-irradiation. Evaluation of the performance of fuels with high boron content has resulted in the current strategy to remove the boron from the fuel plates. Likewise, efforts directed at simplification of the High Flux Isotope Reactor (HFIR) fuel-zone shape are in progress. These efforts are anticipated to result in viable fuel-element conceptual designs in October 2013, after which time the selection of the base-fuel technology can be revisited if necessary.

The Base Fuel Qualification effort is to develop a base fuel for both NRC and DOE research reactors. The 475 W/cm^2 is the current bounding parameter of DOE research reactors; NRC-licensed reactors operate at less than 300 W/cm^2 with a burnup of up to 100% LEU.

It is the purpose of this report to review fuel performance data available as of May 2013, outline data and observations that are significant to informing the down-selection process, review the data against fuel down-select requirements, and update the recommendation on a fuel system suitable for fuel qualification. In addition, this report assesses the potential of a fuel system without a zirconium barrier layer for use in NRC-licensed reactors.

This revised report is issued as a working draft for discussion by the USHPRR program. The document will continue to be revised as additional information becomes available. This may include updates to information about normal and anticipated transient fuel operating conditions as conversion fuel-element designs evolve or additional information on fuel performance gained from U.S. or international testing programs. The current report does not include, for example, important preliminary information from the AFIP–6 MkII, AFIP–7, and RERTR–12 insertion 2 tests, which are currently being examined or are pending post-irradiation examination (PIE), and will be revised as this data becomes available.

2. REQUIREMENTS FOR FUEL DOWN-SELECTION

To be successful, the USHPRR Conversion Program must develop LEU fuels that are:

- QUALIFIED
  - Fuel that has been successfully irradiation tested and is licensable from the point of view of fuel-irradiation behavior
- COMMERCIALLY AVAILABLE
  - Fuel that is available from a commercial manufacturer
- SUITABLE
  - Fuel that satisfies safety criteria
  - Fuel that satisfies criteria for LEU conversion of a specific reactor
  - Fuel with a Service Lifetime comparable to current HEU fuel (e.g., number of fuel elements used per year is the same as or less than with HEU fuel)

High-level requirements for fuel qualification are shown in Figure 1.[17]

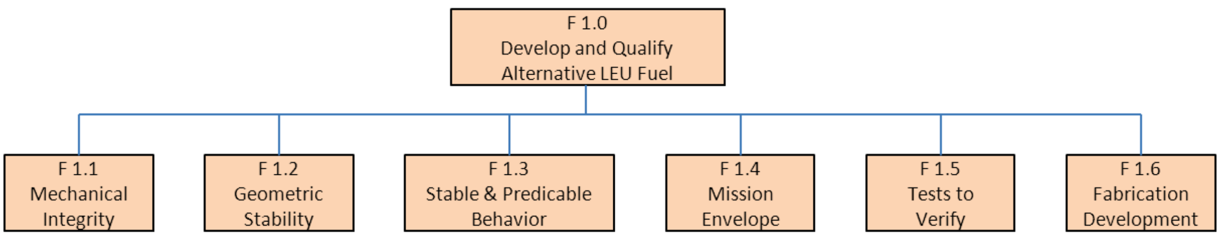

Figure 1. High-level requirements for fuel qualification.

These requirements include ensuring mechanical integrity, geometric stability, and stable and predictable fuel behavior. Performance of the fuel must be demonstrated by testing in conditions that provide adequate margin relative to normal operation and anticipated transient conditions. Fuel that is selected for use must represent a commercially viable product.

The down-selection of the conversion fuel system results from data obtained during the research and development phase of the program. The down-selected fuel design provides the “base fuel” design for subsequent fuel qualification. The down-selection is based on testing that generates a data set that is sufficient to ensure that the broader set of fuel-qualification requirements can be satisfied; down-selection criteria are a subset of the requirements for fuel qualification. Final verification of the selected fuel design against requirements results from the extensive testing conducted during the fuel-qualification process.

Individual requirements relevant to fuel down-selection are listed below, along with a discussion of the data required to demonstrate that the requirements are met.

2.1 Maintaining Mechanical Integrity

For purposes of fuel system down-selection, ensuring that mechanical integrity is maintained requires that:

- The mechanical response of the fuel meat, cladding, and interlayers during normal operations and anticipated transients shall be established
- Diffusion layer performance limits and manufacturing processes shall be established.
- The fuel element shall not delaminate during normal operation or anticipated transients.

Demonstration that these requirements are satisfied requires that:

- The mechanical response of the fuel plate is demonstrated through irradiation testing under prototypic conditions and post-irradiation examination of mechanical integrity through methods such as ultrasonic testing, radiography, gamma scanning, dimensional inspection, and destructive metallography
- Fuel mechanical modeling does not reveal inherent structural issues in the fuel system
- The performance of the diffusion layer is demonstrated through irradiation testing under prototypic conditions and post-irradiation examination using methods such as ultrasonic testing and destructive metallography
- Resistance to delamination is demonstrated through irradiation testing under prototypic conditions, and subsequent post-irradiation examination using methods such as ultrasonic testing, visual examination, radiography, dimensional inspection, and destructive metallography.

The following additional requirements must be satisfied, but are not anticipated to differentiate between down-selection of fuel designs that belong to the aluminum clad U–Mo fuel system:

- Physical properties related to fuel integrity shall be established
- Limits for fabrication defects shall be established
- Water-corrosion limits for the cladding shall be established.

The properties of the fuel-system components have been documented.[18] Some variations in properties are expected to result from changes in processing variables within the U–Mo monolithic fuel system; an example is the change in properties caused by variations in cold work introduced during foil rolling. These variations have not been well documented, but they have been demonstrated to not have a significant impact on fuel performance.

Limitations on fabrication defects were established based on the existing fuel specification for the Advanced Test Reactor (ATR)[19]. These limitations are based on the requirements for heat transfer out of the fuel meat sufficient to provide a margin to cladding failure under all operating conditions. Fuel systems or fabrication routes that could not meet these requirements were rejected and not carried forward in the irradiation testing program. Development of further criteria for defects will be made on the basis of distinct fuel-failure mechanisms when these mechanisms are known.

Corrosion behavior of the aluminum cladding is not anticipated to differentiate between fuel designs that belong to the U–Mo monolithic fuel system. Corrosion behavior is dependent on the coolant chemistry and operating conditions (flow, heat flux, coolant temperature) of individual reactors. The cladding proposed for the U–Mo monolithic fuel system is aluminum alloy 6061. This is the same cladding used for all U.S. High Performance Research Reactors that may be converted using this fuel. Water corrosion behavior is monitored through post-irradiation examination of the oxide layer, including visual inspection, nondestructive measurement of thickness, and destructive metallography.

2.2 Maintaining Geometric Stability

For purpose of fuel system down-selection, ensuring that geometric stability is maintained requires that:

- The geometry of the fuel shall be maintained during normal operation and anticipated transients
- Performance at fission density, heat flux, and fuel geometry shall be maintained to avoid flow instability
- Changes in the channel gap shall not compromise the ability to cool the fuel.

Demonstration that these requirements are satisfied requires:

- That the geometry of the fuel is maintained is verified through irradiation testing under prototypic conditions and post irradiation examination of fuel plate geometric stability using profilometry
- Fuel blister-threshold temperature is established to be adequate based on comparison to existing requirements
- Fuel swelling behavior is evaluated through irradiation under prototypic conditions and post-irradiation methods such as density measurement and dimensional inspection. Reactor Conversion evaluates the effect of swelling on channel-gap closure and plate cooling requirements.

The following additional requirements must be satisfied, but are not anticipated to differentiate between down-selection of fuel designs that belong to the U–Mo monolithic fuel system:

- Data shall be documented for material properties that could affect geometric stability and thermal hydraulic analysis
- Plate movement caused by pressure differential shall not compromise the ability to cool the fuel
- Plate movement is addressed by the flow testing work scope established by the U.S.H.P.R.R. program. See Section 2.1 for discussion of material properties.

2.3 Stable and Predictable Behavior

For purposes of fuel system down-selection, ensuring stable and predictable behavior requires that:

- Fuel performance shall be known and predictable for the expected range of fuel composition and impurities, processing parameters, and microstructure for all credible environmental and irradiation conditions
- Fuel swelling behavior shall be within the stable swelling regime (linear and predictable)
- Uranium-molybdenum corrosion behavior after cladding breach shall be established
- Irradiation behavior on scale-up to larger sized plate and fuel elements shall be predictable.

Demonstration that these requirements are satisfied requires that:

- Sufficient fuel testing is performed to demonstrate acceptable performance for fabrication conditions and the range of specification requirements for composition and impurities anticipated for commercial fabrication
- The effect of the initial fuel microstructural and evolution of the microstructure during irradiation on fuel performance is understood sufficiently to ensure that unstable behavior does not occur for the anticipated range of fuel operating conditions
- The presence of microstructural precursors to fuel failure is documented, and the impact of these precursors on fuel stability is evaluated
- Fuel blister-threshold temperature is established to be adequate based on comparison to requirements for existing HEU fuel
- Fuel swelling behavior is evaluated through irradiation under prototypic conditions and post-irradiation methods—such as density measurement and dimensional inspection—and the regime for stable swelling is established
- Corrosion behavior of the U–Mo system is established through in-reactor or post-irradiation corrosion testing

- Fuel performance of large-scale plates is established through testing under prototypic irradiation conditions and post-irradiation examination methods such as visual examination, ultrasonic testing, radiography, dimensional measurement, gamma scanning, and destructive metallography.

The following additional requirements must be satisfied, but are not anticipated to differentiate between selection of fuel designs that belong to the aluminum clad U–Mo monolithic fuel system:

- Physical properties that are important for the analysis of fuel burn-up limits shall be documented.

See Section 2.1 for discussion of material properties.

2.4 Establish Reactor Mission Performance Envelope

For purpose of fuel system down-selection, establishing that fuel performs within the reactor mission-performance envelope requires that:

- The reactor performance envelope for normal operations and anticipated transients is established
- The fuel elements shall demonstrate peak heat flux of 475 W/cm^2 with a burnup of greater than 50% U–235.

  (Note: The Base Fuel Qualification effort is to develop a base fuel for both NRC and DOE research reactors. The 475 W/cm^2 is the current bounding parameter of DOE research reactors; NRC-licensed reactors operate at less than 300 W/cm^2 with a burnup of up to 100% LEU.)
- Operating costs impacted by fuel conversion shall be minimized and agreed to by the reactor operator and the agency that provides the fuel.

These requirements are satisfied by ensuring that the reactor mission performance envelope is understood and documented. The requirement that, "Operating costs impacted by fuel conversion shall be minimized and agreed to by the reactor operator and the agency that provides the fuel" is currently being addressed through fabrication research, development, and process optimization and by studies of alternative fuel concepts that are outside the scope of this document. Evaluation of reactor operating conditions is based on available reports and differs from the requirements listed here.

The following additional requirements must be satisfied, but are not anticipated to differentiate between down-selection of fuel designs that belong to the aluminum clad U–Mo monolithic fuel system:

- Use-related performance characteristics of the reactor shall not to be degraded significantly by the fuel conversion
- Changes in nuclear capabilities resulting from conversion to LEU fuel shall not result in a significant decrease in the safety margin of the reactor
- The alternative LEU fuel shall have a plausible disposition path
- During transients, two-phase coolant flow is permissible, provided fuel integrity is maintained.

U–Mo monolithic fuel was selected for reactor conversion based on the analysis of neutronic performance in USHPRR reactor systems. The fuel plate design, other than dimensional attributes, does not have a significant impact on these requirements.

2.5 Test to Verify Design Requirements

Testing to verify reactor-specific design requirements is conducted during the qualification phase of the program and is not part of the fuel down-select.

2.6 Fabrication/Manufacturing Development and Qualification

For purpose of fuel system down-selection, requirements for fabrication development are

- Test fuel elements shall be manufactured for evaluation of fuel-element characteristics and performance
- Test fuel elements shall be manufactured for evaluation of fabrication processes, including manufacturing scale-up issues, maximum fuel-loading limits, fabrication porosity (both inside fuel meat and elsewhere in the fuel plate), fuel-meat uniformity (meat thickness and uranium density), dimensional tolerances, fuel-element manufacturing, and hydraulic characteristics:
  - Fabrication of thin clad, thick fuel meat shall be demonstrated
  - Fabrication of thick clad, thin fuel meat shall be demonstrated
  - Fabrication of finned fuel plates shall be demonstrated
  - Fabrication with a diffusion barrier shall be demonstrated.

Demonstration that these requirements are satisfied for fuel down-selection requires that:

- As fuel concepts are developed test specimens that are required to evaluate methods to manufacture the laboratory-scale test elements are fabricated
- Fuel specifications based on anticipated USHPRR requirements are developed, and manufacture of test articles that meet those specifications is demonstrated
- Test articles are characterized for relevant features such as porosity, fuel-meat uniformity, fuel/clad bonding, and dimensional tolerances
- Fabrication scale-up of larger-scale test articles is sufficient to ensure that the selected fabrication processes are viable
- Test articles are fabricated in geometries representative of USHPRR geometries
- Fabrication of fuel plates that incorporate a diffusion barrier sufficient to meet fuel-performance requirements is demonstrated.

The following requirements do not apply to fuel down-selection. These items will be the responsibility of the Design Authority:

- Fabrication process specifications used during fuel qualification and pilot-scale fabrication shall be scalable to full-production process capabilities and quality requirements (applies to qualification phase post down-selection)
- Fabrication process specifications and acceptance criteria for prototype fuel elements shall be based on results of fuel characterization and manufacturing evaluations
- Prior to production, the fabricator shall provide the documentation specified in ANS 15.2, Section 16.1, “Preproduction”
- The fabricator shall provide the documentation specified in ANS 15.2, Section 16.2, “Deliverable Reports,” for each production run of fuel elements.

3. MONOLITHIC FUEL DESIGN ELEMENTS

The primary research and development goal for the monolithic fuel design is to identify a fuel that will provide uranium density adequate for conversion of the USHPRRs while meeting reactor fuel-performance requirements.

Elements of the fuel-plate design important to the fuel performance include:

1. *Cladding material.* The cladding selected for this fuel design is aluminum alloy 6061. This is the fuel-cladding alloy currently used in all USHPRRs. This cladding was selected because of its known compatibility with USHPRR coolant systems. Maintaining an adequate bond (seal) at cladding/cladding interfaces is critical to fuel performance. Debonding can result in water ingress, fuel-meat corrosion, and fission-product release.
2. *Fuel meat*. Based on scoping tests that provided an evaluation of fuel-meat irradiation performance, U–Mo alloys with a molybdenum content of greater than 7 wt.% were considered for the fuel meat. U–10Mo was selected based on consideration of fabrication properties, phase stability, and fuel performance.
3. *Interfaces between the fuel and the cladding.* Maintaining an adequate bond between the fuel meat and the barrier layer, as well as between the barrier layer and the aluminum-based cladding, is critical to fuel performance. Debonding and formation of a gap between the fuel meat and cladding creates a barrier to heat transfer. Under some operating conditions, this may result in failure of the fuel cladding (excessive corrosion, melting, etc) and fission-product release. Fission gas may collect in a debond location, promoting further separation, plate bulging, and subsequent coolant-channel-gap closure. Formation of bulges (or blisters) may also result in fission-gas release from the plates. Consideration of the fuel-to-cladding bond includes minimizing detrimental fuel/cladding chemical interaction. In some cases, alloying additions to cladding/matrix materials have been found to delay the onset of breakaway swelling in dispersion fuels[20]. The interaction layer (IL) that can form between the Al–6061 cladding and U–Mo fuel is similar to that observed at the fuel/matrix interface in dispersion fuels. The interaction layer typically consists of intermetallic phases that may have poor mechanical properties (e.g., they are brittle) [21] and are susceptible to breakaway swelling.[22] For these reasons, a zirconium-diffusion-barrier fuel design was selected.

4. FUEL OPERATING ENVELOPE

The fuel operating envelope under normal operating conditions for the USHPRRs was estimated by GTRI-FD based on available information and is shown in Figure 2 and Figure 3, considered local peak values for key operating parameters. Figure 2 plots fuel-plate operating conditions as a function of fuel-meat peak volumetric power and peak fission density. Figure 3 plots fuel-plate operating conditions as a function of fuel peak surface heat flux and peak fission density. Data plotted in the figures and listed in Table 1 were taken from reports recently provided by the 5 USHPRRs. [23,24,25,26,27] The plots also include a trend line that indicates a margin of approximately 15% as an estimate of the range of testing conditions required to demonstrate sufficient margin to failure. In some cases, this margin may not be applicable because it is not possible for fuel to operate under the conditions represented by the trend line. For example, U–10Mo LEU fuel in research reactors cannot reach fission densities higher than ~7.8E+21 f/cm^3. Because the fuel qualified by the program must be demonstrated to provide robust performance with a very low failure rate under all reactor operating conditions, it is important that these operating conditions be well understood and clearly documented.

It should be noted that these conditions, as estimated by GTRI-FD, differ from those identified in the program requirements document (*U.S. High Performance Research Reactor Project: Functions and Requirements, Global Threat Reduction Initiative Document,* April 2013). As conversion fuel designs and estimation of required operating margin evolve, the design envelope will also evolve.

It should also be noted that the effect of anticipated reactor transients on fuel operating conditions has not been defined. An evaluation of fuel performance under these conditions has not been performed.

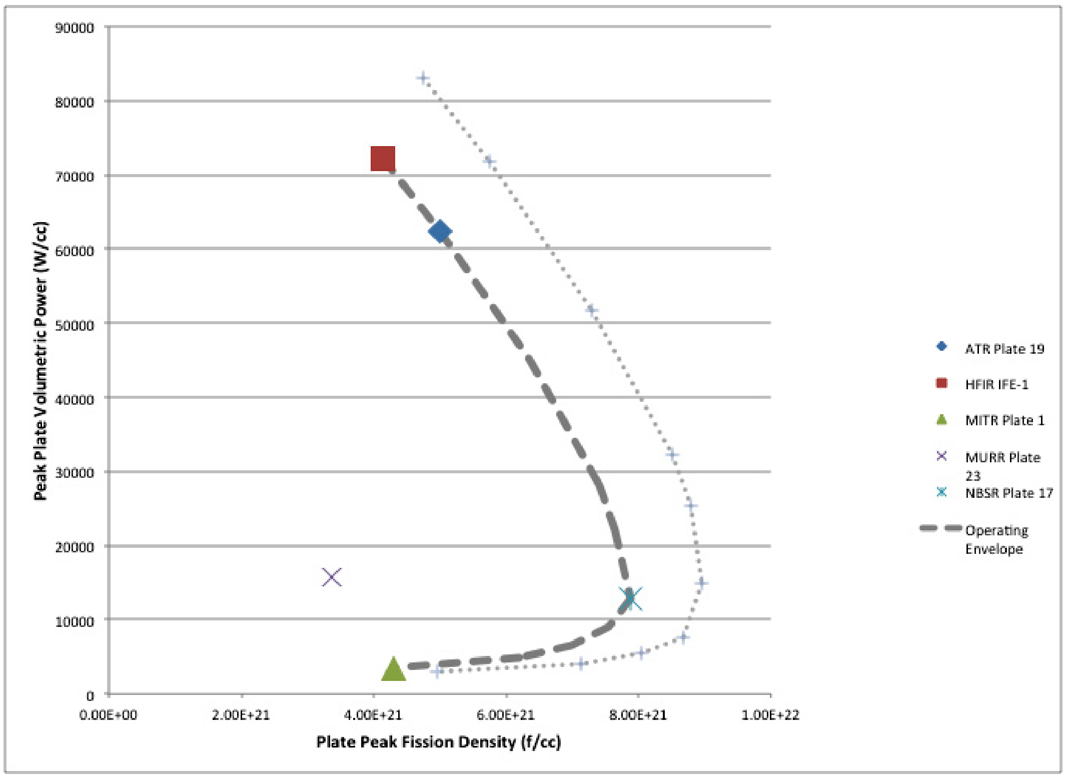

Figure 2. Fuel operating conditions represented as a function of peak fission density and peak volumetric power. Dashed line represents nominal operating envelope. Dotted line represents 15% additional margin.

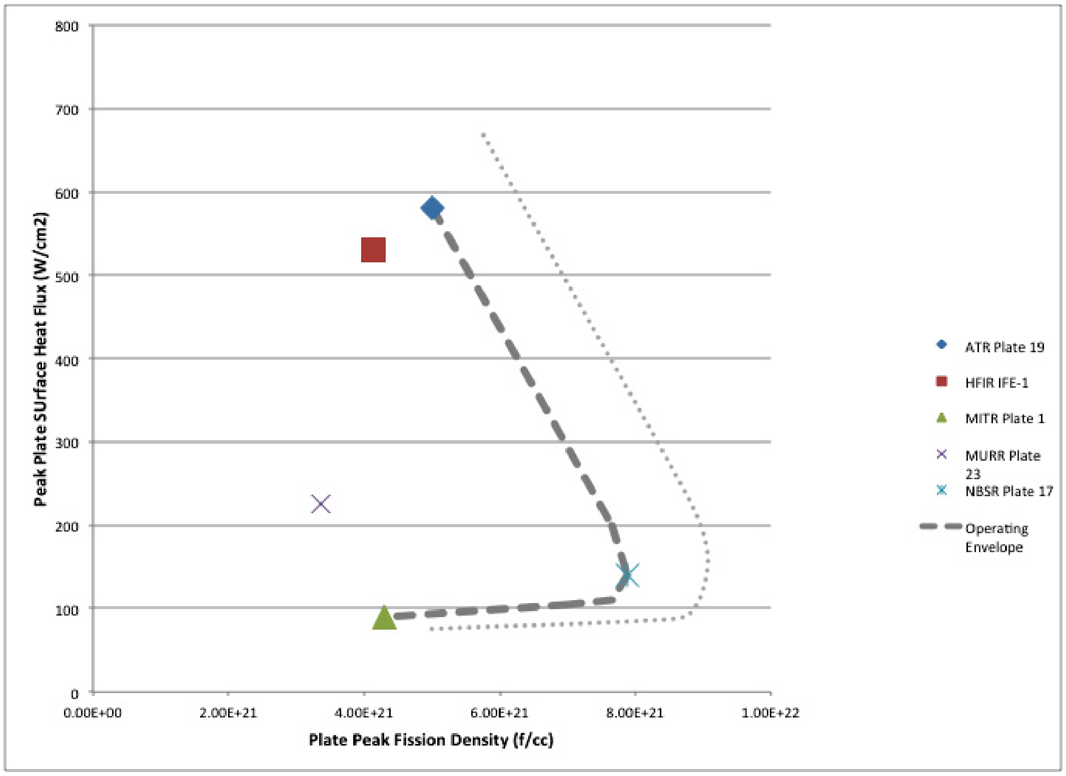

Figure 3. Fuel operating conditions represented as a function of peak fission density and peak surface heat flux. Dashed line represents nominal operating envelope. Dotted line represents 15% additional margin.

Table 1. Fuel plate operating conditions as defined by local peak values (margin not applied).

Reactor/fuel plate	**Peak Power (W/cm^3)**	**Peak Heat Flux (W/cm^2)**	**Peak Fission Density (f/cm^3)**
ATR Plate 19	62485	581	5.0E+21
HFIR IFE-1	72249	531	4.1E+21
HFIR OFE-9	47200	551	2.9E+21
MITR Plate 1	3500	89	4.3E+21
MURR Plate 23	15826	226	3.4E+21
NBSR Plate 17	12910	139	7.9E+21

5. MONOLITHIC FUEL DESIGN CONCEPTS

Many concepts were considered in order to improve early monolithic fuel-plate designs. These concepts included:

- Increasing the Mo concentration from 7 wt % to 10 or 12 wt % to simultaneously simplify foil fabrication and to provide additional fabrication working time by increasing the stability of the U–Mo gamma phase. Stabilizing the γ phase resulted in reduced fuel/Al cladding interaction and the resulting IL thickness.
- Insertion of a barrier material between the fuel and cladding. The barrier was used either to prevent diffusion between the cladding and fuel or to modify the composition of the interaction product such that it was stable under irradiation.
- Fuel-plate bonding technology. The quality of the fuel/clad bond, fuel-plate residual stresses, and fuel/clad interface chemical condition can be greatly influenced by the bonding technique employed.

The process used to fabricate fuel plates can play an integral role in the performance of the fuel during irradiation. The fuel plates irradiation-tested within this development program are typically fabricated by casting a U–Mo alloy ingot (through arc melting or induction melting) in a book mold. The thin ingot is subsequently hot-rolled to an intermediate thickness. The ingot is then either hot- and cold-rolled into a U–Mo foil, hot-rolled with the diffusion barrier (to allow for bonding), or hot- and cold-rolled with a diffusion barrier. Silicon barrier coatings between fuel foil and cladding were alternatively also applied by thermal spray-coating or by the insertion of a silicon-rich aluminum alloy during fuel/clad bonding. The fuel/clad bonding fabrication techniques explored were hot isostatic press (HIP), friction bond (FB) (also known as friction stir weld [FSW]), and transient liquid phase bond (TLPB). The TLPB process is based on applying a silicon spray coating to the fuel/clad interface and hot-pressing the assembly at a temperature above the Si-Al eutectic temperature (577°C), leading to a diffusion bond between the two materials.

A series of seven mini-plate experiments (RERTR–4, 6, 7A, 8, 9,10, and 12), two full-size plate experiments (AFIP–2, and 3), and one intermediate-size experiment (AFIP–4) were conducted in the Advanced Test Reactor to evaluate the performance of monolithic fuel concepts. Variations in the fuel design and fabrication techniques that were tested as part of the monolithic fuel development program are outlined in Table 2. The final down-selection of design variables is a direct outcome of the results of this testing.

Table 2. Fuel-design variables tested for monolithic plates.

Fuel Composition, wt %	Fuel Foil Thickness, mm	Bond Type	Barrier	Other
U–7Mo	0.25	FB	Zr Foil	Split Foil
U–8Mo	0.51	HIP	Si spray coating	Foil with holes to allow Al/Al bonding in fuel meat region
U–10Mo	0.33	TLPB	Al-4043 foil	Zircaloy Cladding
U–12Mo *			Nb Foil	
			None (bare)	

* Limited testing, as U–10Mo provided adequate irradiation performance, and allows for increased U density.

6. Summary of Irradiation Testing Program

Irradiation testing is the only source of fuel-performance data that can be used to differentiate fuel-plate designs. The first irradiation test to include monolithic fuel was the RERTR–4 test, in which two small fuel plates, each containing thin discs of U–10 Mo fuel meat having ~12-mm diameter and 0.25-mm thickness were irradiation tested. The irradiation stability (defined as the absence of breakaway swelling) observed in the RERTR–4 test encouraged further testing of the monolithic fuel plate design[28]. More relevant to down-selection were the RERTR–6 through RERTR–12 test series and the AFIP–2, 3, and 6 tests. Examinations remain to be completed on RERTR–12, and AFIP–4, 6 MkII, and 7 tests, although preliminary examination data are available and increase confidence in the down-selection at this point.

The RERTR–6 irradiation test included both U–7Mo- and U–10Mo-based monolithic mini-plates, with foil thicknesses of 0.25 mm and 0.51 mm. The foil dimensions were increased in size from the original small discs in RERTR–4 to the more representative dimensions shown in Figure 4. This test plate configuration was used for the remainder of the mini-plate test programs (RERTR–8, 9, 10, and 12). Bonding of the fuel/cladding interface in all the fuel plates in RERTR–6 was accomplished using FSWing, also known as friction bonding (FBing). The RERTR–6 mini-plates were irradiated to fission densities consistent with roughly 50% U–235 burnup in LEU fuels. All plates showed generally good irradiation behavior.

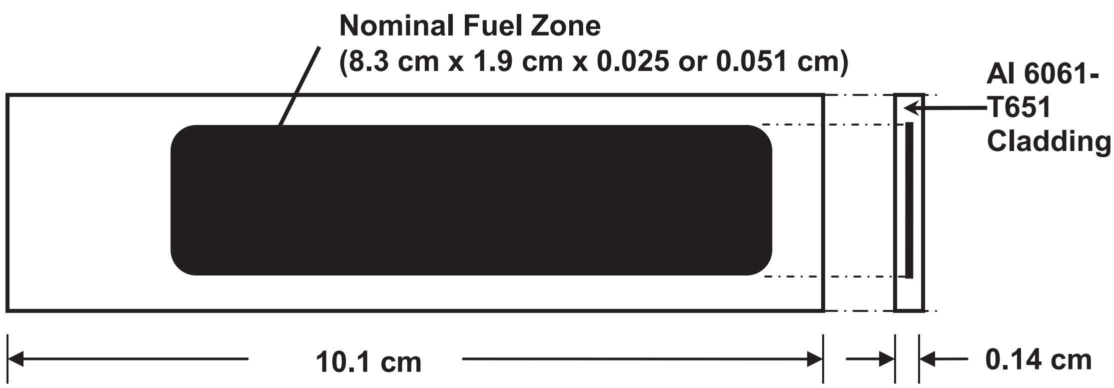

Figure 4. Sketch of RERTR test mini-plate (monolithic fuel).

Based on favorable fuel-performance information from the RERTR–6 experiments, thirteen monolithic fuel plates were incorporated into the RERTR–7A test. Difficulty fabricating U–Mo alloy foils with lower Mo content led to the RERTR–7A experiment's focus on U–10Mo and U-12Mo alloys. The mini-plates were similar to those tested in the RERTR–6 experiment (e.g., the same variations in thickness and bonding by FB). Because life-limiting fuel-performance phenomena were not apparent at low-power and burnup, the peak heat fluxes (and concomitant operating temperatures), and the peak fission density/U–235 depletion were increased for the RERTR–7A experiment and the tests that followed.

In addition to the standard monolithic fuel designs, two Zircaloy–4-clad (hot rolled) U–7Mo fuel plates were included in the RERTR–7 test. These were manufactured in Argentina at the Comisión Nacional de Energía Atómica. Selected post-irradiation examination results for RERTR–6 and RERTR–7 have been previously reported.[29] The RERTR–8 experiment design was similar to that of the RERTR–7 experiment. The test matrix incorporated U–8Mo, U–10Mo and U–12Mo fuel foils and fuel/clad bonding by HIP and FB. All foils were 0.25 mm thick.

The RERTR–9 experiment focused on monolithic fuel plates using only U–10Mo foils. Fuel plates fabricated by FB and HIP bonding were both represented. The key design variable investigated was the incorporation of barrier layers between the fuel meat and cladding. The barriers were inserted to prevent formation of unstable high-aluminum-content intermetallic phases at the interface between the fuel meat and cladding (in the case of zirconium) or to modify the chemistry of the formed interaction layer (in the case of silicon or Al alloys). Barrier types included Zr, Al–4043, and Si-Al mixtures.

The fuel plates tested in RERTR–10A experiment were all fabricated by HIP and were designed to more completely evaluate (1) the effect of various silicon compositions within the interlayer and (2) the thickness of the zirconium-covered fuel foils using 0.25 mm and 0.51 mm nominal foil thicknesses. The fuel plates in RERTR–10B were all fabricated by friction bonding with two different-thickness silicon layers and with niobium or zirconium diffusion barriers. The experiments were also used to test the effect of increased fission rate, operating temperature, and fission density on fuel performance.

All fuel plates included in the RERTR–12 test were fabricated by HIP and were designed to evaluate the effects of fission density and fission rate on fuel performance. All fuel plates contained U–10wt.% Mo alloy monolithic fuel foil with a ~25µm-thick zirconium interlayer. Fifty-six plates with various fuel enrichments and fuel-meat thicknesses (0.25 mm, 0.51 mm, and 0.635 mm) were tested in seven capsules (X1, X2, X3, Y1, Y2, Y3 and Z) over five irradiation cycles. The irradiation hardware used was the same design used in the previous RERTR mini-plate experiments; however, the irradiation assembly was rotated 90 degrees from previous test campaigns. This resulted in a gradient from plate to plate as opposed to across the width of each fuel plate to eliminate a heating/fission-rate gradient across the width of the plate. Post-irradiation examination of the RERTR–12 experiment has not been completed.

The AFIP–2 and 3 experiments were designed to test U–Mo monolithic fuels at more prototypic size. Both experiments were designed to contain two plates, each 57.1 cm long × 5.6 cm wide. The AFIP experiments were irradiated in the center flux trap of the ATR.

The AFIP–2 experiment consisted of two monolithic U–10wt%Mo fuel plates fabricated using the friction bonding process. Each plate contained a nominally 250 µm-thick fuel core of U–10wt%Mo alloy, which was nominally 19.75% enriched uranium. The plate cladding was 6061 aluminum alloy. The plates were designated as 2TT and 2BZ. Plate 2TT was fabricated with a silicon layer thermally sprayed between the fuel and the clad to stabilize any interaction layers present at the fuel/cladding interface. Plate 2BZ was fabricated with a zirconium layer hot-rolled directly onto the fuel foil prior to cladding with aluminum alloy.

The AFIP–3 experiments also consisted of two monolithic U–10wt%Mo fuel plates, but were fabricated using the HIP process. Each plate contained a nominally 250 µm-thick fuel core of U–10wt%Mo alloy, which was nominally 19.75% enriched uranium. The plate cladding was 6061 aluminum alloy. The plates were designated as 3TT and 3BZ. Plate 3TT was fabricated with a silicon layer thermally sprayed between the fuel and the clad to stabilize any interaction layers. Plate 3BZ was fabricated with a zirconium layer hot-rolled directly onto the fuel foil prior to cladding.

The AFIP–4 experiment consisted of two assemblies, each with 6 plates of dimensions 19 cm × 5.6 cm, slightly larger than a mini-plate, as shown in Figure 5. One assembly contained 6 plates fabricated by friction bonding, and the other contained 6 plates fabricated by hot isostatic pressing. All plates contained a zirconium diffusion barrier between the fuel and the aluminum cladding. This experiment was irradiated for the purpose of providing larger-scale specimens for blister testing.

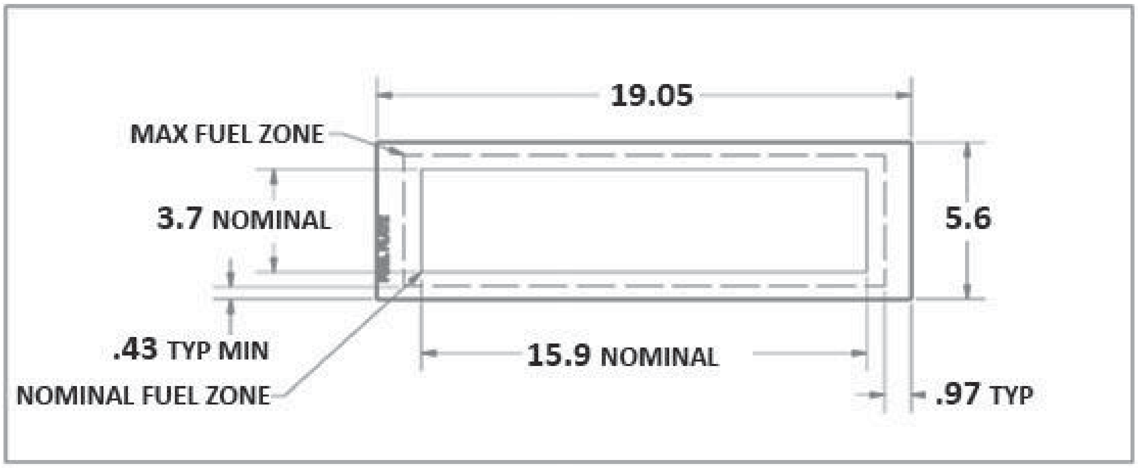

Figure 5. Sketch of AFIP–4 fuel plates (units are cm).

6.1 Irradiation Testing Conditions

Table 3 shows the irradiation conditions for each of the irradiation tests. For all of these tests (except RERTR–6 and RERTR–12), the long edge of the fuel plates was oriented towards core center, creating a large gradient in fission/heating rate and fission density from one edge of the fuel zone to the other (up to 2.2 times the plate average). The gradients provided for a more aggressive environment to test fuel performance as the gradients can induce large, non-uniform thermal and mechanical stresses on the fuel/cladding interfaces. Average fission densities, fission rates, and heat fluxes are listed in Table 3. These are different from the peak values that were used to form the basis of requirements in Figure 2 and Figure 3. The same values are given for the AFIP tests in Table 4.

Table 3. Operating conditions of RERTR mini-plate test fuel plates (monolithic fuel).

Fuel Test	Plate Average* Fission Density (Fd), 10^{21} f/cm^3		Plate Average* Fission Rate, 10^{14} f/cm$^3\cdot$s		Plate Avg.* Heat Flux, W/cm^2 (hottest cycle)		Plate Avg.* Volumetric Power, W/cm^3	
	low	**high**	**min**	**max**	**min**	**peak**	**min**	**peak**
RERTR–6	2.8	3.9	2.9	4.2	116	175	118	174
RERTR–7A	1.7	5.6	3.5	9.0	136	327	163	300
RERTR–8	1.5	7.4	4.1	7.8	164	382	160	380
RERTR–9A	3.1	6.2	3.6	7.2	130	313	150	260
RERTR–9B	5.4	7.5	5.5	7.5	219	330	250	330
RERTR–10A	1.3	4.9	1.7	13	140	350	200	385
RERTR–10B	1.7	4.6	3.0	11	275	370	290	415
RERTR–12	0.4	7.7	2.0	4.5	65	343	150	510

*Plate averages negate the often large gradient across the width of the plate in many experiments. The peak/average can be as high as 2.3 (RERTR–9A).

The time at power for each experiment was:

- RERTR–4, 230 days

- RERTR–6, 109 days
- RERTR–7A , 90 days
- RERTR–8, 104.7 days
- RERTR–9A, 98.1 days, RERTR–9B, 114 days
- RERTR–10A, 82 days, RERTR–10B, 48 days
- RERTR–12 42-149 days.

Table 4. Operating conditions of AFIP test fuel plates.

Fuel Test	**Plate Average Fission Density (Fd), 10^{21} f/cm^3**	**Peak Fission Rate, 10^{14} f/cm^3·s**	**Peak Heat Flux, W/cm^2 (hottest cycle)**
AFIP–2	5.5	12.0	360
AFIP–3	4.0	11.0	351
AFIP–4	3.0–4.5	10.0	315
AFIP–6	3.25	13.0	520
AFIP–6 MkII	4.0	11.8	571
AFIP–7	2.5	12.3	251

Figure 6 and Figure 7 show the conditions tested (surface heat flux and power density) by interlayer type overlaid with the required USHPRR operating envelope. Thin plates used in recent ATR conceptual fuel-element designs and the thin edges used in the HFIR fuel conceptual design result in high local peak volumetric powers (Figure 6). These powers are outside the current range of fuel test conditions. It should be noted that the AFIP–6 MkII irradiation test reached a peak volumetric power of 35,000 W/cm^3, a peak surface-heat flux of 570 W/cm^2, and a peak fission density of 4.3E+21 f/cm^3. The AFIP–6 MkII post-irradiation examination has not been completed and is not represented in the plots. Initial indications from in-canal ultrasonic inspection and visual examination indicate that the fuel plates performed well under irradiation.

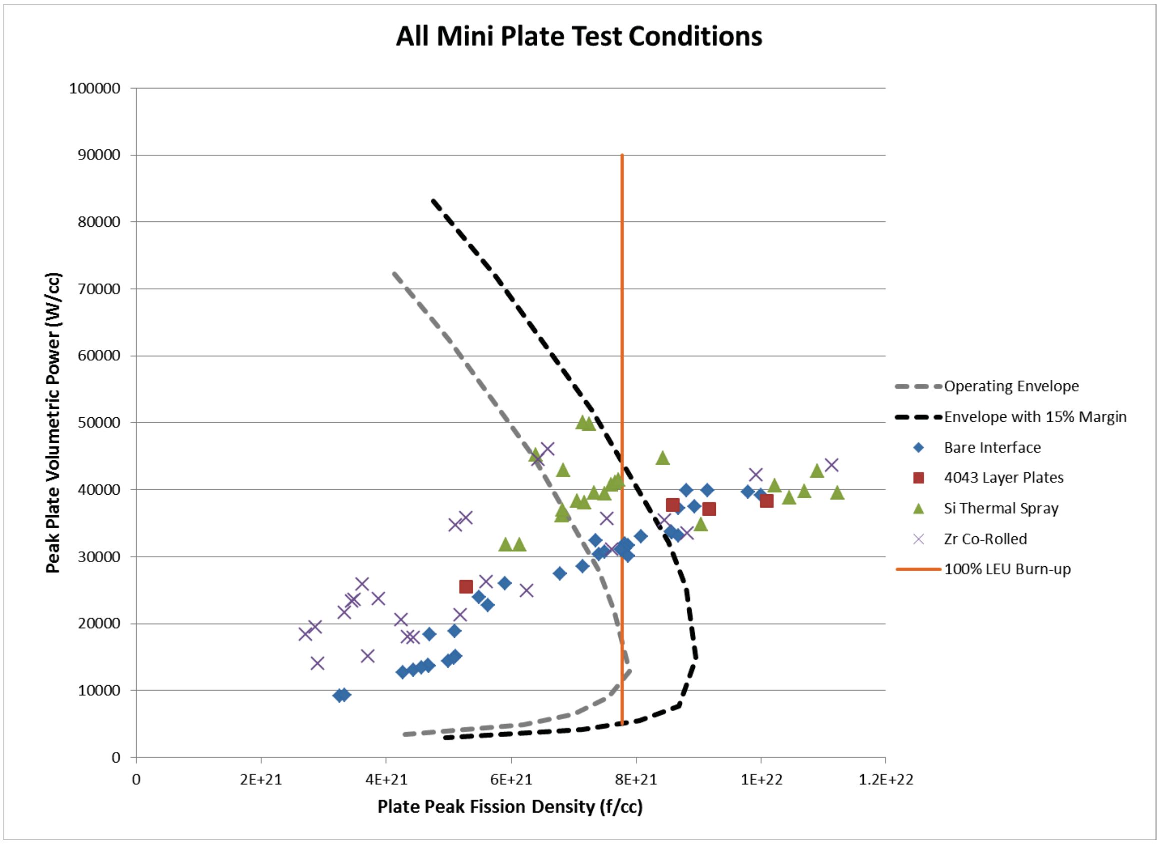

Figure 6. Mini plate test conditions. Peak volumetric power (W/cm^3) vs peak fission density overlaid with required USHPRR operating envelope. The envelope includes HFIR and ATR operating conditions.

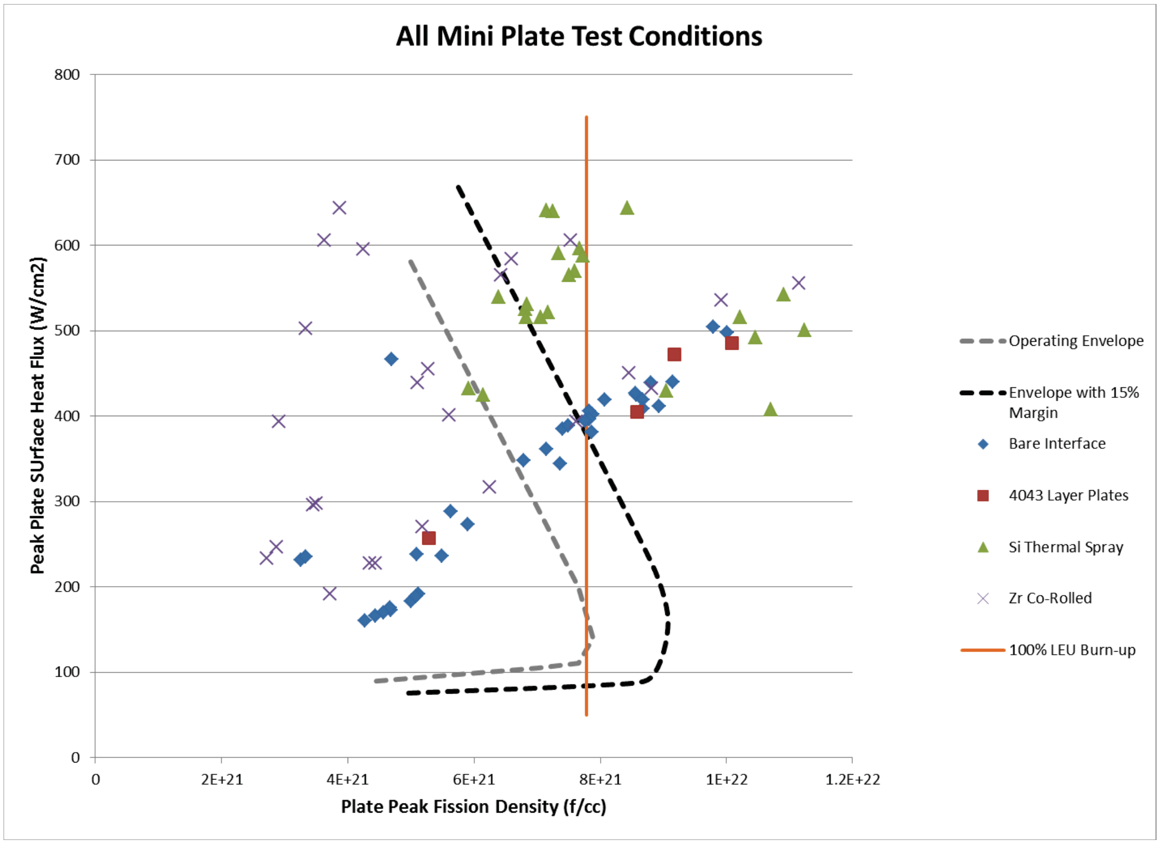

Figure 7. Mini plate test conditions. Peak surface heat flux (W/cm^2) vs peak fission density overlaid on required USHPRR operating envelope. The envelope includes HFIR and ATR operating conditions.

6.2 Summary of Irradiation Testing Results

6.2.1 Initial Experiments: RERTR–4, 6 and 7

Following the limited but encouraging results from the monolithic fuel tested in RERTR–4, the results from the RERTR–6 and 7A demonstrated more completely the fundamental operating characteristics and attributes of U–Mo fuel plates with foil-type fuel meat. [29,30,31]

RERTR–6 provided information that U–7Mo exhibited more pronounced reaction with cladding than did U–10Mo during both fabrication and irradiation. It also provided data that indicated higher swelling in the lower-Mo alloy for foils of the same thickness (see Figure 8) while swelling in the thicker U–10Mo foils was smaller, it was in proportion to the higher uranium loading.[30]

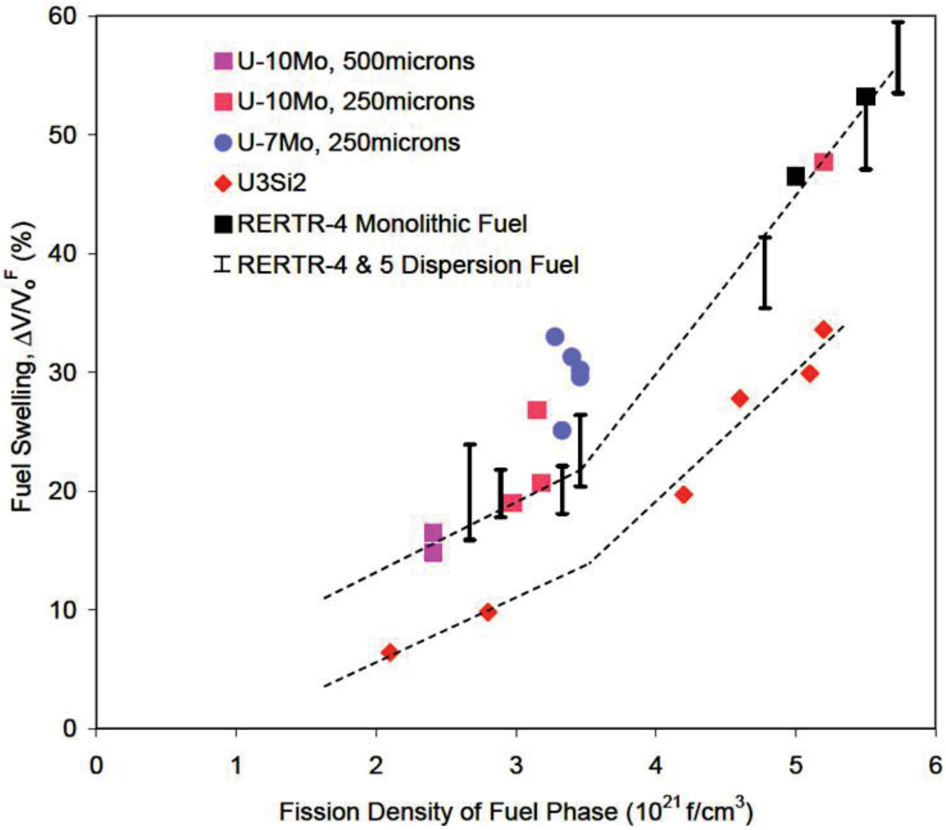

Figure 8. Fuel swelling of RERTR–6 U–Mo foils compared to dispersion fuels from RERTR–4 & 5[32]. Note that 250 micron U–7Mo exhibits higher swelling than plates fabricated with both 250 and 500 micron U–10Mo foils.

The thickness of the interaction layer between the fuel and 6061 aluminum cladding is a key differentiator between the two fuel alloys. The IL of the U–7Mo plates was roughly twice that of the U–10Mo plates (~4–6 µm vs. ~3 µm, respectively).[30] Because the IL contains brittle phases, it was inferred that the IL should be minimized to prevent a potential debonding of the fuel/clad interface if cracks ran along the interface. Figure 9 shows bubbles formed in the interaction layer of a U–10Mo fuel plate in comparison to the unreacted U–10Mo foil. It is assumed this coalescence would eventually lead to delamination of the fuel plate.

Swelling of the unreacted U–Mo fuel foil does not seem to vary significantly as a function of the alloy compositions tested at RERTR–6 and 7 conditions.[33] This is supported by the bulk fuel-swelling correlations developed through analysis of data[34]. There is very little difference predicted in the bulk swelling of U–7Mo and U–10Mo at research-reactor-fuel operating temperatures. More recent swelling models have removed the dependence of Mo concentration. Assuming the same swelling rates for both U–10Mo and U–7Mo alloys, the swelling difference noted in Figure 8 is related to interaction and bubble formation at the fuel/clad interface.

In addition to the aluminum-clad plates, Zircaloy-clad U–7Mo plates, fabricated in Argentina, were included in the RERTR–7 test and performed well to a relatively low fission density (about half of other plates in the RERTR–7 test due to comparatively low enrichment). The integrity of the fuel/clad interface appeared very stable under irradiation (i.e., the post irradiation examination revealed very little interface reaction). This result supported the decision to use Zr barriers to prevent the formation of detrimental high-aluminum interaction layers between the fuel and cladding.

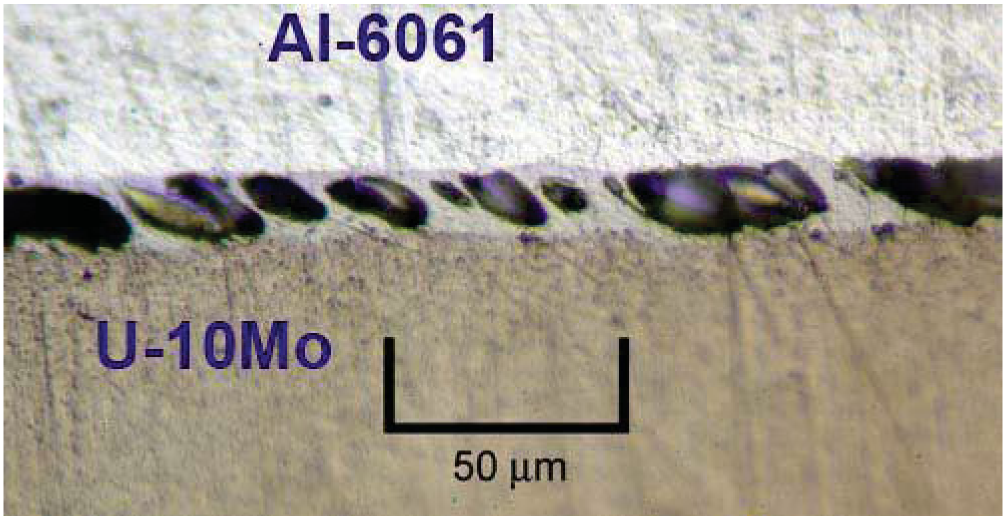

Figure 9. Bubble formation in the interaction layer between a U–10Mo Foil and Al–6061 in plate L1F120 (7E+21 f/cm^3).

In addition to the FB plates, the RERTR–7 experiment included fuel plate fabricated by TLPB. In this process, silicon is applied to the interface and then hot-pressed at a temperature above the Si-Al eutectic temperature of 577°C to form a bond between the fuel and cladding. This process inherently produces an IL at the fuel/cladding interface. U–10Mo and U–12Mo were successfully fabricated for the RERTR–7 experiment using the TLPB process. The U–12Mo plates appeared to swell less than the U–10Mo alloy [35], but the only U–10Mo plate bonded by TLPB that was included in the experiment delaminated during irradiation, so the comparison was made to U–10Mo plates bonded by FB. The observed debonding/delamination of the aluminum-to-aluminum rail region and subsequent release of fission product led to elimination of the TLPB process from the monolithic fuel irradiation-testing program.

The fuel/clad interface in several plates irradiated in the RERTR–7 experiment performed poorly—delamination was observed during post-irradiation sectioning of fuel plates—as shown in Figure 10 and Figure 11. As a result, fabrication FB process improvements were incorporated in later experiments (RERTR–9) and combined with interface modifications to increase the resistance to delamination of the fuel plate.

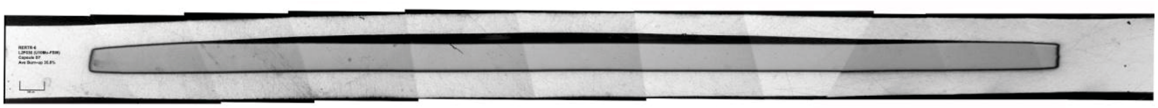

Figure 10. Cross section of friction bonded U–10Mo plate showing fuel/clad delamination (L2F030 average fission density 2.4E+21 f/cm^3).

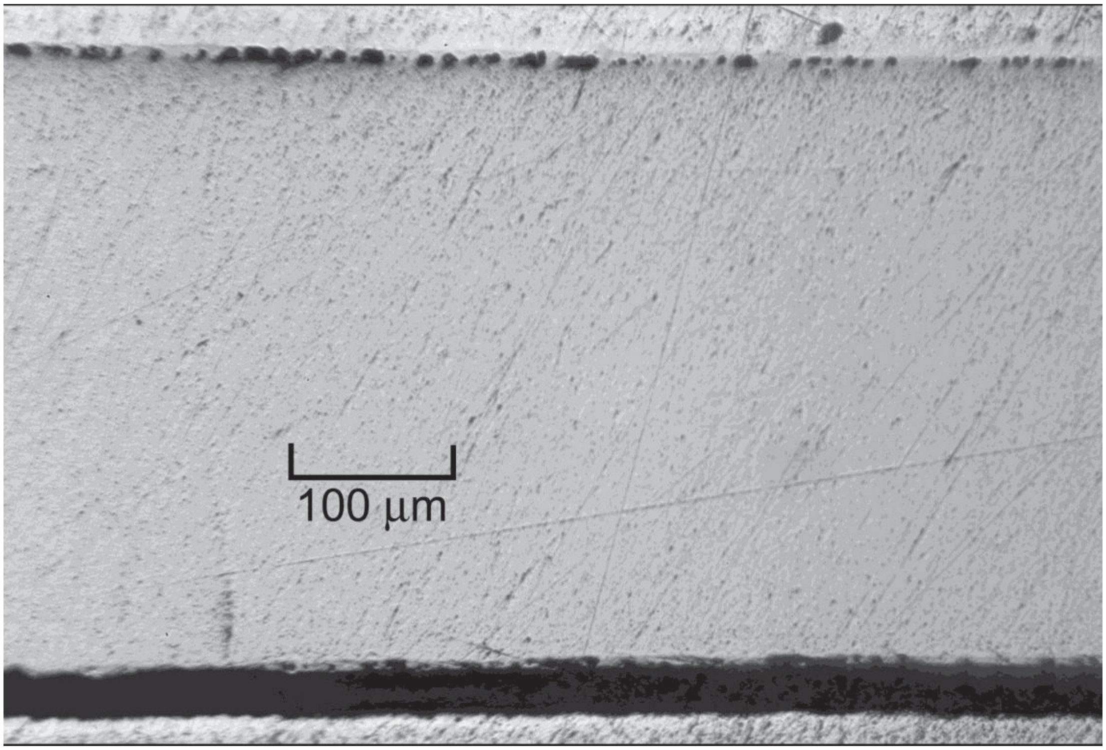

Figure 11. U–10Mo friction bond plate that demonstrated fuel/clad delamination (L1F140, average fission density: 5.15E+21).

6.2.2 RERTR–8: Testing of Hot Isostatically Pressed Fuel Plates

To reduce swelling and prevent debonding, minimization of the (U–Mo)AL_x IL was a primary goal. The RERTR–8 test focused on modifying the IL using Mo concentration in the foil alloy (U–8Mo, U–10Mo and U–12Mo) or by preventing IL formation through modification of the interface chemistry. The RERTR–8 experiment expanded the fabrication options that were tested for monolithic fuel plates so as to include foil barriers (Al–4043), spray-coating barriers of Si-Al (applied to the cladding prior to fabrication), and the HIP bonding process. The HIP processing was added to provide another fabrication option capable of producing a highly reliable, strong bond between the fuel and cladding.

Swelling and debonding were of particular interest in the post-irradiation examination of the monolithic fuel plates in the test, which also contained dispersion fuel plates. The swelling of the monolithic fuel plates, as measured by plate-thickness change, is shown in Figure 12. While the swelling levels are very similar, and trend fairly consistently with the dispersion fuel plates, the alloy with the lowest Mo concentration, U–8Mo, showed the highest swelling rate. Metallography on a plate with U–12Mo fuel showed no large bubbles/porosity at the fuel/cladding interface, although some smaller, non-uniform bubbles were observed near the high-power edge of the plate (the power varies by a factor of 1.9 across the plate width as one edge faces the ATR core). This is shown in Figure 13.

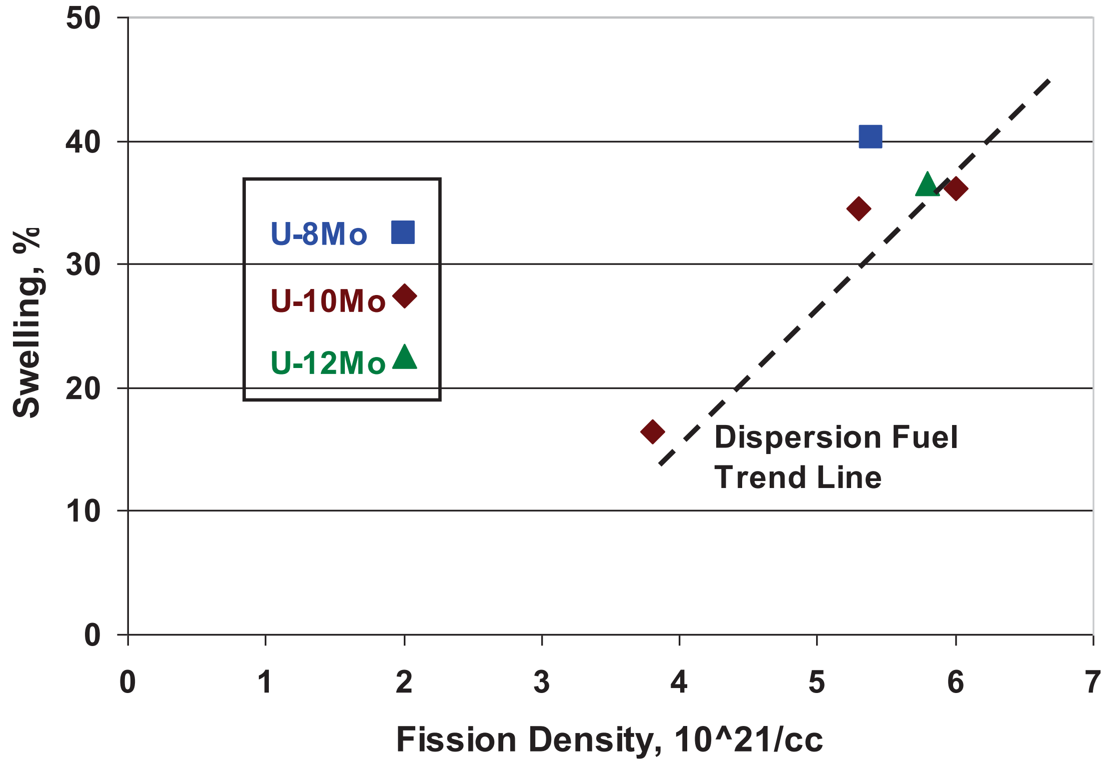

Figure 12. RERTR–8 monolithic fuel swelling as measured by plate-thickness change, compared to U–Mo dispersion fuels.[36]

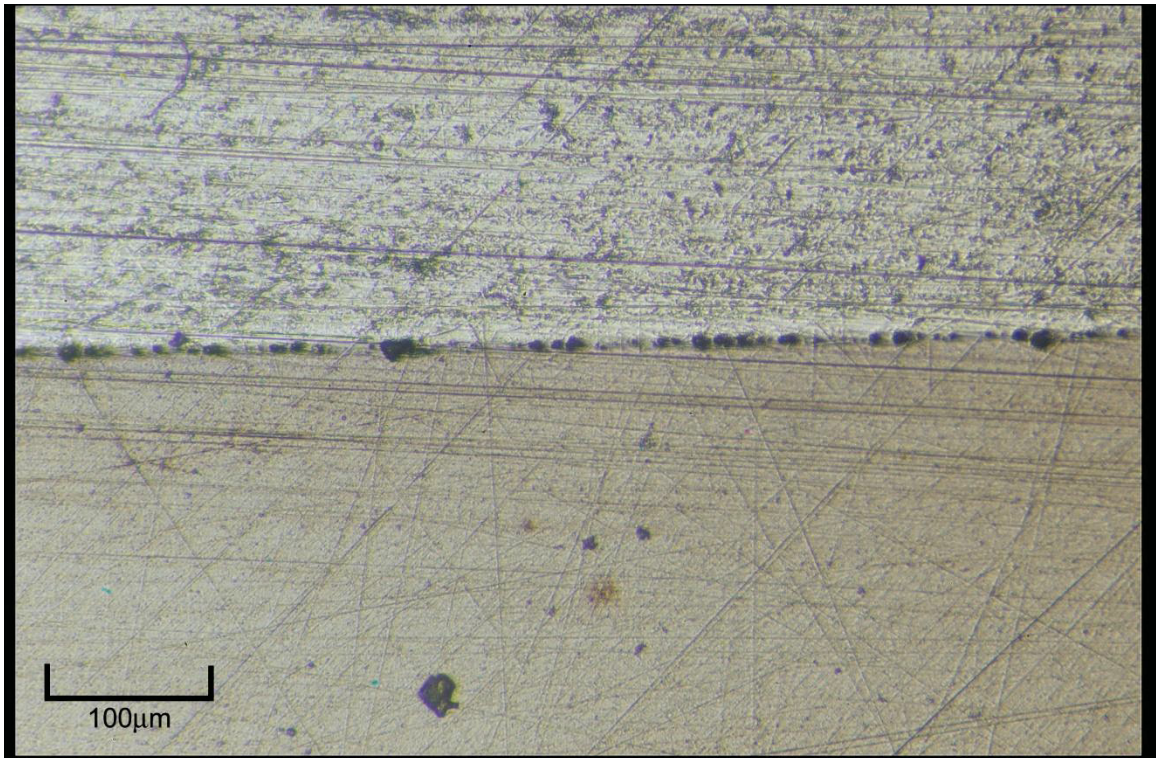

Figure 13. Optical image of metallographic section of monolithic fuel plate from RERTR–8 showing porosity at U–Mo/Al interface (U–12Mo, Al–6061 cladding; H1P010, average fission density: 5.8E+21).

At equivalent fission densities U–10Mo and U–12Mo plates from RERTR–7 exhibited higher swelling rates than the RERTR–8 plates.[37] Note that, while taking into consideration the difference in magnification, the bubbles at the interface in the RERTR–8 plate in Figure 13 (f_d~8.8E+21 f/cm^3, HIP bonded) are much smaller than those in the interface of the plate from RERTR–7, shown in Figure 9 (f_d~8.1E+21 f/cm^3, FB bonded). Although these plates contain different fuel alloys, the swelling was hypothesized as being sensitive to the fuel/cladding bonding method, which impacts the condition of the fuel/cladding interface prior to irradiation. More emphasis was therefore placed upon reducing the impact of fabrication on the interface characteristics, particularly the thickness of the interaction layer.

Fuel with lower molybdenum content was found to be even more sensitive to fabrication variables. A HIP-parameter study showed the instability of fuel foils with lower molybdenum concentration under heat treatments associated with the fabrication processes. Figure 14 shows the results of a 3 hour, 580°C HIP cycle when using both U–7Mo and U–8Mo fuel foils. The U–7Mo foil completely reacted, and the U–8Mo foil almost entirely reacted. Figure 15 shows similar treatments to U–10Mo/Al fuel plates, with much less reaction.

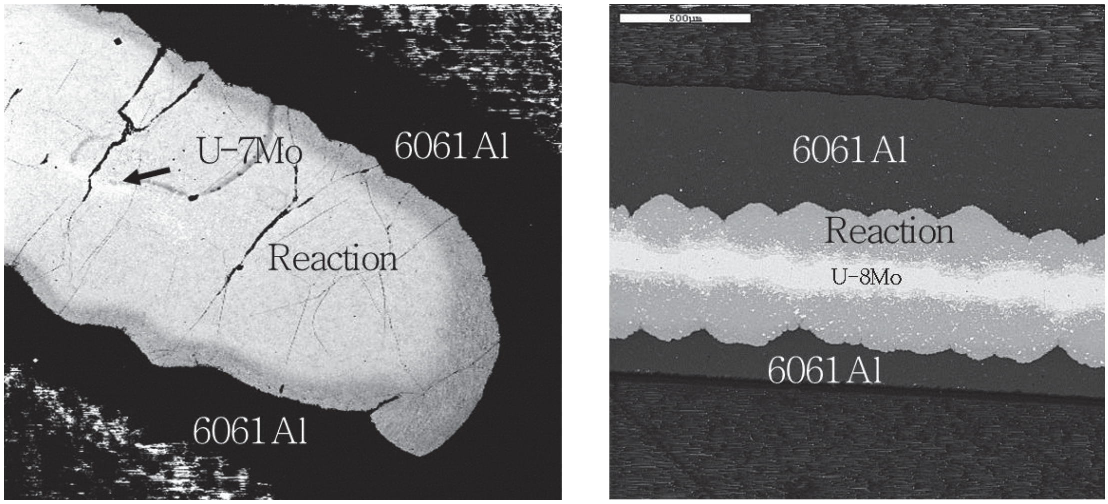

Figure 14. SEM images from fabrication parameter study after HIP'ing (U-7Mo left and U-8Mo right)

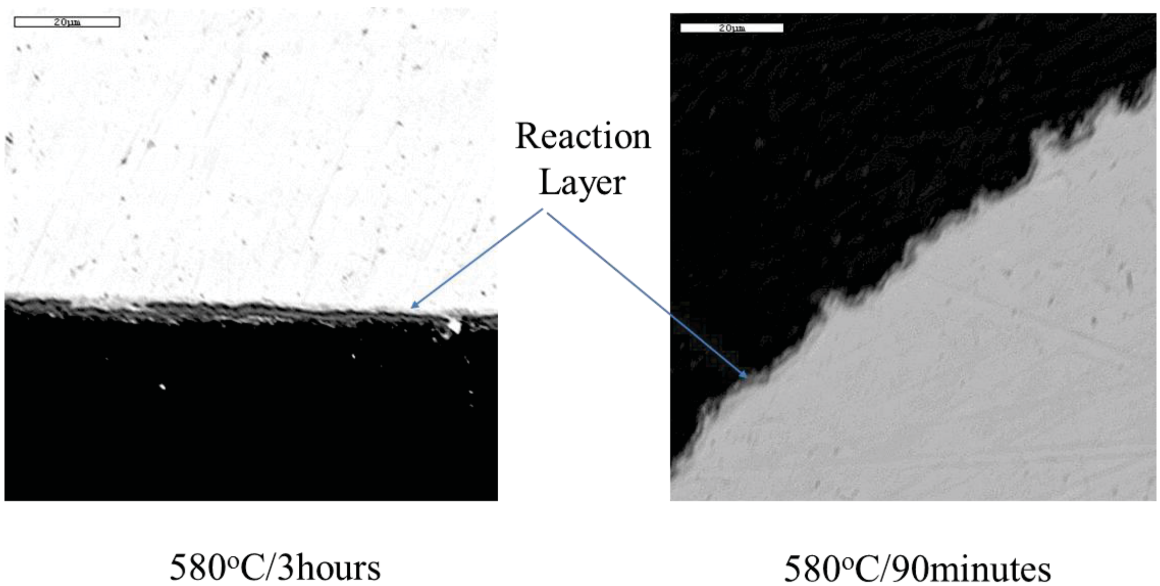

Figure 15. SEM images from fabrication parameter study after HIP of U–10Mo.

HIP bonding was found to be more effective than friction bonding in providing a reliable interfacial bond,[38] but the process required heating to temperatures which led to the formation of significant interaction layers. In order to minimize the formation of the IL formation during the long HIP cycles required for large-plate production, cladding barrier layers were introduced into the fuel-plate design.

6.2.3 Fuel-to-Cladding Interdiffusion Barriers

Barrier layers were developed to control fuel/cladding interaction during fabrication and interface degradation during irradiation. Fabrication development efforts were initiated to investigate several methods to construct fuel plates designed with a barrier material. Potential barrier materials were studied within the context of the previously developed fuel-plate bonding methods (FB and HIP).

The HIP bonding process had previously been found to provide, qualitatively, the strongest bond for as-fabricated test plates (which did not contain barrier layers).[39] HIP bonding also provided more consistent fabrication results. A detailed study of the interfacial bond between the fuel and clad was performed to evaluate Zr, Nb, Ta, C, Si, and Ti barriers.[40] Samples with Zr, Ti and Nb barriers inserted between the foil and cladding and then HIP bonded demonstrated a strong bond between the barrier layer and the cladding, but not between the barrier layer and the fuel. A carbon interface did not bond well to either surface.

Roll-bonding the diffusion barrier (Zr or Nb) to the fuel foil, followed by HIP bonding, resulted in a consistent and qualitatively strong bond between the fuel, barrier, and cladding. Following these demonstrations, Zr, Nb and Ti were considered as viable candidates for an interface barrier. Friction bonding was also shown to produce an acceptable bond between the fuel/barrier foil and the cladding, but these results were somewhat operator dependent and would require considerably more process development to achieve consistent results[10]

6.2.4 Testing of Primary Design Options

6.2.4.1 RERTR–9 Irradiation Test

Additional testing of monolithic fuel plates with improved fuel/cladding interfacial bonds and barrier-modified interfaces (that would ideally not collect fission-gas bubbles during irradiation) was the focus of the RERTR–9 experiment. Fuel plates were fabricated with Zr barrier layer and high Si modified interface. FB and HIP fabrication techniques were used to bond the fuel plates.

Although friction bonding was found in some of the previous testing to produce insufficient bond strength, a fabrication-process modification involving the use of a different tool material (the tool rotates against the surface of the plate to create material flow and bonding), was employed for these tests. The new tool material was shown to produce a significantly stronger as-fabricated fuel/clad bond, demonstrated by pull testing as-fabricated specimens[10,41,42]. Two different tool materials were used to fabricate test plates for the RERTR–9 experiment.

Because the U–10Mo fuel alloy provided a good balance between ease of fuel fabrication and fuel performance, the RERTR–9 experiment contained only fuel plates using the down-selected U–10Mo fuel alloy. Design variables tested included the addition of silicon to the fuel/clad interaction layer and the insertion of a Zr diffusion barrier. Silicon was expected to stabilize the IL fission-gas behavior as it had done in the U–Mo dispersion fuel system. Silicon was incorporated into the experiment through the insertion of an Al–4043 (~5% silicon) interface foil and through the application of a plasma-sprayed Si layer to the interface between the fuel and cladding prior to bonding.

The RERTR–9 test differentiated between these experimental fabrication methods. The FB tool-material change did not provide a sufficiently more stable bond when tested in-reactor; several plates showed evidence of delaminating during irradiation and/or subsequent post-irradiation sectioning (although none released fission product during irradiation), as shown in Figure 16.[42] Plates with an interface sprayed and bonded with Si also showed a tendency to delaminate at the interface, as shown in Figure 17. A thicker Si spray layer before bonding seemed to promote delaminating. A thinner layer had a stronger bond, but a brittle interaction layer was discovered upon post-irradiation examination.[42]

Figure 16. Anvilloy-tool friction-bond plate (L1F32C, average fission density: 5.9E+21 fissions/cm^3).

Figure 17. Silicon thermally sprayed interface on friction-bond plate (L1F37T, average fission density: 7.4E+21fissions/cm^3).

Al–4043 which, as mentioned above, contains ~5 wt% Si, was used to facilitate the formation of an IL rich in silicon, with an expectation of improved irradiation stability, as demonstrated in dispersion fuels using a Si-bearing matrix material. Examination of several test plates, however, showed that the inclusion of Al–4043 foils at the U–Mo/cladding interface had a negative effect on bubble formation (see Figure 18). In fact, the plate examined, which had Al–4043 at one interface and only Al–6061 clad at the other, showed that the bubbles were noticeably larger at the interface that contained the Al–4043 layer (see Figure 19).[42]

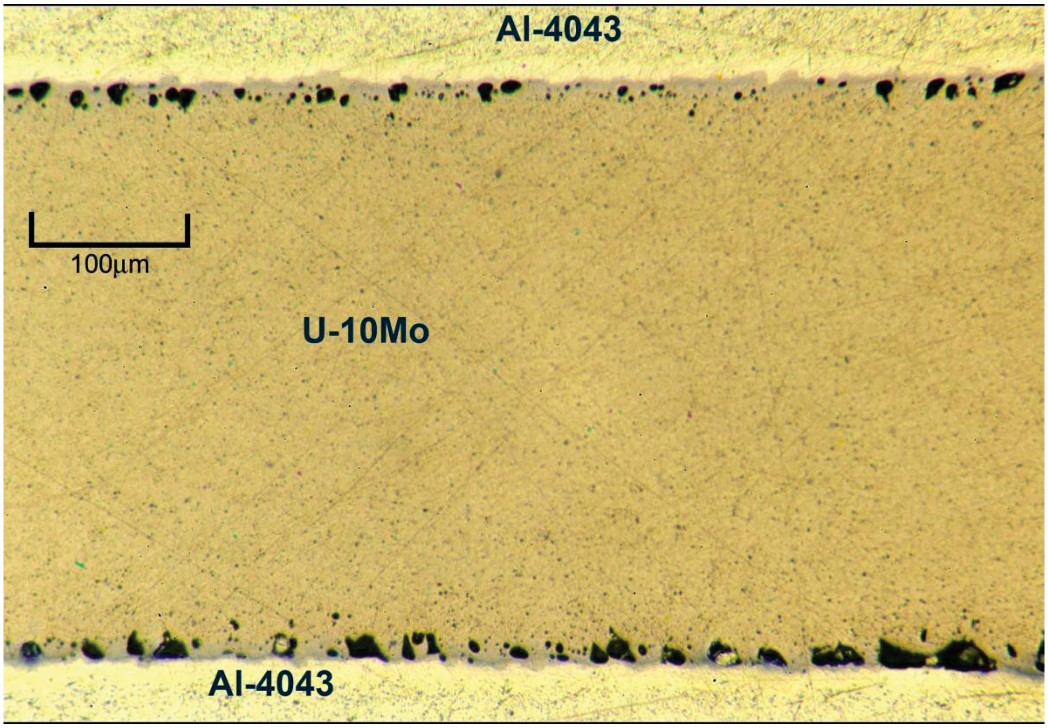

Figure 18. Optical micrograph of RERTR–9 test plate cross-section, which had an inner Al–4043 clad interface (L1P05A, average fission density: 6.1E+21 fissions/cm^3).

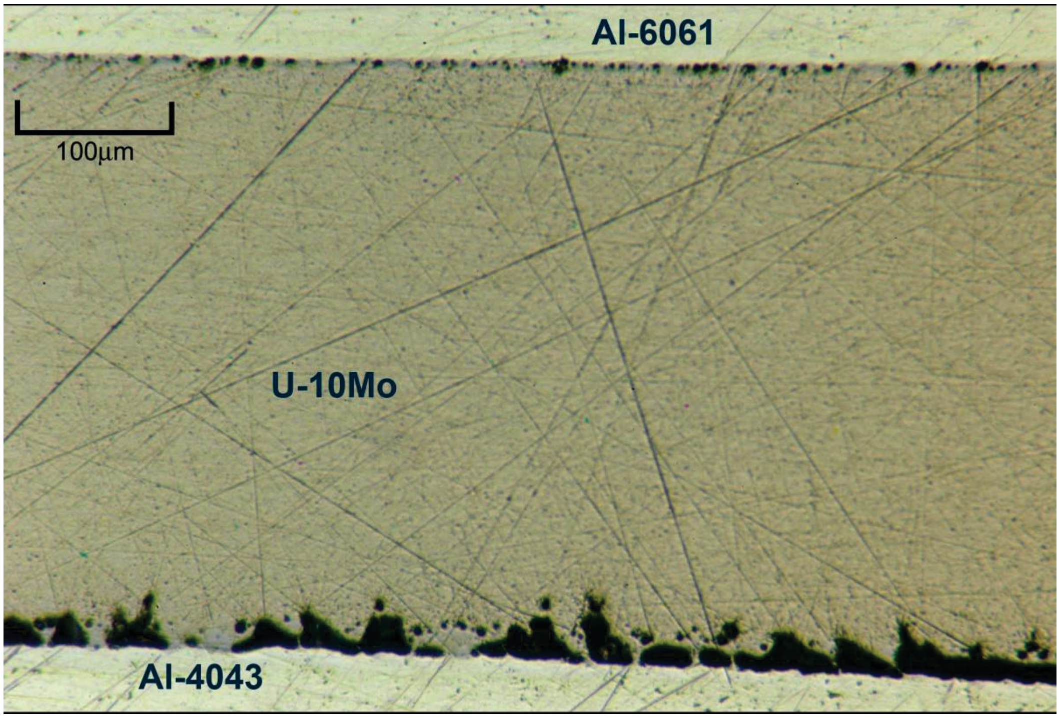

Figure 19. Optical micrograph of RERTR–9 Test plate cross-section, which had an inner Al–4043 clad interface on one surface (lower edge in photo) and Al–6061 on the other (L1P04A, average fission density: 5.3E+21 fissions/cm^3).

Fuel plates that included a zirconium barrier layer between the U–Mo fuel and the Al–6061 cladding exhibited good irradiation behavior. The Zr foil was applied by hot co-rolling in a steel pack with the U–Mo fuel foil prior to the HIP process. Figure 20 shows an optical micrograph of the Al–6061/Zr/fuel/Zr/Al–6061 interfaces following irradiation. The bond remained intact (on both the fuel and clad side), and fission gas bubbles did not form at the U–Mo/Zr interface.

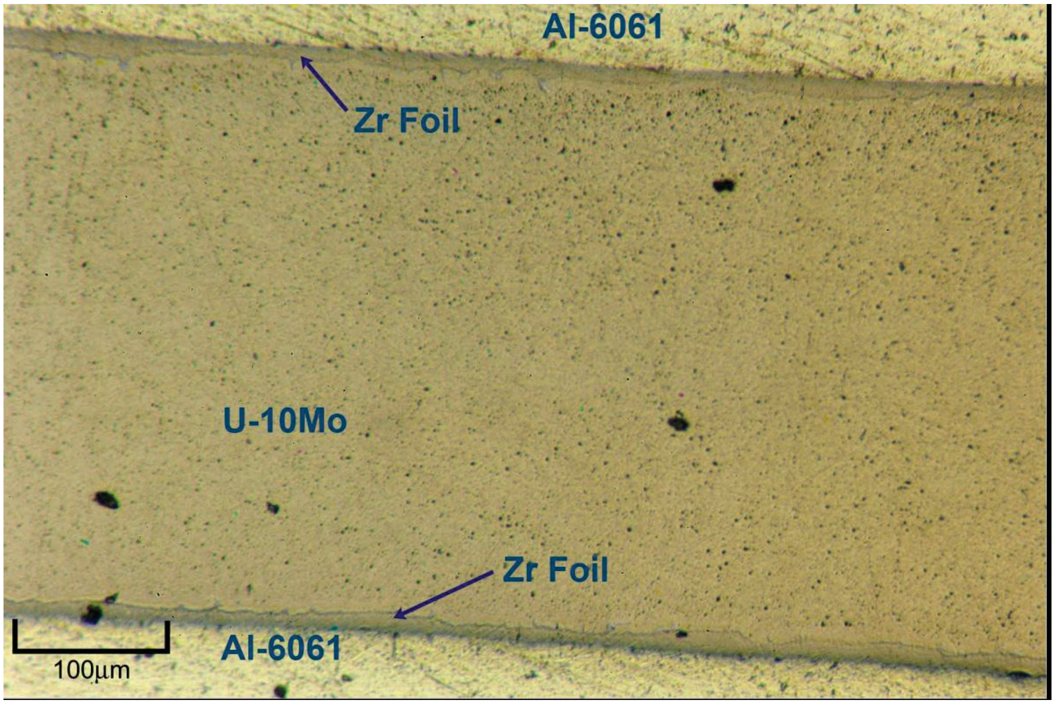

Figure 20. Optical micrograph of an RERTR–9 fuel plate cross-section (L1P09T, average fission density: 7.6E+21 fissions/cm^3).

As shown in Figure 21, the hot side of the fuel plate (local fission density in region of interest 9.0E+21 f/cm^3) developed cracks in the width of the fuel, which may have been caused by rapid cooling of the plate on shut-down of the reactor or during post-irradiation sectioning of the fuel plate. Had the interfaces been weak or very brittle, or had the bond between the layer and the fuel or Zr foil been weak, these cracks would likely have propagated along the interfaces in ways similar to the cracking patterns observed in RERTR–6 and 7 fuel meats. Figure 21 indicates that they did not propagate through the layer or along the interface; instead, the cracking has occurred in the fuel foil, which was saturated with fission-gas bubbles and was likely quite brittle.

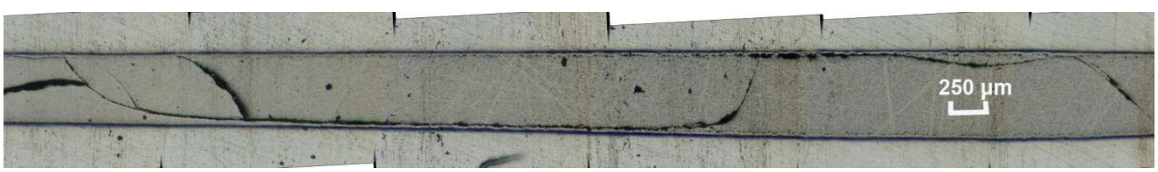

Figure 21. Optical micrograph of an RERTR–9 fuel plate cross-section ('hot' side). The fuel foil has cracked, but the cracks do not propagate along or through the fuel/Zr boundary (L1P09T, plate average fission density: 7.6E+21 fissions/cm^3).

The HIPed U–10Mo plate with a Zr foil barrier seemed to perform reliably with a stable fuel/cladding interface. The conditions for this fuel plate in the RERTR–9 test were very challenging, with an average heat flux near 300 W/cm^2 and a fission density of ~7.5E+21 fissions/cm^3, near the bounding conditions stated in the requirements section. Two plates of this design were tested with consistent results.

6.2.4.2 RERTR–10A and B Irradiation Tests

The RERTR–10 experiment included two reactor insertions. The RERTR–10A experiment contained U–10Mo-based monolithic fuel plates that were bonded using HIP exclusively. The RERTR–10B experiment contained monolithic plates bonded using FB. Zirconium-foil interlayers were included as variables in both tests. A niobium-foil barrier was tested in the RERTR–10B insertion. Several fuel plates were also fabricated to continue to investigate the use of Si at the fuel/cladding interface. The variations in thermal spray included pure Si, and 1, 2, 3.5, 5, and 12 wt% Si in Al. A pre-reacted eutectic material was used for the 12%Si coating; the other coatings were blends of Al and Si elemental powders. On several plates, spray coatings of 12 μm and 25μm (one on either side of the same plate) were applied to test the effect of the thickness of the coating in addition to the composition. Fuel foils of standard thickness (0.25 mm) and double thickness (0.51 mm) were included in the fuel plates containing Zr foils, two of each in the RERTR–10A insertion.

Figure 22 shows the average swelling of fuel plates as documented by fuel-plate thickness measurements (sixteen points laid out in a 3(width) × 6 (length) grid over the surface of the plate). Near the ends of the plates, there was often an area with greater thickness caused by creep of the fuel as it experienced multi-dimensional swelling restraint at the sealed ends of the mini-plates; these data points were disregarded in calculating the averages shown in

Figure 22.

Table 5. RERTR–10 plate information.

Plate ID	Plate Type	Plate Average Fission Density	Fuel Swelling	Notes
L1P30Z	HIP Zr Co-Roll	3.2E+21	0.2097	Not Sectioned
L1P256	HIP Si Thermal Spray	2.88E+21	0.2038	Not Sectioned
L2P15Z	HIP Zr Co-Roll	1.5E+21	0.1189	Not Sectioned
L1P135	HIP Si Thermal Spray	3.42E+21	0.2509	Not Sectioned
L1P234	HIP Si Thermal Spray	3.94E+21	0.762	Pillowed
L1P213	HIP Si Thermal Spray	3.3E+21	0.205	Not Sectioned
L1P192	HIP Si Thermal Spray	3.39E+21	0.219	Not Sectioned
L1P171	HIP Si Thermal Spray	4.15E+21	0.857	Pillowed
L1P12Z	HIP Zr Co-Roll	4.5E+21	0.2743	Acceptable behavior
L1P266	HIP Si Thermal Spray	4E+21	0.3	Not Sectioned
L2P16Z	HIP Zr Co-Roll	2.12E+21	0.093	Acceptable behavior
L1P145	HIP Si Thermal Spray	4.9E+21	1.92	Corrosion Failure
L1P244	HIP Si Thermal Spray	4.67E+21	4.16	Pillowed
L1P233	HIP Si Thermal Spray	3.9E+21	0.2599	Delaminated on sectioning
L1P202	HIP Si Thermal Spray	3.75E+21	0.907	Pillowed
L1P181	HIP Si Thermal Spray	4.78E+21	3.1	Pillowed
L1F44N	HIP Nb Co-Roll	4.34E+21	0.382	Acceptable behavior
L1P381	HIP Si Thermal Spray	4.33E+21	0.358	Delaminated on sectioning
L1F401	HIP Si Thermal Spray	4.58E+21	0.418	Not Sectioned
L1F417	HIP Si Thermal Spray	4.61E+21	0.341	Not Sectioned
L1F427	HIP Si Thermal Spray	4.47E+21	0.6	Not Sectioned
L2F45Z	HIP Zr Co-Roll	2.19E+21	0.086	Acceptable behavior
L2F46Z	HIP Zr Co-Roll	2.25E+21	0.095	Not Sectioned
L2F47Z	HIP Zr Co-Roll	1.75E+21	0.095	Not Sectioned

Table 5 gives pertinent information about plates in RERTR–10. With the exception of the 12% Si eutectic coating, plates with a thermal-spray interface showed instances of significant increase in thickness, with delamination likely contributing to the thickness increase in all cases. Of note is that all silicon thermal-sprayed plates that did not pillow (either under irradiation or upon sectioning) were not sectioned. Had these plates been sectioned, indications of delamination are assumed to be likely.

The plates using the Zr foil barrier showed smaller increases in thickness. Although not labeled, the two Zr barrier plates showing lower swelling were the two with thicker fuel meats.

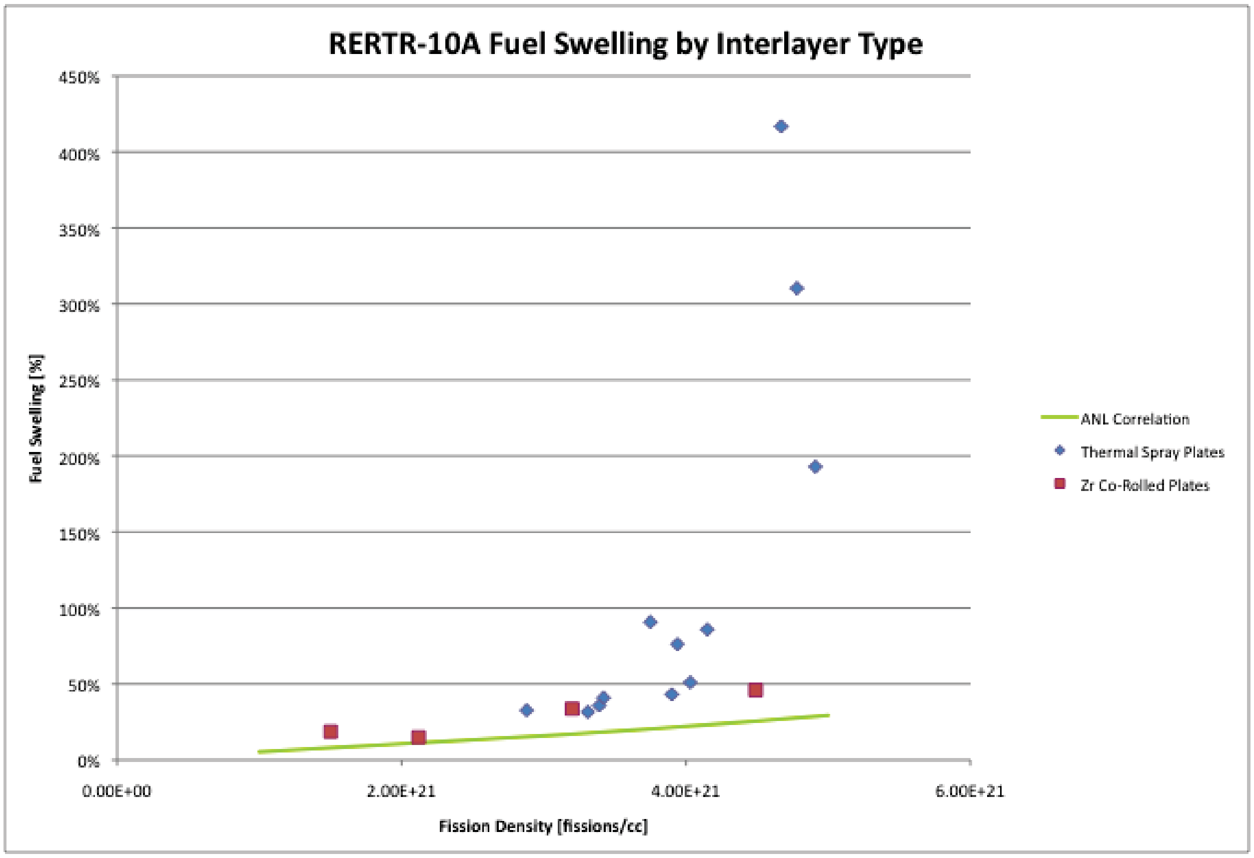

Figure 22. Calculated fuel swelling of RERTR–10A fuel plates. Note several thermal spray plates delaminated or had interface issues resulting in higher than expected swelling values.

The metallographic examinations showed that the thermal-sprayed plates typically had areas of delamination or, in some cases, nearly complete debonding. The implications are that the silicon-enhanced interaction layer may produce layers resistant to fission-gas bubbles, but the fuel/clad bond is either too weak in the as fabricated condition or degrades unacceptably during irradiation, ultimately allowing delamination.

A plate fabricated with co-rolled Zr foil was also examined. The examination demonstrated that the bond of fuel to Zr foil and Zr foil to Al–6061 cladding remained intact. Figure 23 shows an area near the fuel/Zr barrier after irradiation to a fission density of 7.5E+21 f/cm^3. This image shows an accumulation of fission-gas bubbles (significantly larger than those observed in the bulk of the fuel). This image suggests coalescence of these bubbles into a large defect may be one performance-limiting feature in the Zr-barrier fuel design. However, these bubbles are very localized and only seen adjacent to the 'bulge' region of the mini-plates where mechanical stresses are expected to be high. This is not an anticipated condition in full-size plate testing. This hypothesized failure mechanism has not been observed up to a peak fission density of 1.1E+22 fissions/cm^3 based on RERTR–12 data.

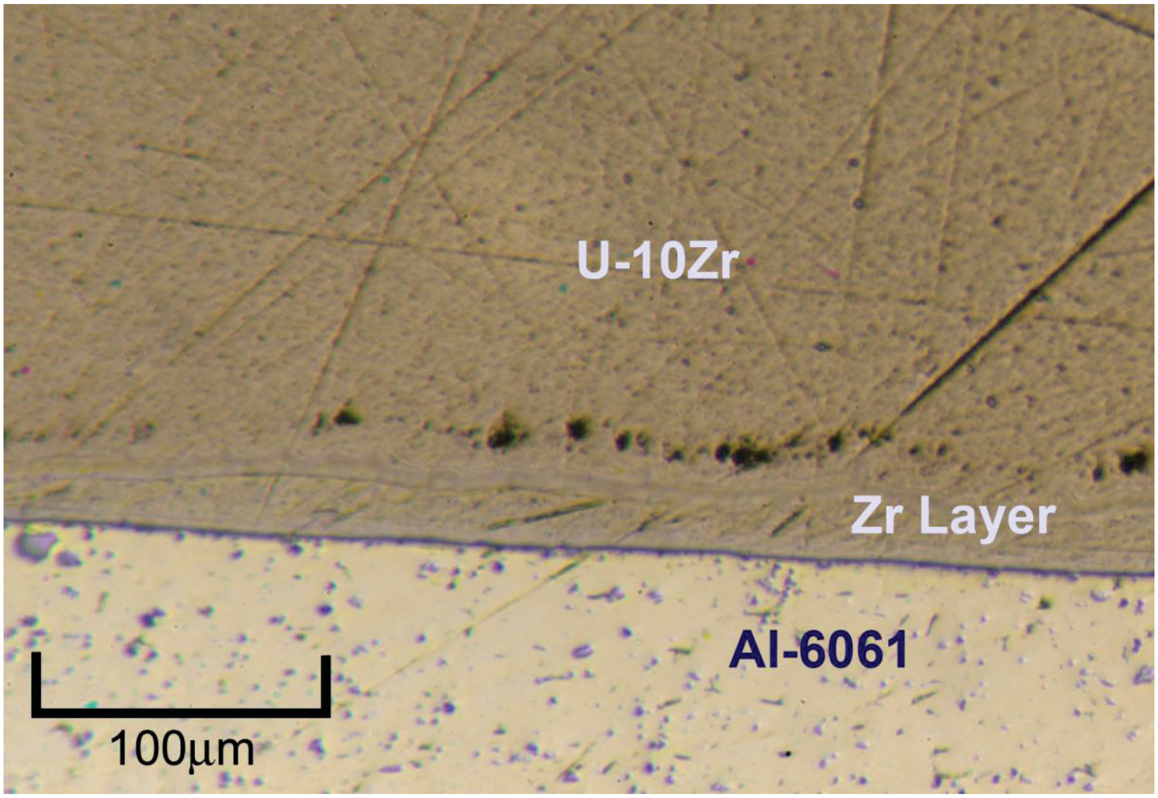

Figure 23. Optical metallography of an RERTR–10A fuel plate, co-rolled with Zr foil barriers and HIP processed (L1P12Z, average fission density: 7.5E+21).

The RERTR–10A test showed that the HIP-bonding process, combined with the Zr-foil barrier design, provided acceptable fuel performance in fuel mini-plates with volumetric peak powers to 36,000 W/cm^3, surface-heat fluxes of 640 W/cm^2, and fission densities of 7.5E+21 f/cm^3. These conditions have not been successfully achieved by any other fuel design.

The RERTR–10B experiment provided further verification that the zirconium diffusion barrier was stable. Although all the zirconium-barrier-layer foils in RERTR–10B were of 500 µm (vs 250 µm) thickness and, therefore, experienced lower burn-up, plate-swelling values are in line with HIP plates and are stable, indicating that delamination did not occur. Calculated swelling values can be seen in Figure 24. It can be seen that all zirconium-coated foil swelling was in line with expected values while the silicon-sprayed plate swelling was higher than predicted. Plate L1F427, which was a plate thermally sprayed with silicon, exhibited notably high swelling, likely due to cracking between the fuel foil and the cladding or delamination of the interface. Metallography was not performed, so verification of this hypothesis cannot be provided at this time.

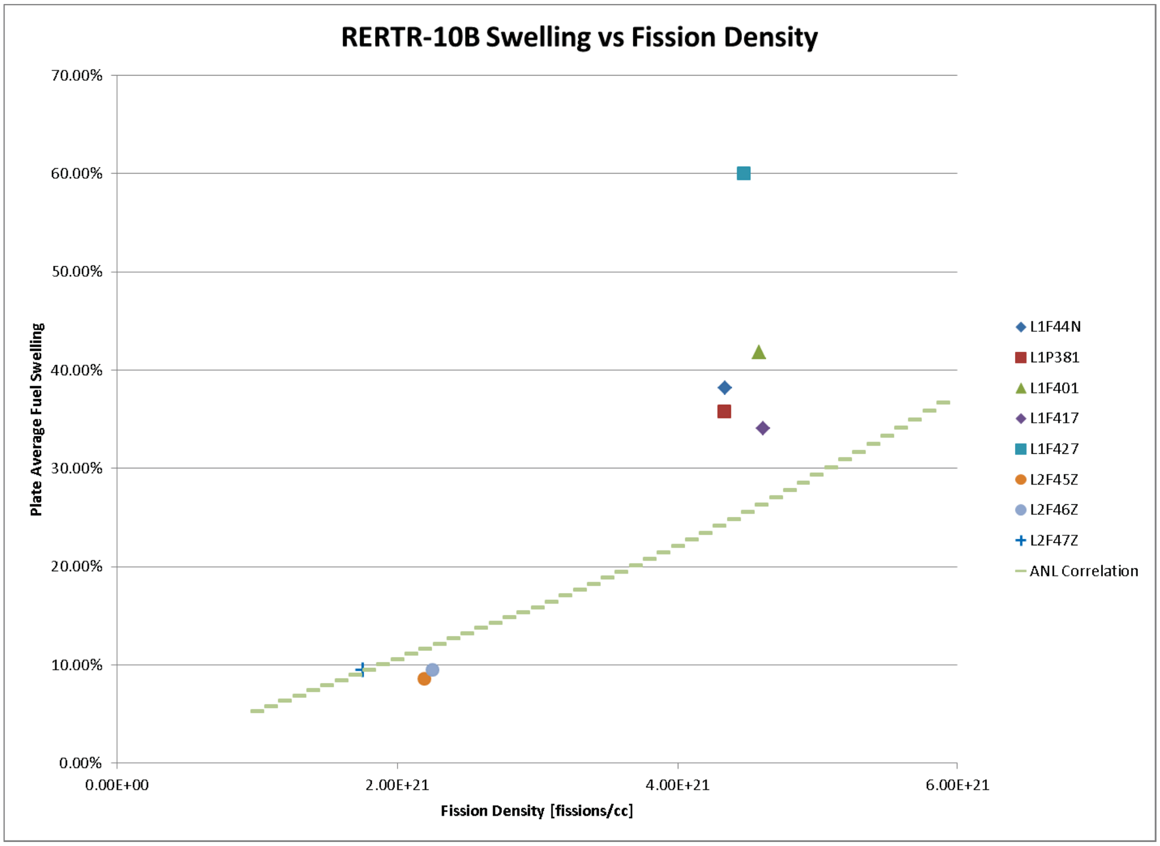

Figure 24. Plate average fuel swelling versus fission density of monolithic U–10Mo RERTR–10B plates and the fuel swelling correlation. L2F47Z, L2F46Z, L2F45Z are zirconium interlayer plates.

6.2.4.3 RERTR–12 Insertion 1

Post-irradiation examinations of the RERTR–12 experiment are currently underway. Results obtained to date from the RERTR–12 first insertion are included here to provide additional information on the irradiation behavior of the zirconium diffusion barrier on U–Mo plates. This document will be revised upon completion of post-irradiation examination of the high-burnup plates that were included in RERTR–12 insertion 2.

Figure 25 shows the plate average fuel swelling (excluding plate L1P754) as a function of average fission density calculated from both thickness measurements and immersion methods. While thickness measurements characterize only one dimension, cladding restrains growth in the plate length and width. The fuel meat creeps as well, so that all swelling is manifested in the plate thickness dimension, and the fractional change in volume is equal to the fractional change in thickness. Both thickness and immersion-density measurements indicate that fuel swelling is stable and predictable to an average fission density of 7.2E+21 f/cm^3. Plate L1P754 (average fission density 8.13E+21 fissions/cm^3, peak fission density 12E+21 fissions/cm^3) pillowed prior to post-irradiation examination. The nature of the pillow, including the crack shape and the fission-gas behavior, indicates that the failure may have occurred upon reactor shutdown.

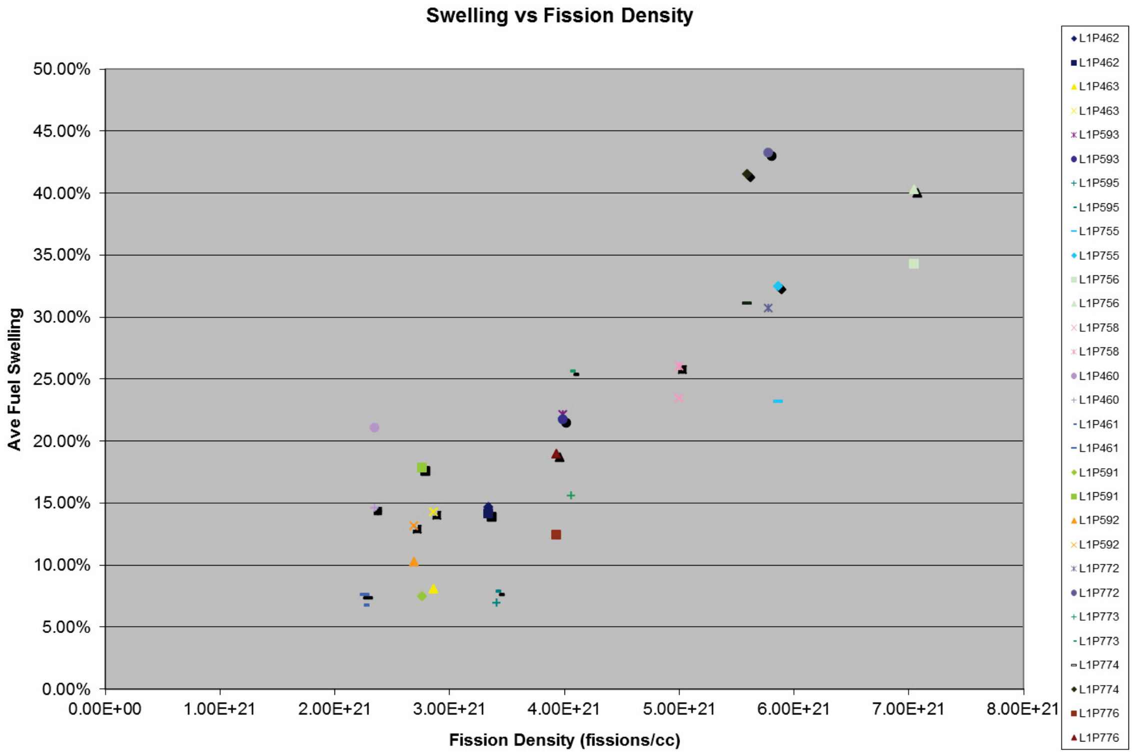

Figure 25. RERTR–12 Insertion 1 fuel-swelling versus fission-density data.

Destructive examinations of plates showed no unusual fuel behavior or interface instability with the exception of L1P754, which exhibited fuel pillowing-type failure at the high-power end of the plate, which accumulated a peak fission density of 1.2E+22 fissions/cm^3. During irradiation, no fission products were detected in the reactor coolant, and examination indicated that the cladding remained intact. A metallographic cross section from the lower-fission-density region of the plate at mid-plane shows no sign of imminent failure, interface degradation, or breakaway fission-gas accumulation, as shown in Figure 26. The mid-plane region of the plate had not pillowed. Figure 27, showing a section from the high-power, pillowed end, indicates that the fuel cracked when the plate pillowed, and that cracks propagated through the regions of the meat saturated with fission-gas bubbles. Figure 28 shows that there is still fuel attached to the zirconium indicating it was not the fuel/zirconium bond itself that failed.

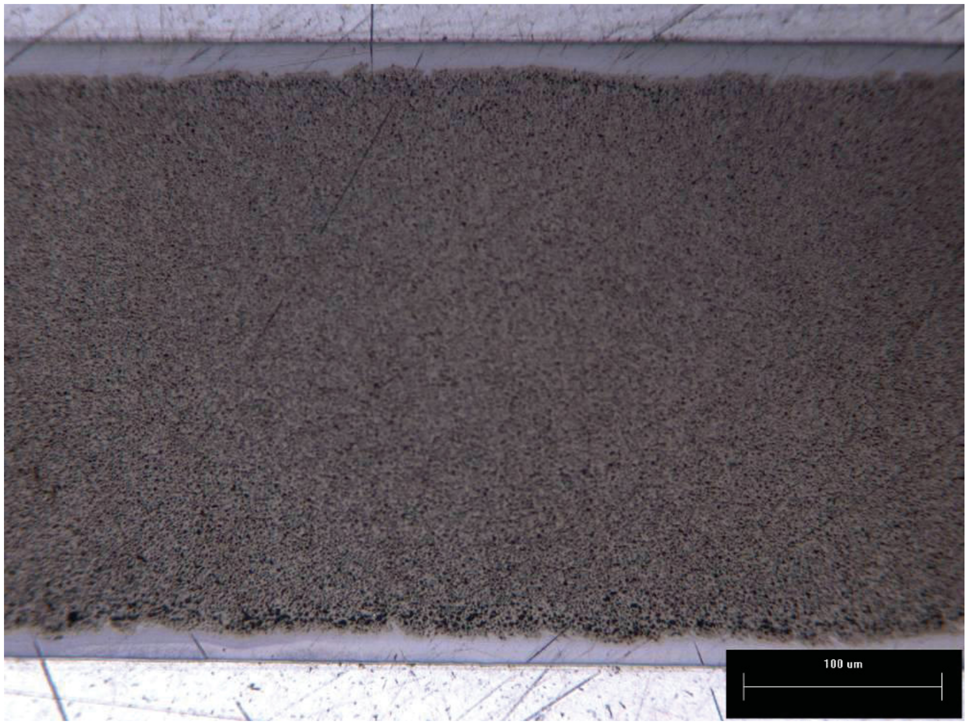

Figure 26. Cross section from mid-plane of L1P754 (fission density of 8.5E+21 fissions/cm^3).


Figure 27. Cross section through failed region of L1P754 (fission density of 9.1–9.9E+21 fissions/cm^3).

Figure 28. Micrograph of interface region of L1P754 (fission density of 9.5E+21 fissions/cm^3).

6.2.4.4 AFIP–2 and AFIP–3 Full-Size Plate Testing

Two irradiation campaigns, AFIP–2 and AFIP–3, were conducted at the prototypic scale (56 mm × 571 mm fuel plates) to ensure that phenomena observed at the mini-plate scale are representative of those expected for fuel plates ultimately placed in service. Tests included both a plate design with a silicon-enhanced fuel/clad interface and a plate with a zirconium diffusion barrier. The AFIP–2 test was fabricated by FB, and the AFIP–3 test by HIP. All four plates showed acceptable dimensional stability (i.e., no plate buckling) during testing conditions that developed prototypic burnup gradients. Fuel-plate thickness measurements following irradiation showed that the overall fuel swelling appeared to be more regular in the Zr-diffusion-barrier fuel plates in both cases than in fuel plates with silicon at the interface. Destructive examinations indicated good interface stability in all four of the plates.

Neither thermally sprayed AFIP plate demonstrated interface failure similar to what was seen at the mini-plate scale. This was likely the result of less-aggressive test conditions as outlined in Section 6.1. In addition to the irradiation conditions, it is currently postulated that the mechanical stresses may be more severe at the mini-plate scale due to non-uniform condition (edge on) and mechanical constraint of the smaller fuel foil. Fuel performance modeling efforts are underway to quantify these effects.

6.2.4.5 AFIP–4 Large-Scale Plate Test

The AFIP–4 experiment was designed to perform fission-gas-release testing on the fuel design. The plates were larger than mini-plate scale (50mm × 190 mm) and provided some information on the stability of the scale-up process to prototypic size. Non-destructive examinations (plate thickness, neutron radiography, eddy current) showed no unusual signs of behavior. No metallography has yet been performed on any of these plates.

Ultrasonic testing (UT) was performed on the AFIP–4 plates during irradiation. No signs of significant deformation, delamination, or gross swelling were observed. The results of the UT data are published in reference [43].

6.2.4.6 *Blister Anneal Testing*

Blister threshold testing of fuel plates is an accepted method through which the safety margin for operation of plate-type fuel in research and test reactors is assessed. The blister-threshold temperature is indicative of the ability of fuel to operate at high temperatures for short periods of time (transient conditions) without failure. Blister annealing studies on the U–Mo monolithic fuel plates began in 2007, with the RERTR–6 experiment, and they have continued as the U–Mo fuel system has evolved through the research and development process.[44]

Blister anneal threshold temperatures for RERTR–6, 7, 9, 10, and 12 range from 400 to 550°C. These temperatures are currently projected to be acceptable for NRC-licensed research reactors, the high-power ATR, and HFIR, based on comparison to blister-threshold temperatures of existing fuels. The blister-threshold temperature is a function of fission density and is not strongly influenced by the zirconium interlayer. Figure 29 shows measured blister-threshold temperatures as a function of fission density.

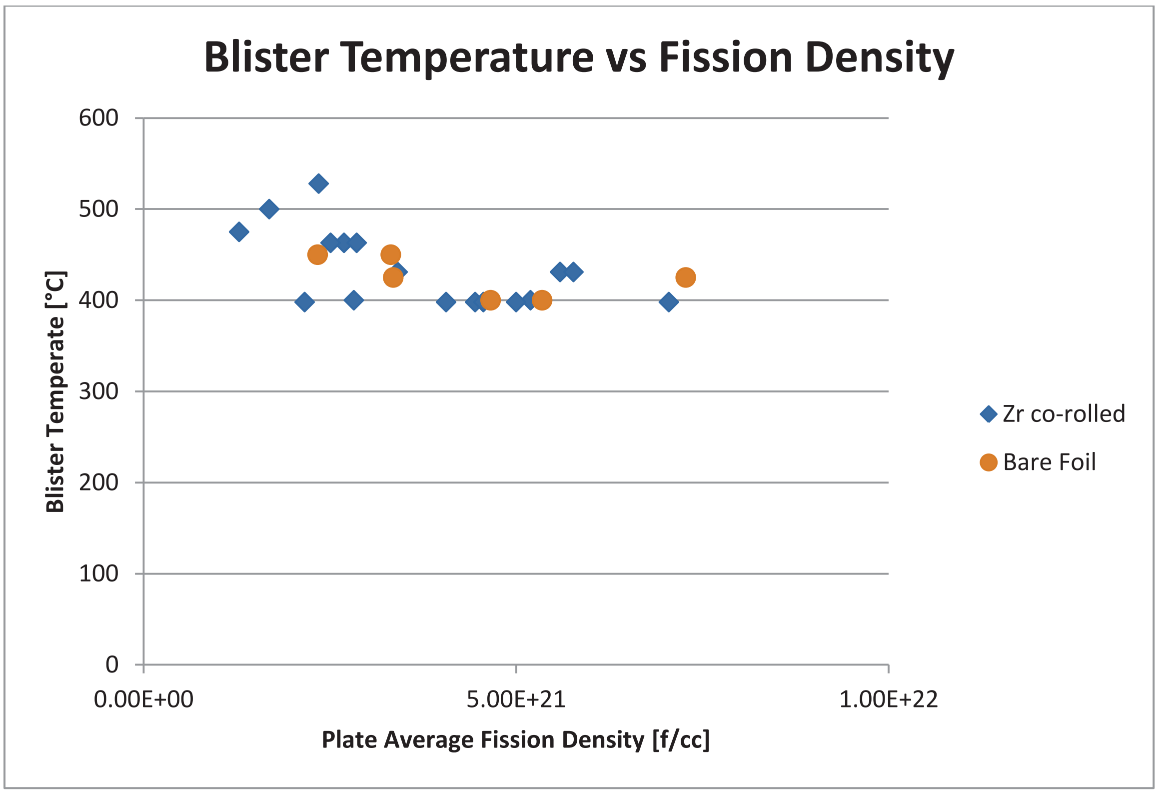

Figure 29. Blister-threshold temperature as function of fission density.

7. DISCUSSION

The research and development program to optimize the design for monolithic very-high-density fuel plates for research reactors has focused on selection of fuel-meat and interface design features that allow the fuel to meet the requirements for down-selection outlined in Section 0. In parallel with the in-reactor testing efforts and fabrication development, a modeling effort has been used to conduct parametric studies of some fuel-design variables[45]. The selection of design variables precedes a focused effort to qualify a fuel design. The testing program described in this report has been used to evaluate a range of design variables and has produced results that indicate fuel designs that are suitable for qualification.

7.1 Composition

U–10Mo alloy fuel plates exhibit smaller fuel-plate-thickness increases than U–7Mo and U–8Mo fuel. It is apparent that this is due, at least in part, to a decreased rate of reaction with the aluminum cladding. Due to the lack of fuel-meat swelling values available from RERTR–6 and 7, in which other fuel compositions were tested, it is difficult to separate thickness increases caused by unreacted U–Mo fuel-meat swelling from those caused by the swelling of the interaction layer. The difference in U–Mo swelling behavior, in the absence of interaction, is likely not a differentiator in fuel performance at intermediate fission densities (5.5E+21 f/cm^3). It is not known molybdenum content will be a differentiator at higher fission density. Other factors, however, influence the choice of fuel-meat alloy composition.

Depending on the fabrication steps used, time at elevated temperatures can lead to decomposition from the metastable U–Mo γ phase to the undesirable alpha phase.[46] Decomposition to the α-phase contributes to increased reaction rate between U–Mo and Al and the formation of a relatively thick interaction layer between the aluminum cladding and the fuel foil[47,48]. Characterization after fabrication indicates U–10Mo, when compared to lower molybdenum content alloys, exhibits slower interdiffusion with Al–6061at times and temperatures relevant to the HIP process [49].

In zirconium-barrier-layer plates, the bulk of the interdiffusion between U–10Mo and zirconium occurs during hot rolling. At HIP processing temperatures (≥550°C), there is very little growth of the interaction layers between the Zr foil and the U–Mo fuel[50]. The bulk of the interdiffusion between U–10Mo and zirconium occurs during hot rolling at a temperature near 650°C. Transmission electron microscope (TEM) examination identified the formation of 5 new phases at the interface.[51] Interaction between the Zr foil and the Al–6061 cladding can produce $(Al, Si)_2Zr$, $(Al, Si)Zr_3$, $(Al, Si)_3Zr$ and $AlSi_4Zr_5$ phases. More or fewer phases can occur as a function of fabrication parameters. Both Mo and U are scavenged from the U–Mo fuel phase during interaction between the fuel and Zr foil, and the result is the formation of a thin layer of Mo_2Zr next to the fuel and a UZr_2 layer on the Zr foil side of the interface. Potentially, some α–U can appear in the fuel near the U–Mo/Zr interface. This has the potential to affect irradiation performance in that region of the fuel, due to the decreased radiation stability of the α-phase. It has been observed that the equilibrium α-U + γ' phases that form in a lamellar structure revert back to γ-phase U–Mo upon irradiation if the alloy contains sufficient molybdenum content. It is not known if the α-U formed in a diffusion zone by depletion of the Mo will revert back to the γ-phase during irradiation. In the case of U–10Mo alloys, Mo concentrations as low a 2–3 wt.% have been measured in local regions near the U–Mo/Zr interface.[52]

The concentration of Mo in U–7Mo and U–8Mo is low enough that Mo depletion near the Zr layer may lead to increased formation of α-U at the interface during fabrication. It is well known that α-U is less stable under irradiation then γ-phase U–Mo. The additional margin to Mo depletion provided by the use of U–10Mo is therefore desirable. HIP fabrication studies of the inter-diffusion between U–10Mo and zirconium show a diffusion layer with molybdenum values between two and three percent. This reduction in molybdenum has an effect on the stability of the fuel during irradiation. This information has been published. [53]

Based on information from fuel-performance evaluation, pre-irradiation characterization, and fabrication studies, U–10wt%Mo was selected as the fuel meat composition.

7.2 Fabrication

Bonding techniques—including hot isostatic pressing (after co-roll bonding of Zr barrier layers to foils), friction bonding, and transient liquid phase bonding (spraying the fuel/cladding interface with Al-Si alloys before bonding with HIP or FB)—were all shown to be successful in some instances. Friction-bonded zirconium interlayer plates were irradiated successfully and could be investigated as a potential alternative to HIP-bonded fuel. The quality and, therefore, the success of the latter two, FB and TLPB, however, are dependent upon fabrication variables that are difficult to control, and delamination between fuel and cladding has been observed as a result.

Both hot rolling and hot rolling followed by cold rolling have been used to apply a diffusion barrier to the U–Mo fuel foil prior to plate fabrication. These methods were selected due to ease of development and the potential for commercial manufacturing using these methods. The irradiation behavior of fuel plates fabricated by both methods appears to be acceptable. The selection of one method over the other depends on the ability of these processes to meet manufacturing tolerances.

Based on the repeatability of the fabrication process and irradiation-testing results, hot isostatic pressing was selected as the primary clad-bonding fabrication method for fuel plates with co-rolling used to apply diffusion barriers (zirconium and niobium having been demonstrated). Other methods of zirconium application are yet to be tested, and the effects of these methods on irradiation behavior are not known at this time.

7.3 Fuel/Cladding interface

7.3.1.1 Unmodified Interface

The RERTR–6, 7, 8, and 9 experiments contained bare (no barrier) U–Mo fuel foils and Al–6061 cladding. Specific examples of plates with either unsatisfactory performance or significant failure precursors are shown in Table 6. Delamination during irradiation or PIE (unstable behavior) and the presence of significant porosity at the U–Mo/Al interface were considered to be unacceptable. These plates had fission densities as low as 2.4E+21 fissions/cm^3. Twelve (12) of the 32 fuel plates fabricated using friction stir welding or HIP failed or exhibited unstable fuel-swelling behavior on examination after irradiation. Two (2) of the 3 plates fabricated by HIP failed. Figure 30 shows a cross section from H1P02B, which delaminated prior to sectioning. Figure 13 shows the interface from H1P010, which exhibits signs of excessive fission-gas-bubble formation, indicating unstable fuel behavior. The surviving bare foil fabricated by HIP processing was L1P020, which was a U–12Mo fuel foil irradiated to a peak fission density of 8.5E+21. This plate was not sectioned and, therefore, may have failure precursors that were not identified. Three additional fuel plates fabricated by TLPB also failed, but these failures are likely related to inadequate bonding during fabrication.

Figure 30. Metallographic cross section of plate H1P02B irradiated to peak fission density of 9.79E+21.

The performance of fuel plates with no interface modifications is plotted along with the USHPRR operating envelope in Figure 31 and Figure 32. Fuel down-selection was made based on the assumption that all five USHPRRs must use the same fuel design. Because of the high failure rate under moderate test conditions, no testing was conducted near USHPRR bounding conditions.

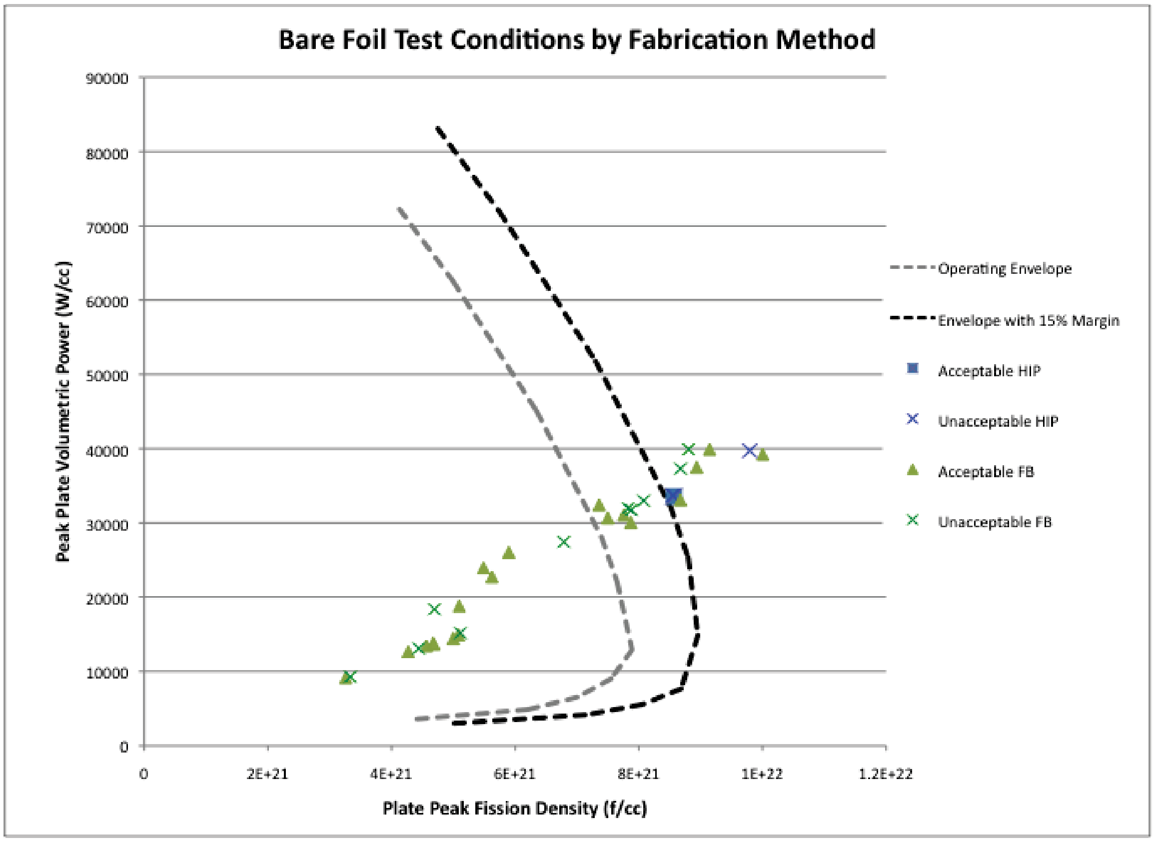

Figure 31. Performance of fuel plates with a U–Mo/Al–6061 interface. Curves indicate the USHPRR operating envelope. X's indicate plate failures or presence of failure precursors. Only plates fabricated using HIP and FB are plotted.

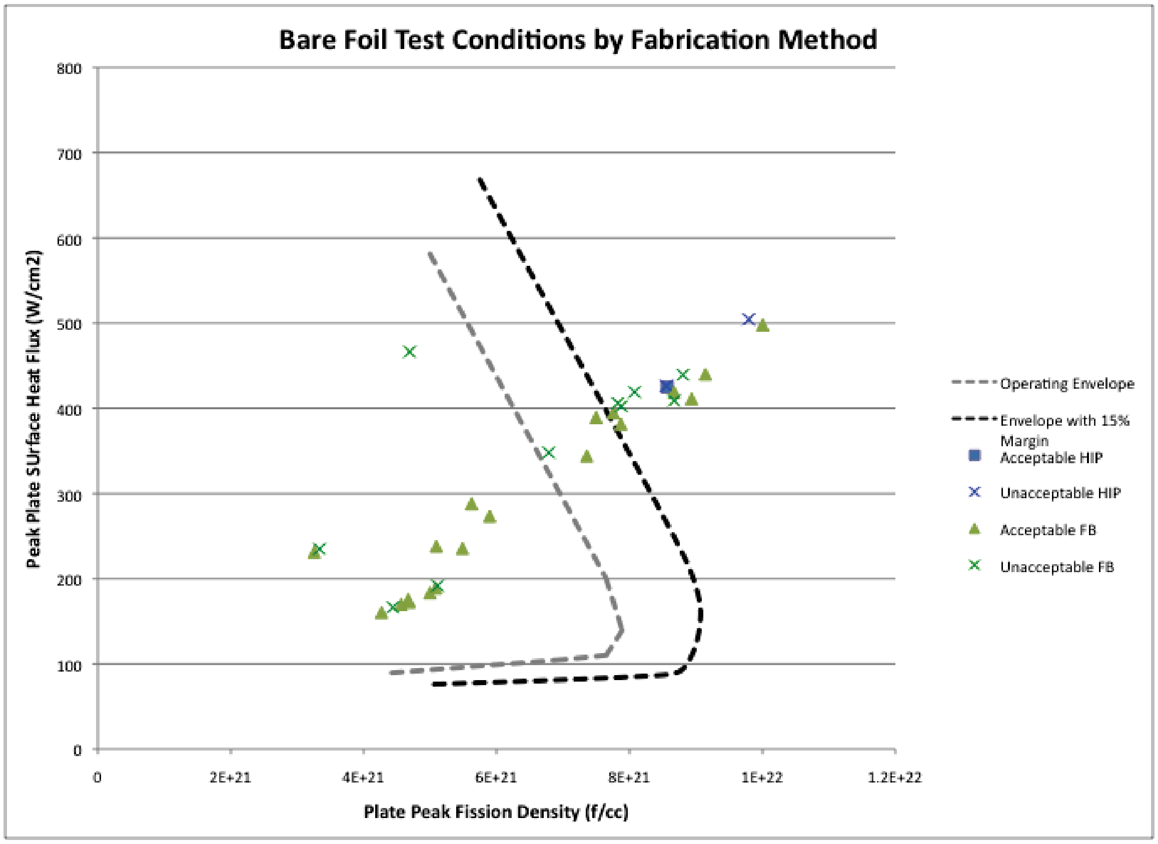

Figure 32. Performance of fuel plates with a U–Mo/Al–6061 interface. Curves indicate the USHPRR operating envelope. X's indicate plate failures or presence of failure precursors. Only plates fabricated using HIP and FB are plotted.

Recent assessment of the baseline fabrication process has indicated that eliminating the zirconium barrier layer may provide manufacturing cost advantages because it allows for the elimination of the difficult-to-recover mixed zirconium/uranium/molybdenum waste stream. Fuel without the barrier layer was evaluated against requirements for the NRC-licensed reactors only to determine whether this fuel would be acceptable for use under a limited set of operating conditions. The performance of plates with an unmodified U–Mo/Al–6061interface is plotted along with the NRC-licensed USHPRR operating envelope in Figure 33. All plates tested under conditions within this envelope were fabricated using friction bonding. Three (3) plate failures occurred (L2F030, N1F030, and N1F090) among the 25 fuel plates tested under these conditions, for a failure rate of 12%. Of these three failures, two exhibited delamination, and one exhibited a heavily reacted interface, considered to be a precursor to failure. Post-irradiation photographs and metallographic images of these plates are shown in Figure 34, Figure 36, and Figure 37.

The high failure rates and unpredictable behavior originally led the program to abandon this design in favor of fuel with modifications to the U–Mo/Al interface to increase fuel reliability.

A gap in the data for fuel plates with an unmodified U–Mo/Al interface exists at high burnup (8E+21 f/cm^3) and moderate power (150 W/cm^2) conditions, representative of the National Bureau of Standards Reactor (NBSR) peak conditions. If HIP is chosen as the fabrication method for this type of monolithic fuel, further assessment of the performance of HIP plates without a diffusion barrier layer within the range of operating conditions that bound the NRC-licensed reactors would be required.

Consideration of this fuel design for qualification for use in NRC-licensed USHPRRs would require that this gap be filled through mini-plate and full-size plate experiments under these conditions and demonstration of lower failure rates for fuel fabricated with an optimized fabrication process.

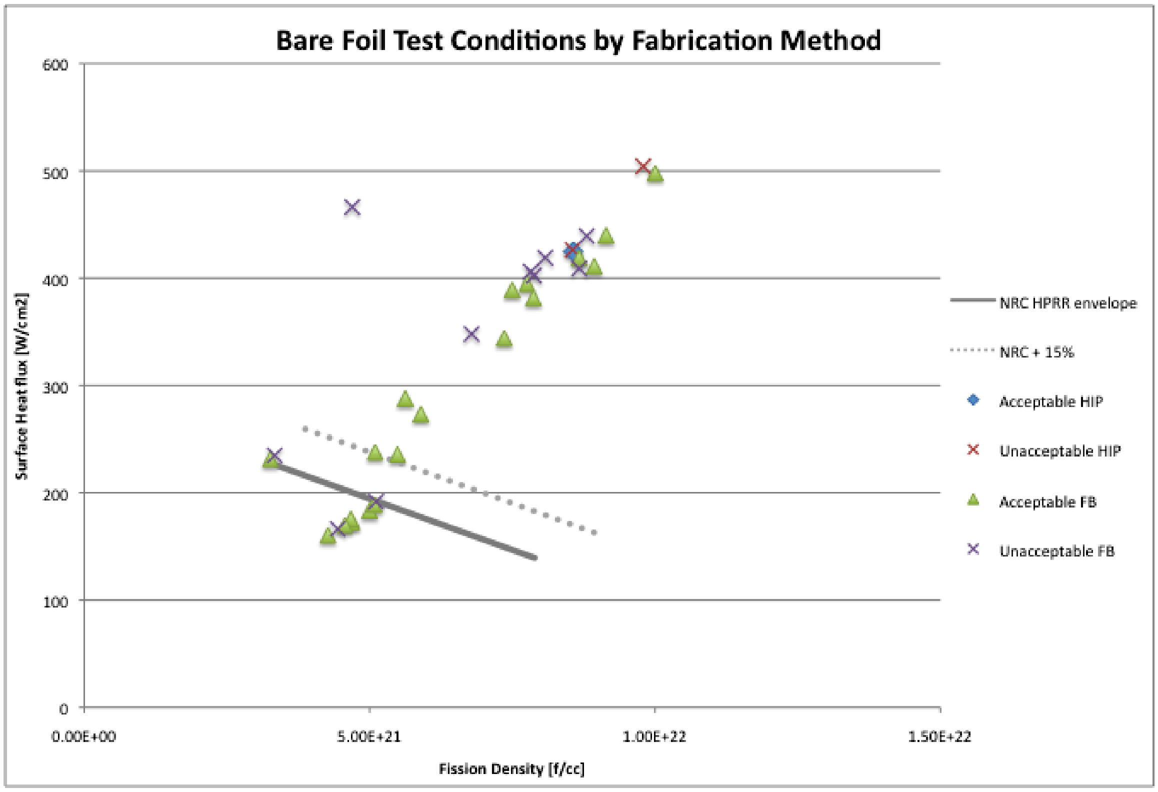

Figure 33. Performance of fuel plates with an unmodified U–Mo/Al–6061 interface. Curves indicate the NRC-licensed USHPRR operating envelope (lower power). Open symbols indicate plate failures or failure precursors. Only plates fabricated by friction bonding and hot isostatic pressing are shown.

Figure 34 and Figure 35 show examples of failure or failure precursors observed during destructive examination of U–Mo/Al–6061 plates with an unmodified interface.

Table 6. Plates with U–Mo/Al–6061 interface and unsatisfactory behavior.

Plate Name	Experiment #	Position	Fabrication	Plate Composition	Average Surface Heat Flux (w/cm^2)	U–235 depletion	Plate Average Fission Density (f/cm^3)	Performance notes
N1F090	RERTR–6	A8	Friction Bond	250 µ U7Mo/Al	127	45.30%	3.39E+21	Heavily Reacted Interface
N1F030	RERTR–6	C4	Friction Bond	250 µ U7Mo/Al	141	47.27%	3.76E+21	Delaminated & Interface Porosity
L2F030	RERTR–6	D7	Friction Bond	500 µ U10Mo/Al	169	35.79%	2.40E+21	Delaminated
H1F020	RERTR–7	A8	Friction Bond	250 µ U12Mo/Al	240	25.19%	4.68E+21	Delaminated
H1T010	RERTR–7	B4	TLPB	250 m U12Mo/Al	269	28.80%	5.18E+21	Interface Instability
L1F140	RERTR–7	B7	Friction Bond	250 µ U10Mo/Al	263	27.09%	5.15E+21	Delaminated
H1F030	RERTR–7	C1	Friction Bond	250 µ U12Mo/Al	288	29.30%	5.55E+21	Interface Porosity
L1F120	RERTR–7	C5	Friction Bond	250 µ U10Mo/Al	289	28.67%	5.57E+21	Interface Porosity
L2F040	RERTR–7	D7	Friction Bond	500 µ U10Mo/Al	297	17.24%	2.99E+21	Porosity/cracking
H1P010	RERTR–8	C4	HIP	250 µ U12Mo/Al	292	31.31%	5.86E+21	Interface Porosity
H1P02B	RERTR–8	B5	HIP	250 m U12Mo/Al	382	38.74%	7.42E+21	Delaminated
L1F32C	RERTR–9A	C5	Friction Bond	250 m U10Mo/Al	260	31.55%	5.90E+21	Delaminated
L1F26C	RERTR–9A	C1	Friction Bond	250 µ U10Mo/Al	269	32.76%	6.20E+21	Delaminated

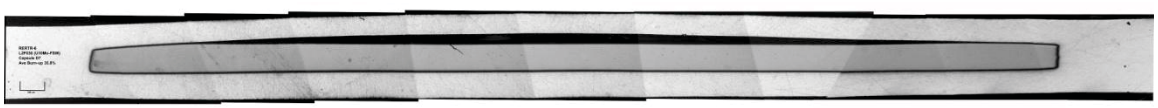

Figure 34. Plate L2F030, showing delamination along one interface.

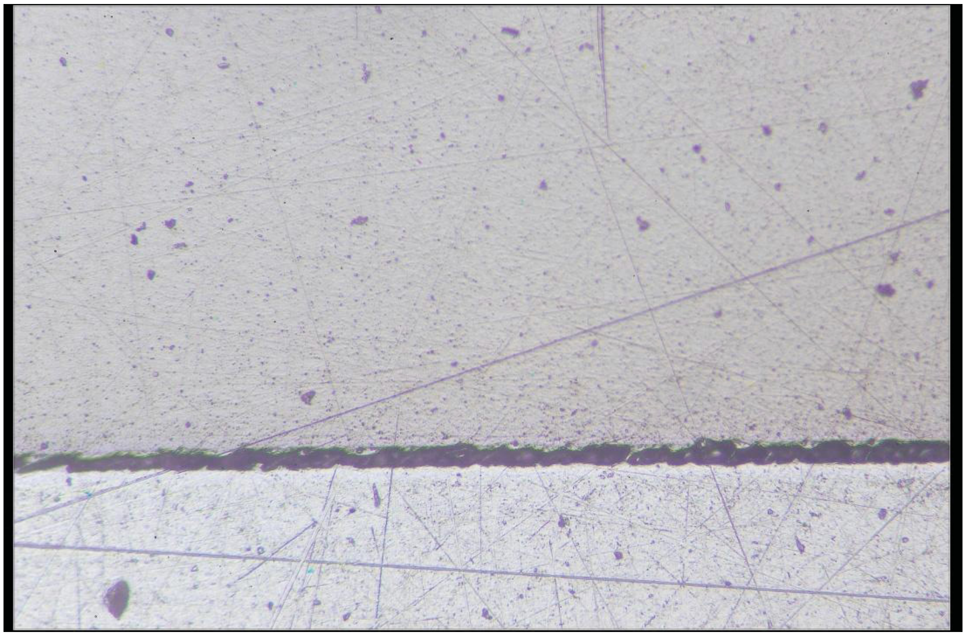

Figure 35. Interface porosity seen between the bare U–Mo foil and the aluminum cladding (H1F030, local fission density: 6.1E+21).

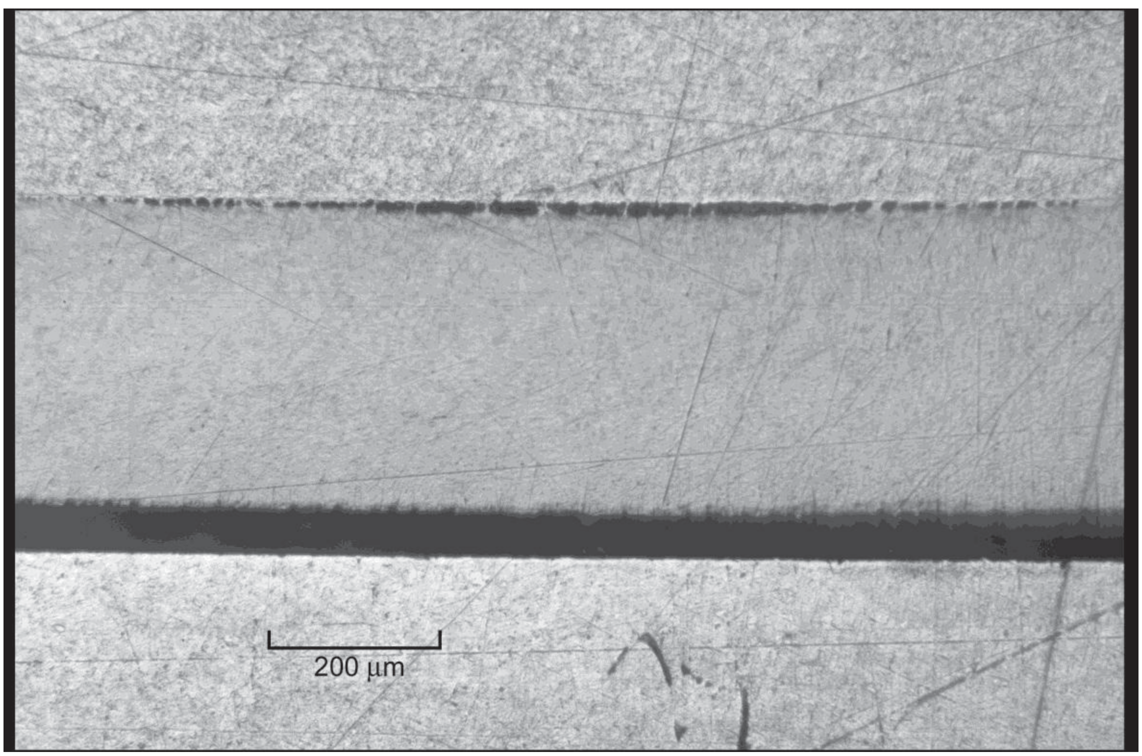

Figure 36. Post-irradiation micrograph of plate N1F030, irradiated to a peak fission density of 3.76E+21 f/cm^3 in the RERTR–6 irradiation test.

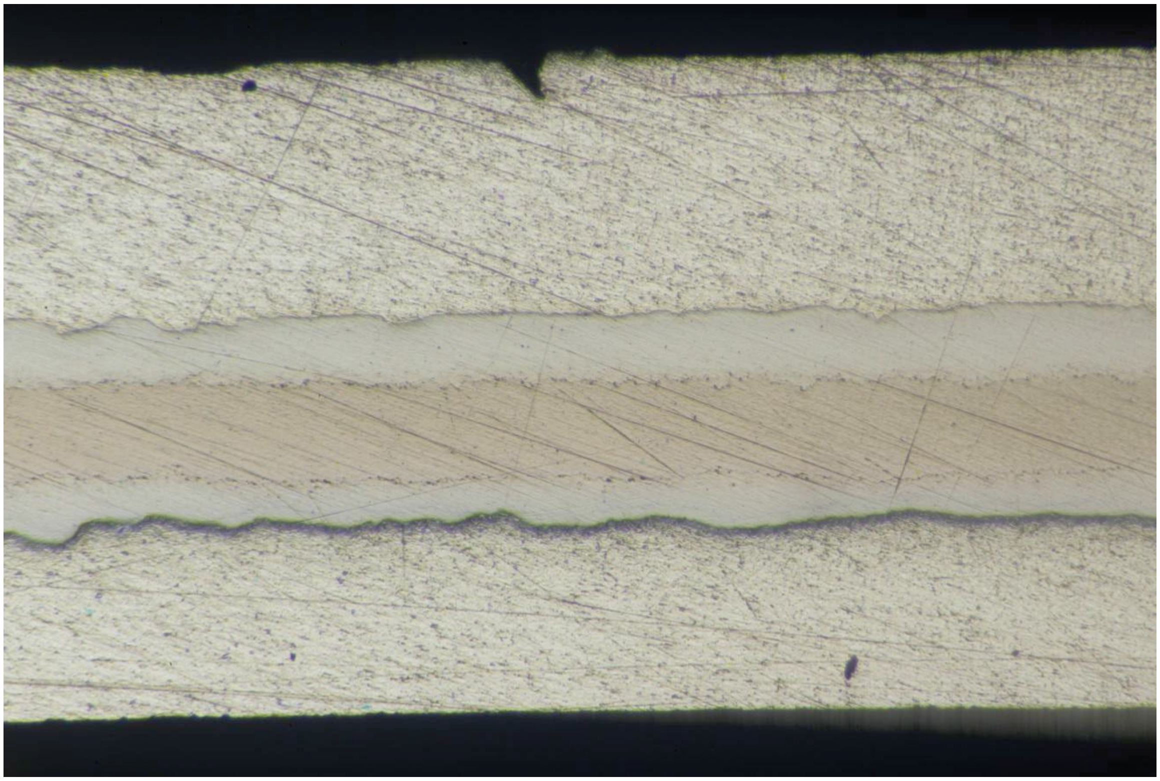

Figure 37. Post-irradiation micrograph, showing excessive fuel-clad interaction, of N1F090, irradiated to peak fission density of 3.4E+21 f/cm^3 in the RERTR–6 experiment.

7.3.1.2 Silicon Modified Interface

Fuel/cladding interface modifications included the application of silicon or aluminum alloys containing silicon to the cladding prior to fabrication of the plates. These plates were tested as part of the RERTR–9 test and, again more systematically, in the RERTR–10 tests. The use of an aluminum–4043 interface was eliminated due to the formation of excessive porosity at the interface during irradiation (Figure 18). The plates with interfaces modified using silicon thermal spray displayed better results, but resulted in total delamination of the interface in 5 out of 22 plates tested. Table 7 lists the plates that behaved unsatisfactorily or exhibited precursors to failure. Ten of the 21 plates, 48% tested with this configuration, failed or behaved unsatisfactorily. Figure 41 and Figure 42 show examples of a delaminated plate with a silicon-enhanced interface.

Both the AFIP–2 and AFIP–3 silicon-sprayed plates behaved satisfactorily up to peak fission densities of 5.6E+21 and 4.9E+21 fissions/cm^3 respectively. The test conditions of the AFIP experiments are not considered to be as aggressive as the mini-plate tests due to the uniform irradiation configuration and the associated more prototypic mechanical stresses within the plate.

The performance of plates with a silicon-modified interface is plotted along the USHPRR operating envelope in Figure 38. It can be seen that five of the plates failed below the 15% margin for the HPRR operating envelope.

Figure 40 shows the performance of plates with a silicon-modified interface against the operating envelope for NRC reactors only. It can be seen that all of the plates were tested at significantly higher than the operating envelope for these lower-power reactors.

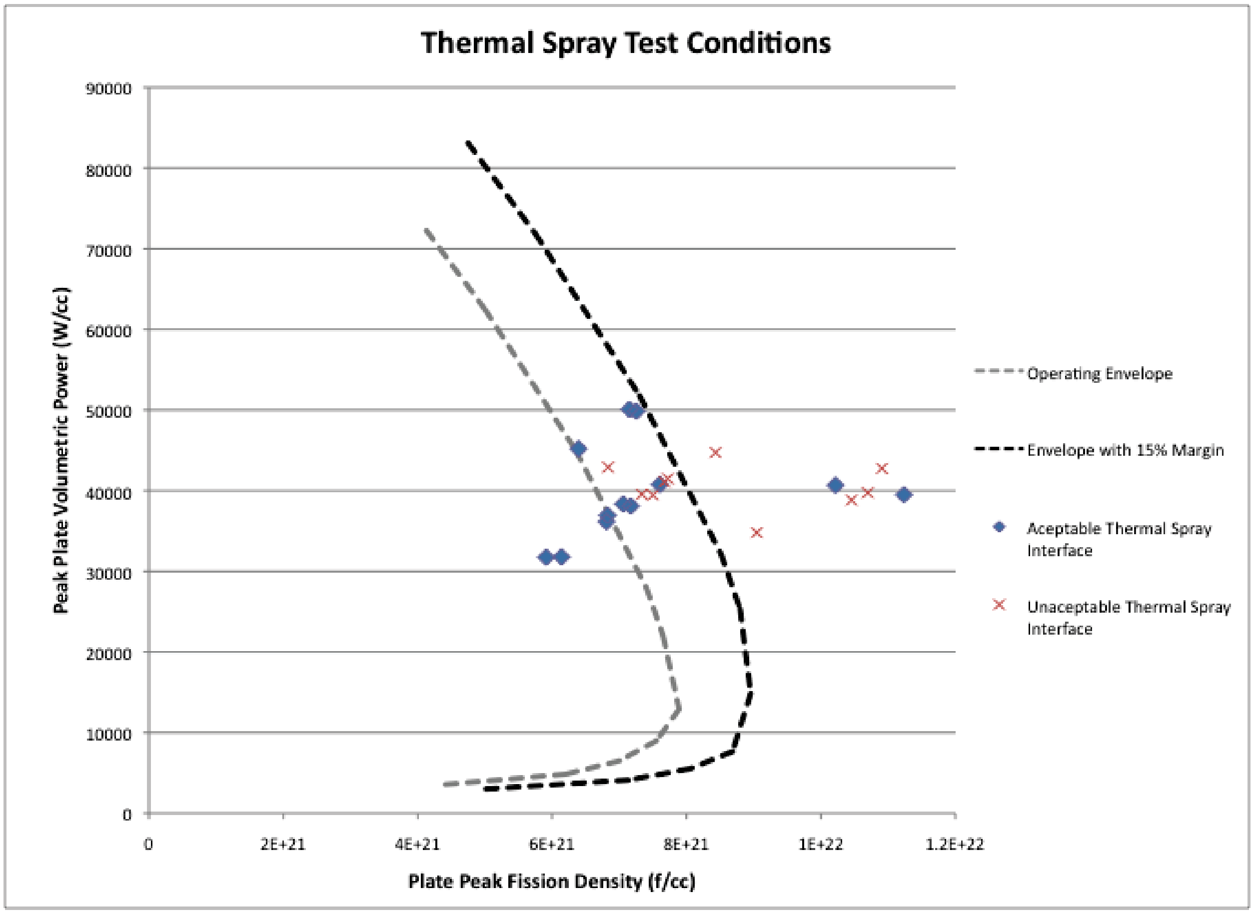

Figure 38. Performance of fuel plates (W/cm^3) with silicon-modified interface compared to the USHPRR operating envelope.

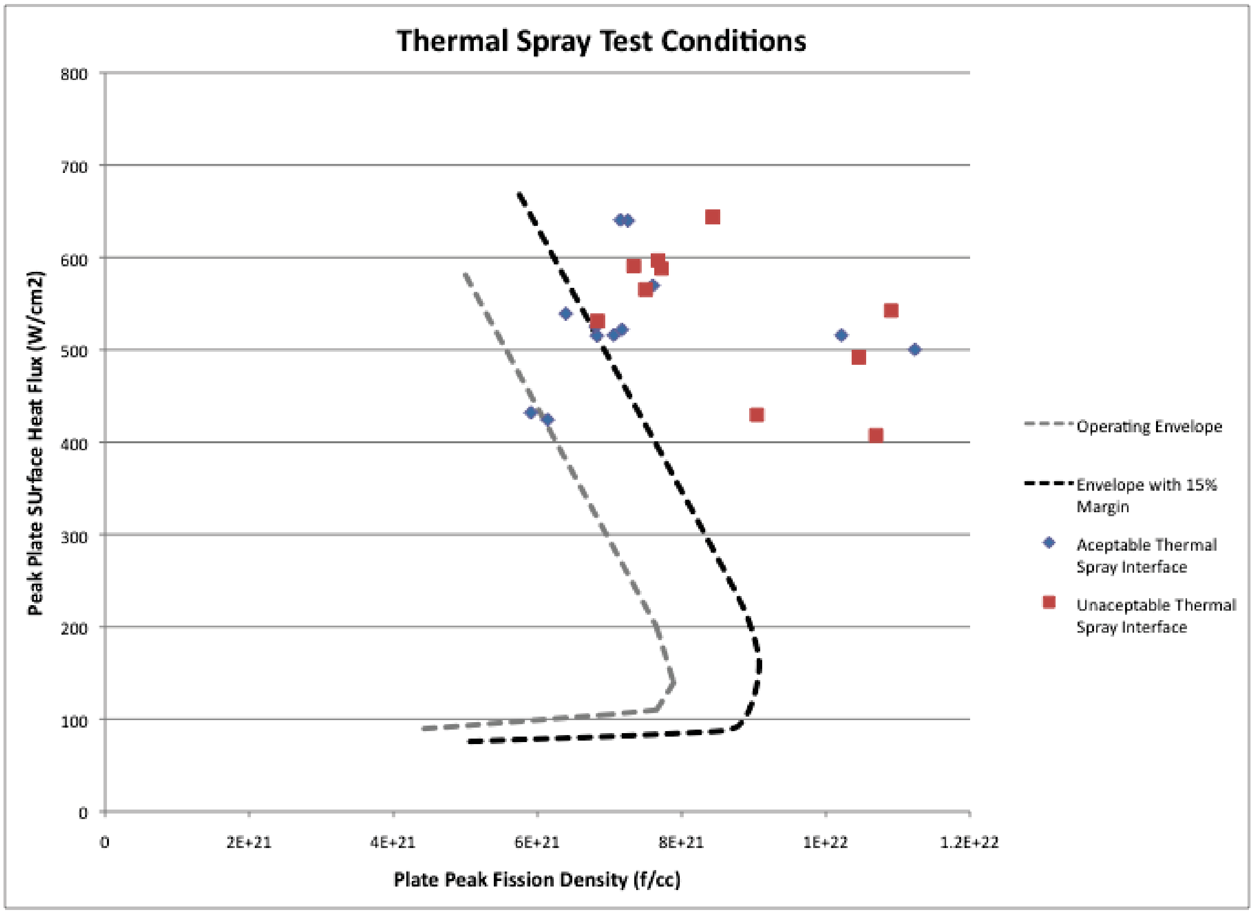

Figure 39. Performance of fuel plates (W/cm^2) with silicon-modified interface compared to the USHPRR operating envelope.

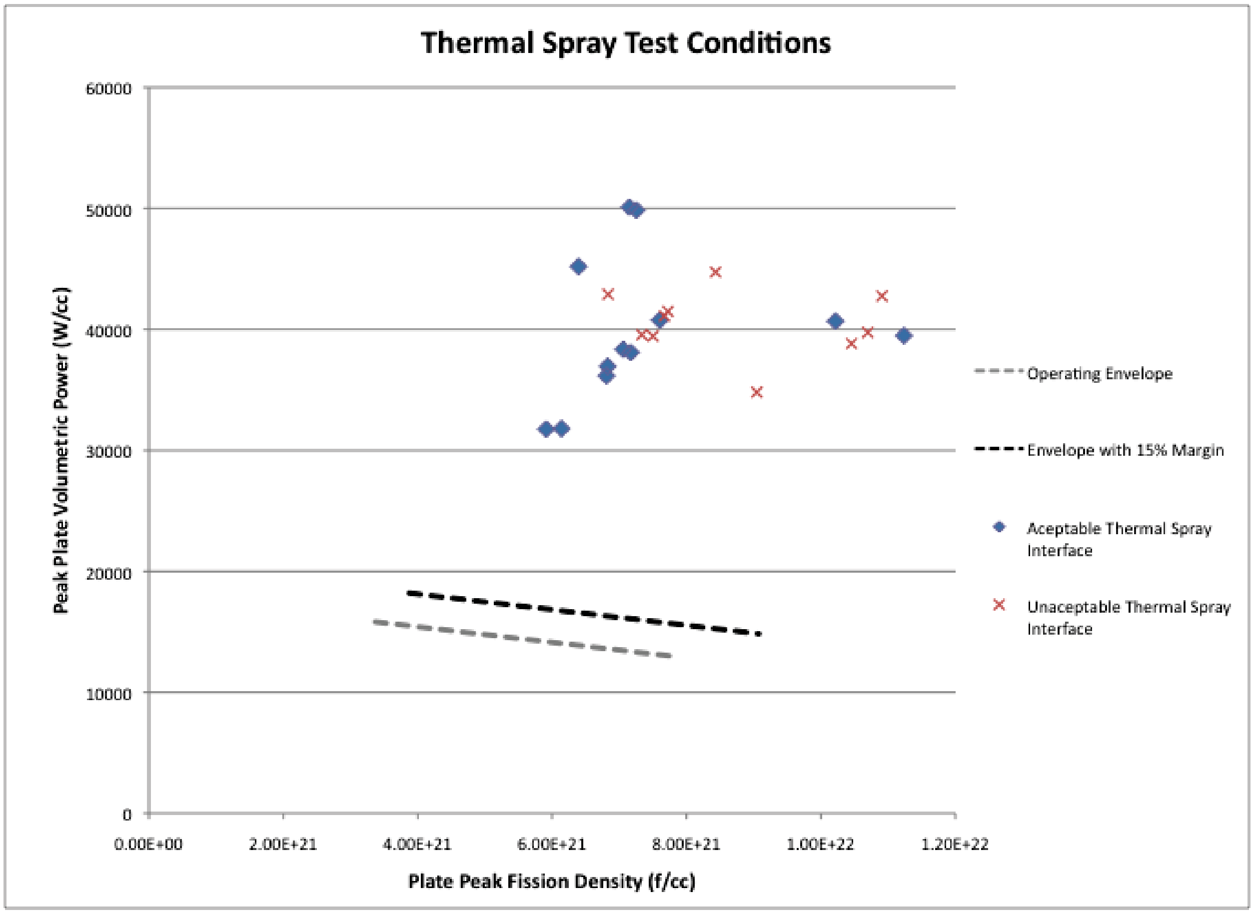

Figure 40. Performance of fuel plates (W/cm^2) with silicon-modified interface with NRC operating envelope.

The high failure rate of the silicon interlayer fuel design led to discontinuation from the test program. Consideration of this fuel design for NRC licensed USHPRRs would require optimization of the fabrication process using quantitative measurement of bond strength and irradiation of test plates within the operating envelope of this class of lower-power reactors.

Table 7. Silicon modified interfaces with unsatisfactory behavior.

Plate Name	Experiment #	Position	Fabrication	Plate Composition	Surface Heat Flux (w/cm^2)	U–235 depletion	Fission Density (f/cm^3)	Performance notes
L1F37T	RERTR–9B	B4	Friction Bond	250 µ U–10Mo/Al	259	38.65%	7.48E+21	Delaminated
L1P07T	RERTR–9B	B8	HIP	250 µ U–10Mo/Al	331	40.17%	7.32E+21	Delaminated
L1F36T	RERTR–9B	D4	Friction Bond	250 µ U–10Mo/Al	295	36.51%	7.21E+21	Porosity/delamination
L1P08T	RERTR–9B	D8	HIP	250 µ U–10Mo/Al	249	32.10%	6.03E+21	Cracked Interface + Porosity
L1P244	RERTR–10	C5	HIP	250 µ U–10Mo/Al	352	20.84%	4.67E+21	Delaminated
L1P223	RERTR–10	C6	HIP	250 µ U–10Mo/Al	297	17.44%	3.90E+21	Delaminated
L1P202	RERTR–10	C7	HIP	250 µ U–10Mo/Al	302	17.61%	3.75E+21	Delaminated
L1P181	RERTR–10	C8	HIP	250 µ U–10Mo/Al	365	21.54%	4.78E+21	Delaminated
L1F381	RERTR–10	D2	Friction Bond	250 µ U–10Mo/Al	374	19.53%	4.81E+21	Interface Porosity

Figure 41. Cross section of L1P202 and delaminated interface (3.75E+21 f/cm^3).

Figure 42. Cross section of L1P244 and delaminated interface (4.67E+21 f/cm^3)

7.3.1.3 Zirconium Barrier Layer

Both zirconium and niobium barrier layers were fabricated and irradiated as part of the RERTR–9, 10, and 12 experiments. None of the plates tested to date at either the mini-plate or full-size–plate scales have demonstrated failures at the fuel/clad interface of the fuel plates.

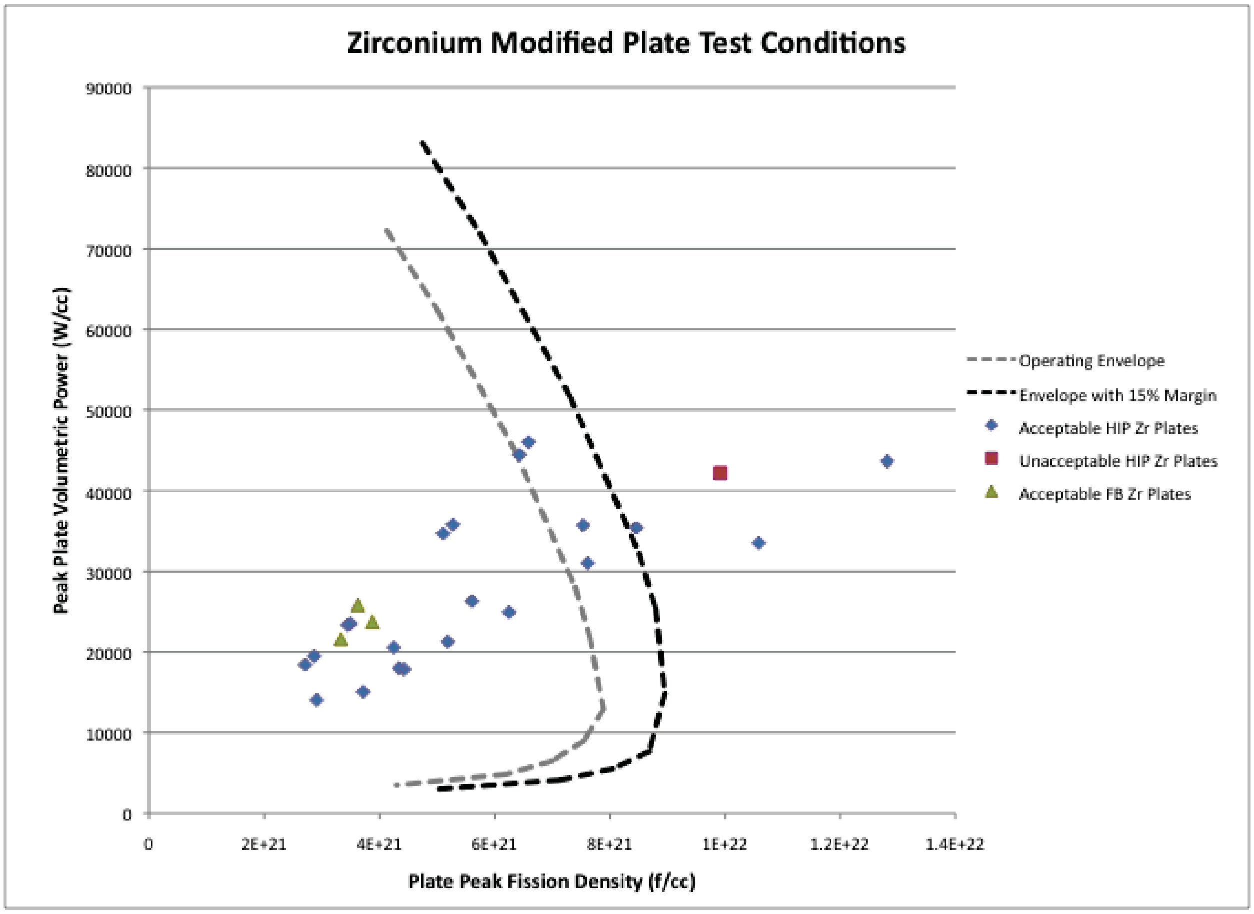

Figure 43. Performance of fuel plates with a zirconium barrier layer by fabrication method against HPRR operating envelope (W/cm^3). The failure shown for RERTR-12 was in plate L1P754. RERTR-12 insertion 2 data are preliminary and not included in this plot.

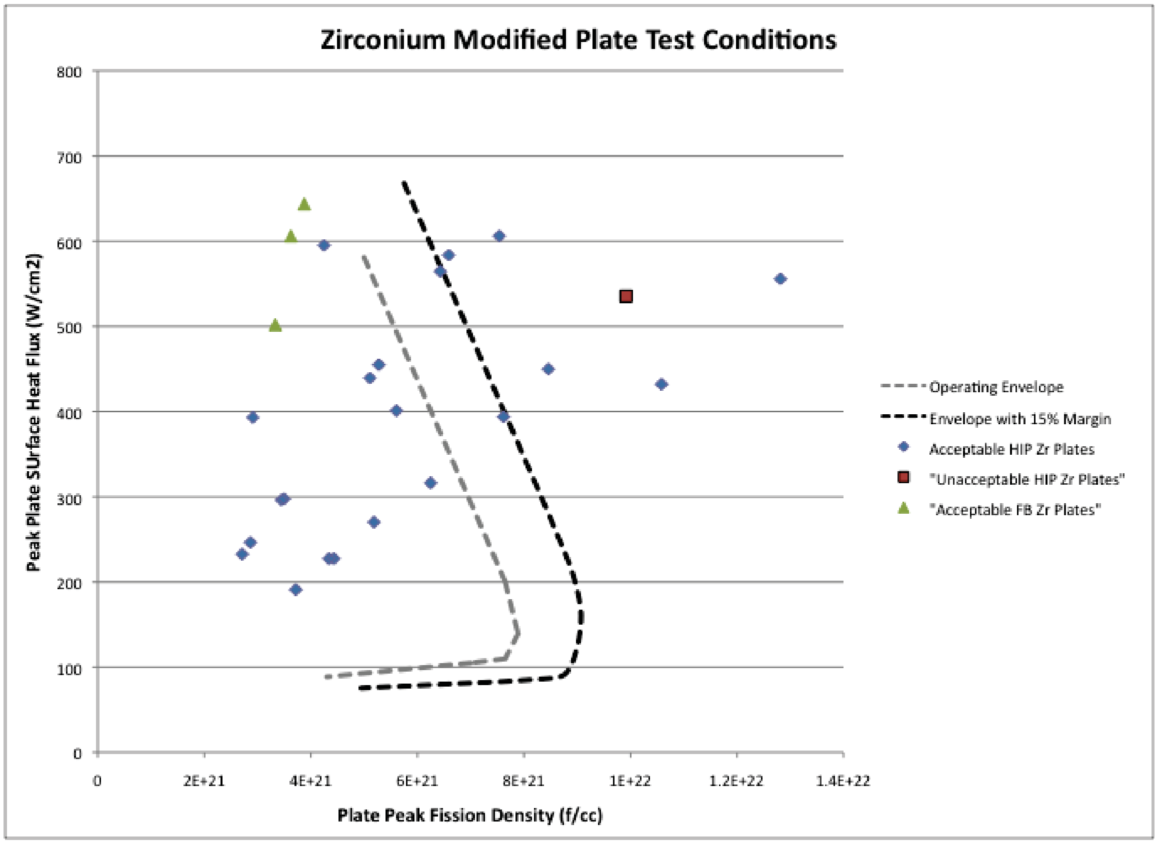

Figure 44. Performance of fuel plates with a zirconium barrier layer by fabrication method against HPRR operating envelope (W/cm^2). The singular failure was plate L1P754. RERTR-12 insertion 2 data are preliminary and not included in this plot.

Several zirconium-interface plates have failed (including preliminary RERTR–12 insertion-2 plates), at very high fission densities (average fission density >7.2E+21 fissions/cm^3), and the failures were contained within the fuel meat while the fuel meat remained bonded to the cladding. Visual examination of the fuel plates indicates that the failures all appear to be of a singular mode, characterized by the formation of a pillow. All failed plates attained peak fission densities higher than possible for LEU, and no fission-gas release occurred. The pattern of fuel cracks and the fission–gas-bubble morphology observed in plate L1P754 leads to the preliminary conclusion that the brittle failures occurred within the fuel meat during the final power shutdown as a result of accumulated strains and thermal stresses.

Figure 43 and Figure 44 show the zirconium-barrier-plate test conditions as a function of fabrication method. It can be seen that the only failure was at a peak fission density outside of the range of the USHPRR operating envelope with the 15% margin applied.

Only three friction-bonded zirconium barrier layer plates have been irradiation tested. All three of these plates exhibited acceptable fuel performance. Although there are few data points, a comparison of the data presented in Figure 43 and Figure 44 to the data presented in Figure 31 and Figure 32 indicates that the friction-bonding process applied to Zr-barrier-layer plates results in no failures, while friction bonding applied to non-barrier-layer plates results in failures under similar irradiation-testing conditions.

Failure rates among bare foils fabricated by both HIP and FB do not indicate a dependence upon the fabrication method. Two of the bare foil FB plates fabricated with the improved FB process (Anvilloy tool) (L1F26C [average fission density 6.2E+21] and L1F32C [average fission density 5.9E+21]) both pillowed in reactor. The zirconium barrier plates fabricated by FB with the same tool—L2F46Z (average fission density 2.5E+21), L2F47Z (average fission density 1.94E+21), and L2F45Z (average fission density 2.19E+21)—none failed, exhibited precursors to failure, or delaminated upon post-irradiation sectioning.

Regardless of fabrication method, fuel plates fabricated using zirconium-diffusion-barrier fuel foils have demonstrated stable and predictable irradiation behavior under all conditions tested to date. Fuel plates fabricated without a zirconium barrier layer exhibit a higher failure rate regardless of the fabrication method.

Zirconium-diffusion-barrier plates do exhibit the formation of discrete fission-gas bubbles in the interaction layers at the interface between the zirconium and the U–Mo at higher fission densities (FD>6E+21 f/cm^3). Figure 46 through Figure 50 show the evolution of this microstructural feature as a function of increasing burnup.

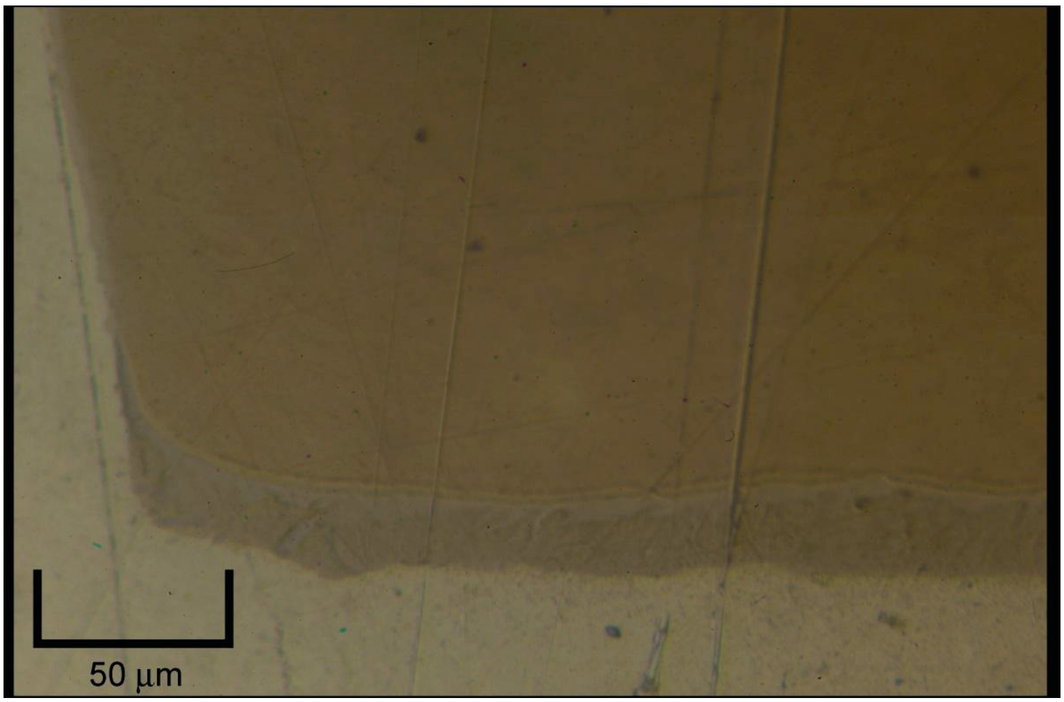

Figure 45. Metallographic mage of zirconium barrier plate (L2F45Z) irradiated to 2.2E+21 f/cm^3.

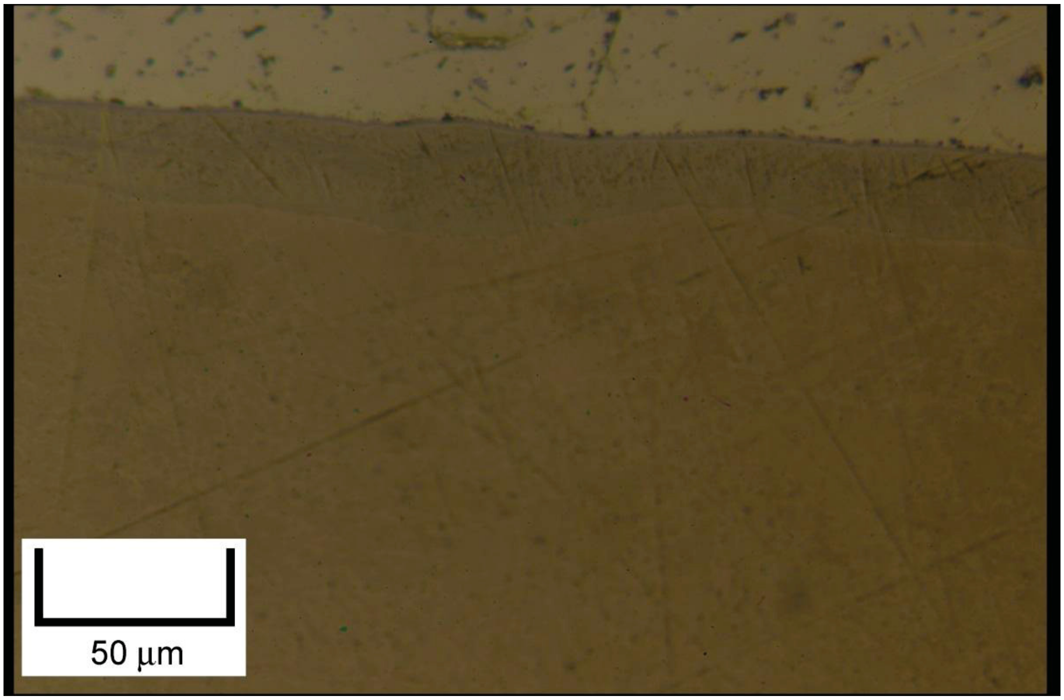

Figure 46. Metallographic image of a zirconium barrier plate (L2P16Z) at fission density of 4E+21 fissions/cm^3.

Figure 47. Metallographic image of a zirconium barrier plate (L1P773) at fission density of 5E+21 fissions/cm^3.

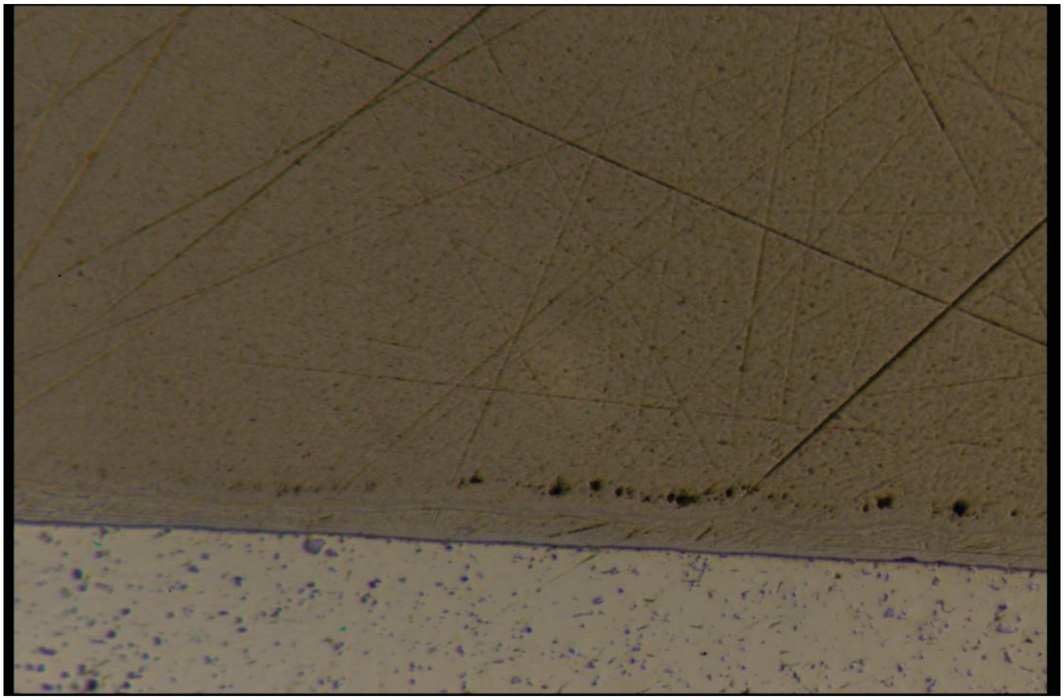

Figure 48. Porosity forming in Zr/U–Mo interaction layer at fission density of L1P12Z ~6.2E+21 fissions/cm^3).

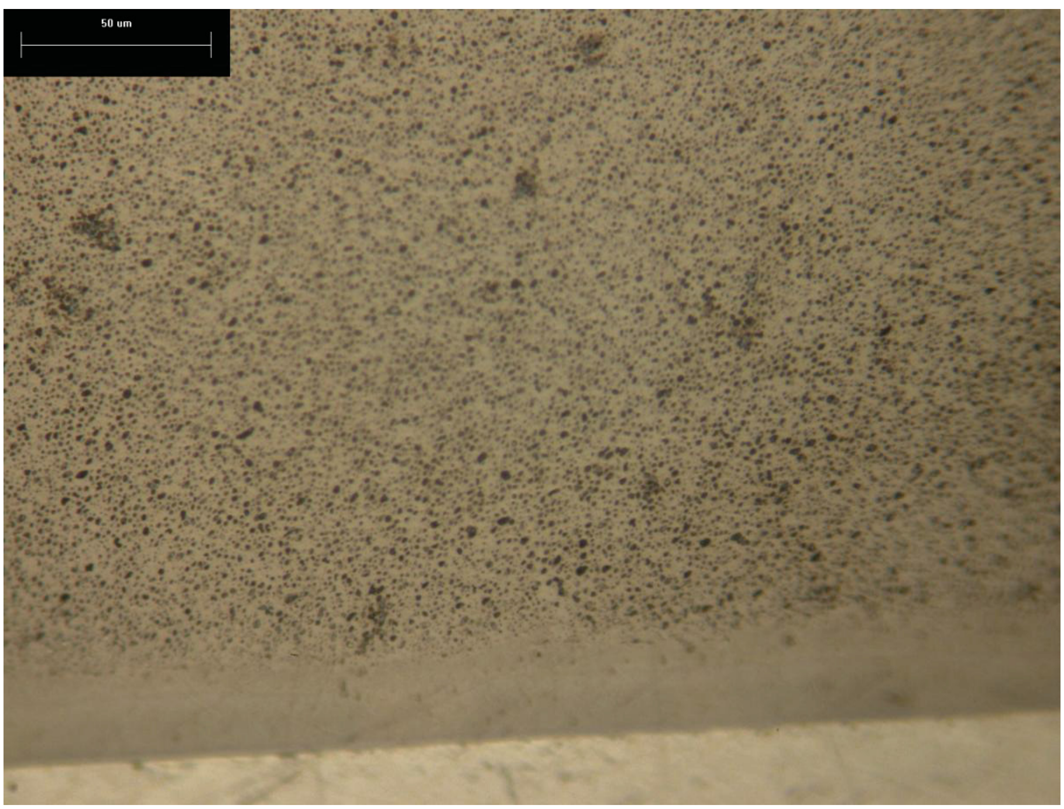

Figure 49. Metallographic image of a zirconium barrier plate (L1P755) at fission density of 7.2E+21 fissions/cm^3.

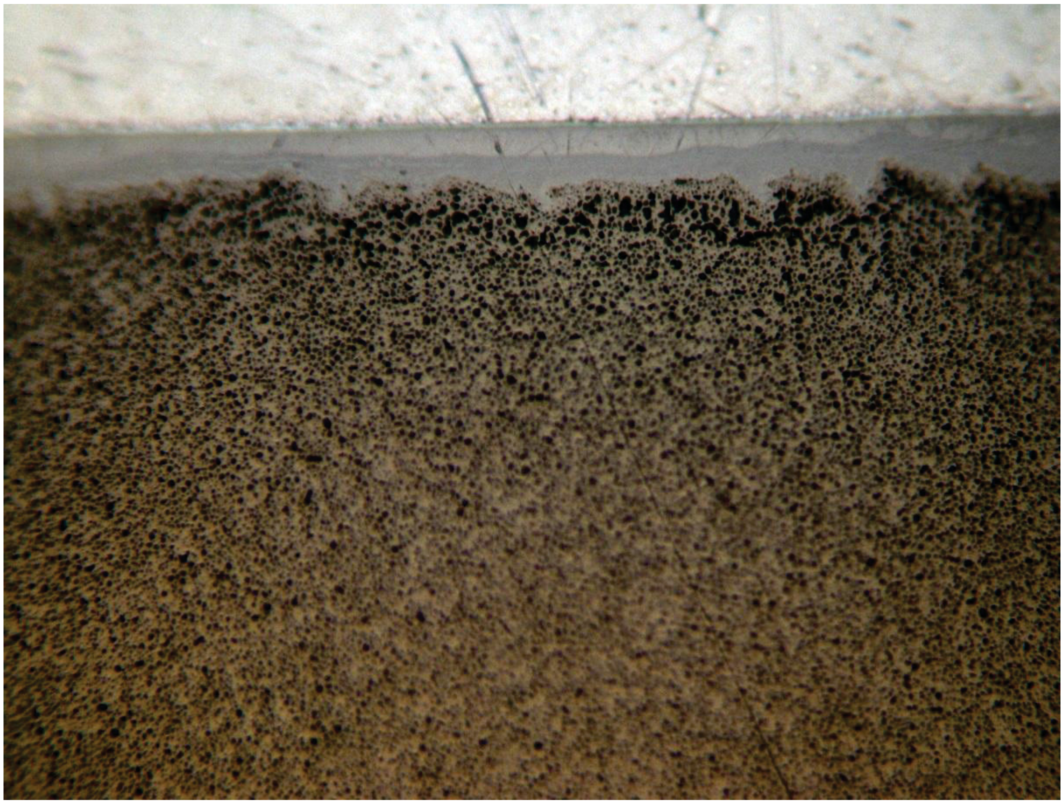

Figure 50. Metallographic image of a zirconium-barrier plate (L1P754) at fission density of 8.4E+21 fissions/cm^3.

Fuel swelling is stable and predictable to an average fission density of at least 7.2E+21 f/cm^3. Plate L1P754 (average fission density 8.13E+21 fissions/cm^3, peak fission density 1.2E+21 fissions/cm^3) pillowed in-reactor at the high fission density end of the plate, which indicates significant margin to failure relative to the USHPRR operating envelope.

Because of its robust performance to high fission densities under a wide range of conditions relevant to USHPRRs, a U–10Mo monolithic fuel plate, using a Zr foil barrier between fuel and cladding, and fabricated by co-rolling the layers and then bonding by a HIP process is recommended for qualification by the USHPRR program. Recent information on ATR and HFIR fuel-operating conditions indicates that an expanded testing envelope is required to ensure that the selected fuel performance under this extended range of conditions.

8. VERIFICATION OF FUEL SELECTION AGAINST DOWN-SELECTION REQUIREMENTS

8.1 U–Mo/Zr barrier layer fuel

A summary of fuel-plate performance against down-select requirements is provided in Table 8. Gaps are identified where additional data or additional work is required to verify that the selected fuel design meets requirements for fuel qualification. The irradiation conditions tested using the zirconium diffusion barrier system can be seen in Figure 51. No failures have occurred to date within the operating envelope (that includes the estimated operating envelope and 15% operating margin) for the five USHPRRs.

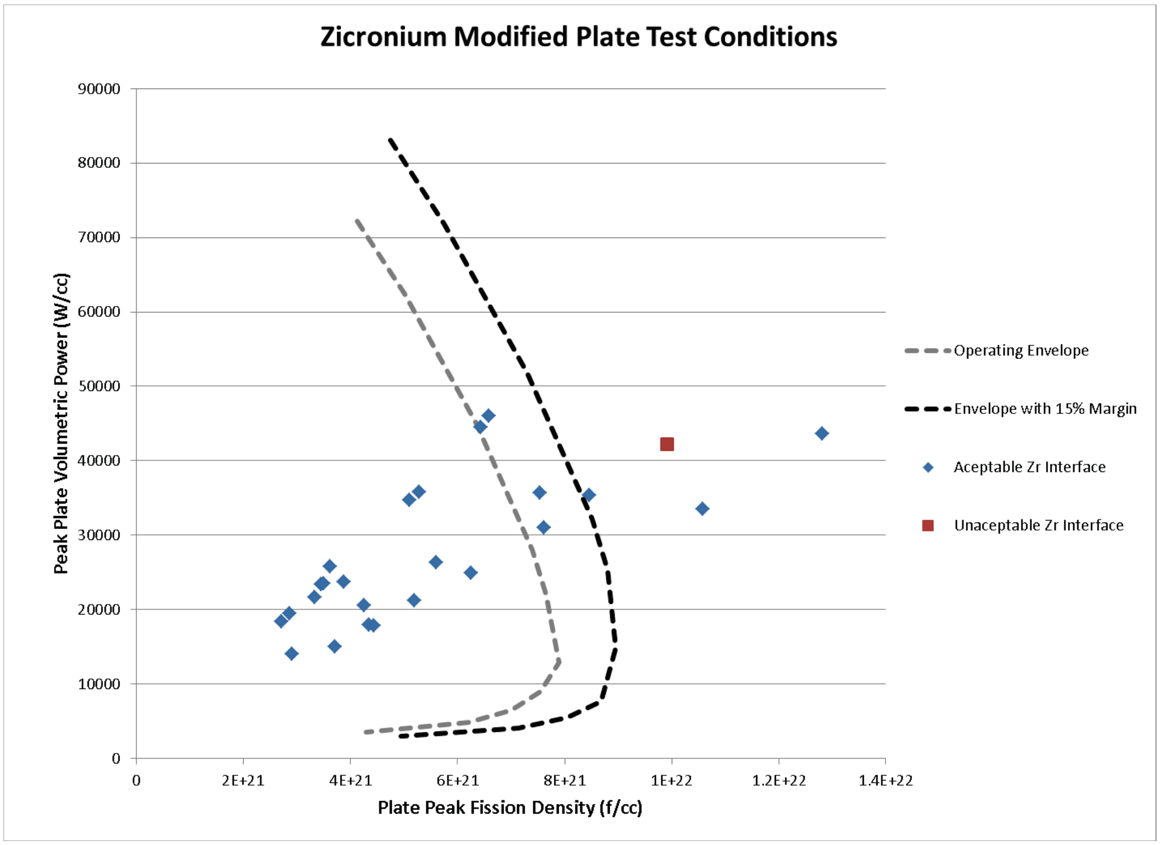

Figure 51. Test conditions of zirconium-modified-interface plates compared to operating-envelope requirements in terms of power density (W/cm^3) and fission density (f/cm^3).

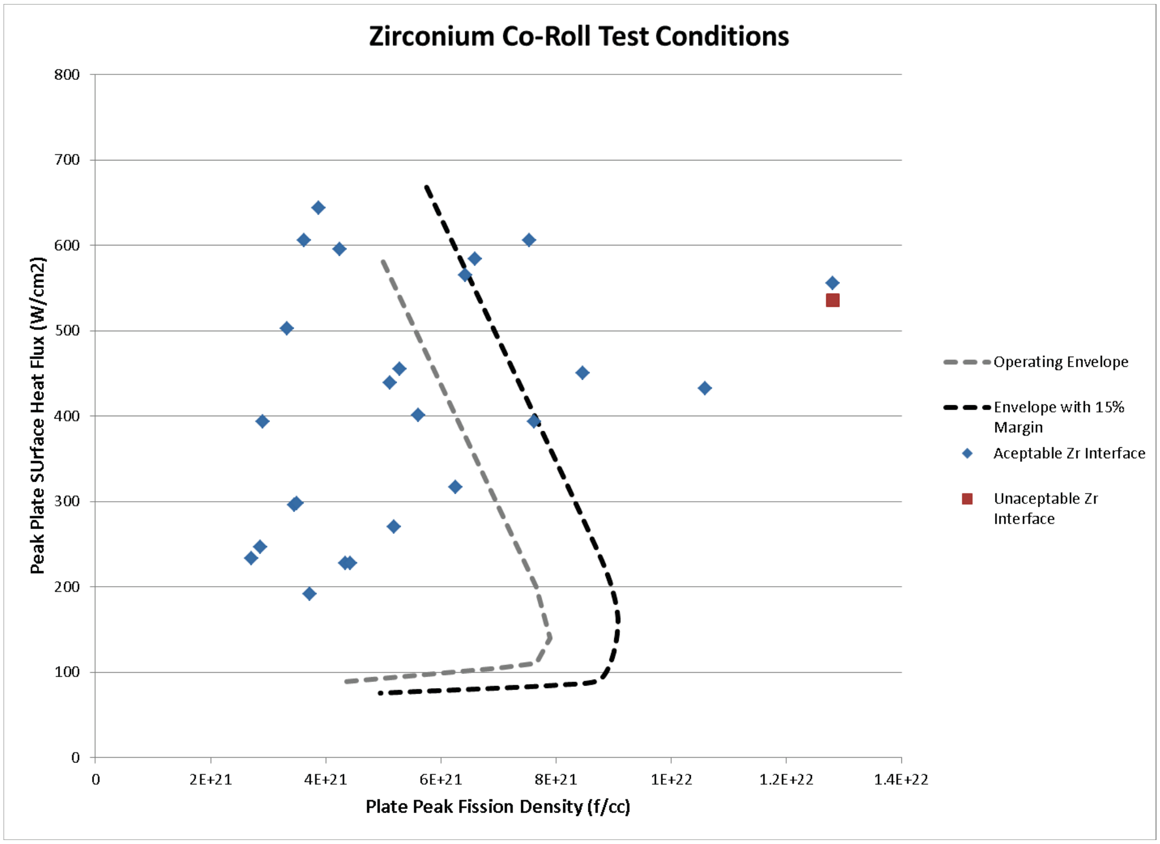

Figure 52. Test conditions of zirconium-modified-interface plates compared to operating-envelope requirements (W/cm^2).

Table 8. Summary of Zr-barrier-layer fuel attributes against down-select requirements.

Requirement	Evidence Requirement Is Met	Gaps
	Maintain Mechanical Integrity	
The mechanical response of the fuel meat, cladding, and interlayers during normal operations and anticipated transient conditions shall be established	The mechanical response of the fuel design was demonstrated to be acceptable through irradiation testing of mini-plates and full-size plates to an average fission density of 7.7E+21 and a peak fission density of 1.3E+22 fissions/cm^3. There were no in-reactor mechanical failures.	Fuel operating conditions during anticipated transient conditions have not been identified, with the exception of blister-threshold testing, and evaluation of fuel performance under these conditions has not been evaluated.
		Preliminary examination of irradiated RERTR–12 insertion 2 plates have shown a failure threshold at a plate average fission density of 7.7E+21 f/cm^3 (peak fission density of this plate in the location of the failure was 1.3E+22 f/cm^3). These plates require examination to determine the failure mode and the impact of the failure mode on fuel plate operation within the reactor mission envelope.
Diffusion-layer performance limits and manufacturing processes shall be established	The performance of the Zr diffusion barrier layer on U–10Mo fuel has been demonstrated to be acceptable through irradiation testing to a plate average fission density of 7.7E+21 (peak fission density of this plate in the location of the failure was 1.3E+22 f/cm^3). There were no in-reactor diffusion-barrier failures. Manufacturing processes as defined are technically acceptable for fuel research and development.	Zirconium-barrier-layer performance has only been evaluated for fuel plates fabricated using the current baseline fabrication process, which may not be optimal for scaled-up processes. Zirconium-barrier thickness and microstructure may change if a different fabrication process is adopted. The effect on fuel performance of this potential change has not been tested. The effects of cold rolling versus hot rolling on the fuel-swelling behavior have not been methodically evaluated to date.

Requirement	Evidence Requirement Is Met	Gaps
The fuel element shall not delaminate during normal operation and anticipated transients.	The resistance of the base fuel design to delamination was demonstrated to be acceptable through irradiation testing to an average fission density of 7.7E+21 (peak fission density of this plate in the location of the failure was 1.3E+22 f/cm^3). There were no in-reactor delamination failures of the down-selected fuel design.	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated.
		Preliminary examination of irradiated RERTR–12 insertion 2 plates have shown a failure threshold at an average fission density of 7.7E+21 f/cm^3 (peak fission density of this plate in the location of the failure was 1.3E+22 f/cm^3). These plates require examination to determine the failure mode and the impact of the failure mode on fuel-plate operation within the reactor mission envelope.
Maintain Geometric Stability		
The geometry of the fuel shall be maintained during normal operation and anticipated transients.	The resistance of the base fuel design to delamination was demonstrated to be acceptable through irradiation testing to an average fission density of 7.7E+21 fissions/cm^3 (peak fission density of this plate in the location of the failure was 1.3E+22 f/cm^3). There were no in-reactor delamination failures of the down-selected fuel design.	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated.

Requirement	Evidence Requirement Is Met	Gaps
		Preliminary examination of irradiated RERTR–12 insertion 2 plates have shown a failure threshold at an average fission density of 7.7E+21 f/cm^3 (peak fission density of this plate in the location of the failure was 1.3E+22 f/cm^3). These plates require examination to determine the failure mode and the impact of the failure mode on fuel-plate operation within the reactor mission envelope. Full-sized plates of geometries, reflecting the designs used in the USHPRRs, have not been tested.
		Verification of the geometric stability of formed plates after irradiation is pending completion of AFIP–7 PIE.
Performance at fission density, heat flux, and fuel geometry shall be maintained to avoid flow instability	The resistance of the base-fuel design to changes in geometry has been demonstrated through irradiation testing to an average fission density of 7.6E+21 and a peak fission density of 1.3E+22 f/cm^3. There were no in-reactor failures that would lead to flow instability within that range.	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been performed.
Changes in the channel gap shall not compromise the ability to cool the fuel.	The AFIP–7 irradiation test was conducted to demonstrate the stability of fuel coolant channels against irradiation driven fuel plate geometry changes. This test operated at an average power of 9,300 and peak power of 15,200 W/cm^3 and average fission density of 2.1E+21and a peak fission density of 3.0E+21 f/cm^3. In-canal analysis of the channel gap widths after irradiation indicates that the coolant channel gaps are stable under these conditions	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated.

Requirement	Evidence Requirement Is Met	Gaps
	References: INL/EXT-12-25915 "AFIP–7 Irradiation Summary Report"; TEV-1280 "AFIP–7 Ultrasonic Channel Gap Test Results."	Full-sized plates of complex geometries reflecting the designs used in the USHPRRs have not been tested.
	Stable and Predictable Behavior	
Fuel performance shall be known and predictable for the expected range of fuel composition and impurities, processing parameters, and microstructure for all credible environmental and irradiation conditions	Swelling behavior of the U–10Mo fuel has been characterized to an average fission density of 7.6E+21 and a peak fission density of 1.3E+22 f/cm^3, average heat fluxes of 490, and peak heat fluxes of 640 w/cm^2. The swelling rate has been found to be near linear and predictable over this range, and represented by the equation: $\left(\frac{\Delta V}{V}\right)_f = 15 + 6.3(f_d - 3) + 0.33(f_d - 3)^2$ Where f_d is fission density.	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated. Limiting values for fuel swelling have not been identified for evaluation of swelling rate against reactor cooling requirements.
	References: Y.S. Kim, G.L. Hofman, J. *Nucl. Mater.*, 419 (2011) 291	Preliminary examination of irradiated RERTR–12 insertion 2 plates have shown a failure threshold at an average fission density of 7.7E+21 f/cm^3 (peak fission density of this plate in the location of the failure was 1.3E+22 f/cm^3). These plates require examination to determine the failure mode and the impact of the failure mode on fuel-plate operation within the reactor mission envelope.

Requirement	Evidence Requirement Is Met	Gaps
Fuel swelling behavior shall be within the stable swelling regime (linear and predictable).	Swelling behavior of the U–10Mo fuel has been characterized to an average fission density of 7.6E+21 and a peak fission density of 1.3E+22 f/cm^3, average heat fluxes of 490 and peak heat fluxes of 640 w/cm^2. The swelling rate has been found to be near linear and predictable over this range, and represented by the equation: $\left(\frac{\Delta V}{V}\right)_f = 15 + 6.3(f_d - 3) + 0.33(f_d - 3)^2$ Where f_{d} is fission density.	
Uranium-molybdenum corrosion behavior after breach is established.	Several in-reactor experiment cladding breaches have occurred during the GTRI-FD irradiation testing program. Catastrophic failure of the fuel test plates did not result from these breaches. The breaches and resulting fuel loss and fission-product release have been documented. The lack of zirconium barrier is not expected to influence fuel corrosion bahavior. **References:** INL/EXT-11-21110 “AFIP–6 Breach Assessment Report”; INL/INT-10-17854 “RERTR–10A Test: Overview and Breach Assessment.”	
Irradiation behavior on scale-up shall be predictable	The AFIP irradiation test series provides fuel-performance data on larger-scale monolithic fuel test plates. The AFIP–2, 3, 4, 6, 6 MkII, and 7 tests demonstrate acceptable fuel performance over fission densities ranging to 6.0E+21 f/cm^3, surface heat fluxes ranging to 570 W/cm^2, and power densities ranging to 35,000 W/cm^3.	A decrease in blister-threshold temperature of AFIP–4 plates relative to RERTR series experiment mini-plates was noted during blister testing in 2011. Eight AFIP–4 plates were blister tested using the standard anneal procedure. Additional blister testing data is required to evaluate the presence of this phenomenon in full-size fuel test plates.

Requirement	Evidence Requirement Is Met	Gaps
	References: INL/EXT-12-26305, "AFIP-6 Mark II Irradiation Summary Report"; INL/LTD-11-21851, "AFIP-2 Post-Irridation Examination Summary Report."	Verification of the performance of full-size plates is pending completion of AFIP–6 MkII and AFIP–7 PIE. Irradiation at power densities required for current HFIR and ATR conversion fuel element designs have not been demonstrated.
Reactor Mission Performance Envelope		
The reactor performance envelope for normal operations and anticipated transients is established	Operating conditions based on available information have been documented in this report.	The reactor performance envelope established in GTRI reactor-conversion-requirements documents does not appear to be adequate to encompass peak fuel-plate operating conditions for HFIR and ATR. Fuel performance has not been verified for anticipated transients.
The selected fuel elements shall demonstrate peak heat flux of 475 W/cm^2 with a burn-up of greater than 50%.	Evaluation of fuel plate test conditions against the estimated reactor-performance envelope (Figure 6) indicates that test conditions are adequate to envelope all fuel operating conditions except for HFIR and ATR peak plates.	The reactor performance envelope established in GTRI reactor-conversion-requirements documents does not appear to be adequate to encompass peak fuel-plate operating conditions for HFIR and ATR.
Operating costs impacted by fuel conversion shall be minimized and agreed to by the reactor operator and the agency that provides the fuel	This requirement has not been satisfied.	Fuel production costs are not known at this time

Requirement	Evidence Requirement Is Met	Gaps
Fabrication/Manufacturing Development and Qualification		
Test fuel elements shall be manufactured for evaluation of fuel-element characteristics and performance (Basis: As fuel concepts are developed test specimens are required to evaluate the characteristics and performance of the fuel).	Test fuel elements have been manufactured and characterized, including bench-scale fabrication for the RERTR series experiments and prototype-scale fabrication of the AFIP experiments. The performance of these elements has been evaluated through irradiation testing and post-irradiation examination.	
Test fuel elements shall be manufactured for evaluation of fabrication processes, including manufacturing scale-up issues, maximum fuel-loading limits, fabrication porosity (both inside fuel particles and elsewhere in the fuel meat), fuel meat uniformity (meat thickness and uranium density), dimensional tolerances, fuel element manufacturing, hydraulic characteristics. (Basis: IAEA NF-T-5.2, Section 5.2.3. As fuel concepts are developed, test specimens are required to evaluate methods to manufacture the laboratory scale and full scale elements).	Test fuel elements have been manufactured and characterized, including bench-scale fabrication for the RERTR series experiments and prototype-scale fabrication of the AFIP experiments	Pilot-scale manufacturing of foils for the RERTR–FE test elements was not successful. Gaps in fabrication process development are currently being addressed by the Fuel Fabrication Capability pillar.
Fabrication of thin-clad, thick fuel meat shall be demonstrated.	Thin-clad, thick-fuel-meat plates were fabricated and irradiated as part of the RERTR–12 insertion 2 experiment.	PIE of the insertion 2 plates has not been completed to date. Blister anneal testing has not been performed on thick-fuel-meat plates.
Fabrication of finned fuel plates shall be demonstrated.		Finned plates have not yet been irradiated with the U–Mo monolithic fuel.
Fabrication with a diffusion barrier shall be demonstrated.	Both zirconium and niobium diffusion barriers have fabricated on a bench and prototypic scale and have been irradiation tested.	

8.2 Fuel with a U–Mo/Al–6061 interface

Eliminating the zirconium barrier layer may provide manufacturing cost advantages because it allows for the elimination of difficult-to-recover mixed zirconium/uranium waste stream. Fuel without the barrier layer was evaluated against requirements for the NRC-licensed reactors only to determine if this fuel would be acceptable for use under a limited set of operating conditions. The conditions evaluated are provided in Table 9.

Table 9. Peak fuel plate operating conditions for NRC-licensed reactors.

Reactor/fuel plate	Peak Power (W/cm^3)	Peak Heat Flux (W/cm^2)	Peak Fission Density (f/cm^3)
MITR Plate 1	3500	89	4.3E+21
MURR Plate 23	15826	226	3.4E+21
NBSR Plate 17	12910	139	7.9E+21

Figure 53 and Figure 54 show the conditions to which bare-foil plates have been tested along with the operating envelope for NRC-licensed HPRRs. Bare-foil plates tested were fabricated almost entirely using friction bonding, with the exception of the TLPB plates and three HIP-fabricated plates. Two HIP-fabricated plates that were tested outside of the operating envelope for NRC-licensed reactors showed signs of failure (pillowing in H1P02B and porosity accumulation on the interface of H1P010, see Figure 13).

Down-selection of fuel with an unmodified interface for conversion of these NRC reactors would require additional research and development, including irradiation testing designed to encompass these lower powers and fission densities to verify the stability of the interface and establish the margin to failure. Advances that have been made in the fabrication processes since the discontinuation of testing of plates fabricated with an unmodified U–Mo/Al interface could potentially decrease the failure probability, although there is no data that suggest that this would be the case.

Fuel performance (swelling, fission-gas retention, and mechanical stability) of the unmodified interface has been shown to be stable in previous experiments (select plates in RERTR–6 and 7). However, resistance to delamination has not been demonstrated to be consistent beyond 6,500 W/cm^3, 166 W/cm^2, and 3.3E21 f/cm^3.

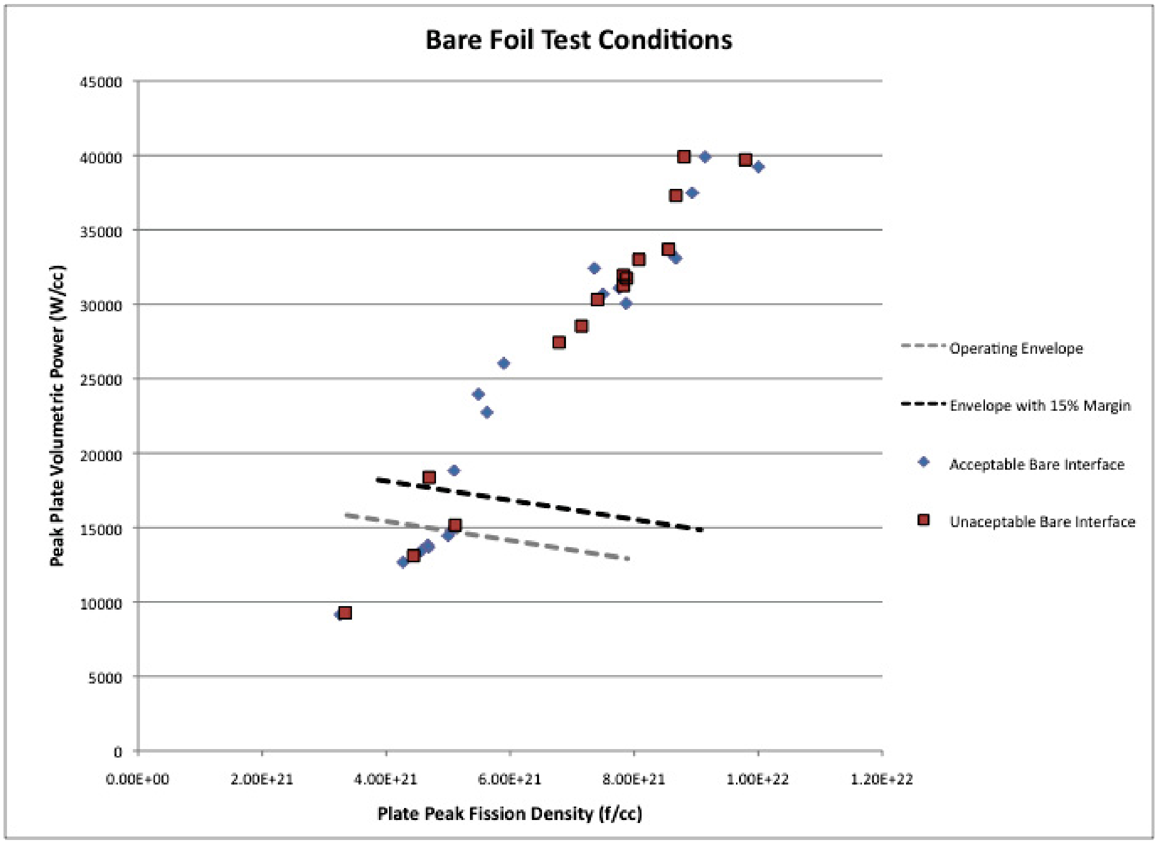

Figure 53. Bare-foil plate performance against operating envelope for NRC reactors requirements (W/cm^3).

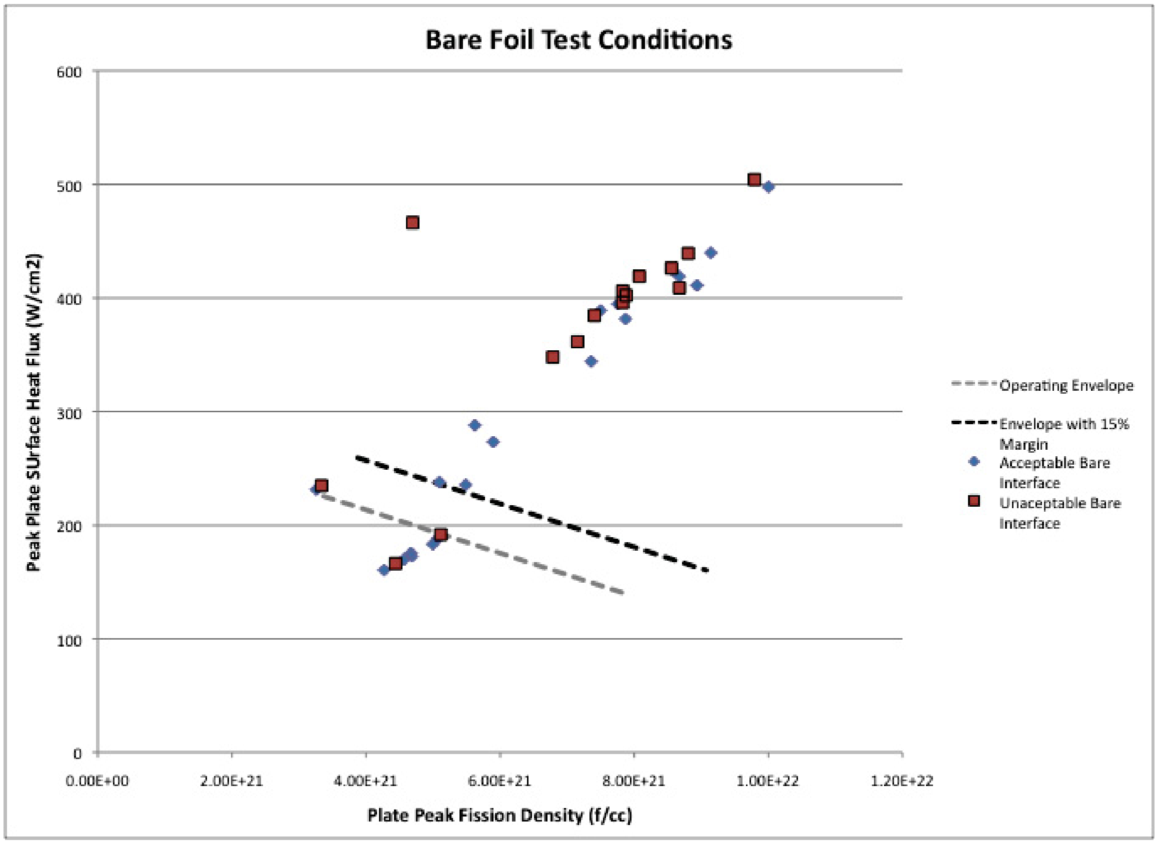

Figure 54. Bare-foil plate performance against operating envelope for NRC reactors requirements (W/cm^2).

It is not anticipated that corrosion behavior would be negatively impacted by the removal of the barrier layers.

Table 10. Summary of U–Mo/Al–6061 fuel attributes against down-select requirements for NRC-licensed reactors.

Requirement	Evidence that Requirement Is Met	Gaps
	Maintain Mechanical Integrity	
The mechanical response of the fuel meat, cladding, and interlayers during normal operations and anticipated transient conditions shall be established.	Requirement is not met. High occurrence of failure for fuel plates fabricated by friction bonding. High failure probability for HIP plates outside of bounding USHPRR conditions.	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated.

Requirement	Evidence that Requirement Is Met	Gaps
Diffusion layer performance limits and manufacturing processes shall be established	The diffusion between the U–Mo foil and the aluminum cladding has demonstrated acceptable behavior up to fission densities of 2.4E+21 f/cm^3. Above this fission density delaminations and fission gas accumulation has been observed. Requirement not met. Performance limits and an acceptable manufacturing process for non-barrier layer plates has not been established.	Thorough testing of the unmodified plates has not been performed at NRC reactor conditions (low power).
The fuel element shall not delaminate during normal operation and anticipated transients.	The unmodified interface fuel plates have demonstrated delaminations at plate average fission densities as low as 2.4E+21 using friction bonding and 7.4E+21 f/cm^3 for HIP bonding. Requirement not met. Fuel has exhibited delamination failures within the envelope of NRC-reactor operating conditions.	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated.
The geometry of the fuel shall be maintained during normal operation and anticipated transients.	Fuel operating conditions during anticipated transient conditions have not been identified, and fuel performance under these conditions has not been evaluated.	Preliminary examination of irradiated RERTR–12 insertion 2 plates have shown a failure threshold at an average fission density of 8.7E+21 and a peak fission density of 1.3E+22 f/cm3. These plates require examination to determine the failure mode and the impact of the failure mode on fuel plate operation within the reactor mission envelope.
Maintain Geometric Stability		
		Geometric stability of formed plates with an unmodified interface after irradiation is currently not known.

Requirement	Evidence that Requirement Is Met	Gaps
Performance at fission density, heat flux, and fuel geometry shall be maintained to avoid flow instability.	Requirement not met. The resistance of the unmodified interface system to changes in geometry has been demonstrated through irradiation testing to an average fission density of 2.4E+21 f/cm3. Some bare foil plates have survived to fission densities up to 6.2E+21 f/cm3. Within this range there were several in-reactor failures that could lead to flow instability.	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated. Large-size irradiation tested of unmodified interface plates has not been performed. Geometric stability at the large scale is not known.
Changes in the channel gap shall not compromise the ability to cool the fuel.		No tests have been performed with bare foils with the goal of demonstrating the stability of the fuel coolant channels. Previously seen delaminations would be an indication of concern. Further testing would be required. Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated.
	Stable and Predictable Behavior	
Fuel performance shall be known and predictable for the expected range of fuel composition and impurities, processing parameters, and microstructure for all credible environmental and irradiation conditions.	The swelling behavior of U–10Mo fuel has been characterized to an average fission density of 7.6E+21 and a peak fission density of 1.3E+22 f/cm^3 and average heat fluxes of 490 w/cm^2 and peak heat fluxes of 640 w/cm^2. Swelling rate has been found to be nearly linear and predictable over this range, and represented by the equation: $\left(\frac{\Delta V}{V}\right)_f = 15 + 6.3(f_d - 3) + 0.33(f_d - 3)^2$ Where fd is fission density.	Limiting values for fuel swelling have not been identified for evaluation of swelling rate against reactor cooling requirements.

Requirement	Evidence that Requirement Is Met	Gaps
Fuel performance shall be known and predictable for the expected range of fuel composition and impurities, processing parameters, and microstructure for all credible environmental and irradiation conditions	Swelling behavior of the U–10Mo fuel has been characterized to an average fission density of 7.6E+21 and a peak fission density of 1.3E+22 f/cm^3, average heat fluxes of 490 w/cm^2, and peak heat fluxes of 640 w/cm^2. The swelling rate has been found to be near linear and predictable over this range, and represented by the equation: $\left(\frac{\Delta V}{V}\right)_f = 15 + 6.3(f_d - 3) + 0.33(f_d - 3)^2$ Where f_{d} is fission density.	Preliminary examination of irradiated RERTR–12 insertion 2 plates have shown a failure threshold at an average fission density of 8.7E+21 f/cm^3 and a peak fission density of 1.3E+21 f/cm^3. These plates require examination to determine the failure mode and the impact of the failure mode on fuel plate operation within the reactor mission envelope.
	Processing parameters, specifically time at elevated temperatures, have a more dramatic effect on the performance of the bare foil system. Diffusion between the bare U-Mo fuel and aluminum cladding has been observed to create interaction layers that behave unsatisfactorily within the operating envelope of the HPRR's.	Low-power testing targeting only the NRC-licensed reactors of the bare-foil fuel system has not been performed.
	References: Y.S. Kim, G.L. Hofman*, J. Nucl. Mater.*, 419 (2011) 291	Fuel operating conditions during anticipated transient conditions have not been identified, and evaluation of fuel performance under these conditions has not been evaluated.
Fuel swelling behavior shall be within the stable swelling regime (linear and predictable).	The swelling behavior of U–10Mo fuel has been characterized to an average fission density of 7E+21 and a peak fission density of 1.3E+22 and average heat fluxes of 490 w/cm2 and peak heat fluxes of 640 w/cm2. Swelling rate has been found to be nearly linear and predictable over this range, and represented by equation $\left(\frac{\Delta V}{V}\right)_f = 15 + 6.3(f_d - 3) + 0.33(f_d - 3)^2$ Where fd is fission density.	The effects of hot and cold against only hot rolling have not been thoroughly studied.

Requirement	Evidence that Requirement Is Met	Gaps
		Calculated fuel swelling values based on plate thickness values for unmodified interfaces can be effected by fuel/cladding interaction as well as delamination.
Uranium-molybdenum corrosion behavior after breach is established.	Several in-reactor experiment cladding breaches have occurred during the GTRI FD irradiation testing program. Catastrophic failure of the fuel test plates did not result from these breaches. The breaches and resulting fuel loss and fission product release have been documented.	It is assumed that the zirconium barrier or its absence does not change the U–Mo corrosion behavior.
Irradiation behavior on scale-up shall be predictable.	The AFIP irradiation test series provides fuel performance data on larger-scale monolithic fuel test plates. The AFIP–2, AFIP–3, AFIP–4, AFIP–6, AFIP–6 MkII, and AFIP–7 tests demonstrate acceptable fuel performance over fission densities ranging to 6.0E+21 F/CM^3, surface heat fluxes ranging to 570 W/cm^2, and power densities ranging to 35,000 W/cm^3. **References:** INL/EXT-12-26305, INL/LTD-11-21851.	None of the larger scale tests have tested unmodified interface plates. It is not currently understood how an unmodified fuel foil would behave at large scale. A decrease in blister temperature of AFIP–4 plates relative to RERTR series experiment mini-plates was noted during blister testing in 2011. Eight AFIP–4 plates were blister tested using the standard anneal procedure. Additional blister testing data is required to evaluate the presence of this phenomenon in full-size fuel test plates. Irradiation at power densities required for current HFIR and ATR conversion fuel element designs have not been demonstrated.

Requirement	Evidence that Requirement Is Met	Gaps
Reactor Mission Performance Envelope		
The reactor performance envelope for normal operations and anticipated transients is established.	Operating conditions based on available information have been documented in this report.	The program has not formally documented the reactor operating envelope for normal operation and anticipated transients for the USHPRRs at this time.
The selected fuel elements shall demonstrate peak heat flux of 475 W/cm^2 with a burn-up of greater than 50%	Evaluation of fuel plate test conditions against the reactor performance envelope (Figure 31 and Figure 32) indicate that test conditions are inadequate to envelope all fuel operating conditions. Figure 33 indicates they may be acm3eptable under some NRC reactors only conditions.	Further testing at low power would be required to demonstrate this fuel type for use at NRC reactor conditions.
Operating costs impacted by fuel conversion shall be minimized and agreed to by the reactor operator and the agency that provides the fuel.	This requirement has not been satisfied.	Fuel costs are not known at this time.
Fabrication/Manufacturing Development and Qualification		
Test fuel elements shall be manufactured for evaluation of fuel element characteristics and performance. (Basis: As fuel concepts are developed test specimens are required to evaluate the characteristics and performance of the fuel.)	Test fuel elements have been manufactured and characterized, including bench scale fabrication for the RERTR series experiments and prototype scale fabrication of the AFIP experiments. The performance of these elements has been evaluated through irradiation testing and post-irradiation examination.	

Requirement	Evidence that Requirement Is Met	Gaps
Test fuel elements shall be manufactured for evaluation of fabrication processes, including manufacturing scale up issues, maximum fuel loading limits, fabrication porosity (both inside a fuel foil and elsewhere in the fuel meat), fuel meat uniformity (meat thickness and uranium density), dimensional tolerances, fuel element manufacturing, hydraulic characteristics. (Basis: IAEA NF-T-5.2, Section 5.2.3. As fuel concepts are developed, test specimens are required to evaluate methods to manufacture the laboratory scale and full scale elements)	Test fuel elements have been manufactured and characterized, including bench scale fabrication for the RERTR series experiments and prototype scale fabrication of the AFIP experiments	
Fabrication of thin clad, thick fuel meat shall be demonstrated		Thin clad, thick fuel meat plates have not been fabricated or irradiated for this fuel design.
Fabrication of finned fuel plates shall be demonstrated.		Finned plates have not yet been irradiated with the U–Mo monolithic fuel.

9. CONCLUSIONS

Based on data available in 2009, a decision was made to focus U–Mo monolithic fuel development and qualification efforts on a single fuel design. This fuel design consists of a U–10Mo (wt.%) monolithic fuel foil, a zirconium barrier layer applied to the faces of the foil by a co-rolling process, and 6061 aluminum cladding bonded to itself and to the fuel foil by a HIP process. This update to the report includes fuel performance data available as of May 2013, outlines data and observations that are significant to informing the down-selection process, reviews the data against fuel down-select requirements, and updates the recommendation on a fuel system suitable for fuel qualification.

Data from the RERTR–6, 7, 9, 10, and 12 (insertion 1) and AFIP–2, 3, and 4 irradiation tests performed in the ATR, as well as bench-scale fabrication research and development, have been used to down-select a monolithic fuel design for qualification.

Additional preliminary data from the AFIP–6, 6 MkII, 7, and RERTR–12 insertion 2 tests are not presented here, but do add confidence to the down-select decision.

Because of inherently better gamma-phase stability and qualitatively better rolling behavior, U–10Mo was selected as the fuel alloy of choice, exhibiting reasonable uranium density and good irradiation performance. A zirconium foil barrier between the fuel and cladding was chosen to provide a predictable, well-bonded, fuel-cladding interface, greatly reducing fuel-cladding chemical interaction. The fuel plate testing conducted to inform this selection was based on the use of U–10Mo foils fabricated by hot and cold co-rolling with a Zr foil. The foils were subsequently bonded to the Al–6061 cladding by hot isostatic press or friction stir weld.

Since the original down-select decision was made, additional information has become available on fuel performance and on the cost of the fuel system. Preliminary cost estimates indicate that U–Mo-Zr waste generated during fabrication of the current fuel system is difficult to recycle and adds substantially to production cost.

This revision to the report is issued as a draft for discussion by the USHPRR program. The document will be revised as additional information becomes available. This additional information may include either updates to information about fuel-normal and anticipated transient conditions as conversion fuel-element designs evolve or additional information on fuel performance gained through the U.S. or international testing programs. In particular, information from the RERTR–12 experiment will be included when available.

This update also includes an assessment of the suitability of a fuel system without a zirconium barrier layer for use in USNRC-licensed research reactors. Testing to date has shown a high failure rate for monolithic fuel plates that do not include a modified U–Mo/Al interface. Recent assessment of the baseline fabrication process has indicated that eliminating the zirconium barrier layer may provide manufacturing cost advantages because it allows for the elimination of a difficult-to-recover mixed zirconium/uranium/molybdenum waste stream. Fuel without the barrier layer was evaluated against requirements for the NRC-licensed reactors only to determine if this fuel would be acceptable for use under a limited set of operating conditions. All plates tested within this limited operating envelope were fabricated using friction bonding. Three (3) plate failures occurred (L2F030, N1F030, and N1F090) among the 25 friction-bonded fuel plates tested under these conditions, exhibiting a failure rate of 12%. Of these three failures, two exhibited delamination, and one exhibited a heavily reacted interface, considered to be a precursor to failure.

Two (2) of the 3 bare foil plates fabricated by HIP failed, both during testing outside of the NRC-reactor operating envelope. The surviving bare foil fabricated by HIP processing was L1P020, which was a U–12Mo fuel foil, irradiated to a peak fission density of 8.5E+21. The high failure rates and unpredictable behavior originally led the program to abandon this design in favor of fuel with modifications to the U–Mo/Al interface to increase fuel reliability.

A gap in the data for fuel plates with an unmodified U–Mo/Al interface exists at high burnup (8E+21 f/cm^3) and moderate power (150 W/cm^2) conditions representative of NBSR peak conditions. If HIP is chosen as the fabrication method for this type of monolithic fuel plates, further assessment of the performance of HIP plates within the range of conditions that bound the NRC-licensed reactors would be required. Consideration of this fuel design and fabrication process for qualification for use in NRC-licensed USHPRRs would require both a filling of these gaps through mini-plate and full-size-plate experiments under these conditions and demonstration of insignificant failure rates for fuel fabricated with an optimized fabrication process.

Fuel plates fabricated with a zirconium barrier layer exhibited no failures within the peak USHPRR operating envelope, regardless of the fabrication method (HIP or friction bonding). Because of its robust performance to high fission densities under a wide range of conditions relevant to USHPRRs, a U–10Mo monolithic fuel plate, using a Zr foil barrier between fuel and cladding, fabricated by co-rolling the layers and then bonding by a HIP process, is recommended for qualification by the USHPRR program. Recent information on ATR and HFIR fuel-operating conditions indicates that an expanded testing envelope is required to ensure that the selected fuel performs under this extended range of conditions.

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Appendix A

Data Plots

Appendix A

Data Plots

Figure 55 shows all plate failures by interlayer type along with the reactor requirements as a function of power density versus fission density.

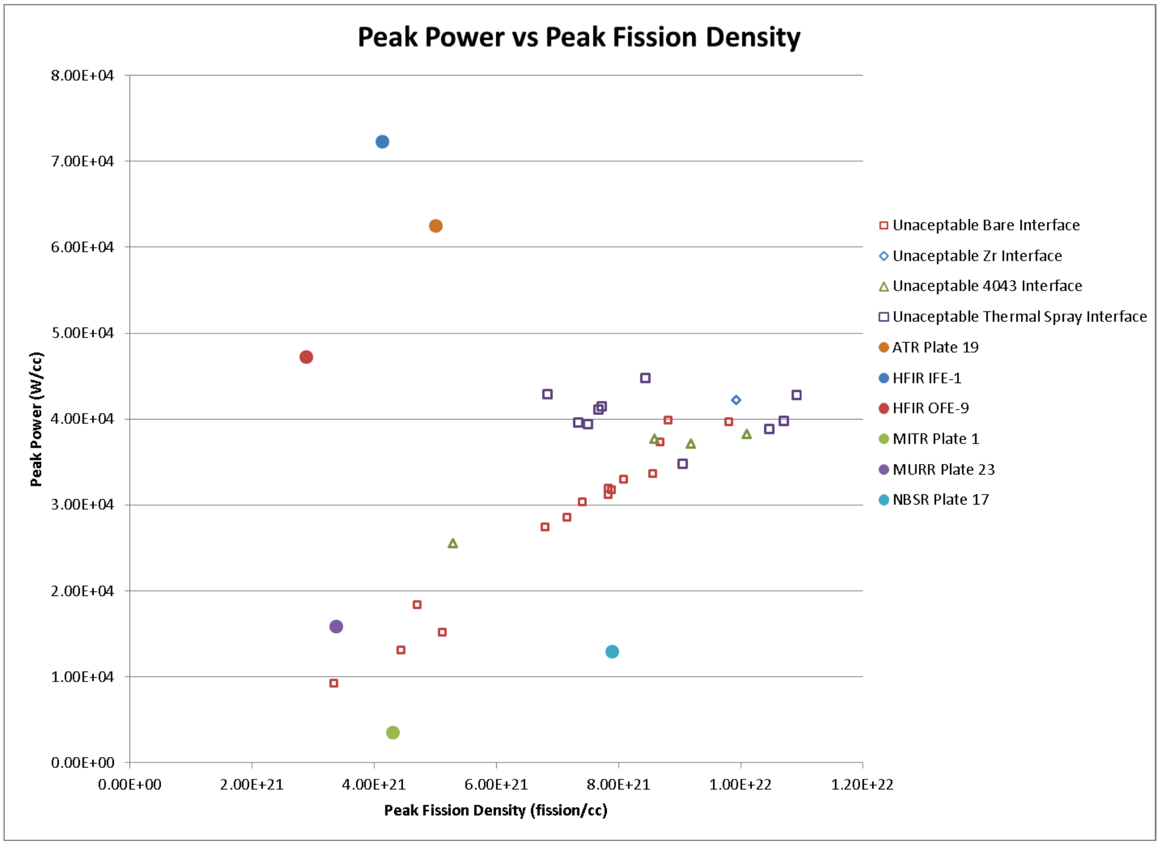

Figure 55. All fuel failures versus reactor operation conditions.

Appendix B

Failure Analysis

Appendix B

Failure Analysis

Failures that have occurred in the U–Mo monolithic system (except plates with various silicon treatments) have been analyzed to determine a failure probability as a function of fission density. Of the plates containing a zirconium interlayer at high fission density, one failure was from RERTR–12 insertion 1; three were from RERTR–12 insertion 2. Failures in the bare foil plates exclude plates that delaminated during sectioning. A preliminary analysis is presented here, with the objective of determining the fission density at which the probability of failure is insignificant. The analysis system used is based on the Weibull failure model, represented by Equation 1:

Equation 1 Weibull equation used for failure analysis

$$F(t) = 1 - \exp\left[-\left(\frac{t - t_o}{\eta}\right)^{\beta}\right]$$

Where:

F(t) = Cumulative probability of failure (the area under the distribution from t_o to t).

β = Weibull slope

η= Characteristic life

t = Random variable (time, stress, size, cycles, etc.)

t_o = Origin of the distribution

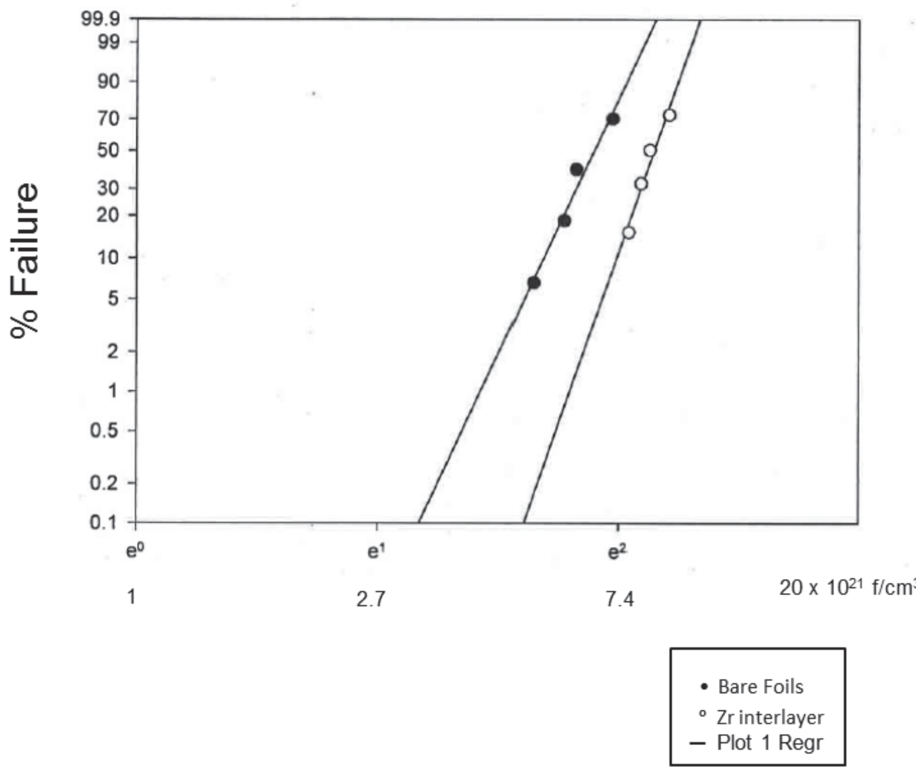

Figure 56. Failure probability of all monolithic HEU mini-plates from RERTR–7, 8, 9, 12.

This preliminary analysis points to a failure mode that depends on both accumulated fission damage (burnup) and thermal stresses (ΔT). There appears to be a clear separation between bare foils and foils with a zirconium interlayer. The data indicate that bare foils may be viable to low fission densities (fd <3E+21 f/cm^3). Although no failures have occurred at fission densities lower than 7.6E+21 f/cm^3 in Zr-barrier-layer plates, the Weibull analysis indicates that the probability of failure is not insignificant. The plates containing various applied coatings of silicon represent a challenging number of variables and need further analysis.

The failure mode for monolithic fuel plates is fundamentally different from that for dispersion fuel plates, which is determined by fission-gas behavior. Cyclic reactor behavior may be an important issue for monolithic fuels. The present analysis does not include power-peaking factors in certain fuel plates and uncertainties. It is based on limited post-irradiation examination data from mini-plates only.