CROSS SECTION GENERATION AND PHYSICS MODELING IN A FEASIBILITY STUDY OF THE CONVERSION OF THE HIGH FLUX ISOTOPE REACTOR CORE TO USE LOW-ENRICHED URANIUM FUEL

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Abstract

A computational study has been initiated at ORNL to examine the feasibility of converting the High Flux Isotope Reactor (HFIR) from highly enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. The current study is limited to steady-state, nominal operation and are focused on the determination of the fuel requirements, primarily density, that are required to maintain the performance of the reactor. Reactor physics analyses are reported for a uranium-molybdenum alloy that would be substituted for the current fuel—U_3O_8 mixed with aluminum. An LEU core design has been obtained and requires an increase in ^{235}U loading of a factor of 1.9 over the current HEU fuel. These initial results indicate that the conversion from HEU to LEU results in a reduction of the thermal fluxes in the central flux trap region of approximately 9 % and in the outer beryllium reflector region of approximately 15%. Ongoing work is being performed to improve upon this initial design to further minimize the impact of conversion to LEU fuel.

KEYWORDS: HFIR, Research Reactor, LEU fuel feasibility, cross sections, reactor physics calculations.

1. Introduction

As discussed in a companion paper, Ref. [1], the U.S. nonproliferation policy "to minimize, and to the extent possible, eliminate the use of highly enriched uranium (HEU) in civil nuclear programs throughout the world" has resulted in the conversion (or scheduled conversion) of many of the U.S. research reactors from HEU to low-enriched uranium (LEU)—i.e., having a ^{235}U enrichment less than or equal to 20 wt%. In support of this activity, a study has been initiated in 2005 to study the feasibility of converting the High Flux Isotope Reactor to low-enriched uranium fuel [2].

Five high-performance reactors are operating in the United States with HEU that have not converted to LEU because there is currently no available suitable LEU fuel that will allow these reactors to meet their mission requirements. These reactors include the High Flux Isotope Reactor (HFIR), the Advanced Test Reactor (ATR), the National Institute of Standards and Technology (NIST) research reactor, Missouri University Research Reactor (MURR), and Massachusetts Institute of Technology Reactor (MITR-II). Of these, the highest power density core and the most challenging to convert to LEU, with its involute-shaped fuel plates, is HFIR.

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In order to perform this conversion, fuels with uranium densities will be required that are not currently available in aluminum-dispersed oxide and silicide fuels.

One of the most important activities under the Reduced Enrichment for Research and Test Reactors (RERTR) Program [3] has been the development of high-density LEU fuels. Recent efforts have focused on the development dispersed (in aluminum) and monolithic uranium-molybdenum (U-Mo) alloy fuels. The monolithic alloy fuel has a density of approximately 15–17 gU/cm^3. As discussed in Ref. [1], the requirements of LEU fuels in HFIR include more than just obtaining high fuel densities. In 1997 the RERTR Program performed a neutronics feasibility study of an LEU conversion of HFIR [4]. The study concluded that fuels with densities of up to 9 gU/cm^3 would be required for the conversion; however, the core power peaking was significantly higher than for the HEU core, and no thermal analysis was performed to determine if the core met the required thermal margins. A more complete study is required to determine the feasibility of converting HFIR to LEU fuels.

The purpose of the current study is to perform the analysis and assessment of the feasibility of LEU conversion for HFIR using the analytical tools qualified for HFIR and the ORNL expertise most knowledgeable of HFIR operations and fuel supply. Key top-level assumptions and constraints guide the current study [1,2].

2. Description of HFIR

The HFIR (Fig.1) is a pressurized light-water-cooled and -moderated, flux-trap type reactor that uses fuel highly-enriched in ^{235}U (93 wt. %) and is currently operateing at 85 MW. The reactor core (Fig. 2) consists of two annular fuel elements, each approximately 61 cm high (fueled height is 51 cm). At the center of the core is a 12.70-cm-diam cylindrical hole, referred to as the “flux trap target” region, which contains 37 vertical experimental target sites.

Figure 1: The basic layout schematic of HFIR.

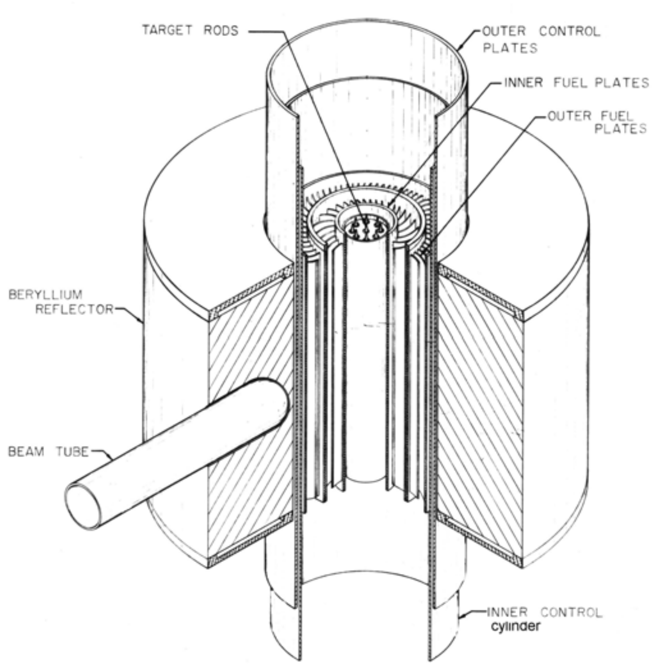


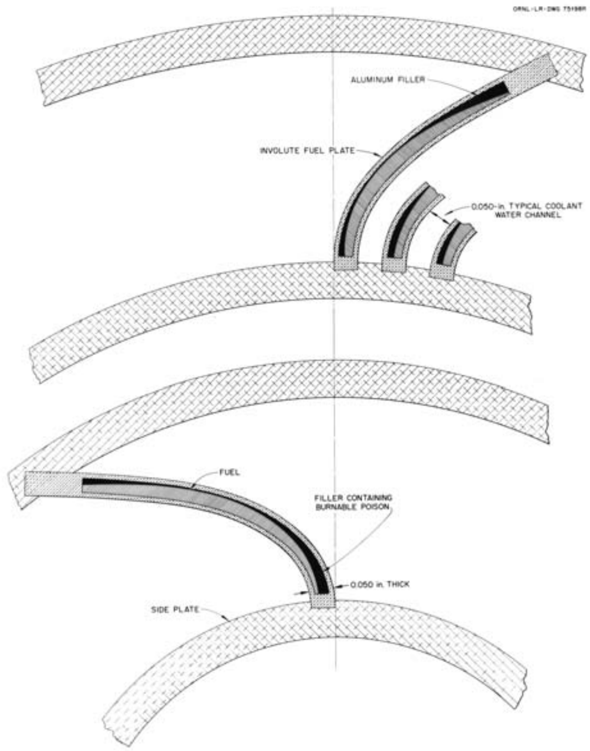
Figure 2: The core of HFIR, showing the inner (IFE) and outer (OFE) fuel elements.

This HFIR fuel elements, surrounding the flux trap, contain vertical, curved plates extending in the radial direction. The fuel elements are separated by a narrow water gap. The inner element contains 171 involute-shape fuel plates, and the outer element contains 369 involute-shape fuel plates, as detailed in Fig. 3. The fuel plates are a sandwich-type construction with a fuel-bearing cermet bonded to a cladding of type-6061 aluminum. The highly enriched uranium oxide is distributed (graded) along the arc of the involute aluminum plate, as seen schematically in Fig. 3.

Control plates, in the form of two thin, europium/tantalum-bearing concentric cylinders, are located in an annular region between the outer fuel element and the beryllium reflector (see Fig. 1). These plates are driven in opposite directions. Reactivity is increased by downward motion of the inner cylinder, which is used only for shimming and regulation. The outer control cylinder consists of four separate quadrants, each having an independent drive and safety release mechanism. Reactivity is increased as the outer plates are raised.

The control plates and fuel elements are surrounded by a concentric ring of beryllium that serves as a reflector and is approximately 30-cm thick. This Be reflector is subdivided into three regions: the inner removable reflector, the middle semipermanent reflector, and then the outer permanent reflector. The beryllium is surrounded by a light water reflector of effectively infinite radial thickness. In the axial direction, the reactor is reflected by light water.

Figure 3: Schematic of the current HEU fuel plates in the HFIR fuel elements.



3. Details of the LEU Feasibility Study

As mentioned, for HFIR LEU feasibility studies the current uranium form, U_3O_8, will be replaced by a uranium molybdenum alloy with uranium at 19.75 wt. % ^{235}U enrichment. Per direction from the RERTR Program Office, the neutronics/thermal-hydraulics studies are based on the assumption that uranium/molybdenum metal alloy with 10 wt% Mo (U-10Mo) is the fuel (density of 17.02 g/cm^3). Neutronics studies will be conducted for two variants of LEU fuel – mixtures of U-10Mo and Al powders and discrete, U-10Mo and Al layers (termed monolithic fuel). For this paper, the results will be shown for monolithic U-10Mo fuel in the plates.

The performance of HFIR with LEU fuel was analyzed using the standard set of computational tools that are currently used at ORNL to support the operation of the reactor. These tools include those for neutronics, thermal-hydraulics, and dose assessments. The methods and computer codes are an extension of the experience base at ORNL developed for the conceptual core design for the Advanced Neutron Source (ANS) project. The computer codes used for these analyses included MCNP5 [5], SCALE 5 [6], AMPX [7], and BOLD VENTURE [8] for reactor physics calculations and assessments. The nuclear data libraries used with the neutronics codes were based on ENDF/B-V and VI nuclear data. The libraries used with BOLD VENTURE are ISOTXS libraries prepared using SCALE cross-section generation and the SCALE 238-group ENDF/B-V master libraries. The few-group cross section library for the BOLD VENTURE analysis is created using modules for resonance processing followed by a one-dimensional radial calculation using SCALE sequences to obtain the appropriate neutron flux spectrum for collapsing the cross sections to 20 neutron energy groups (sequence includes (BONAMI, NITAWL, and XSDRNPM modules). Table 1 shows the neutron energy structure of the 20-group set of collapsed energy groups with a comparison to the groups of the 238-group nuclear data master library.

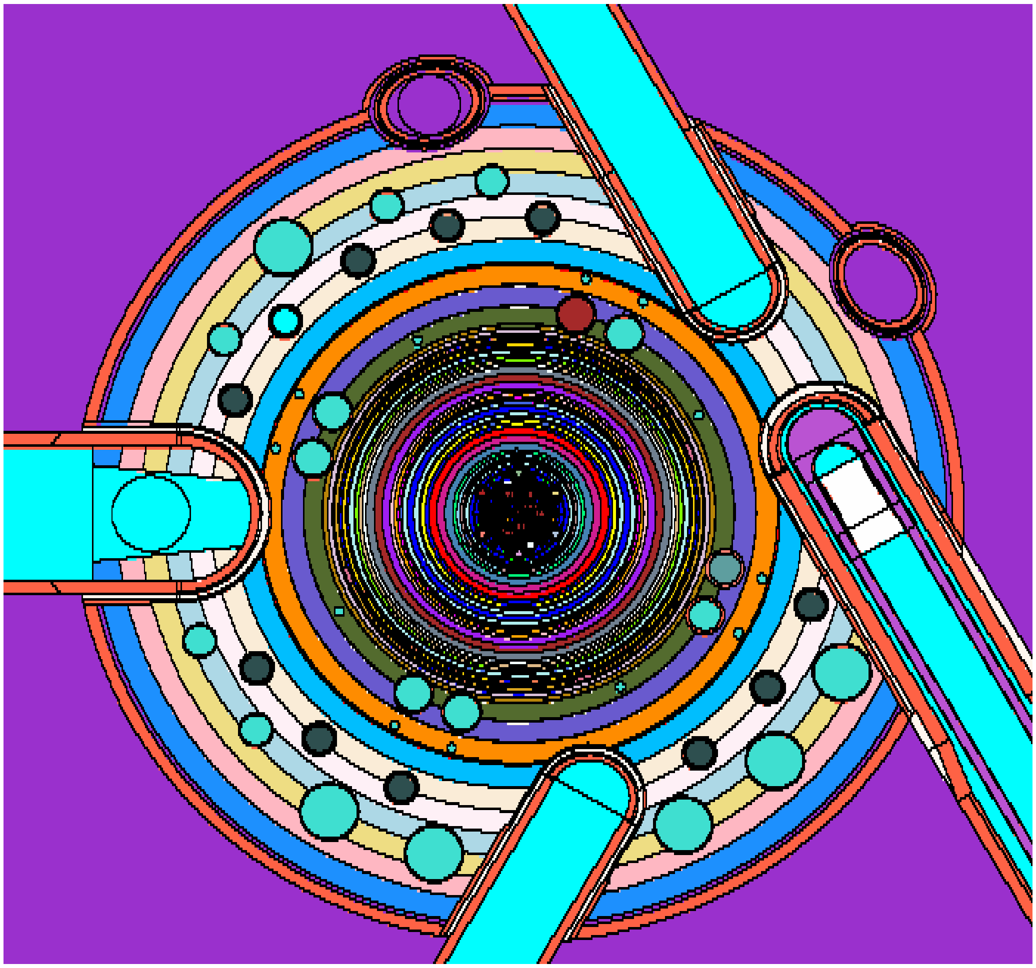
Table 1: Structure of the collapsed neutron energy group

20-Group	238-Group	Lower Energy Limit (eV)
1	12	2.479E+06
2	15	1.50E+06
3	25	8.75E+05
4	45	8.50E+04
5	63	2.58E+03
6	86	9.00E+01
7	116	2.75E+01
8	132	9.10E+00
9	149	2.97E+00
10	163	1.68E+00
11	190	9.75E-01
12	199	6.25E-01
13	205	3.75E-01
14	210	2.50E-01
15	215	1.25E-01
16	222	4.00E-02
17	226	7.50E-03
18	230	2.50E-03
19	232	1.50E-03
20	238	1.00E-05

In addition to models of HFIR for BOLD VENTURE to represent the operational behavior of the reactor, MCNP models have been prepared to serve as detailed references and models for predicting neutron flux distributions and HFIR core reactivity [9,10]. Fig. 4 presents a cross-sectional view of the HFIR at midplane showing the detailed spatial representation of HFIR in the MCNP5 model.

The BOLD VENTURE model of HFIR provides the ability to perform the depletion calculations of the HFIR core, using the BURNER module, and to provide detailed power distributions based on the input fuel distribution. As described above, the multigroup neutron cross-section libraries are obtained using SCALE. The fuel depletion calculations are performed; and the control plate positions are adjusted for each step to provide an approximate critical configuration. The output of the calculation is the detailed power distribution in the fuel region, the isotopic composition of the fuel, neutron flux distribution, and the effective multiplication factor. These calculations are fast, so they can be used to perform the numerous fuel grading calculations needed to provide a flat power profile, as well as indicate impacts on the peak fluxes in the target and reflector regions.

Figure 4: MCNP model of HFIR: cross section of reactor core at horizontal midplane.



The determination of the distribution of the fuel in the fuel plate (radial grading profile) requires an iterative calculation process that is similar to fuel management calculations performed for commercial power reactors. Upon obtaining a reasonable fuel grading profile, the calculated power distribution results from BOLD VENTURE will be used as input in the steady-state thermal-hydraulics code (SSTHC) to obtain thermal margins. The goal in obtaining the power distribution is to maintain the current thermal margins. In a feasible HEU-to-LEU fuel conversion, the margins of safety in the bases of the currently approved Technical Safety Requirements [11] shall be maintained.

The approach in the current work to determine the appropriate fuel loading and fuel grading profile is as follows:

1. An initial fuel loading profile and uranium mass is assumed (starting with a uniform profile, for example).
2. A fuel cycle calculation is performed with BOLD VENTURE to obtain the cycle length and power distribution.

3. If the fuel cycle length criterion is not met, the overall fuel loading will be increased (but will not exceed the maximum local loading determined by the existing plate geometric design).
4. The relative loading of fuel in each local region is adjusted to minimize variation among local power densities.
5. Steps 2 through 4 are repeated as necessary for a number of iterations resulting in the best grading profile for the fuel being considered.
6. Perform the thermal-hydraulic analysis to determine the maximum reactor operating power to stay within the required thermal and safety margins.

4. Results and Discussion

The initial constraint on the thickness (0.0127 cm) of the fuel material (monolithic U-10Mo) made it challenging to obtain acceptable power densities at the inner and bottom edge of the IFE fuel plates. As discussed in Ref. [2], an iterative process was performed between the neutronics results from VENTURE cases and the HFIR heat transfer analysis software as the LEU fuel grading was performed. Table 2 compares the fuel meat thickness for the standard HEU core and the LEU core developed during this study.

Table 2: Comparison of fuel meat thicknesses for LEU and HEU fuel plates.

Distance along inner element plate (cm)	Thickness of fuel meat (mils)		Distance along outer element plate (cm)	Thickness of fuel meat (mils)	
	LEU	HEU		LEU	HEU
0.252	5.0	10.2	0.191	8.5	15.3
0.448	5.0	11.6	0.216	9.4	15.6
1.203	6.0	15.5	0.395	15.3	16.9
2.439	8.6	20.5	1.134	16.1	23.0
3.811	10.0	24.4	2.256	18.0	27.1
5.314	10.5	24.6	3.449	15.9	25.5
6.969	9.6	21.5	4.655	10.2	20.7
7.985	7.9	18.6	5.908	6.4	14.7
8.091	7.6	18.3	6.731	5.1	11.5

In Fig. 5, plots of k_{eff} as a function of the days of full-power operation are shown for the reference HEU core and the proposed LEU core. For this plot, the HFIR control absorbers are fully-withdrawn so that the comparisons only include the neutron physics associated with fuel depletion. As can be seen in Fig. 5, both curves exhibit the initial drop in k_{eff} because of the buildup of equilibrium levels of ^{135}Xe. The k_{eff} curve for HEU eventually drops off at a faster rate than the LEU curve. The curves have about the same excess reactivity at 26 days, the target core lifetime for this LEU feasibility study.

In Fig. 6, the simulated k_{eff} curves are shown for the HEU and LEU cores with the control absorber insertion modeled in the two BOLD VENTURE cases. In these calculations, the control rods are adjusted to obtain a near-unity multiplication factor during the reactor operation. These results show that LEU and HEU cores can be operated for the 26-day irradiation with similar control rod movement requirements.

Figure 5: Comparison of k_{eff} for the LEU and HEU HFIR cores (without control absorber insertion).

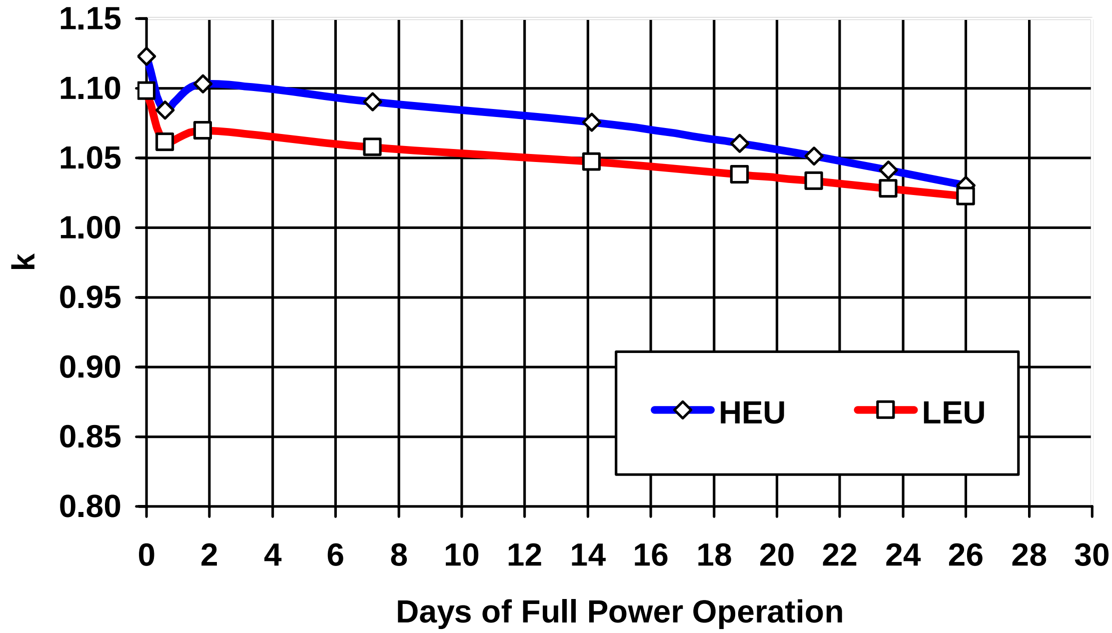
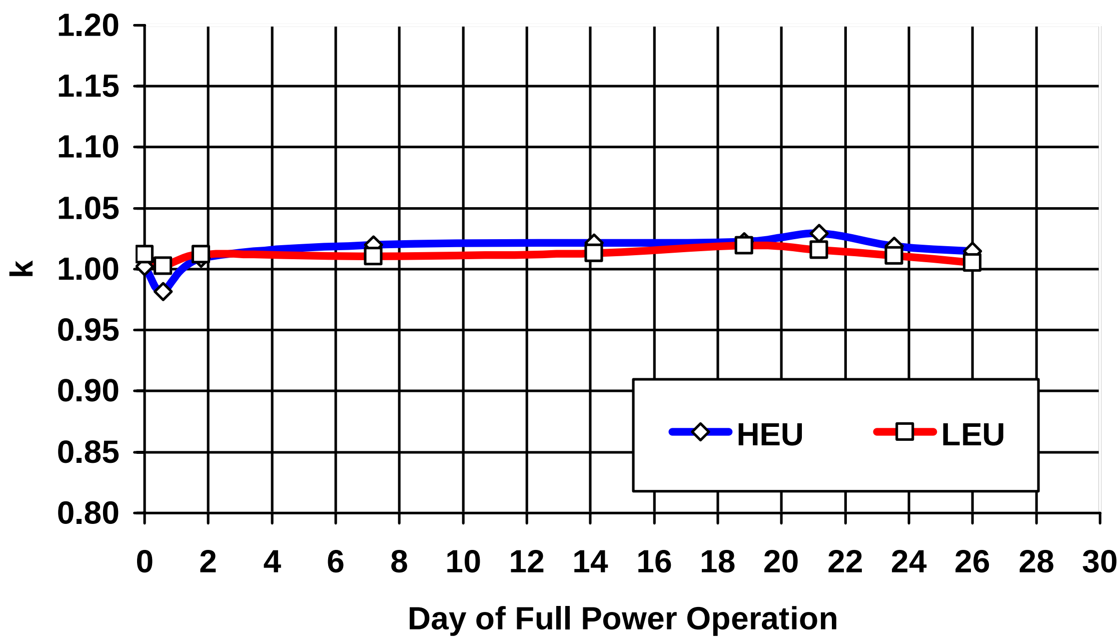


Figure 6: Comparison of k_{eff} as a function of days of operation in the simulation of the LEU and HEU HFIR operational history (with control absorber insertion).



MCNP5 models of HFIR were used to analyze the HEU and LEU fuel cores, and the performance of the reactor in terms of power densities and neutron flux distributions. F4 flux tallies and F7 fission energy tallies were employed in the MCNP cases to obtain information on the power distribution in the core, and the flux levels in the flux trap target (FTT) and Be reflector regions (see Figs. 1 and 4). Table 3 shows good agreement between the MCNP tally results and BOLD VENTURE calculations for a representative LEU HFIR model simulation. There is very good agreement in k_{eff} and the overall power produced in the IFE and OFE.

Table 3: Comparison of VENTURE and MCNP5 results for an LEU core with no control absorbers inserted.

Code	Initial k_{eff}	IFE Power	OFE Power
VENTURE	1.0888	28.86 MW	56.14 MW
MCNP5	1.0868±0.0001	28.80 MW	56.20 MW

To assess the performance of HFIR with the LEU core, comparisons are made to the peak thermal flux levels in the standard HEU core in the important FTT region and the peak in the Be reflector. Table 4 presents the LEU and HEU peak thermal flux level in these locations, for the cases with and without control absorber insertion.

Table 4: Peak thermal neutron flux densities (< 0.625 eV) as predicted by VENTURE for HEU and LEU fuels.

HFIR Core	Peak Thermal Neutron Flux in FTT Region ($n/cm^2s/W$)	Peak Reflector Thermal Neutron Flux ($n/cm^2s/W$)
HEU ref (no control absorbers)	3.08×10^7	2.00×10^7
LEU (no control absorbers)	2.67×10^7	1.70×10^7
HEU ref (w/control absorbers)	3.18×10^7	2.03×10^7
LEU (w/control absorbers)	2.89×10^7	1.72×10^7

For the cases with control absorber insertion, the FTT thermal flux level is reduced by approximately 9% in the LEU case compared to the HEU reference case. The peak thermal neutron flux level in the Be reflector is seen to be reduced by approximately 15% in the LEU core case compared to the HEU core case. These VENTURE results are consistent with MCNP5 case studies.

Table 5 shows the consistency of the neutron flux spectral shape in comparisons of fast-to-thermal neutron flux levels at BOL and EOL from the LEU and HEU HFIR core simulations in VENTURE with the control elements at their critical positions and with them fully withdrawn for comparison. These results indicate a minimal impact on the overall neutron spectrum in the both flux trap and the Be reflector regions.

To assess the behavior and performance of the LEU and HEU cores, the concentrations of several important nuclides are tabulated in Tab. 6 at BOL and EOL. During the simulated 26-d full-power cycles, the HEU case indicates that 2.8 kg of ^{235}U is used during the reactor operation, while 2.7 kg of ^{235}U is used up in the LEU core. The LEU core consumes 80 g less of ^{235}U than the HEU case because of the additional production of ^{239}Pu in the fuel plates. The net production of ^{239}Pu in the HEU case is only 12.6 g compared to a net production of ^{239}Pu in the LEU case of 269 g. Additionally, 18.4 g of ^{240}Pu is produced in the LEU core fuel plates. In the HEU core, about 21 g of ^{238}U is consumed, while about 400 g of ^{238}U is used up in the LEU core; most of it was converted to Pu.

Table 5: Ratios of fast/thermal neutron flux levels at selected locations in LEU and HEU, at BOL and EOL. (fast > 0.875 MeV; thermal < 0.625 eV)

	HEU		LEU	
	BOL	**EOL**	**BOL**	**EOL**
W/Control Absorbers				
FTT	0.248	0.217	0.257	0.244
Be Reflector	0.102	0.110	0.099	0.096
No Control Absorbers				
FTT	0.241	0.222	0.251	0.246
Be Reflector	0.103	0.109	0.097	0.094

Table 6: Comparison of selected HFIR nuclide content (g) in LEU and HEU cores, at BOL and EOL.

Nuclide	HEU		LEU	
	BOL	**EOL**	**BOL**	**EOL**
^{235}**U**	9,470	6,650	17,900	15,200
^{238}**U**	555	534	72,800	72,400
^{239}**Pu**	-	12.6	-	269
^{240}**Pu**	-	2.06	-	18.4
^{10}**B**	2.78	0.19	2.78	0.74

In the HEU core, about 93% of the ^{10}B is burned out of the IFE plates compared to about 73% in the LEU core IFE plates. As discussed in Ref. 13, the thinness of the U-10Mo material in the fuel plate necessitates a relatively thick and fairly uniform "filler" region between the fuel plate cladding and the fuel meat surface. The filler contains a uniform mixture of Al and B_4C; in the HEU design, the amount of ^{10}B at the inner and outer edge of the IFE fuel plates served to reduce the power at the radial edges of the fuel plates. This desired effect is not present in the LEU core because there is no preferential increase in ^{10}B at the edges.

It should be noted that the work presented here updates the previous results presented in Ref. 13. Results presented in that reference were based on VENTURE calculations performed with cross section libraries in 20 groups for the LEU models and 7 groups for the HEU model generated from 99 group, ENDF/B-V based master libraries. In this work, the HEU and LEU VENTURE calculations used cross section libraries, in 20 groups, generated from SCALE 238-group ENDF/B-V and –VI master library data. Corrections were made to some old errors in HFIR zone materials in the HEU and LEU models. Five additional finer spatial meshes were added at the top and bottom of the fuel region to try to improve the modeling of the power production at the edges of the core.

MCNP tallies were also used to compare the power densities at the edges of the fuel plates. The results showed that VENTURE calculated edge power densities that were between 5 and 20% larger than the corresponding MCNP energy tally predictions for power density. When the edge power densities are properly accounted for, the operational power level of the HFIR with an

LEU core can be in the 80- to-90-MW range, which meets the current full power operation conditions at HFIR of 85 MW.

5. Future Work and Plans

The results presented above are still preliminary in nature and a full assessment and comparison of fuel plate power density distributions will be made between MCNP5 results and BOLD VENTURE calculations. Further improvements and developments will be made to the generation of cross section libraries (ISOTXS) for use in BOLD VENTURE simulations using SCALE methods and modules. In particular, the issue of modeling the power generation at the edges of the fuel plates will be resolved by explicitly modeling the spectral zones at the top and bottom of the core-regions in the cross section processing methodology. Additionally, improvements to the overall models and other techniques will be employed to lessen the reductions in performance factors (peak neutron flux, isotope production, beam tube performance, and core lifetime). Furthermore, studies will be conducted regarding the core lifetime to explore the maximization of the life cycle of an LEU core. It may be advantageous to optimize the reactor operational procedures such that the reactor could extend its life cycle by small reductions in power level, to increase the availability of HFIR for scattering and neutron science experiments.

6. Conclusions

A new core design has been obtained for HFIR utilizing LEU U-10Mo fuel. The limiting minimum concentration of ^{235}U at the inner and bottom edges of the IFE fuel plates makes it challenging to obtain acceptable power densities and peak temperatures [1] at these locations. From a neutronics viewpoint, the use of a LEU core in HFIR can be expected to result in a 9% reduction in the peak thermal neutron flux level in the flux trap target region, and about a 15% reduction in the peak thermal neutron flux level in the Be reflection region. The lifetime of the LEU core attained the 26 day target with approximately 17.9 kg of ^{235}U in the core, about a factor of 1.9 more than the current ^{235}U loading of HFIR.

Acknowledgements

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