Advances in Nuclear Fuel Management IV (ANFM 2009)
Hilton Head Island, South Carolina, USA, April 12-15, 2009, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2009)

DESIGNING A NEW FUEL FOR HFIR—PERFORMANCE PARAMETERS FOR LEU CORE CONFIGURATIONS

Germina Ilas, R. T. Primm III, and Jess C. Gehin
Oak Ridge National Laboratory
Oak Ridge, Tennessee, USA
ilasg@ornl.gov, primrtiii@ornl.gov, gehinjc@ornl.gov

Keywords: research reactor, low enriched fuel, HFIR

ABSTRACT

An engineering design study for a fuel that would enable the conversion of the High Flux Isotope Reactor from highly enriched uranium to low enriched uranium fuel is ongoing as part of an effort sponsored by the U.S. Department of Energy's National Nuclear Security Administration through the Global Threat Reduction Initiative. Given the unique fuel and core design and high power density of the reactor and the requirement that the impact of the fuel change on the core performance and operation be minimal, this conversion study presents a complex and challenging task, requiring improvements in the computational models currently used to support the operation of the reactor and development of new models that would take advantage of newly available simulation methods and tools. The computational models used to search for a fuel design that would meet the requirements for the conversion study and the results obtained with these models are presented and discussed. Estimates of relevant reactor performance parameters for the low enriched uranium fuel core are presented and compared to the corresponding data for the currently operating highly enriched uranium fuel core.

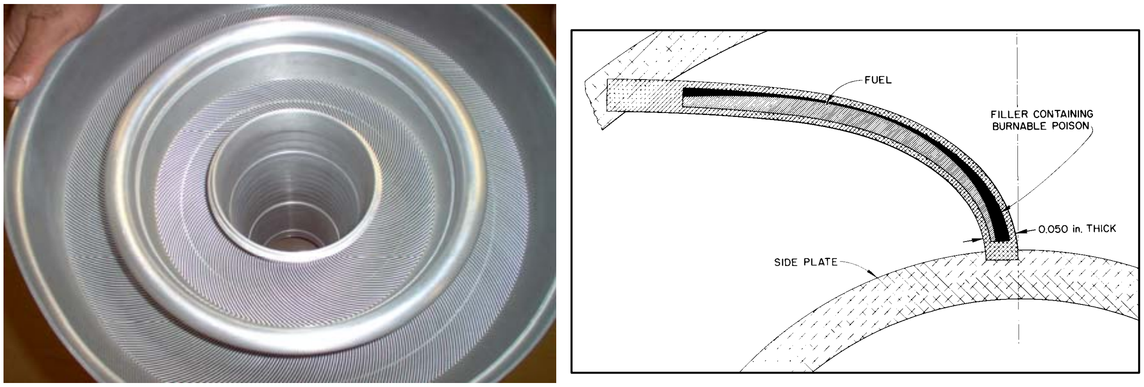
1. INTRODUCTION

The High Flux Isotope Reactor **(**HFIR) is an 85 MW, very high flux, pressurized light-water cooled and moderated flux-trap type reactor, which is operated at the Oak Ridge National Laboratory (ORNL). It is currently fueled with highly enriched uranium (HEU). The primary mission of HFIR is to support neutron scattering experiments; however, it is also used for medical, industrial, and research isotope production and materials irradiation research. The core of the reactor consists of a series of concentric annular regions: a central flux trap containing vertical experimental targets surrounded by two annular fuel elements separated by a thin water region; a region containing two control plates; a beryllium reflector; and a water region to the edge of the pressure vessel, which is located in a pool of water. Details of the reactor configuration and operation can be found elsewhere.[1,2]

There are no fuel management issues with HFIR as occur in power reactors. There is no fuel shuffling. The two fuel elements are required to have the same burnup, meaning they are almost always loaded fresh or, if discharged mid-cycle, are kept paired. Only once during the past six years have fuel elements been removed from the core during a mid-cycle shutdown and then subsequently reloaded several months later. Consequently, all HFIR fuel management issues are addressed at the design stage.

An engineering design study for a fuel that would enable the conversion of HFIR from HEU to low enriched uranium (LEU) is ongoing[2] as part of an effort sponsored by the U.S. Department of Energy's National Nuclear Security Administration through the Global Threat Reduction Initiative. Given HFIR's unique fuel and core design and high power density, and the requirement that the impact of the fuel change on the core performance and operation be minimal, this conversion study presents a complex and challenging task. Such a task requires improvements in and extensions of the computational methodologies and tools that are currently used to support the operation of the reactor.

The two fuel elements in HFIR, identified as inner fuel element (IFE) and outer fuel element (OFE), are composed of numerous, involute-shape, 1.27 mm thick fuel plates that are separated by 1.27 mm thick coolant channels, as illustrated in Fig. 1(a). The fuel plates have a sandwich type design, with a fuel region enclosed in an aluminum-based clad, as illustrated in Fig. 1(b). The fuel meat inside the fuel region is characterized by variable thickness along the width of the fuel plate (radial grading) and a uniform thickness along the height of the fuel plate for a given radius (no axial grading). The fuel meat is the region that will need to be changed when replacing the current HEU fuel with LEU fuel, with no changes to the geometry of the fuel plate. The fuel region for the HEU fuel contains a mixture of aluminum powder and uranium oxide (U_3O_8) with 93 wt % ^{235}U enrichment. The LEU fuel currently under consideration is a high-density monolithic uranium-molybdenum alloy, U-10Mo, which contains 90 wt % uranium and 10 wt % natural molybdenum and has uranium enrichment of 19.75 wt % ^{235}U.



(a) inner and outer fuel elements (b) fuel plate profile

Fig. 1 HFIR fuel elements.

The neutronics analyses of HFIR performance for LEU fuel were carried out initially[3] using the Monte Carlo transport code MCNP[4] and the SCALE[5] and BOLD-VENTURE[6] code systems. BOLD-VENTURE, a three-dimensional (3-D) multigroup diffusion theory code with depletion capabilities, was used to obtain the power profile and the peak fluxes in the target and reflector regions for various initial LEU fuel distributions as a function of time during the reactor cycle. The data calculated with BOLD-VENTURE were normalized based on results from 3-D MCNP simulations for the beginning of cycle (BOC) configuration.

Two main directions were pursued recently to improve the computational methodologies used for simulations of HFIR LEU configurations. Recognizing the efficiency of using a very fast code such as BOLD-VENTURE for scoping studies, to quickly perform numerous fuel grading calculations, and provide estimates of the reactor safety and performance parameters, efforts were focused on improving the methodology for generating multigroup cross section data for use with BOLD-VENTURE. A new cross section processing methodology for LEU fuel configurations was developed[7] to ensure a more appropriate representation of the cross sections; this is of particular importance for the fuel regions located near the edges of the fuel element, which are characterized by large leakage from fuel regions to surrounding moderator regions. This new methodology is based on the two-dimensional (2-D) arbitrary polygonal mesh discrete ordinates transport code NEWT in SCALE.[8] Use of this methodology for a preliminary LEU fuel design and for a simplified fresh fuel core model resulted in a significant improvement in predicting the distribution of power across the fuel elements compared with the one-dimensional cross section processing methodology previously used. The new cross section methodology can also be used to prepare multigroup cross section data for use with other deterministic transport codes.

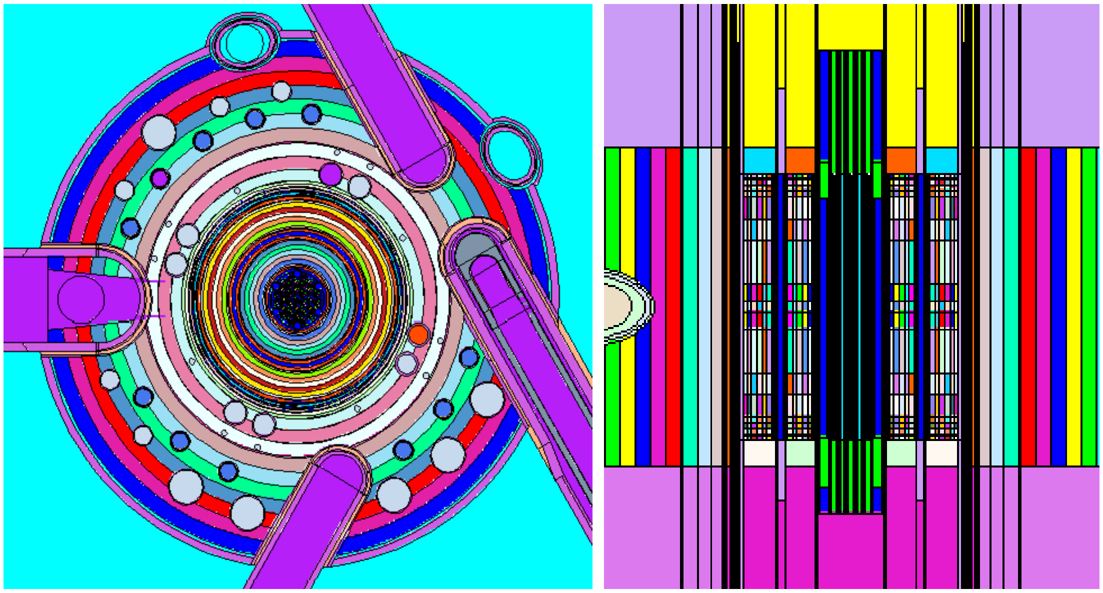
A second direction for improving the neutronics analysis was to develop a Monte Carlo-based depletion model for studies of HFIR LEU configurations using the ALEPH[9] computational tool. This tool has been used in the search for a fuel grading that would meet the conversion study requirements. Once a near optimal fuel load and grading are established from BOLD-VENTURE and ALEPH studies and coupled thermal-hydraulics analyses, the distribution is confirmed and refined with Monte Carlo depletion calculations (ALEPH). Results of the Monte Carlo depletion studies for LEU fuel are presented and discussed in this paper.

2. MONTE CARLO DEPLETION MODEL FOR HFIR LEU CONFIGURATIONS

ALEPH is a Monte Carlo-based depletion tool, developed at SCK-CEN in Belgium, that couples a Monte Carlo transport code from the MCNP family of codes (e.g., MCNP, MCNPX) and the point depletion and decay code ORIGEN 2.2.[10] It is a relatively user-friendly code; if an appropriate MCNP model of the configuration to be analyzed is available, the changes and/or additions to this model are minimal. At each depletion step, the transport flux solution from MCNP is used to generate the cross section data for the ORIGEN 2.2 depletion calculation; the isotopic composition data resulting from ORIGEN 2.2 are used in the subsequent MCNP transport calculation to obtain cross sections for the next depletion step; and so on, in an iterative manner. ALEPH has a particular approach, compared to other Monte Carlo depletion tools, for determining from MCNP the data needed for the ORIGEN 2.2 depletion calculation. Whereas other tools obtain the cross sections for depletion based on reaction rate tallies in the Monte Carlo transport calculation, ALEPH requires only flux tallies in a fine-group structure. The one-group cross sections for ORIGEN 2.2 are obtained by weighting available pointwise cross section data with the MCNP-calculated fine-group flux. These pointwise cross section data are consistent with the cross section data used in the MCNP transport calculation as both sets are precomputed based on the same ENDF/B data files. The user must obtain sufficient convergence in flux spectra to accurately collapse cross section data, but of course, the same is true for accurately determining cross sections for depletion with other Monte Carlo-based depletion codes. By calculating only the flux with MCNP, as compared to calculating reaction rates in other codes, execution times for ALEPH are considerably less than for other Monte Carlo depletion tools.

2.1 MCNP model for ALEPH

The MCNP model used for HFIR LEU configurations is based on the 3-D MCNP model for HFIR HEU cycle 400.[11] The model is illustrated in Fig. 2, which shows both a radial cross section at the core midline and an axial cross section through the center of the core; various regions and materials are shown in different colors. The model explicitly represents the experiment locations in the central target region and beryllium reflector region. As used in the MCNP model for cycle 400, the fuel in the IFE region is modeled by homogenizing the fuel meat and aluminum cladding of the fuel plates and the water in between the fuel plates. To approximate the variation of the ^{235}U content in the radial direction of the fuel plate (i.e., radial fuel grading), eight radial regions with different ^{235}U concentrations are used in the IFE modeling. A similar model is used for the OFE, but with nine radial regions. Axially, the fuel element region is divided into 19 axial layers for calculation purposes. The concentration of ^{235}U in the axial direction was initially considered uniform, though axial grading was used in some of the studied cases, as discussed below.



(a) radial view (b) axial view

Fig. 2 MCNP model for HFIR LEU core.

The selection of the axial layer dimensions was carried out by studying trends in the thermal neutron flux and microscopic thermal fission cross section of ^{235}U as a function of radial and axial location in the fuel element to establish an optimal axial zoning of the fuel elements in the Monte Carlo model in ALEPH that would better represent the axial variation across the core of the neutron flux and fission density. A simplified 3-D MCNP model of HFIR, representing half of the core, was used for this trend study; the simplification being in the representation used for the central target, reflector, and control regions, whereas the IFE and OFE were modelled with the same level of detail as in the model for HFIR cycle 400. The variation of the thermal flux as a function of axial location for a given radial region in the IFE and OFE is illustrated in Fig. 3. As observed, the regions at the top (or bottom) edge of the fuel elements are characterized by large leakage from fuel-bearing to non-fuel-bearing regions and neutron flux spectra much different from the average flux in the fuel element. A more refined axial grid was therefore used for the end regions of the fuel elements to better represent the flux variation in these regions.

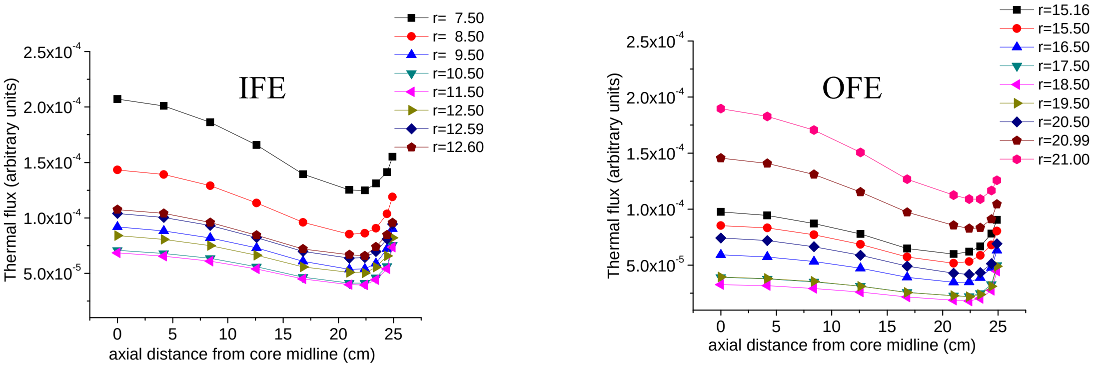


Fig. 3 Axial variation of thermal flux in fuel elements.

2.2 Depletion with ALEPH

In addition to the MCNP model of the configuration to be simulated, the input data for ALEPH include information about the depletion mixtures (i.e., materials whose composition varies during simulation due to depletion and decay) and irradiation history. A total of 152 fuel regions in the IFE (8 radial by 19 axial) and 171 fuel regions (9 radial by 19 axial) in the OFE were used in the model. For cases with no axial grading of the fuel, a number of 80 and 90 depletion mixtures were specified in the IFE and OFE, respectively, for flux calculation with MCNP. As previously mentioned, this flux serves to weight the cross section data for obtaining the one-group cross sections for use in the ORIGEN 2.2 depletion calculation. Eight of the 80 depletion mixtures in the IFE are specified in the central (i.e., core midline) axial layer of the IFE, one for each of the 8 radial regions. A unique depletion mixture is specified for fuel regions with the same radial region number and with the same axial distance with respect to the core midline. Similarly, there are 90 depletion mixtures in the OFE, which gives a total of 170 depletion mixtures in the fuel elements. The material in the curium targets of the central target region is also considered as a depletion mixture.

The value used for power during the irradiation was, as will be discussed, either 85 or 100 MW. The cross section libraries used in the simulations were based on data from ENDF/B-VI release 8. All cross sections were considered at 300 K temperature. The movement of the control elements during irradiation was included in the simulation, using a control plate movement scenario similar to that for HFIR HEU cycle 400.

3. LEU FUEL LOAD AND FUEL GRADING

One of the main requirements for HFIR conversion is that facility capabilities be maintained at current levels. To test whether a proposed LEU design meets this requirement, the following performance parameters were studied first for the LEU core and compared to their values for the current HEU core: cycle length, power distribution across the core, neutron flux in the central target region, reflector, and cold source location.

3.1 Variation of k_{eff} as a Function of the LEU Fuel Load

As a first step, the fuel load in the ALEPH model for an LEU core was varied to determine the loading that would ensure a core lifetime to that of the HFIR HEU core. As previously mentioned, no changes in the fuel plate geometry are to be made; therefore, the maximum thickness of the fuel meat region and the thickness for the fuel clad or cooling channel are the same as for the HEU fuel design. The initial LEU fuel design considered had a radial fuel grading profile as illustrated in Fig. 4 for a 17.9 kg ^{235}U load and uniform grading in the axial direction. This initial grading was used for five values for the ^{235}U load: 17.0, 17.9, 20.0, 25.0, and 30.5 kg. The maximum total load (i.e., fuel meat thickness less than the maximum possible value of 0.762 mm) corresponding to this radial grading shape is 30.5 kg. For computation speedup, the depletion simulations in all these five cases were carried out with seven depletion steps to reach a total irradiation time of 26 days. The movement of the control elements during the cycle was not simulated; the control elements were considered at their fully withdrawn end of cycle (EOC) locations. A value of 85 MW was used for the operating total thermal power.

The variation of the effective multiplication constant (k_{eff}) with the irradiation time for this fuel configuration is shown in Fig. 5 for each of the five values considered for the ^{235}U load. As seen, to reach a core lifetime (i.e., total irradiation time for which k_{eff} is greater than 1.0) of around 26 days, an initial ^{235}U load of about 25 kg would be necessary. The corresponding uranium load would be about 127 kg, a large increase compared with the current HEU uranium load of 10.1 kg.

3.2 Search for an Optimal Fuel Grading

Iterative reactor core physics/thermal-hydraulics calculations were performed to search for an optimal fuel load and fuel grading that would ensure a core performance similar to that of the currently operating HEU core. Iterations were carried out on both total ^{235}U load and radial fuel grading profile by searching around the 25 kg value for the total ^{235}U load. Depletion simulations with ALEPH were performed initially for a thermal power of 85 MW as used in the current HEU core. For each assumed fuel grading profile, the corresponding core power distribution was applied in steady-state thermal-hydraulics analysis[12] to verify whether the thermal margins were maintained. Later, for the LEU cases, the power had to be increased to 100 MW in order to keep both flux performance at the target locations and thermal margins similar to that for the HEU core. The first iterations were carried out for various profiles of the radial grading only, starting with the profile illustrated in Fig. 4. Axial grading was included later due to thermal-hydraulics limits.

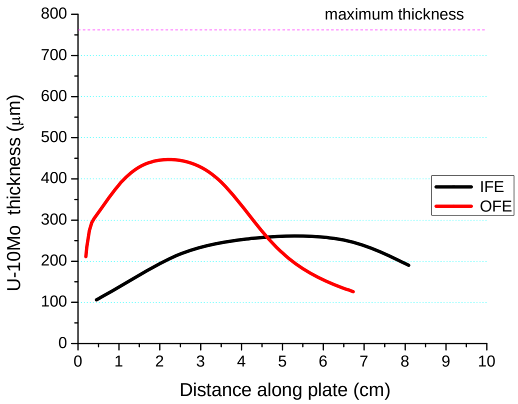


Fig. 4 Initial fuel element radial profile.

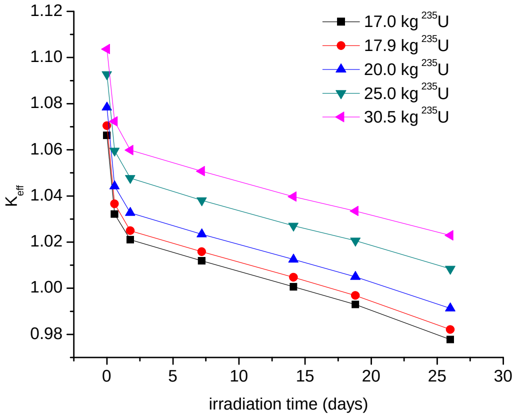


Fig. 5 K_{eff} vs. irradiation time.

The results of the search indicated as the "best" candidate a core with a total ^{235}U load of 25.3 kg with a radial fuel grading profile as illustrated in Fig. 6 and with axial grading applied to the bottom 3 cm of the fuel elements. The concentration of ^{235}U in the bottom 3 cm of the fuel elements for a given radial location was considered to be half of the value used in the rest of the axial regions of the fuel elements for that radial location. The use of this axial grading was based on the observation that, as the water coolant enters the top of the core and flows from the top to the bottom of the core, the occurrence of a power "spike" at the bottom of the fuel elements would be more problematic than one at the top of the fuel elements, and therefore this occurrence should be the first thing to be avoided. The results of a thermal-hydraulics analysis showed, as discussed elsewhere,[13] that the reduction of plate thickness at the bottom 3 cm would ensure a predicted maximum operating power of 103 MW for HFIR with LEU at BOC, with the same margin-to-incipient-boiling as exists for the current HEU fuel configuration.

In addition to the axial grading discussed above, two other axial grading cases were studied, in which the width of the graded axial layer at the bottom of the fuel element was changed from 3 cm to 2 cm and 1 cm. In each of these two cases, the relative power distribution was calculated for BOC and used in the thermal-hydraulics analysis to estimate the power that would meet the thermal margins requirement. The result was that both cases were insignificantly different from the 3 cm axial grading case. This would indicate that the decision on tapering the bottom end of the fuel plates becomes a fabrication issue, decided by minimizing the cost of manufacturing. At this time, the "best" fuel grading configuration, selected for further analysis, corresponds to the radial fuel grading shown in Fig. 6, with additional axial grading for the bottom 3 cm of the fuel plate as discussed above.

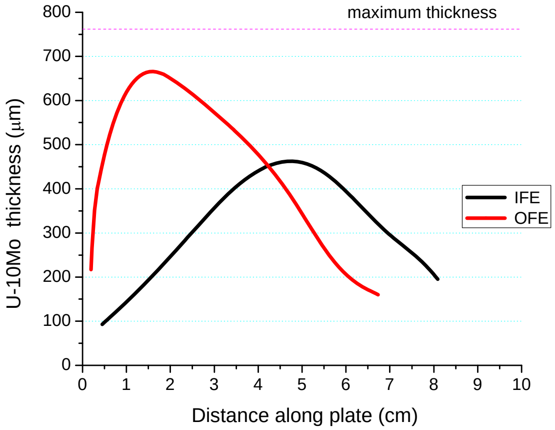


Fig. 6 Fuel element plate radial profiles for 25.3 kg ^{235}U load.

4. PERFORMANCE PARAMETERS FOR THE LEU CORE

Depletion simulations were carried out with ALEPH for the fuel grading and load established as discussed above to calculate relevant performance parameters for the core. A total of 25 depletion steps—24 of one day and one of 0.3 days duration for a total of 24.3 days irradiation time—were used, the same as used for the HFIR HEU cycle 400 simulation with ALEPH. The same history of the control elements movement as for cycle 400 was considered. The power used for the depletion simulation was 85 MW for the HEU cycle 400 core and 100 MW for the LEU core. The value estimated for k_{eff} at EOC for the LEU core was 0.9963 ± 0.0001, which is in good agreement with the 0.9990 ± 0.0002 obtained for the HFIR HEU cycle 400 simulation.

The relative fission density data were calculated for each of the defined regions in the fuel elements based on flux and fission density tallies in MCNP for both BOC and EOC (at 24.3 days). These data served as input for the thermal-hydraulics analysis that showed that the corresponding operating power is 103 MW at BOC and 100.5 MW at EOC; these values of the power are the maximum power levels at which all thermal limits are met. The relative fission density data at BOC are illustrated in Figs. 7 and 8 for the IFE and OFE, respectively.

As mentioned previously, the neutron flux level, one of the key parameters for characterizing core performance, should be maintained when replacing HEU by LEU so that facility missions will not be affected. A comparison of three-group flux data estimated based on MCNP flux tallies for the current HEU core at 85 MW power and the proposed LEU core at 100 MW power at BOC and EOC, respectively, is presented in Table 1. The energy structure for the shown three-group data is: thermal < 0.625 eV; epithermal 0.625 eV–100 keV; fast 100 keV–20 MeV. As the flux tallies provided by MCNP are normalized to the source (i.e., 1 fission neutron), the values for the flux in n/cm^{2}s were obtained by multiplying the flux tally values by the total source. The total source *S* was approximated as[11]

$$S = \frac{\nu P}{Ee} \quad , \tag{1}$$

where ν is the average number of neutrons per fission, *P* the reactor power in MW, *E* the average energy per fission in MeV, and *e* is a conversion factor. An approximate value of 200 MeV was used for E, whereas the ν value was directly taken from the MCNP output. The value of the total source at BOC is 6.47×10^{18} n/s for HEU at 85 MW power and 7.65×10^{18} n/s for LEU at 100 MW power. The flux is shown for three important experiment locations: the central target in the flux trap, the edge of the cold source, and the beryllium reflector at 27 cm from the center at core midline. As observed from data in Table 1, the fluxes corresponding to the LEU core are comparable to those for the HEU core.

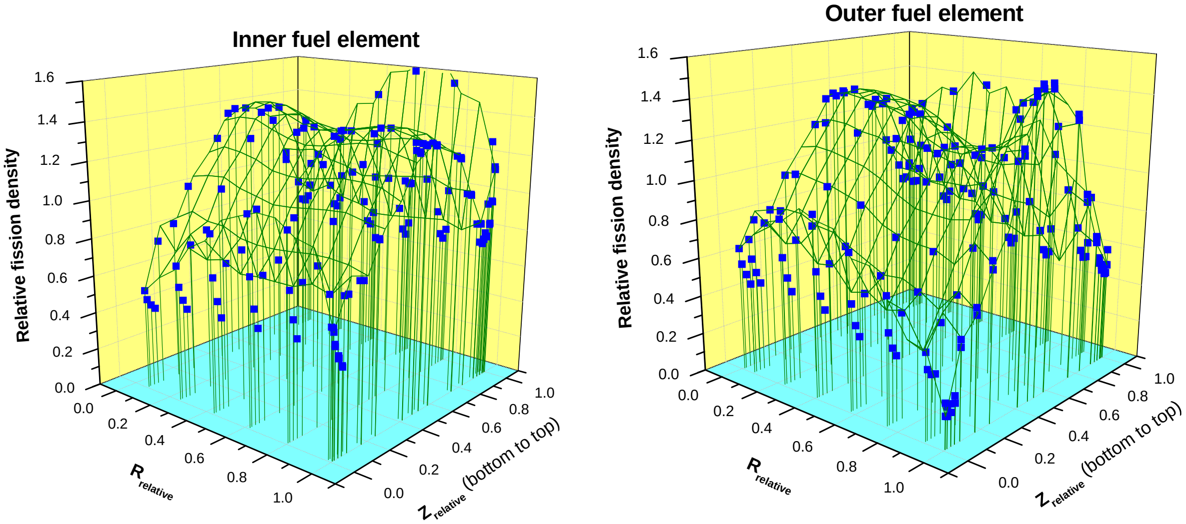


Fig. 7 Relative power density in fuel elements.

The isotopic composition of the spent fuel, which is important to safeguards, reactor safety, and waste management, the plutonium inventory in particular, is another core parameter to be assessed for the LEU fuel. A comparison of the HEU and LEU cores with respect to the total mass in the core at EOC for main actinides and some fission products is presented in Table 2. As expected, the production of plutonium increases, given the large fraction of ^{238}U present in LEU fuel.

Other important safety parameters such as decay heat generation, void reactivity, control element worth, and dose rates will be calculated based on the results of ALEPH simulations for the current "best" configuration.

Table 1. Neutron flux—comparison of HEU cycle 400 and LEU cores.

Location	Time	Fuel	Thermal flux (n/cm^{2}s)	Epithermal flux (n/cm^{2}s)	Fast flux (n/cm^{2}s)
Central target	BOC	HEU	2.2×10^{15}	1.3×10^{15}	1.1×10^{15}
		LEU	2.3×10^{15}	1.2×10^{15}	1.0×10^{15}
	EOC	HEU	2.3×10^{15}	1.1×10^{15}	9.9×10^{14}
		LEU	2.5×10^{15}	1.2×10^{15}	1.0×10^{15}
Cold source edge	BOC	HEU	6.8×10^{14}	2.4×10^{14}	9.0×10^{13}
		LEU	8.1×10^{14}	2.8×10^{14}	1.0×10^{14}
	EOC	HEU	8.3×10^{14}	2.4×10^{14}	8.9×10^{13}
		LEU	8.3×10^{14}	2.7×10^{14}	9.9×10^{13}
Reflector r=27cm	BOC	HEU	6.0×10^{14}	6.5×10^{14}	4.1×10^{14}
		LEU	7.0×10^{14}	7.7×10^{14}	4.8×10^{14}
	EOC	HEU	8.1×10^{14}	6.5×10^{14}	4.0×10^{14}
		LEU	7.2×10^{14}	7.3×10^{14}	4.5×10^{14}

Table 2. End of cycle inventory data (grams) for HEU and LEU cores.

Nuclide	HEU	LEU	Nuclide	HEU	LEU
^{234}U	88.0	232.1	^{135}I	1.3	1.4
^{235}U	6,785.0	22,250.0	^{131}Xe	18.6	22.7
^{236}U	502.3	740.3	^{133}Xe	23.3	27.9
^{238}U	532.0	101,700.0	^{135}Xe	0.1	0.3
^{237}Np	6.2	9.4	^{133}Cs	50.2	60.3
^{238}Np	0.1	0.1	^{134}Cs	1.5	1.3
^{239}Np	2.8	76.2	^{135}Cs	2.9	12.3
^{238}Pu	0.3	0.6	^{141}Ce	58.8	68.8
^{239}Pu	11.4	390.9	^{143}Nd	26.3	32.2
^{240}Pu	1.4	25.4	^{145}Nd	49.4	58.3
^{241}Pu	0.6	8.1	^{147}Nd	14.1	17.5
^{242}Pu	<0.1	2.8	^{86}Kr	15.8	18.1
^{149}Sm	0.4	1.9	^{93}Zr	53.5	61.8
^{150}Sm	13.2	14.1	^{97}Mo	51.4	60.2
^{151}Sm	1.1	3.3	^{99}Tc	43.6	51.6
^{152}Sm	7.0	7.1	^{101}Ru	47.0	55.7
^{153}Sm	0.6	0.6	^{103}Ru	24.3	29.9
^{147}Pm	12.0	15.4	^{103}Rh	5.1	6.5
^{148}Pm	0.3	0.3	^{105}Rh	0.5	1.0
^{148m}Pm	0.1	0.2	^{10}B	0.2	0.7
^{149}Pm	2.1	2.4	^{11}B	12.5	10.3
^{143}Pr	40.9	48.1			

5. CONCLUSIONS

A Monte Carlo-based depletion model for a HFIR core with LEU fuel has been developed. This model served as an aid in the design of an LEU fuel that would ensure core performance would be equivalent to that of the currently operating HEU core. The comparison of relevant core performance parameters for the LEU and HEU cores, such as neutron flux at important experiment locations and fuel cycle length, indicates that the objective of maintaining the current level of reactor performance appears to have been achieved. Studies are ongoing to refine the established design and to estimate other parameters of importance to reactor safety and spent fuel disposal.

ACKNOWLEDGMENTS

The authors would like to thank the U.S. Department of Energy's National Nuclear Security Administration for sponsoring this work through its Global Threat Reduction Initiative program.

REFERENCES

1. R. D. CHEVERTON and T. M. SIMS, "HFIR Core Nuclear Design", ORNL-4621, Oak Ridge National Laboratory (1971).

2. R. T. PRIMM III, R. J ELLIS, J. C. GEHIN, D. L. MOSES, J. L. BINDER, and N. XOUBI, "Assumptions and criteria for performing a feasibility study of the conversion of the High Flux Isotope Reactor core to use low-enriched uranium fuel," CD Proceedings, PHYSOR 2006, Vancouver, Canada (2006).

3. R. J. ELLIS, J. C. GEHIN, and R. T. PRIMM III, "Cross section generation and physics modelling in a feasibility study of the conversion of the High Flux Isotope Reactor core to use low-enriched uranium fuel," CD Proceedings, PHYSOR 2006, Vancouver, Canada (2006).

4. "MCNP—A general Monte Carlo n-particle transport code," Version 5, LA-CP-03-0245, Los Alamos National Laboratory (2003).

5. "SCALE: A modular code system for performing standardized computer analyses for licensing evaluations," Version 5.1, Vols. I–III, ORNL/TM-2005/39. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-732 (2006).

6. D. R. VONDY, T. B. FOWLER, and G. W. CUNNINGHAM III, "The BOLD VENTURE computation system for nuclear reactor core analysis," Version III, ORNL-5711, Oak Ridge National Laboratory (1981).

7. G. ILAS, J. C. GEHIN, and R. T. PRIMM III, "New cross section processing methodology for HFIR core analysis," CD Proceedings, PHYSOR 2008, Interlaken, Switzerland (2008).

8. M. D. DEHART, "NEWT: A New Transport Algorithm for Two-Dimensional

Discrete Ordinates Analysis in Non-Orthogonal Geometries", NUREG/CR-0200, Rev. 7, Vol. 2, Sec. F21 of reference 5, U.S. Nuclear Regulatory Commission (November 2006).

9. W. HAECK and B. VERBOOMEN, "An optimum approach to Monte Carlo burnup," *Nuclear Science and Engineering*, **156**, 180 (2007).
10. RSICC Computer Code Collection CCC-371, ORIGEN 2.2. Available from RSICC, Oak Ridge National Laboratory (2002).
11. N. XOUBI and R. T. PRIMM III, "Investigation of beryllium internal reflector installation on the fuel cycle length of the High Flux Isotope Reactor," ORNL/TM-2004/252, Oak Ridge National Laboratory (2005).
12. H. A. MCLAIN, "HFIR Fuel Element Steady State Heat Transfer Analysis," Revised Version, ORNL/TM-1904, Oak Ridge National Laboratory, 1967, as appended by T. E. Cole, L. F. Parsly, and W. E. Thomas, "Revisions to the HFIR Steady State Heat Transfer Analysis Code," ORNL/CF-85/68 (1986).
13. R. T. PRIMM III, G. ILAS, and C. GALVEZ VELIT**, "**Status of design effort for LEU fuel for the High Flux Isotope Reactor," RERTR-2008 International Meeting on Reduced Enrichment for Research and Test Reactors, Washington, D.C. (2008).