U.S. HIGH PERFORMANCE RESEARCH REACTOR CONVERSION PROGRAM: AN OVERVIEW ON ELEMENT DESIGN

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ABSTRACT

Based on favorable irradiation behaviour, the U10Mo “monolithic” fuel has been selected for qualification in the low-enriched uranium (LEU) conversion of the U.S. high performance research reactors (USHPRRs). Irradiation behaviour has previously been demonstrated in test plate geometries across a range of irradiation conditions similar to those found in the current USHPRRs. Based on the initial success of this fuel system, LEU fuel element designs of the USHPRR LEU conversion cores have been (or are being) optimized by each reactor facility to allow the reactors to meet mission, operational, and safety basis requirements using monolithic LEU fuel. The paper will provide an overview of the U.S. effort on the development of LEU design for its domestic fleet. Connection with other parts of the USHPRR conversion program – namely fuel qualification and fuel fabrication – will also be described.

1. Introduction

The U.S. National Nuclear Security Administration (NNSA) Office of Material Management and Minimization (M^3) developed an integrated approach to address the persistent threat posed by unintentional proliferation of nuclear materials. As explained in [1], *“The primary objectives of M^3 is to achieve permanent threat reduction by minimizing and, when possible, eliminating weapons-usable nuclear material around the world.”*

One of the M^3 missions is reactor conversion. The M^3 Conversion Office works around the world to convert (or verify shutdown of) civilian facilities that use or produce weapon-usable nuclear materials. Since the U.S. Department of Energy Reduced Enrichment for Research and Test Reactor (RERTR) program began in 1978 [2], Argonne National Laboratory has been involved in reactor conversions and associated fuel development activities along with many domestic and foreign partners. Following the formation of NNSA, the domestic RERTR program has been integrated into the missions of M^3. This program has currently completed 69 reactor conversions to the use of low-enriched uranium (LEU) fuel. In addition, 26 reactor facilities have been verified to have been permanently shut down. These 95 reactors include conversion of 20 U.S. reactors among the 37 countries on six continents where conversions have occurred.

Within M^3 reactor conversion, the goal of the U.S. High Performance Research Reactor (USHPRR) conversion program is to convert the six domestic high performance reactors (including one critical facility) that still use, and regularly refuel with, Highly Enriched Uranium (HEU) fuel. Based on demonstrated favorable irradiation behaviour, the USHPRR program focuses on the development of the LEU uranium-molybdenum (UMo) “monolithic” alloy fuel [3]. Feasibility analyses [4-8] as well as further analyses have shown that the UMo monolithic fuel system would allow the conversion of all the domestic high performance reactors.

The goal of this paper is to provide an overview of the U.S. LEU designs and the integration of the design effort within the USHPRR program. The program is divided into four pillars, as shown in Figure 1, each responsible for a specific aspect of the conversion effort:

- Fuel Qualification (FQ): characterize and document the fuel performance and properties, and design and execute the test and qualification irradiation campaigns (led by Idaho National Laboratory)
- Fuel Fabrication (FF): deploy viable industrial processes for the commercial production of LEU UMo monolithic elements for the six facilities under the scope of the program (led by Pacific Northwest National Laboratory)
- Reactor Conversion (RC): perform along with the facilities all the necessary activities to convert the reactors such as fuel element design, reactor core safety analysis, licensing or other regulatory submittals and implementation (led by Argonne National Laboratory)
- Cross-Cutting (CC): address cross-program activities including, but not limited to, transport and back-end planning (led by Savanah River National Laboratory)

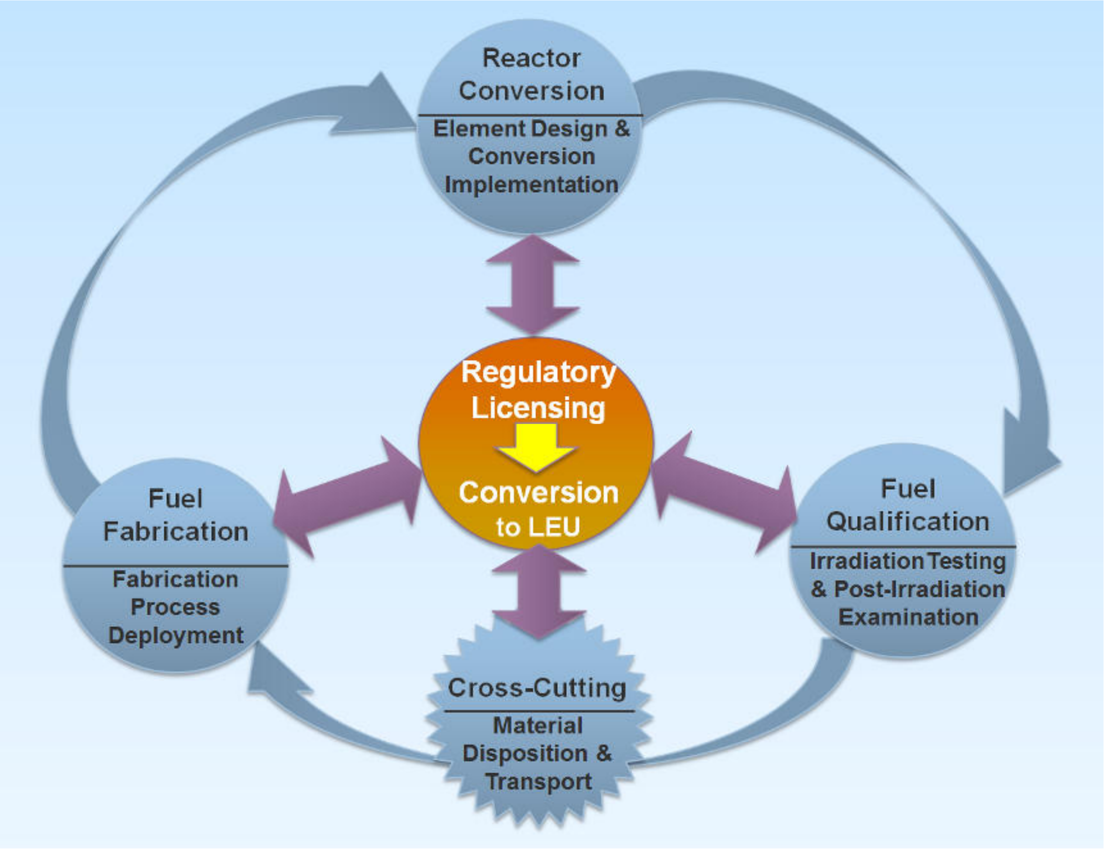

Figure 1 – Structure of the four pillars of the program to convert USHPRRs to the use of LEU fuel.

Through activities conducted within each pillar, the program works cooperatively with many organizations including the USHPRR reactor facilities, regulators, national laboratories and plants, and a commercial fuel supplier. This paper discusses the RC pillar design efforts and status.

2. The U.S. High Performance Research Reactors

The following provides a brief description of the six reactors (including one critical facility) constituting the USHPRR fleet.

2.1. The Massachusetts Institute of Technology Reactor [MITR]

The Massachusetts Institute of Technology Reactor (MITR) is a research reactor located in Cambridge, Massachusetts, designed primarily for experiments using neutron beam and in-core irradiation facilities. Upgraded from MITR-I and relicensed as MITR, the MITR reactor has been in operation since 1958. It delivers a neutron flux comparable to current LWR power reactors in a compact 6 MW core using HEU dispersion fuel enriched at 93 wt% ^{235}U. More details on the facility can be found in [9, 10].

The MITR facility, shown in Figure 2, is currently licensed to operate at 6 MW. The hexagonal core contains twenty-seven fuel locations. The core is light water moderated and cooled, and is surrounded by a heavy water (D_2O) reflector. The MITR HEU fuel element is rhomboid-shaped with fifteen flat plates of the same fuel and plate thickness. The fuel is made of uranium aluminide (UAl_x) cermet dispersed in an aluminum matrix. To increase heat transfer, the cladding has 110 vertical grooves (or fins) on each side of the plate.

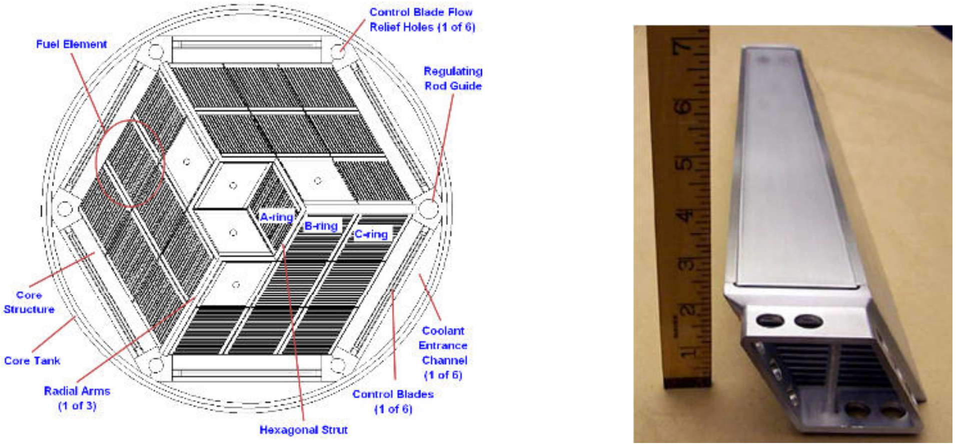

Figure 2 – Schematic of the MITR reactor core configuration (left) and MITR reactor fuel element (right). Source: [9]

2.2. The University of Missouri Research Reactor [MURR]

The Missouri University Research Reactor (MURR) is a multi-disciplinary research and education facility providing a broad range of analytical and irradiation services to the research community and the commercial sector. The facility is situated in the central portion of the University of Missouri Research Park. It first achieved criticality in 1966. MURR refuels weekly and, operates with a high availability over 90% in order to meet its mission needs. More details on the facility can be found in [11].

The MURR core is a fuel region made of eight fuel elements. The fuel elements are placed vertically around an annulus between two cylindrical aluminum reactor pressure vessels, as depicted in Figure 3. The MURR is currently fueled with UAl_x HEU fuel enriched at 93 wt% ^{235}U. Each HEU fuel element has 24 curved plates (all of similar fuel and plate thickness) that form a 45-degree arc. Schematics of the MURR core and fuel element are shown in Figure 3.

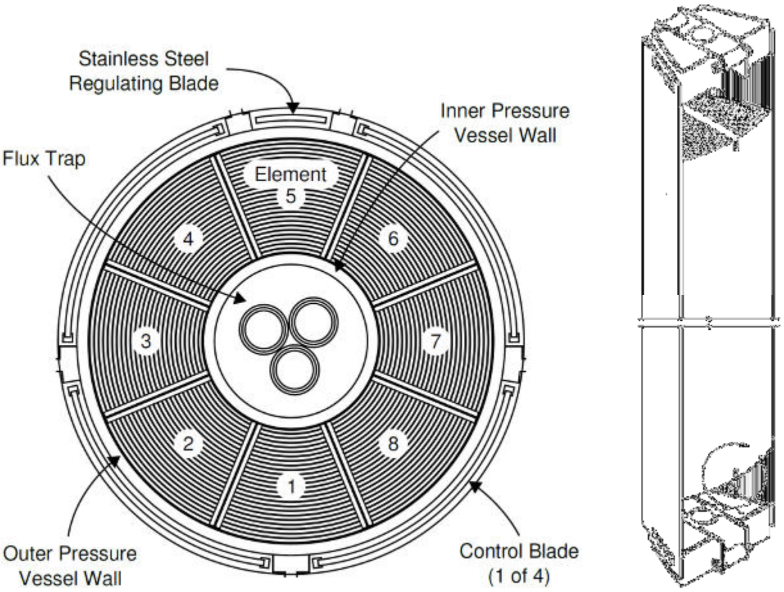

Figure 3 – Schematic of the MURR core configuration (left) and HEU element (right).

2.3. The National Bureau of Standard Reactor [NBSR]

As explained in [12], the Department of Commerce National Institute of Standards and Technology (NIST) operates NBSR, which is a heavy water moderated and cooled reactor operating at 20 MW. The reactor has been in operation since 1967, provides intense neutron beams and has been upgraded to include a liquid hydrogen cold source.

The HEU fuel for NBSR, enriched to 93 wt% ^{235}U, is made of U_3O_8 dispersed in an aluminum matrix. There are 30 fuel elements in the core on a triangular pitch. The fuel elements are split axially into two halves with a gap located between the two halves at the vertical mid-plane of the core. This gap allows the beam tubes to be pointed directly at the mid-plane of the core so that thermal neutrons can escape for use in thermal and cold neutron scattering research devices while minimizing contamination from fast neutrons and gamma rays. Each half-element encapsulates 17 curved identical fuel plates in the materials test reactor (MTR) geometry. The control elements within the NBSR core consist of four semaphore-type shim safety arms and a single automatic regulating rod. Schematics of the NBSR core and fuel element are shown in Figure 4. The NBSR is operated for 38.5 day cycles. At the end of each cycle, four fuel elements are removed from the core. The remaining 26 fuel elements are moved to new positions and four fresh fuel elements are inserted into the core. Of the 30 fuel elements, 14 remain in the core for seven cycles and 16 for eight cycles.

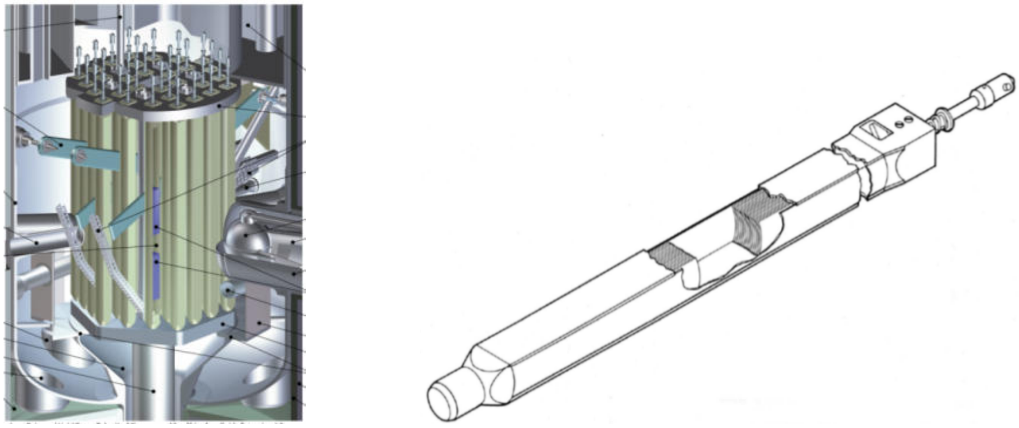

Figure 4 – NBSR vessel internals, reactor core (left) and NBSR fuel element (right). Source: [12]

2.4. The Advanced Test Reactor [ATR] and its Critical Facility [ATRC]

As explained in [13, 14], the Advanced Test Reactor (ATR) located at the Reactor Technology Complex of the Idaho National Laboratory (INL), is a 250-MW (thermal) high flux test reactor. Full power operation began in August 1969. Since then, the ATR has been utilized to study the effects of radiation on reactor structural and fuel materials, and to produce medical and industrial isotopes.

The ATR core, represented in Figure 5, contains 40 fuel elements arranged in a serpentine annulus between and around nine main flux traps. The fuel element consists of 19 curved plates of different widths (but all of identical fuel thickness), attached to side plates, forming a 45-degree sector of a circular annulus in cross section. The fuel meat consists of highly enriched (93 wt% ^{235}U) UAl$_x$ fuel powder dispersed in aluminum. The fuel plates are moderated by light water and reflected by beryllium blocks.

In each element, all 19 fuel plates are loaded with 93 wt% enriched uranium in an aluminum matrix. The eight inner and outer plates (1 through 4 and 16 through 19) contain boron as a burnable poison to flatten the power distribution.

ATRC is a critical mockup of ATR used to test and verify the worth of experimental devices and samples before being inserted and irradiated in ATR.

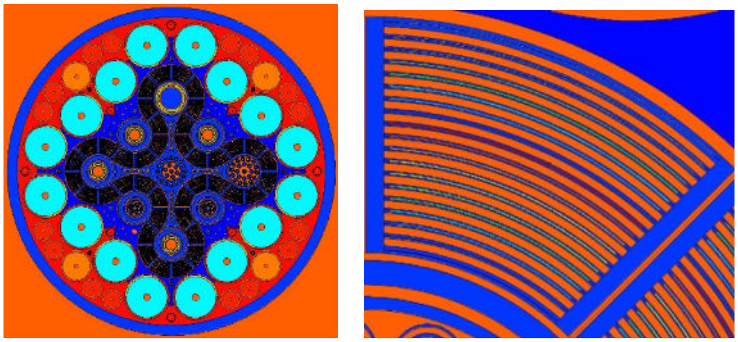

Figure 5 – ATR MCNP full core model (left) and element detail (right).

2.5. The High Flux Isotope Reactor [HFIR]

The High Flux Isotope Reactor (HFIR), located at Oak Ridge National Laboratory (ORNL), achieved first criticality in 1965. The reactor fulfills a wide range of missions, including providing intense neutron beams for science, and continues its original mission of trans-plutonium isotope production. After a shutdown and modification made to the operating pressure of the primary coolant, the reactor was re-started in 1989 and reached its new full power of 85 MW thermal (100 MW before) in 1990. Significant upgrades of the neutron scattering experiment equipment was carried out and completed in 2007 including the addition of a liquid hydrogen cold source (see [15] for more details).

HFIR is a light water moderated and cooled, beryllium reflected, flux-trap type reactor. The reactor core assembly is contained in a pressure vessel located in a light water pool. A cross-sectional view of the reactor is given in Figure 6 on the left. At the center of the core, a hole forms the flux trap target region. The flux trap is surrounded by two concentric annular fuel elements. Each consists of a sandwich-type fuel plate (fuel surrounded by cladding) curved as a circle involute. The involute shape allows the coolant gap between the plates to have a constant thickness. A picture of the fuel element is shown in Figure 6 on the right. Within a plate, the fuel thickness varies along the fuel width from one edge to the other to flatten the power. In addition, boron is present in the inner element plates to reduce the initial excess reactivity.

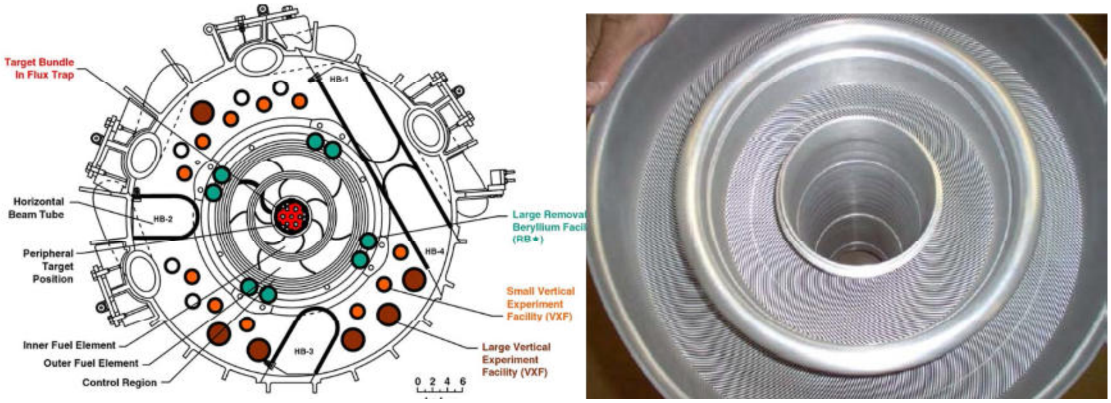

Figure 6 – Schematic of HFIR core (left) and HFIR fuel elements (right). Source: [15]

2.6. Reactor Characteristics

Table 1 below provides some key characteristics on the USHPRR fleet. It can be seen that significant geometric and operating conditions differences exist between the reactors preventing the development of a single LEU design solution to convert all of them. Instead a unique approach must be developed for each individual reactor.

Table 1 – Key characteristics of the HEU USHPRR fleet. Sources [4-15]

Reactor	**ATR**	**ATRC**	**HFIR**	**MURR**	**MITR**	**NBSR**
HEU Power	≤ 250 MW	≤ 600 W	85 MW	10 MW	6 MW	20 MW
Operating Cycle (days)	1 to 60	—	26	7	up to 70	38.5
HEU Fuel Type	U-Al$_x$	U-Al$_x$	U_3O_8-Al	U-Al$_x$	U-Al$_x$	U_3O_8
Fuel Enrichment	93%	93%	93%	93%	93%	93%
Fission Density Limit (f/cm^3)	2.3E+21	2.3E+21	1.5E+21	2.3E+21	1.8E+21	2.6E+21
Peak Heat Flux (W/cm^2)	476*	—	320	198	71	107
Element Shape	45° sector	45° sector	Cylindrical	45° sector	Rhomboid	Square
Number of Fuel Elements	40	40	2	8	22-24 (27 max)	30 (37 max)
Burnable Poison	Boron-10	Boron-10	Boron-10	None	None	None
Plates / element	19	19	171/369	24	15	17
Fuel Plate Shape	Curved	Curved	Involute	Curved	Flat + fins	Curved
Coolant	H_2O	H_2O	H_2O	H_2O	H_2O	D_2O
Inlet Pressure (bar abs.)	25.5	Atmospheric	32.3	4.7	1.3	1.2
Inlet Temp (°C)	52	Ambient	29	49	38	40
Flow Velocity (m/s)	14.4	None	15.5	7.6	2.6	5.8
Flow Direction	Downward	None	Downward	Downward	Upward	Upward
Reflector	Beryllium	Beryllium	Beryllium	Beryllium/ Graphite	D_2O / Graphite	D_2O

* Note that this value refers to a lobe operating at a power of 60 MW.

3. UMo Element Design Overview

3.1. UMo "monolithic" Fuel System

Due to its very high uranium density and its stable and predictable behaviour during irradiation, the uranium-10 wt% molybdenum metallic alloy (U10Mo) fuel system has been selected to convert the USHPRRs [16].

Unlike dispersion fuels (i.e. U-Al_x, U_3O_8, U_3Si_2), the U10Mo monolithic fuel meat is only made of U10Mo in an alloy form and does not have an aluminium matrix, as depicted in Figure 7. This allows the fuel meat – called a foil – to reach a uranium density of 15.3 gU/cm^3 when the molybdenum content is equal to 10% in weight. A plate is formed by cladding the UMo foil with aluminium.

To prevent the undesirable reaction between the aluminium and the U10Mo foil, and to avoid delamination, a protective layer is added between the foil and the aluminum alloy (AA 6061) cladding. The reference U10Mo fuel system uses a 25.4 µm-thick (1 mil) layer of zirconium. It has been demonstrated that such a layer prevents undesirable Al-U reaction layers even at high burnup and fission rate.

Designers can, to some extent, vary the fuel volume fraction in dispersion fuel to achieve specific goals (i.e., to reduce the volume fraction to decrease power peaking or to increase the volume fraction to increase the core reactivity). Due to its alloy form, the same approach cannot be used with U10Mo monolithic foil; the foil thickness can be varied instead to achieve the same goals.

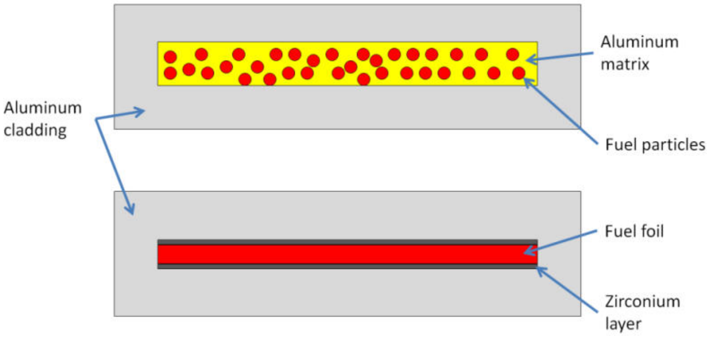

Figure 7 – Schematic cross-section of a dispersion fuel plate (top) and monolithic fuel plate (bottom).

3.2. Fuel Properties and Fabrication Control

Significant data exist on UMo metallic alloy fuel relative to fast reactors starting from the 1950's and 60's; However, work on this fuel at lower operating temperatures typical of MTR-type plate reactors was not initiated until the RERTR program resumed testing in 2001 **Error! Reference source not found.**. Because the UMo monolithic fuel system is still under development, design analyses are built on a number of assumptions that will eventually be replaced by qualified data and a final fuel specification.

The four the USPHRR pillars interact with one another to converge to a solution satisfying all conversion requirements. The FQ pillar provides data to the RC pillar on the behavior and properties of the fuel system. Some of these properties depend directly or indirectly on the fabrication process developed and used by the FF pillar. As a result, the RC pillar can provide designs having unique irradiation conditions and assumed fabrication tolerances. The FQ pillar designs and runs experiments to test the fuel system in prototypic conditions while the FF pillar deploys a commercial fabrication process capable of meeting the required specifications. Along with cross-cutting activities led by the CC pillar, the conversion process therefore requires a high degree of coordination.

3.3. Constraints and Requirements

Each reactor has its own specific set of facility constraints and various mission requirements that frames the conversion problem in a unique way. However, reactor-specific constraints and requirements are derived from more generic ones listed below. For the USHPRRs, in a general sense, the engineering problem to be solved is to find U10Mo-based LEU designs for all USHPRRs that:

1) Meet or exceed performance requirements (No significant loss with respect to HEU capability. Improved performance for key mission(s) may be possible during conversion.)
2) Meet or exceed safety requirements
3) Minimize element/plant modifications
4) Minimize conversion-related impacts, including impacts to the operational cycle

3.4. Design Options

Conversion problems are complex by nature because they have to satisfy multiple requirements simultaneously. Often, these requirements call for design changes that go in opposite directions. For instance, in order to maximize performance one may want to increase fissile mass while in order to satisfy safety requirements, one may need to decrease fissile mass. It is, therefore, often impossible to maximize performance and safety requirements simultaneously without one or more design changes. This is the reason why stakeholders and designers must frequently discuss trade-offs and define requirement in order to arrive at a design that meets all reactor mission and safety requirements with the simplest design.

3.4.1. Performance

A critical impact that designers face when trying to establish an LEU design for a given HEU reactor is to mitigate the neutron flux losses induced by the fuel change. Due to the significantly larger amount of ^{238}U in LEU fuel, parasitic captures occurring in this isotope increase, causing some neutron flux losses. For the USHPRRs which all currently use 93 wt% enriched fuel, going to U10Mo monolithic enriched at 19.75 wt% typically leads to a flux penalty of about 15% prior to mitigation. There are several ways to mitigate the neutron flux loss penalty. Those most frequently used in conversion analysis are briefly discussed below:

Neutron fluence conservation

The neutron fluence is defined as the neutron flux integrated over time. The fluence appears explicitly in the Bateman equations that describe the transformation of the nuclide exposed to a flux of particles. Although it is not as commonly implemented, a reactor that could benefit from having additional days of continuous operation could elect to design towards an extension of a particular fuel cycle. This constraint would tend to minimize the annual number of fuel elements fabricated annually. If the design compensates through the addition of operating days to equal or exceed the fluence delivered, isotope production and other scientific missions could be met without implementing other, potentially more costly, options.

Reactor power increase

The magnitude of the neutron flux is linearly proportional to the reactor power. Increasing the reactor power is therefore a very straightforward way to compensate for neutron flux losses.
Increasing the reactor power, however, is not always an easy solution to implement. On the design side, higher power typically means smaller thermal-hydraulic margins. On the operation side, a power uprate may lead to costly plant modifications (new piping, additional pumps, increased pressure…) and licensing hurdles.

Out-of-core improvements

Instead of perturbing reactor operations (by increasing power and/or cycle length), a facility may elect to pursue an upgrade of the experimental devices. This solution – to upgrade the facility other than the core design – is particularly well suited for reactors specialized in neutron scattering. The approach can potentially carry less risk, cost less and be easier to implement than other options.

3.4.2. Geometric Change and Complex Features

Though modifications are minimized, it is nonetheless often necessary to make geometric changes to a fuel element in order for the reactor to achieve criticality, to preserve operational cycle length, and to meet performance and safety requirements. Geometric modifications can be limited to changes within the plate (thickness, length and width of the fuel) or extend to the fuel element external geometry (plate thickness, length, number of plates…). In addition, HEU fuel elements may have features considered difficult to conserve with LEU (e.g., neutronic absorber mixed with dispersion fuel for ATR and HFIR or fins on MITR cladding for instance). Therefore, reactor-specific solutions have, therefore, to be developed to overcome various reactor-specific challenges.

3.5. Design Status

3.5.1. MITR LEU Design

For MITR, the current strategy is to mitigate the neutron flux losses by increasing the reactor power (from 6 MW to 7 MW). In addition, current fabrication capability is unlikely to allow for thin vertical grooves (fins) on the cladding currently used to improve heat transfer. The current LEU design is a 7 MW design with at least 305 µm-thick (12 mil) cladding with no fins. The number of plates has increased from 15 in the current HEU design to 19 plates to partially compensate for the eliminated fins. The meat thickness is graded to further reduce the power peaking: thin in the outer plates and gradually increased in the inner plates. Table 2 lists additional details, where the fabrication dimensions are expressed in mil (1 mil = 25.4 µm). Overall, the current LEU design preserves performance and present thermal-hydraulic margins. Currently, the MITR conversion team is drafting a preliminary safety analysis report that will be submitted to the regulator to evaluate methods and preliminary analysis. Details on this design, labelled 19B25, can be found in [17].

Table 2 – Characteristics of the MITR HEU and LEU fuel elements

Parameter	**HEU [17]**	**LEU [17]**
Reactor power (MW)	6	7
^{235}U / U mass per element (g)	508 / 545	968 / 4900
Number of plates / element	15	19
Plate thickness (mil)	60 (at the fin base)	49
Fuel / AA 6061+Zr thickness plates 1 and 19 (mil)	30 / 15	13 / 18
Fuel / AA 6061+Zr thickness plates 2-3 and 17-18 (mil)		17 / 16
Fuel / AA 6061+Zr thickness all other plates (mil)		25 / 12

Note 1: Plates 1 and 19 are the outermost plates

3.5.2. MURR LEU Design

The MURR strategy is to increase the reactor power (from 10 MW to 12 MW) to mitigate the neutron flux losses. The current LEU design is a 12 MW design, each element having 23 fuel plates (compared to 24 in the HEU element). Fuel thickness varies among the plates (as shown in Table 3) to flatten the power profile, reducing the magnitude of the heat flux and power peaking and, thus, allowing for acceptable thermal-hydraulic margins. More details on this design, labelled CD35, can be found in [11].

Overall, the current MURR LEU design preserves the reactor performance and present thermal-hydraulic margins. The MURR conversion team is currently drafting a preliminary safety analysis report that will be submitted to the regulator to validate models and tools used in the analysis.

Table 3 – Characteristics of the MURR HEU and LEU fuel elements.

Parameter	**HEU [11]**	**LEU [11]**
Reactor power (MW)	10	12
^{235}U / U mass per element (g)	775 / 833	1507 / 7630
Number of plates / element	24	23
Fuel / AA 6061+Zr / plate thickness plate 1 (mil)	20 / 15 / 50	9 / 17.5 / 44
Fuel / AA 6061+Zr / plate thickness plate 2 (mil)		12 / 16 / 44
Fuel / AA 6061+Zr / plate thickness plate 3 (mil)		16 / 14 / 44
Fuel / AA 6061+Zr / plate thickness plate 23 (mil)		17 / 16 / 49
Fuel / AA 6061+Zr / plate thickness all other plates (mil)		20 / 12 / 44

Note 1: Plate 1 is closest to the reactor core center.

3.5.3. NBSR LEU Design

The current NBSR strategy is to stay at its current power (20 MW) and compensate for the flux losses induced by conversion by upgrading a key component of their experimental devices, the cold source. In addition, no changes to the external dimensions of the fuel element are considered (17 50 mil-thick fuel plates). Only an adjustment of the meat thickness is proposed (from 20 mil to 8.5 mil). The NBSR has already submitted a preliminary safety analysis report to the regulator.

3.5.4. ATR LEU Design

The ATR fuel element currently contains boron in the outer plates to flatten the power profile. The boron is mixed in the uranium that forms the meat. It is currently considered too difficult – with respect to fabrication– to have a neutronic absorber such as boron within the plate along the fuel. The ATR design team has been able to identify a design that, like the MURR design, overcomes this difficulty by varying fuel meat thickness between plates as described in Table 4. To satisfy all performance requirements, it might be necessary to raise the ATR power when used in a specific high-power mode. The determination on the power uprate has not been made yet. Detailed safety analyses are underway to address the different aspects of the conversion of ATR and the critical mockup ATRC. More details on this design, labeled ELF Mk 1A, can be found in [18].

Table 4 – Characteristic of the ATR HEU and LEU fuel elements.

Parameter	**HEU [13]**	**LEU [18]**
^{235}U / U mass per element (g)	1075 / 1156	1648 / 8344
Number of plates / element	19	
Fuel / AA 6061+Zr / plate thickness plate 1 (mil)	20 / 30 /80	8 / 36 /80
Fuel / AA 6061+Zr / plate thickness plate 2, 3, 16 (mil)	20 / 15 / 50	13 / 18.5 / 50
Fuel / AA 6061+Zr / plate thickness plate 19 (mil)	20 / 40 / 100	8 / 46 / 100
Fuel / AA 6061+Zr / plate thickness plate 17, 18 (mil)	20 / 15 / 50	8 / 21 / 50
Fuel / AA 6061+Zr / plate thickness all other plates (mil)		16 / 17 / 50

Note 1: Plate 1 is closest to the reactor core center

3.5.5. HFIR LEU Design

In the HEU design, the fuel within the HFIR plates is contoured along the fuel width, being thinner on the edges than in the middle. In addition, the inner element plates contain a layer of aluminium-boron in the form of a mixed powder. These two features make the conversion problem particularly difficult to solve.

One of the most mature HFIR LEU designs assumes that the neutron absorber will be removed from the plate and relocated into the side-plate. This design still requires fuel contouring. The conversion program is currently assessing the best fabrication process to contour the fuel. The reactor power would have to increase from 85 MW to 100 MW to compensate all performance losses. Other design options are being considered and the program will assess soon the benefits of pursuing these.

3.6. Design Summary

To summarize, each USHPRR has established LEU fuel element designs using the LEU UMo monolithic fuel system. Under the RC pillar, reactors are working with the FD and FQ program pillars to mature and finalize the designs. Different LEU design strategies have been developed for all USHPRRs, considering their unique constraints and requirements. Figure 8 illustrates the variety of LEU plate designs based on the number of different fuel foil thicknesses required (A, B, C,…) for each USHPRR. It should be noted that these are extremely high aspect ratios since the scale of the fuel length (m) is on the order of 1000 times the thickness of the fuel (mm) in this figure. Other than HFIR's contoured fuel, the thinnest fuel required to reduce power peaking, typically on the outer plates of the fuel element, is planned for ATR at 8 mil (0.203 mm). Current LEU fuel designs have up to 25 mil (0.635 mm) thick fuel. This is for MITR which also has the thickest fuel among the HEU USHPRR designs. For LEU MITR cores this is loaded into the interior plates, and is needed for this reactor which ordinarily burns to 3 effective full power years. Length of the fuel also covers a significant range from 0.28 m for NBSR's high-performance split heavy water core to ATR's serpentine arrangement with fuel that is 1.2 m in length and has a similar radial core extent in excess of 1 m.

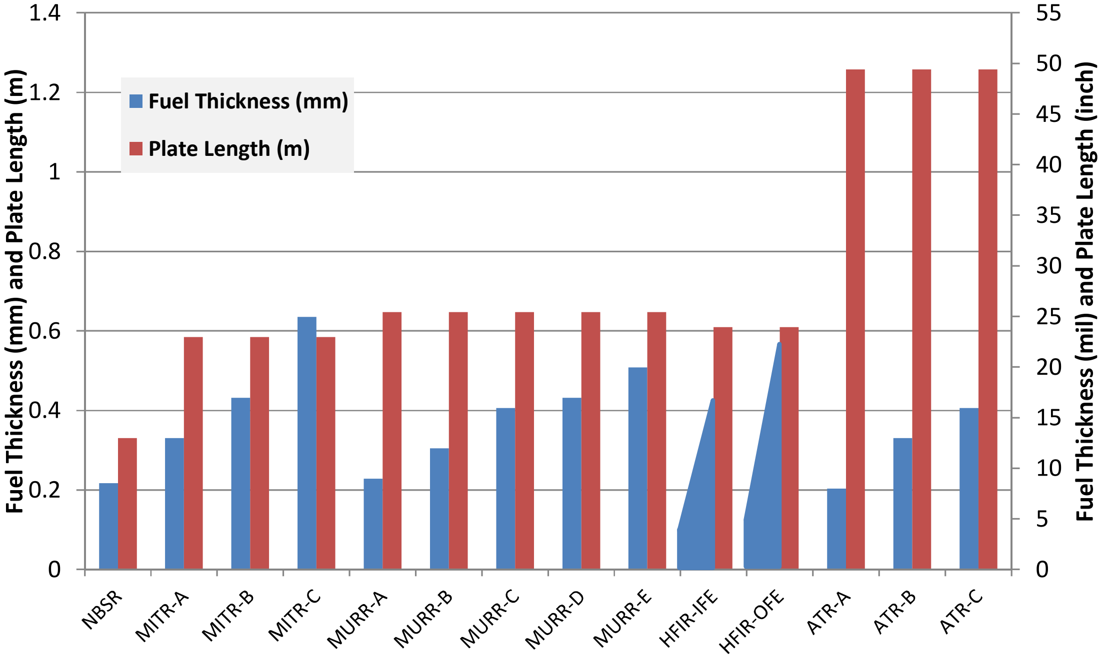

Figure 8 – Fuel thickness and plate length for LEU USHPRR designs (letters indicate different plate types for a given reactor design; HFIR inner and outer element fuel thickness is not constant along the fuel width and range of variation is illustrated with triangles).

The next irradiation campaigns, which are aimed at qualifying the plates in prototypic LEU conditions, are illustrated in Figure 9. Notice that in addition to the variety of fuel plate geometries shown in Figure 8, it is clear that no single reactor is limiting in design space. For example, whereas the thin NBSR fuel has the highest burnup, the thicker fuel in MIT plates is combined with a significant burnup. Irradiation testing the fuel behavior across the range of designs is therefore important to capture the effects that impact design. Among the more important considerations found to date for LEU U10Mo fuel are burnup effects on fuel swelling, fuel thermal conductivity, and establishing the fuel plate temperature safety limit based on sufficient number of fuel plates blister annealed at various burnups. Although these phenomena are present in currently deployed fuel systems, the impact of the very high fuel density requires that these are carefully considered in designing cores with LEU U10Mo monolithic alloy fuel.

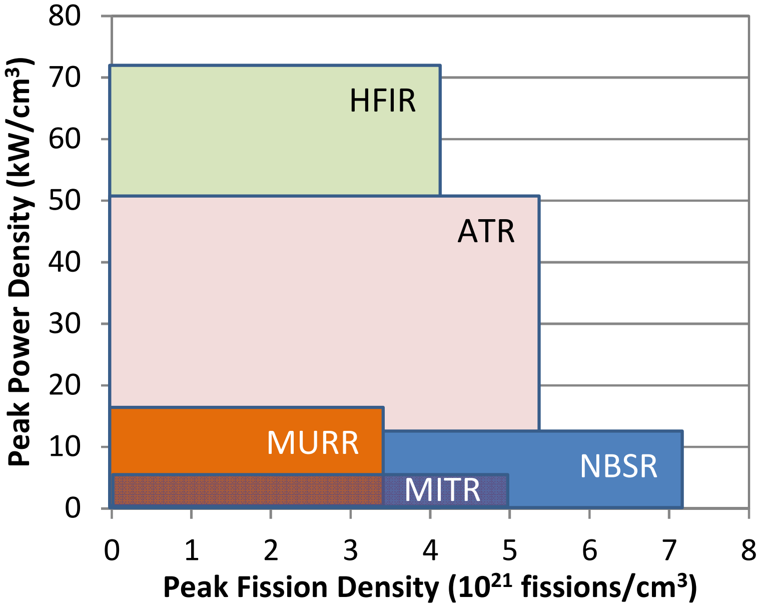

Figure 9 – Maximum local power density and fission density requirements for the USHPRR.

These peak values are the maximum local values found in any of the plates. Combined with the anticipated LEU prototypic operational conditions through the life of the fuel element, including within plate shape of power distributions, these form the USHPRR design parameters for a comprehensive plate-level irradiation campaign, and subsequent lead test assembly demonstrations. These are planned in ATR or BR2 experiment positions for each of these element designs. These fabrication and irradiation campaigns are required to confirm or update element design assumptions to allow for reactor conversion of the USHPRR to LEU fuel.

4. Conclusions

The NNSA Office of Material Management and Minimization (M^3) developed an integrated approach to address the persistent threat posed by the unintentional proliferation of nuclear materials. One of the three M^3 subprograms is reactor conversion. One of the goals of the M^3 reactor conversion program is to convert the six domestic high performance research reactors or USHPRRs (including one critical facility) that still use HEU fuel. Based on demonstrated favorable irradiation behaviour, the USHPRR program focuses on the development of the LEU UMo “monolithic” fuel.

The USHPRR conversion program is working to deploy fabrication capability, qualification of the fuel system, and design of LEU fuel elements that meet reactor missions, operational, and safety basis requirements. The FF, FQ and RC pillars of the USHPRR conversion program are working together to converge to solutions that meet all the conversion requirements.

The UMo “monolithic” fuel system selected for the conversion of the USHPRRs represents a significant technological departure from the dispersion fuel form used currently. While the adoption of a new technology presents challenges, it also considerably extends the design possibilities to make the conversion of these high-performance reactors feasible. All the reactor design teams have been able to find creative solutions to overcome challenges encountered and have been successful in proposing designs. Designs, which are currently preliminary, will be finalized once fuel qualification data is available to confirm and update design analysis.

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