NEUTRONIC AND THERMAL-HYDRAULIC FEASIBILITY STUDIES FOR HIGH FLUX ISOTOPE REACTOR CONVERSION TO LOW-ENRICHED URANIUM U_3SI_2-AL FUEL

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ABSTRACT

Oak Ridge National Laboratory (ORNL) operates the High Flux Isotope Reactor (HFIR) for the US Department of Energy Office of Science to perform neutron scattering experiments, produce isotopes, and conduct materials irradiation research. ORNL is funded by the National Nuclear Security Administration (NNSA) Office of Materials Management and Minimization to evaluate the conversion of HFIR's fuel from high-enriched uranium (HEU) to low-enriched uranium (LEU). Due to potential fabrication issues with the complex UMo fuel design required to convert HFIR, NNSA requested ORNL to evaluate an alternate LEU fuel form – U_3Si_2-Al dispersion fuel. Neutronic and thermal-hydraulic feasibility analyses were performed with Shift and HSSHTC, respectively, to predict reactor performance and thermal safety margins. A number of designs were proposed and evaluated using an iterative approach in an effort to show that reactor performance could match that obtained using HEU fuel and that thermal safety margins were adequate. This study concludes that conversion of HFIR with U_3Si_2-Al LEU (19.75 wt.%) fuel with 4.8 gU/cm^3 is feasible if, among other requirements, the fuel meat region is centered and symmetric about the fuel plate thickness centerline, the active fuel zone length is increased from 50.80 cm to 55.88 cm, the proposed fabrication tolerances can be met, and the fuel can be qualified for HFIR conditions.

KEYWORDS: HFIR, LEU, U_3Si_2-Al, Shift, HSSHTC

1. INTRODUCTION

The High Flux Isotope Reactor (HFIR) is a versatile research reactor that is operated at the Oak Ridge National Laboratory (ORNL) and serves a broad range of science and technology communities. Neutrons produced in HFIR are used to support cold and thermal neutron scattering experiments, isotope production, and materials irradiation research. ORNL is funded by the National Nuclear Security Administration (NNSA) Office of Materials Management and Minimization to evaluate the conversion of HFIR's fuel from high-enriched uranium (HEU) to low-enriched uranium (LEU) in support of international nuclear nonproliferation objectives. For over a decade, the US LEU conversion program has pursued conversion of the remaining domestic high-performance research reactors with a UMo monolithic fuel. However, due to potential fabrication issues with the complex fuel design needed to convert HFIR while maintaining its performance, NNSA requested ORNL to evaluate an alternate LEU fuel form – U_3Si_2-Al dispersion fuel. Advantages of this silicide dispersion fuel over monolithic are that it (1) has been fabricated for several years by several vendors using a process that is similar to the one currently used to fabricate the complex fuel design of the HFIR HEU U_3O_8-Al dispersion fuel and (2) was qualified and approved by the US Nuclear Regulatory Commission for use in medium-power research reactors. Following the conversion of HFIR from HEU to LEU fuel, (1) the ability of HFIR to perform its scientific missions must be maintained, (2) all safety requirements must be met, and (3) annual operating costs must not increase. Design studies are ongoing to support the conversion of HFIR [1-10].

Research previously performed on the feasibility of converting HFIR with U_3Si_2-Al LEU fuel [10] concluded that it could not be converted with the highest density U_3Si_2-Al fuel qualified for research reactor use, which is 4.8 gU/cm^3. However, significant advances have been made over the past few years in the neutronics modeling and simulation of the HFIR core [6,8,11]. The advanced fuel plate modeling technique, which allows for the explicit modeling of the fuel plate internals and coolant channel, currently applied in the neutronics tools, has provided enhanced fidelity relative to the previous fuel modeling approach that homogenized the fuel plates and coolant channels into concentric rings. Explicitly modeling the fuel meat region reduces the overall core self-shielding effect introduced with the increased ^{238}U loading relative to the homogenized approach. More realistic predictions of reactor performance and cycle length should be achieved with this higher fidelity modeling approach. For this reason, the conversion of HFIR with U_3Si_2-Al LEU fuel is again being considered. The purpose of the work described in this paper is to study the feasibility of converting HFIR to U_3Si_2-Al LEU fuel. The conclusion of this work is not to present an optimal design, but to gain a better understanding of the reactor performance and fuel requirements if HFIR were to convert to U_3Si_2-Al fuel.

2. BRIEF OVERVIEW OF THE HIGH FLUX ISOTOPE REACTOR

HFIR is a pressurized, light-water cooled, light-water moderated, beryllium reflected, flux-trap-type research reactor. HFIR is fueled by two concentric fuel annuli referred to as the inner fuel element (IFE) and the outer fuel element (OFE). The IFE and OFE contain 171 and 369 fuel plates, respectively, and each of the 540 involute-shaped plates contain fuel meat in the form of U_3O_8-Al, encapsulated in Al-6061 clad. The fuel is HEU enriched to ~93 wt.% ^{235}U and approximately 2.6 and 6.8 kg of ^{235}U are respectively loaded into the IFE and OFE (9.4 kg total). The fuel meat is radially contoured to reduce edge power peaking and the IFE filler region contains B_4C poison for reactivity hold-down and edge power suppression. Each fuel plate is 60.96 cm in length and the active fuel zone length is 50.80 cm.

The inlet coolant temperature and pressure are approximately 48.89° C and 3.33 MPa, respectively. Down-flowing water coolant enters the pressure vessel through two inlets above the reactor, passes through the core, and then exits the vessel through the outlet below the core. Approximately 0.820 m^3/s of the total 1.009 m^3/s coolant flow passes through the core where a ΔT and ΔP of ~20° C and 0.69 MPa are nominally maintained. Because HFIR experiences core exit power peaking where the greatest bulk water temperatures and lowest pressures exist, thermal safety margin at the exit is typically limiting.

The unique geometry and high power density core (1.7 MW/l average) result in one of the world's greatest neutron fluxes and one of the world's brightest cold neutron sources. A central flux trap target provides 37 target positions that are utilized for isotope production and materials irradiation. A large concentric ring of beryllium reflector surrounding the core provides numerous experiment facilities including two pneumatic tubes that are used for neutron activation analysis. Four horizontal beam tubes penetrate the reflector and terminate at instruments in the beam room and guide hall where thermal and cold neutron scattering is performed, respectively.

3. MODELS, METHODS, AND CONSIDERATIONS

3.1. Neutronic Toolset and Performance Metrics

Previous HFIR analyses [2,6-9,11,12] used the VESTA [13] computational tool to perform Monte Carlo-based depletion modeling and simulation. VESTA couples the 3-D, continuous-energy Monte Carlo neutron transport code MCNP5 [14] and the point depletion code ORIGEN 2.2 [15]. Shift [16-17] has recently been adapted for HFIR LEU design studies because of its state-of-the-art features, which enable coupled high-fidelity and time-efficient neutronics calculations. The continuous-energy massively parallel Monte Carlo code is able to perform neutron transport, simulate isotopic depletion, and calculate important key metrics and fission densities for follow-on thermal-hydraulic calculations.

Shift uses advanced domain decomposition and rebalancing algorithms to scale efficiently on high-performance machines as well as on small clusters. By using Shift, the effective calculation time for a given design is significantly reduced and metrics can be calculated on the fly. A major advantage of using Shift is that the code is able to read in the existing HFIR MCNP5 geometry model (Fig. 1), with only a few minor changes and additions to the MCNP5 input. Comparisons between Shift and VESTA for HFIR Cycle 400 showed close agreement for *k-eff* (within 100 pcm), cycle length (within 0.5 days), and major actinides (e.g., uranium and plutonium within 1%), providing confidence in Shift results [16].

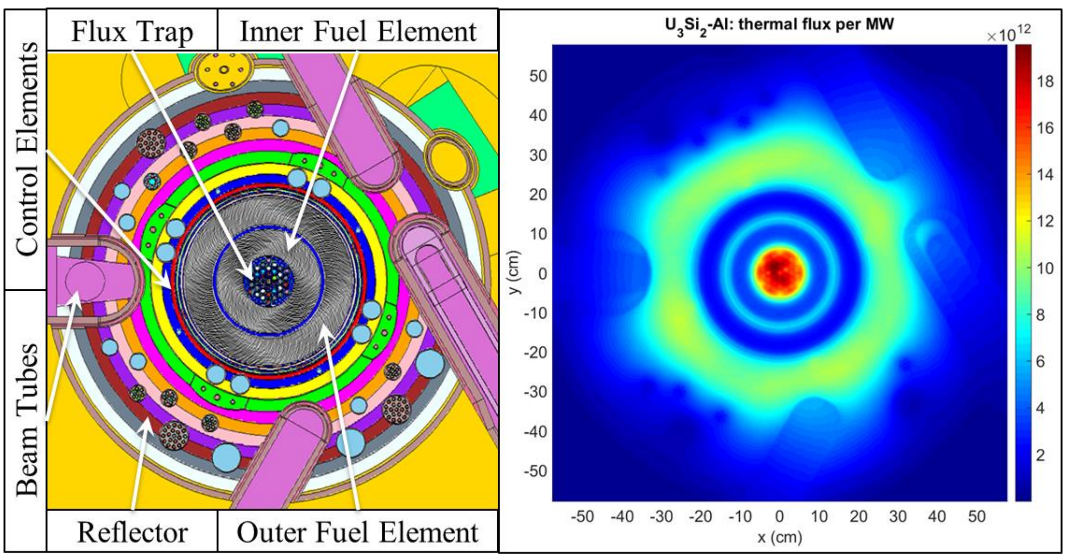

Figure 1. HFIR MCNP model (left) and U_3Si_2-Al core thermal flux per unit power (right).

For a given design, the time-dependent critical control element positions are determined via an external critical search script that iteratively runs Shift. This script is similar in functionality to the one used for VESTA simulations [6]. Once the script determines that the configuration is sufficiently critical (e.g., *k-eff* is within 50 pcm of 1.0 at each day), a final Shift depletion simulation is then performed. This final simulation generates data necessary to calculate key performance metrics to assess the capabilities of the design and data needed for follow-on thermal-hydraulic analyses. Due to some limitations in Shift, the methodology implemented in VESTA, via the external script, to account for fuel swelling due to burnup [2] is not transferrable. However, U_3Si_2-Al fuel swelling is small and has minimal reactivity effects relative to the UMo fuel. The cross sections used with Shift are based on ENDF/B-VII.0 data [18]. All cross sections are considered at 300 K temperature, with the water thermal scattering data at 293.6 K. The fission yield and nuclear decay data used for depletion are based on ENDF/B-VII.1 evaluations [19].

Conversion to LEU fuel is required to have a minimal impact on reactor performance. Thus, the ability of the reactor to perform its primary missions should not be compromised due to conversion. Key performance metrics are defined as a means of capturing data essential for HFIR's primary missions and are being used in studies to assess the impact of HFIR conversion from HEU to LEU fuel. Details regarding these key metrics and the results for the HEU and LEU UMo Interim designs are provided in [5]. The key performance metrics considered include reactor cycle length, cold neutron flux at the cold source moderator vessel, ^{252}Cf production in the flux trap target region, fast neutron flux in the flux trap target region, and the fast neutron flux in the removable beryllium reflector. Results for the HEU and LEU UMo Interim designs, as calculated with the MCNP and VESTA tools, are summarized in Table I. A successful U_3Si_2-Al design must meet or exceed these HEU performance metrics.

Table I. HEU and LEU UMo Interim design key performance metrics.

Parameter	HEU at 85 MW	LEU UMo Interim design at 100 MW
Cycle length [d]	26.2	34.0
Cold source moderator vessel cold flux [n/cm^2-s]	4.48E14	4.70E14
Cold source cold-to-total flux ratio	0.736	0.747
^{252}Cf Production [mg/day]	1.388	1.691
Flux trap fast flux in material irradiation locations [n/cm^2-s]	1.07E15	1.25E15
Flux trap fast-to-total flux ratio	0.290	0.318
Reflector fast flux in material irradiation locations [n/cm^2-s]	2.89E14	3.18E14
Reflector fast-to-total flux ratio	0.192	0.204

3.2. Steady State Heat Transfer Code and Safety Margin Metrics

The HSSHTC [20,21] is a steady-state channel code with 2-D R-Z geometry nodes that was specifically developed for and tailored to analyze HFIR with focus on thermal analysis necessary to produce reactor limiting control settings (LCSs) and safety limits (SLs), which are equivalent to limiting safety system settings and SLs, respectively, for NRC-regulated reactors. The analysis solves the integral mass, momentum, and energy equations on each axial coolant channel. The code searches the entire core for the worst hot streak and worst hot spot in that hot streak over the course of a fuel cycle. At the last time step, the power is increased to cause hot spot incipient boiling or burnout (depending on the input choice).

The core thermal-hydraulics model includes numerous physical models that are connected and iteratively solved together with the mass, momentum, and energy balance to consider the effects on local coolant channel width caused by oxide layer growth, plate thermal expansion, plate radiation swelling, fuel meat fission-induced swelling, plate deflections due to axial thermal expansion differences between the hot plate center and cool plate edges, plate deflection due to axial temperature gradients between an adjacent hot plate (overloaded with fuel) and cold plate (underloaded with fuel), and azimuthal pressure gradients caused by velocity difference between adjacent wide and narrow coolant channels. With the exception of the radiation-induced swelling correlation, all of the aforementioned models were unchanged for applicability to the U_3Si_2-Al designs. The fraction of total power deposited in the U_3Si_2-Al fuel elements is assumed to be 0.965, which is slightly conservative with respect to the LEU UMo Interim design [3].

The steady-state heat transfer analysis for the HFIR fuel design considers 26 uncertainty factors to capture the uncertainties in reactor process conditions, tolerances and uncertainties in fuel manufacturing, and analysis/correlation uncertainties. Most of the U_3Si_2-Al uncertainty factors were assumed to be the same as those currently used for the U_3O_8-Al HEU fuel (e.g., fissile loading, power distribution, axial track integral overloading) due to similarities in the manufacturing process.

Uncertainty factors U_{18} and U_{19}, both of which are 1.27, represent the local heat flux effects of a fuel particle agglomeration, for the hot and cold sides of the fuel plates, respectively. Uncertainty factors U_{20} (hot side) and U_{21} (cold side) represent fuel distribution and non-bond peaking effects. For U_3Si_2-Al, fuel segregation flux peaking uncertainty and newly generated fuel distribution and non-bond peaking uncertainty are combined and the resultant uncertainty factor is termed UBAR [22].

Uncertainty factor U_{25} is used to capture the uncertainty associated with the bottom axial location of the fuel zone. Current HEU specifications allow (1) a ±0.635 cm variation in the end location of the fuel meat on either end of the nominally 50.80 cm long active region and (2) the fuel plate to be axially misaligned within the fuel element up to 0.0381 cm. Thus, the total maximum allowable axial misalignment of the fuel-bearing material beyond the nominal boundary is 0.6731 cm. To account for this effect, Shift-calculated (or MCNP) distributions are applied to the last row of nodes to provide additional exit power peaking. This maximum misalignment is less conservative than that assumed and approved in the current HFIR SAR [23] and has not been approved for regulatory safety-basis analysis.

All safety requirements must be met following the conversion. The safety design criteria provided in the HFIR SAR establish a bounding approach for calculating the reactor safety system settings, SLs, and LCSs for the key process variables of reactor power, primary coolant flow, core inlet coolant pressure, and core inlet coolant temperature. For HFIR, the reactor safety system limits on core thermal power and primary coolant flow are coupled into the flux-to-flow ratio, which is the ratio of percent of full power divided by the percent of full primary coolant flow.

For these feasibility studies, reactor operating Mode 1 full flow safety limit calculations for flux-to-flow are analyzed because this condition is typically limiting for HFIR steady-state heat transfer analyses. With the flux-to-flow ratio at its SL (1.36), all other variables at their LCS (inlet temperature = 57.2° C and inlet pressure = 2.41 MPa), and all uncertainties in the technical knowledge of the process resolved unfavorably (uncertainty factors), no hot spot burnout can occur. Burnout margin is the safety metric used to determine the acceptability of the studied U_3Si_2-Al LEU fuel designs, and therefore, the burnout margin variations with time into cycle for the studied designs are compared with the SL value of 1.36 and the HEU-calculated burnout margin in the results section of this paper. The studies performed in this paper therefore do not cover the full extent of steady-state bounding calculations or RELAP transient calculations required by the SAR.

3.3. Design Considerations

This feasibility study considers 4.8 gU/cm^3 U_3Si_2-Al dispersion fuel because [24] "concludes that plate-type fuels suitable and acceptable for use in research and test reactors can be fabricated with U_3Si_2-Al dispersion compacts with uranium densities up to 4.8 g/cm^3." With an assumed LEU enrichment of 19.75 wt.%, the ^{235}U density in the U_3Si_2-Al fuel meat is 0.948 g/cm^3 (3.025 g ^{235}U/cm^3 for UMo).

When considering conversion to a LEU core, the physical dimensions of the fuel and core geometry are to be preserved to the extent possible. With the exception of the active fuel zone length and plate internals, the HEU core and studied U_3Si_2-Al LEU cores are identical. The fuel plates consist of U_3Si_2-Al fuel and Al-1100 filler regions clad in Al-6061. Boron-10 in the form of B_4C is used in the IFE filler for reactivity hold-down and to reduce radial power peaking. For these studies, a fixed ^{10}B density of 1.74 mg/cm^3-filler is used, which is similar to that of the HEU design. The U_3Si_2-Al fuel length for all designs is assumed to be 55.88 cm, which is a 5.08 cm increase relative to the HEU fuel zone. Elongating the fuel increases the volume available for fuel by 10%, resulting in a greater fuel loading and longer cycle length. Fuel zones longer than 55.88 cm are not considered because they likely won't allow for establishing hydrodynamic stability of the coolant at the entrance region and, more importantly, longer fuel will significantly increase exit power peaking where thermal safety margin is typically limiting.

4. SILICIDE DESIGN FEASIBILITY STUDIES

A scoping study was performed in [4], which led to a 92 MW HFIR fuel design with 4.8 gU/cm^3 U_3Si_2-Al LEU fuel encased in 304.8 µm thick Al-6061 clad. A 304.8 µm clad thickness, which is an increase of 50.8 µm relative to the current HFIR HEU plates, was chosen primarily because U_3Si_2-Al fuel is denser and harder to roll in comparison to U_3O_8-Al. Thus, a thicker clad was analyzed with fabrication and plate acceptance rates (i.e., plate yields) in mind, in an attempt to mitigate particle penetration.

As a starting point for the ORNL feasibility study, this design was incorporated into the HFIR UMo Interim design representative MCNP model [2,3] and modified for use with Shift. One change made to the design included centering the inner fuel element fuel meat within the fuel plate thickness such that the fuel is symmetric about the plate thickness centerline. Herein this design is referred to as the LEUA1 design. This change enhances the heat transfer within the fuel plate. Cross-sections of the HEU and LEUA1 fuel plates prior to being curved into involute plates are provided in Figure 2. A Shift depletion simulation of the LEUA1 design with ~13 kg ^{235}U at 92 MW resulted in a reactor cycle length of ~27 days. The LEUA1 performance metrics are comparable to those of the HEU core but the cold source cold neutron flux and ^{252}Cf production rate, which are two of HFIR's primary missions, are reduced relative to the HEU core.

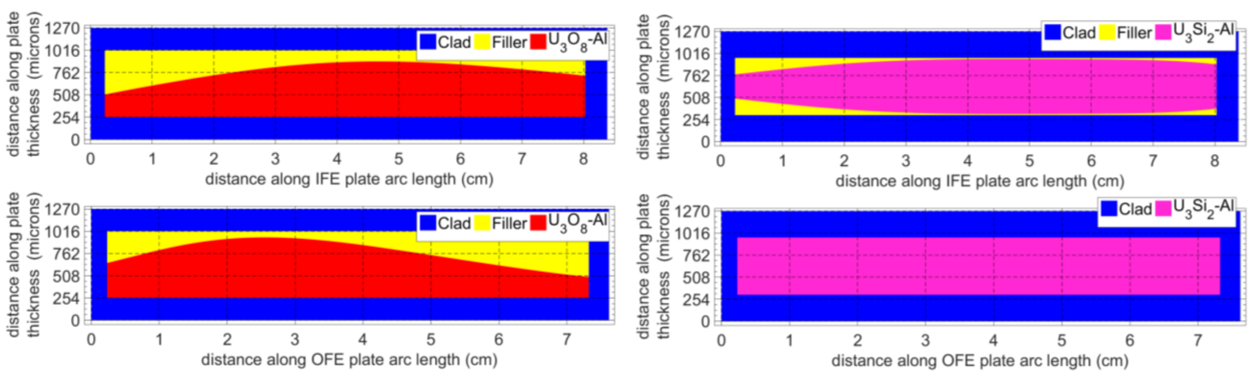

Figure 2. HEU fuel (left) and LEUA1 design (right) flat plate cross-sections.

The Shift-derived neutronics data was then processed for insertion into the HSSHTC, which was executed to assess the variation in burnout margin with time. HSSHTC results indicate that the LEUA1 design does not meet the 1.36 SL requirement at BOC and its thermal safety margin is less than that for the HEU core. A detailed mesh study was performed on this design at BOC. The power input method (continuous and step-wise shape profiles) and mesh size (very fine to very coarse), specifically at the radial edges, were varied and the burnout margin for the seven cases analyzed varied between 1.26 and 1.38.

4.1. Iterative Approach to U_3Si_2-Al Design

The LEUA1 design does not meet requirements set forth for the conversion of HFIR. Extreme radial and axial power peaking proved to be unfavorable for thermal-hydraulic analyses. This design made the best use of the fuel plate interior volume, so the obvious design constraint needing to be relaxed is the fuel plate clad thickness. Reducing the clad thickness back to the HEU design thickness of 254 µm will increase the available plate interior by 15.4%. This will allow for increased fuel meat thicknesses in the center of the plate, which will help to flatten the radial power profile. In turn, the power may be able to be increased to enhance reactor performance while simultaneously gaining needed thermal safety margin. The negative impact associated with reducing the clad thickness is the potential increase in the number of rejected plates due to the rolling performance of the denser silicide fuel.

A step-by-step approach is taken to design a U_3Si_2-Al LEU HFIR core to maximize the burnout margin and meet or exceed the HEU key performance metrics. More emphasis is focused on improving the thermal-hydraulic performance of the fuel than the neutronics performance because thermal safety margin is more difficult to attain as the minimal fuel plate interior volume limits the designer's ability to contour the fuel. Both axially flat and axially contoured fuel designs are assessed; however, more emphasis is placed on designing an axially flat fuel because an axially contoured dispersion fuel may not be feasible to fabricate. There is no attempt to generate an optimized axially contoured fuel design in these feasibility studies, but axially contoured fuels making use of fuel meat thickness profiles of non-axially contoured designs are analyzed to estimate the impact on reactor performance and thermal safety margin.

The approach taken to increase the thermal safety margin throughout this study consisted primarily of analyzing previous designs' radially dependent fuel meat thickness and maximum local-to-critical heat flux ratio profiles. The local-to-critical heat flux ratio profiles indicate regions where margin is needed and where margin can be sacrificed. For example, if a radial streak's maximum ratio is 0.99, no more fuel should be loaded into this region because additional power in this streak would cause burnout in this streak. However, if a radial streak's maximum ratio is 0.70, more fuel should be loaded into this region because it has plenty of margin. A desired design will have a flat local-to-critical heat flux ratio profile consisting of "ones" across the plate because this design will provide the best performance in terms of balancing thermal and nuclear performance.

4.2. Summary of Evaluated U_3Si_2-Al Designs

Brief descriptions of the designs assessed are provided in Table II along with a high-level summary of the physics metrics included in this feasibility study. This table indicates whether the design meets the key performance metrics requirements or not. It also provides the minimum burnout safety margin and the minimum equivalent channel thickness predicted throughout the cycle. As previously discussed, the current HFIR burnout margin SL is 1.36, but a greater margin is required to render a LEU design option feasible because the minimum HEU margin is 1.61 and the assumptions made for the U_3Si_2-Al uncertainty factors (e.g., U_{25} calculation approach) may need to be refined, making it prudent to favor a more conservative calculation at this early stage of silicide design evaluation. The minimum equivalent channel thickness, which is the smallest hot streak axially integrated average channel thickness predicted during the reactor cycle, is provided in Table II because it is an important input to the RELAP transient analysis. Table III provides a summary of the key performance metrics calculated for all the designs analyzed. Fast neutron fluxes in the material irradiation locations are greater for all U_3Si_2-Al designs, whereas cycle length and ^{252}Cf production rates are typically the limiting performance metrics.

Two preferred designs are identified. The LEUB5 and LEUB9 designs are the favored non-axially contoured and axially contoured designs, respectively, because they both met the key performance metrics and yielded satisfactory thermal safety margins, which were greater than the 1.36 SL but less than the HEU margins. It is believed that additional thermal safety margin can be gained by adding a Gd or Sm neutron poison into the IFE filler along with the ^{10}B poison, which itself could be further optimized. A burnable poison in the OFE may also help to increase thermal safety margin. Loading neutron poisons that deplete quickly will help to reduce radial power peaking early in the cycle (first few days) when thermal margin is least. However, the reactor cycle length and other performance metrics could potentially be reduced depending on the amount and type of burnable absorber used.

The designs presented in this paper with axial contouring at the bottom of the active fuel zone show huge benefits in increasing burnout margin. These designs were not optimized and a few have thermal safety margins greater than those for the HEU core. Thus, if silicide is deemed a candidate fuel option for HFIR and the LEU conversion program's Fuel Fabrication pillar concludes that an axial contour is possible, further optimization will enhance the performance of the silicide fuel with axial contouring.

Table II. Description and results summary of evaluated U_3Si_2-Al designs.

Design	P [MW]	Fuel length [cm]	Axial contour [a)]		^{235}U loading [kg]	symmetric [b)] or asymmetric	pass or fail key performance metrics	Min. burnout margin [-]	Min. eq. ch. thickness [mm] [c)]
			length [cm]	bottom thickness [μm]					
HEU	85	50.80	N/A		9.4	asymmetric	N/A	1.61	0.939
LEUA1	92	55.88	N/A		13.0	symmetric	fail	1.31	0.966
LEUB1 [d)]	95	55.88	N/A		13.7	symmetric	N/A	1.38	N/A
LEUB2 [d)]	95	55.88	3	150	13.4	symmetric	N/A	1.74	N/A
LEUB3	95	55.88	N/A		13.8	symmetric	pass	1.43	0.922
LEUB4	95	55.88	3	150	13.5	symmetric	fail	1.65	0.967
LEUB5 [e)]	95	55.88	N/A		13.9	symmetric	pass	1.47	0.927
LEUB6	95	55.88	3	200	13.6	symmetric	fail	1.64	0.967
LEUB7	95	55.88	N/A		13.6	symmetric	fail	1.45	0.928
LEUB8	95	55.88	N/A		13.9	asymmetric	pass	1.38	0.926
LEUB9 [e)]	95	55.88	1	200	13.9	symmetric	pass	1.59	0.965
LEUB10 [e)]	95	55.88	1	200	13.9	asymmetric	pass	1.48	0.966

a) Designs LEUB2, B4, and B6 are the same as LEUB1, B3, and B5, respectively, with the exception of the axial contour.
b) Fuel meat profile is either assumed to be symmetric or asymmetric about the fuel plate thickness centerline.
c) Minimum equivalent channel thickness.
d) Only BOC conditions analyzed. Therefore, only BOC burnout margin results are presented.
e) Designs with sufficient key performance metrics.

Table III. Key performance metrics summary of U_3Si_2-Al designs.

Design		Cycle length [d]	Cold source moderator vessel cold flux [10^{14}n/cm^2-s]	Cold source cold-to-total flux ratio	^{252}Cf production [mg/day]	Flux trap fast flux [10^{15}n/cm^2-s]	Flux trap fast-to-total flux ratio	Reflector fast flux [10^{14}n/cm^2-s]	Reflector fast-to-total flux ratio
HEU		26.2	4.48	0.736	1.388	1.07	0.29	2.89	0.192
LEUA1		27.2	4.45	0.723	1.369	1.13	0.308	3.10	0.199
LEUB	1	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	2	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	3	27.1	4.64	0.724	1.398	1.15	0.308	3.22	0.198
	4	25.9	4.68	0.724	1.351	1.16	0.308	3.24	0.198
	5	27.1	4.65	0.724	1.383	1.14	0.306	3.23	0.198
	6	25.8	4.69	0.724	1.337	1.15	0.307	3.25	0.198
	7	26.3	4.66	0.725	1.371	1.14	0.307	3.23	0.197
	8	27.0	4.65	0.724	1.389	1.13	0.306	3.24	0.198
	9	27.2	4.64	0.724	1.398	1.15	0.308	3.22	0.198
	10	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Although an exhaustive design and optimization study was not performed as part of this feasibility assessment, many conclusions can be drawn with respect to conversion of HFIR with silicide fuel. Table IV provides a qualitative summary on the radial and axial contouring effects on thermal safety margin. HEU-like thermal safety margin and performance can likely be achieved with U_3Si_2-Al fuel if the fuel plates are fabricated with (1) 254 µm thick cladding, (2) fuel meat that is radially contoured, (3) fuel meat that is centered and symmetric about the fuel plate thickness centerline, (4) fuel meat that is axially contoured at the bottom 1-3 cm of the active fuel zone, (5) fuel meat that is 55.88 cm in length, and (6) IFE filler regions with a burnable poison. A thermal safety margin sufficiently greater than the 1.36 SL, but less than that of the HEU core, may be obtained with a fuel design incorporating (1) centered and symmetric fuel with no axial contour or (2) off-centered and asymmetric fuel with a short axial contour.

Table IV. Qualitative thermal safety margin summary for silicide radial and axial contour options.

<table>
<tr><td colspan="2"></td><td colspan="3">Radial Contour Description</td></tr>
<tr><td colspan="2"></td><td>none (flat)</td><td>centered and symmetric</td><td>off-centered and asymmetric</td></tr>
<tr><td rowspan="2">Axial Contour Description</td><td>none (flat)</td><td>Insufficient Margin</td><td>Sufficient Margin</td><td>Insufficient Margin</td></tr>
<tr><td>short contour (1-3 cm)</td><td>Insufficient Margin</td><td>HEU-like Margin</td><td>Sufficient Margin</td></tr>
</table>

The designs and results provided in this section show that, with the design constraints and caveats discussed in this paper, U_3Si_2-Al may be a feasible fuel option for HFIR conversion. It is also clear that current HFIR HEU fuel fabrication procedures are not directly applicable to producing these silicide fuel plate designs. Feedback and testing by the Fuel Fabrication and Fuel Qualification pillars will be needed to supplement this paper because many assumptions regarding fuel fabrication and performance are made in the calculations that need to be verified or modified. It is also important to keep in mind that this feasibility study considered only neutronics and steady-state reactor operating Mode 1 flux-to-flow burnout safety margin calculations. The full suite of steady-state calculations has not yet been performed and no RELAP transient or kinetics calculations have been performed.

4.3. Physics Results Summary for the LEUB5 Design

Physics results calculated for the LEUB5 design using the Shift neutron transport and depletion code are compared to those calculated for the HEU design using the VESTA and MCNP tools in this section. The estimated cycle lengths for the HEU and LEUB5 (Fig. 3) designs are ~26 and 27 days, respectively; the symmetrical critical control element withdrawal curves for these designs are illustrated in Fig. 4. The initial critical positions for both designs are approximately the same (~44.5 cm), which indicates that the fresh, submerged subcritical reactivity worths are likely similar. On average, the control elements are more withdrawn for the silicide design in comparison to the HEU core and the withdrawal rate for the HEU core is greater at the end of the cycle.

The initial ^{235}U loading for the HEU and LEUB5 cores is 9.44 and 13.91 kg, respectively. At the end of the cycle, the HEU and LEUB5 cores contain ~6.68 and 10.80 kg ^{235}U. Thus, ~29 and 22% of the total initial ^{235}U is removed during the 26-day HEU cycle and the 27-day LEU cycle, respectively. Only a small amount of ^{239}Pu (~11.55 g) is generated in the HEU core because the fuel only contains a small amount of ^{238}U. However, due to the increased ^{238}U loading in the LEU core, the LEUB5 design generates ~276 g ^{239}Pu during the reactor cycle. The HEU core initial ^{10}B loading and percent depletion are 2.71 g and 94%, whereas the LEUB5 initial loading and percent depletion are 2.47 g and 87%, respectively.

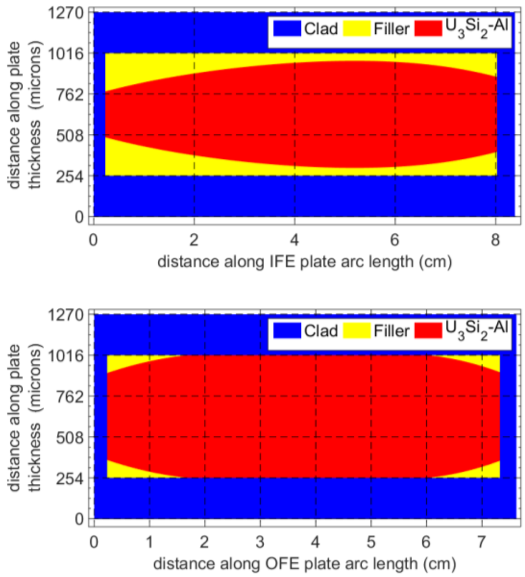

Figure 3. LEUB5 design flat plate cross-sections.

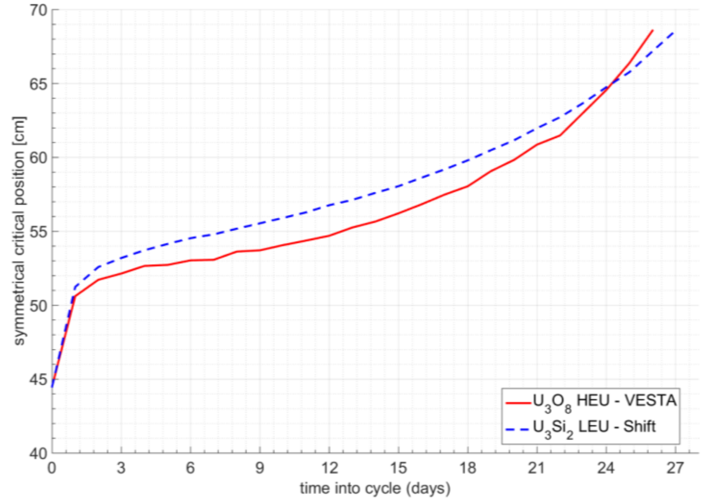

Figure 4. Symmetrical critical control element withdrawal curves.

Figs. 5 and 6, respectively, illustrate the LEUB5 core end-of-cycle ^{235}U depletion and cumulative fission density distributions. The maximum IFE and OFE ^{235}U depletion values are 62.3 and 47.3%, respectively, and their maximum cumulative fission densities are 1.38x10^{21} and 1.02x10^{21} fissions/cm^3-U_3Si_2-Al. The location of the maxima in the IFE is at the inner radial edge on the core midplane and the location of the maxima in the OFE is at the outer radial edge on the core midplane.

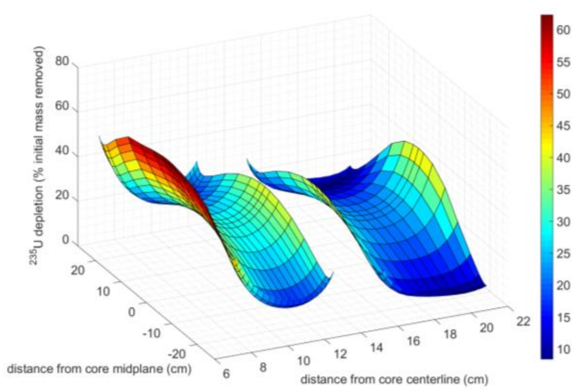

Figure 5. LEUB5 end-of-cycle ^{235}U depletion distribution.

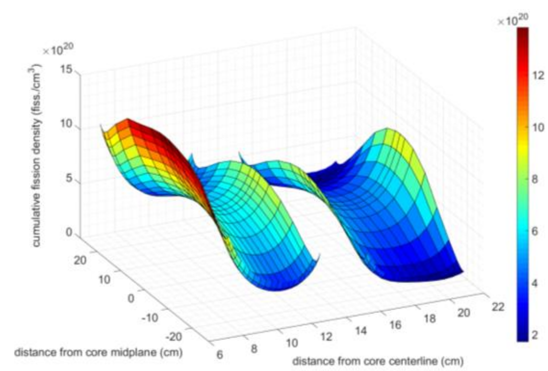

Figure 6. LEUB5 end-of-cycle cumulative fission density distribution.

During the depletion simulation, five full length ^{252}Cf production targets in the flux trap target region and ^{238}Pu production targets in the inner small, outer small, and large vertical experiment facilities are activated. These targets are irradiated for several cycles, but the purpose of this work was to model a single reactor cycle. Approximately 36.1 and 37.3 mg ^{252}Cf are produced during the HEU and LEUB5 reactor cycles, respectively. For the HEU core, ~47, 19, and 33 g ^{238}Pu is present in the inner small, outer small, and large facilities at end-of-cycle, whereas ~51, 21, and 36 g is produced for the LEUB5 core in the same facilities. These production values do not account for the decay of ^{238}Np (2.117 day half-life) after reactor shutdown. Thus, the total production of ^{252}Cf and ^{238}Pu is slightly better for the LEUB5 design in comparison to the HEU core.

5. CONCLUSIONS

Key performance metrics and thermal safety margins were evaluated for HFIR using neutronic and steady-state thermal-hydraulic analyses for silicide dispersion fuel in this feasibility study. In order to maximize the available volume for fuel within the existing plate geometry, a longer (50.80 cm to 55.88 cm) fuel zone was proposed. A number of designs were proposed and evaluated using an iterative approach in an effort to show that reactor performance could match that obtained using HEU fuel and that thermal safety margins were adequate. Two designs, one with (LEUB9) and one without (LEUB5) fuel zone axial contouring, were identified that met the required performance metrics and had adequate thermal safety margins. However, there is a strong interrelationship between the analyzed designs and the ability to fabricate them with specified tolerances and cost-effective yields. Fabrication features important to the selected designs that remain to be demonstrated include fuel zone shape and location, "filler" shape and location, burnable absorber location, cladding minimum thickness and alloy, and potentially higher uranium density loadings. It also remains for silicide fuel to be qualified for the more severe conditions of higher volumetric power and ensuing heat flux and temperatures inherent in a high-power research reactor such as HFIR.

It should be noted that the present feasibility study was conducted without verification of codes or analyses, that no transient analyses were performed, and that material properties and behavior correlations were not verified. It must also be emphasized that this study does not provide a basis for eliminating UMo monolithic LEU fuel from selection as the preferred fuel for HFIR conversion. Except for the aforementioned potential fabrication issues associated with HFIR's complex fuel design, UMo monolithic fuel exhibits better performance and safety margins. If NNSA Office of Materials Management and Minimization determines U_3Si_2-Al fuel is worth pursuing as a candidate HFIR LEU fuel, the ORNL team will continue to optimize the fuel design based on knowledge gained in this feasibility study and feedback provided by the LEU conversion program's Fuel Fabrication and Fuel Qualification pillars.

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