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A NEUTRONIC FEASIBILITY STUDY FOR LEU CONVERSION OF THE BUDAPEST RESEARCH REACTOR*

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ABSTRACT

A neutronic feasibility study for conversion of the Budapest Research Reactor (BRR) from HEU to LEU fuel was performed at Argonne National Laboratory in cooperation with the KFKI Atomic Energy Research Institute in Hungary. Comparisons were made of the reactor performance with the current HEU (36%) fuel and with a proposed LEU (19.75%) fuel. Cycle lengths, thermal neutron fluxes, and rod worths were calculated in equilibrium-type cores for each type of fuel. Relative to the HEU fuel, the LEU fuel has up to a 50% longer fuel cycle length, but a 7-10% smaller thermal neutron flux in the experiment locations. The rod worths are smaller with the LEU fuel, but are still large enough to easily satisfy the BRR shutdown margin criteria. Irradiation testing of four VVR-M2 LEU fuel assemblies that are nearly the same as the proposed BRR LEU fuel assemblies is currently in progress at the Petersburg Nuclear Physics Institute.

INTRODUCTION

A neutronic feasibility study was conducted for the potential conversion of the Budapest Research Reactor at the KFKI Atomic Energy Research Institute from HEU to LEU fuel. The study focused on comparison of the reactor performance with HEU (36%) and LEU (19.75%) fuels. Calculations were made of the equilibrium fuel cycle lengths, the thermal neutron fluxes in the in-core and ex-core experiment locations, and the control- and safety-rod reactivity worths. Multigroup diffusion theory calculations of the neutron fluxes and the rod worths were benchmarked to detailed Monte Carlo calculations. All calculations were made using ENDF-B/VI nuclear data.

BRR DESCRIPTION

Reactor Model

The BRR is an upgraded[1] (December 1992) 10 MW research reactor that uses VVR-type fuel assemblies and is cooled and moderated with light water. The reactor core is reflected radially with three separate materials: an inner replaceable beryllium reflector, an outer fixed beryllium reflector and a light-water reflector. The axial reflector above and below each radial reflector material is light water. A plan view of the BRR is shown in Fig. 1.

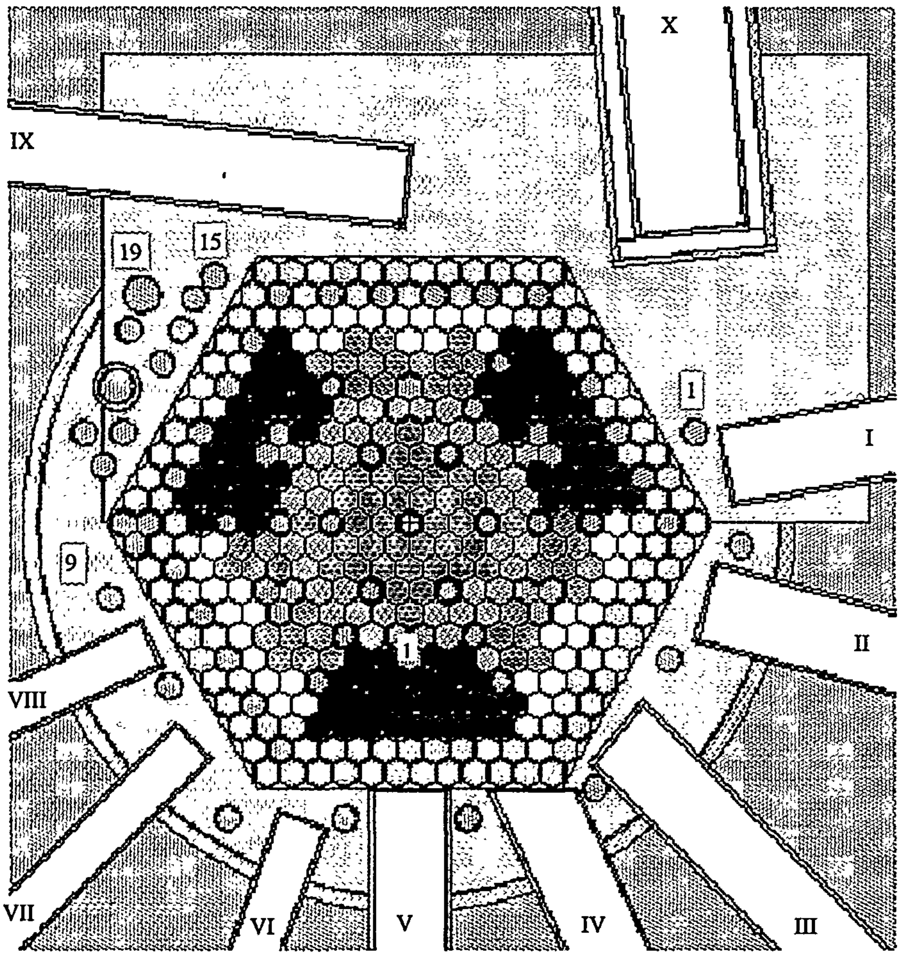

Figure 1. Plan View of the BRR.

The BRR has ten horizontal beam tubes consisting of eight radial tubes (I-VIII), a tangential tube (IX) and a cold neutron source tube (X). An aluminum tank (radius 115 cm) surrounds the radial water reflector; a second aluminum tank (radius 50 cm) separates the water and fixed beryllium reflectors.

There are 397 assembly positions in 12 hex-rings that can be used to build the desired reactor core. The core size depends upon the number of fuel assemblies, the number and location of the in-core experiments, and the control- and safety-rod locations. The remaining assembly positions are filled with replaceable beryllium assemblies. For purposes of the reactor calculations, the reactor is modeled to a radius of 50 cm. The core and radial reflector active heights are 60 cm, with 60 cm thick light-water reflectors above and below the active height. When rods are fully inserted (0.0 cm), the rod-tip is 5 cm below the active core height and when fully withdrawn (70 cm), the rod-tip is 5 cm above the active height.

There are 19 ex-core experiment locations numbered in a clockwise direction from beam tube I; the first ring is numbered from 1 to 15 and the second ring from 16 to 19. There are three sizes of large. The beam tube radii are: I to V, 5 cm; VI to VIII, 3 cm; IX, 5 cm; and X, 11 cm. The horizontal beam tubes I to VIII are identified as HBT*n*, the tangential tube IX as TAN, and the cold neutron source tube X as CNS. All of the tubes are located at the reactor midplane. Additionally, there are six in-core flux traps (FT) numbered 1 to 6 in a clockwise direction, and 18 rod locations.

In the diffusion theory calculational model, the fixed beryllium reflector that surrounds the 10 horizontal beam tubes and the 19 outer irradiation channels were represented with hexagons that have the same 3.5 cm pitch as the core assemblies. The hexagonal geometry used for diffusion theory calculations does not permit the exact modeling (size and location) of the beam tubes, the ex-core experiment locations or the tank boundary. In the Monte Carlo calculational model, the horizontal beam tubes and vertical irradiation channels were modeled in detail as shown in Fig. 1.

Fuel Assembly Model

The HEU fuel assemblies that are used in this study are VVR-SM and VVR-M2 types which were designed and manufactured in Russia. The fuel assemblies consists of two inner cylindrical fuel elements and an outer hexagonal fuel element. A model of the fuel assembly is shown in Fig. 2. (Note: the hexagon is represented here as having sharp corners, but they are actually rounded.)

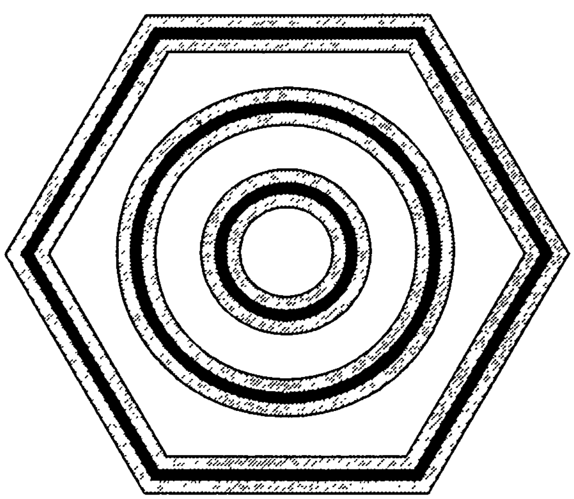

Figure 2. VVR-SM and VVR-M2 Fuel Assembly Model.

Table 1 list the physical specifications of the current HEU (36%) fuel assemblies and the proposed LEU (19.75%) fuel assembly. Each fuel element is 2.5 mm thick including the fuel meat and the (SAV1) aluminum clad thicknesses; the flat-to-flat distance of the hexagon is 32 mm. The BRR currently uses two types of 36% enriched HEU fuel assemblies. One type contains UAl alloy fuel (VVR-SM) and a second type contains UO_2-Al dispersion fuel (VVR-M2). The HEU fuels have different clad and meat thicknesses. The 19.75% enriched LEU fuel assembly contains UO_2-Al dispersion fuel with a uranium density of 2.47 g/cc and a ^{235}U mass of 52.3 g. (Note: this LEU fuel type with an active height of 50 cm instead of 60 cm is currently being irradiation tested[2] at the Petersburg Nuclear Physics Institute in Russia.) Nuclear cross sections of the three fuel assembly types were generated using the WIMS-ANL cross section code[3] in the 7-group energy structure shown in Table 2.

Table 1. BRR Fuel Element Specifications.

Fuel Element Type	HEU UAl Alloy	HEU UO_2-Al	LEU UO_2-Al
^{235}U Enr., %	36.8	36.2	19.75
U Density, g/cc	1.37	1.02	2.47
Element[a]: 1/2/3, mm El./Clad/Meat, mm	32/22/11 2.5/0.9/0.7	32/23.48/11.54 2.5[b]/0.76[c]/1.02	32/22/11 2.5/0.78/0.94
^{235}U Mass, g Element 1 (hex) Element 2 (cyl) Element 3 (cyl) Assembly Total	 21.6 13.0 5.6 40.2 ± 0.8	 23.1 14.9 6.4 44.3 ± 0.6	 28.1 16.9 7.3 52.3 ± 2.5

[a] Outside dimensions of the three fuel elements; each element has an active height of 60 cm.
[b] Element thickness is 2.533 mm with different inner and outer clad thicknesses.
[c] Average clad thickness is 0.7565 mm; the inner clad is 0.753 mm and the outer clad is 0.760 mm.

Table 2. BRR Energy Group Structure.
(Lower-energy group boundaries; Group-1 upper energy is 10 MeV)

1- 0.821 MeV	2- 5.53 keV	3- 4.0 eV	4- 0.625	5- 0.25	6- 0.058	7- 1.0×10^{-5}

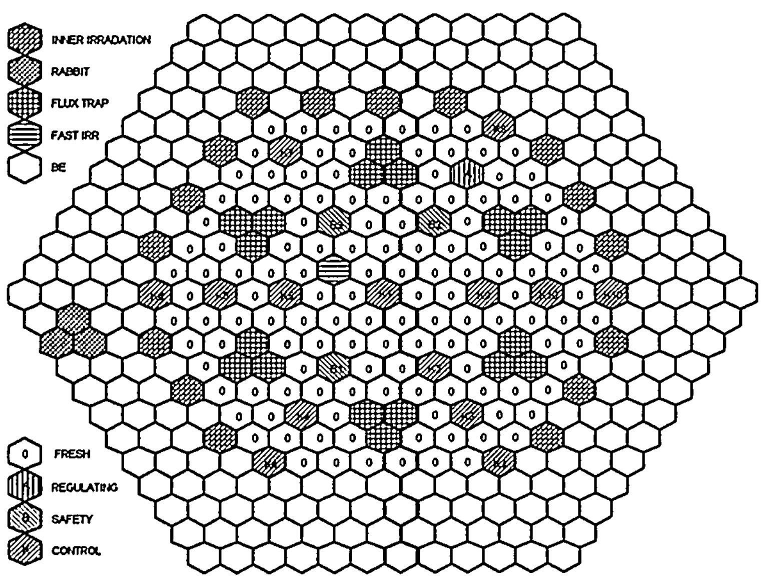

Figure 3. BRR Initial Core Configuration.
(132 Fuel Assemblies)

Reactor Operation

The BRR in its upgraded state began operation in 1992 with an initial core that contained 132 fresh HEU UAl alloy fuel assemblies[1]. This core was expanded over the years to about 228 fuel assemblies with six burnup-level groups of about 38 fuel assemblies each. Based upon the data presented in Ref. 1, an idealized 228-assembly equilibrium fuel cycle model was made for this study. (Note: in the initial all fresh fuel core the HEU UAl alloy fuel assemblies had a ^{235}U enrichment of 35.7%, a uranium density of 1.40 g/cc, and an average ^{235}U content of about 39.8 g. These data are slightly different than the data in Table 1 for the HEU UAl alloy fuel now in use.)

Figures 3 and 4 show respectively, the 132- and 228-assembly core configurations. The fresh fuel in Fig. 3 is labeled with the number 0, and the fuel burnup levels in Fig. 4 are labeled 1-6. In the equilibrium model at the end of a fuel cycle, fuel assemblies in position 1 are moved to position 2, position 2 fuel is moved to position 3, etc. Fuel is discharged from position 6 and fresh fuel is inserted in position 1.

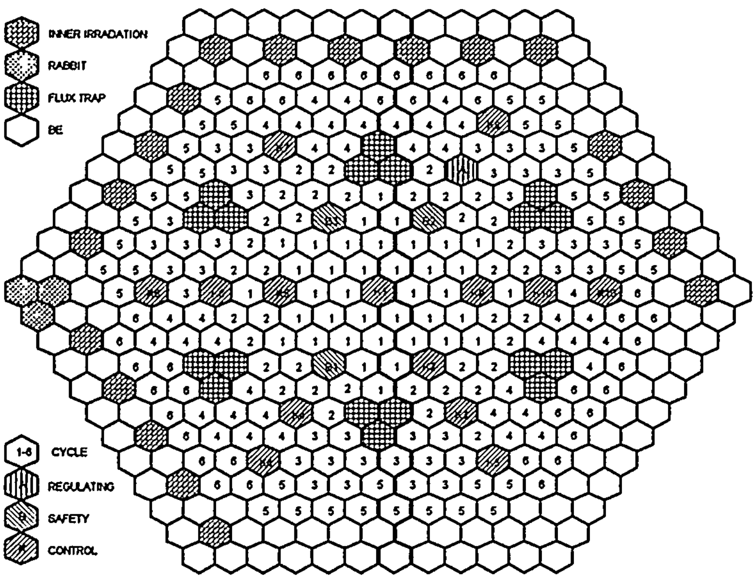

Figure 4.. BRR Equilibrium Core Configuration. (228 Fuel Assemblies)

FUEL CONVERSION STUDY RESULTS

Fuel Cycle Length

Equilibrium fuel cycle calculations, using the REBUS burnup code[4], were made for the HEU UAl alloy, HEU UO_2-Al and LEU UO_2-Al fuel types in the equilibrium core configuration. The results of these calculations are shown in Table 3 and in Fig. 5.

At a power level of 10 MW, the fuel cycle lengths are 84.5 and 98.8 full-power-days (FPD) with the two HEU (36%) fuels, and 127.8 FPD with the LEU (19.75%) fuel. The average ^{235}U discharge burnup of the three fuel types are 67, 71 and 74%, respectively. The excess reactivity at the end-of-the-equilibrium-fuel-cycle (EOEC) is chosen to be about 2% $\Delta k/k^2$ (k-eff $\approx$ 1.02). The EOEC corresponds to a critical reactor, at power, with the central control rod (K1) inserted and the 13 other control and 3 safety rods withdrawn. Assuming 4000 full-power-hours (FPH) of reactor operation per year, the number of fuel assemblies discharged are: 75 assemblies with HEU UAl alloy fuel, 64 assemblies with HEU UO_2-Al fuel, and 50 assemblies with LEU UO_2-Al fuel.

Table 3. Equilibrium Fuel Cycle Characteristics for the Reactor Model With 228 Assemblies and Six Burnup Level Groups of 38 Assemblies Each.

Fuel Assembly Type	Enr., % / ^{235}U Mass, g	Equilibrium Fuel Cycle Length, d	Average ^{235}U Discharge Burnup, %	Fuel Assemblies Used per Year[a]
HEU UAl alloy	36.8 / 40.2	84.5	67.4	75
HEU UO_2-Al	36.2 / 44.3	98.8	70.8	64
LEU UO_2-Al	19.75 / 52.3	127.8	74.4	50

[a] Assuming 4000 full power hours or 167 full power days of operation per year.

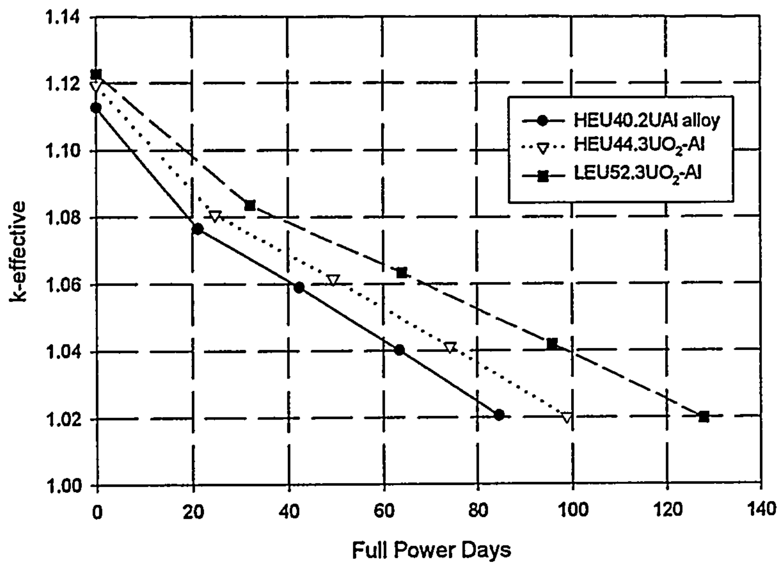

Figure 5. BRR Reactivity Rundown Curves.

Rod Reactivity Worth

Control- and safety-rod worths were calculated with the DIF3D diffusion theory code[5] and benchmarked with the MCNP Monte Carlo code[6]. In the diffusion theory calculations when rods are inserted, internal boundary conditions were used to represent the rods instead of using B_4C-absorber nuclear cross sections. (All rods, including the regulating rod, are now aluminum-clad B_4C; in the initial core configuration the regulating rod was stainless steel.) The multigroup internal boundary conditions that are used were generated from an MCNP calculation and are assumed to be the same for all rod locations. Reactor calculations with MCNP and DIF3D were made assuming all fresh fuel in the initial 132-assembly core and in a hypothetical 228-assembly all fresh fuel core. The all fresh fuel cores were chosen to benchmark diffusion theory to Monte Carlo calculations. Table 4 list the rod worths for the critical, 132 assembly core configuration.

Table 4. BRR Critical Configuration Eigenvalue and Reactivity.

Core/Fuel Assemblies	Fuel Type	Enr., % / ^{235}U Mass, g	Data Source	Excess Reactivity	Critical[a] Rod Worth
Fresh/132	HEU UAl alloy	35.7 / 39.8	Ref. 1	1.2089 (17.3%)	1.0000 (17.3%)
			MCNP[b]	1.1748 (14.9%)	1.0037 (14.5%)
			DIF3D	1.1833 (15.5%)	1.0288 (12.7%)

[a] Critical rod positions: K1 withdrawn, K8 withdrawn 15 cm, other (12) control rods inserted, regulating rod withdrawn 36.4 cm, and (3) safety rods withdrawn. (Rod inserted (withdrawn) is 5 cm below (above) the active core; rod travel is 70 cm.)
[b] MCNP eigenvalue uncertainties are of the order of ± 0.0005.

The data in Table 4 shows that the calculated MCNP and DIF3D eigenvalues and rod worths are smaller than the results reported in Ref. 1. The excess reactivity from Ref. 1 is 17.3% $\Delta k/k^2$, and from MCNP and DIF3D, they are 14.9% and 15.5%, respectively. The larger excess reactivity from Ref. 1 may be due to summing of individually measured rod worths compared to the calculated rod worths that include shadowing effects. The MCNP and DIF3D critical rod worths are in reasonably good agreement except for the critical eigenvalue which is 1.0288. (Note: the DIF3D critical eigenvalue without internal boundary conditions is 1.0039 and the critical rod worth is 15.1% $\Delta k/k^2$. While these DIF3D results would appear to agree better with MCNP, other data suggests that is better to use group dependent internal boundary conditions to assess control- and safety-rod worths.)

Table 5 shows the calculated excess reactivity, shutdown margin and total rod worth for the initial 132- assembly core with fresh fuel, the hypothetical 228-assembly core with fresh fuel, and the beginning-of-equilibrium-cycle (BOEC) cores for the three fuel assembly types. These data indicate that the DIF3D results are in reasonable agreement with the MCNP results and that the required 2% $\Delta k/k^2$ shutdown margin with the safety rods withdrawn[1] are met with either HEU or LEU fuel at BOEC.

Table 5. Reactor Model Eigenvalues and Reactivities.

Core/Fuel Assemblies	Fuel Type	Data Source	Excess Reactivity[a]	Shutdown Margin[b]	All Rod Shutdown[c]
Fresh/132	HEU 39.8	Ref. 1	1.2089 (17.3%)	0.9670 (3.42%)	
	UAl alloy	MCNP[d]	1.1748 (14.9%)		0.8855 (27.8%)
		DIF3D	1.1833 (15.5%)	0.9893 (1.08%)	0.9056 (25.9%)
Fresh/228[e]	HEU 40.2	MCNP[d]	1.2648 (20.9%)		1.0265 (18.4%)
	UAl alloy	DIF3D	1.2736 (21.5%)		1.0311 (18.5%)
BOEC/228	HEU 40.2 UAl alloy	DIF3D	1.1100 (9.91%)	0.9235 (8.29%)	0.8518 (27.3%)
BOEC/228	HEU 44.3 UO_2-Al	DIF3D	1.1154 (10.3%)	0.9264 (7.95%)	0.8530 (27.6%)
BOEC/228	LEU 52.3 UO_2-Al	DIF3D	1.1197 (10.7%)	0.9416 (6.21%)	0.8695 (25.7%)

[a] Excess Reactivity = all (18) rods withdrawn.
[b] Shutdown Margin = (14) control rods inserted, regulating rod inserted, and (3) safety rods withdrawn. Regulating rod is SST in the 132-assembly model and B4C in the 228-assembly models. The shutdown margin[1] must be greater than 2.0% $\Delta k/k^2$ (2.5$) with $\beta = 0.8\%$ $\Delta k/k^2$.
[c] All Rod Shutdown = all (18) rods inserted.
[d] MCNP eigenvalue uncertainties are of the order of ± 0.0005.
[e] Hypothetical core with fresh fuel for calculational model validation only.

Thermal Neutron Flux

Calculated thermal neutron flux results for selected in-core and ex-core experiment locations in a 228-assembly all fresh fuel core are shown in Table 6. The reactor experiment locations include 19 outer irradiation channels (OIC), 20 inner irradiation channels (IIC), 4 flux traps (FT), 8 horizontal beam tubes (HBT), the tangential beam tube (TAN), the cold neutron source tube (CNS), and the pneumatic rabbit facility (RAB). The in-core experiment location fluxes were calculated for a small hexagonal region defined near the axial midplane of the core, and the beam tube fluxes were calculated for a small hexagonal region defined at the tip of the beam tube. As noted above, the circular beam tubes and the outer irradiation channels are approximate as they can not be exactly represented in hexagonal geometry using DIF3D. The MCNP fluxes shown in Table 6 however, are calculated for the exact beam tube and OIC geometry's.

A comparison of the DIF3D and MCNP thermal neutron fluxes for most experiment locations are in fairly good agreement. The CNS tube flux (1.83×10^{13} n/cm^2-s) is in very good agreement and the flux in other locations in the fixed beryllium reflector vary by about 25% or less. These larger flux differences can be attributed to the modeling differences between DIF3D and MCNP. The relative fluxes in all experiment locations are expected to be similar with either DIF3D or MCNP.

Table 6. Thermal Neutron Flux Comparison Between DIF3D and MCNP in Selected BRR Experiment Locations for a Hypothetical 228-Assembly Core with Fresh Fuel.

HEU UAl Alloy Fuel (40.2g ^{235}U), kΦ_{th} (10^{13} n/cm^2-s), $\Phi_{th} < 0.625$ eV								
Loc.	DIF3D	MCNP[a]	Loc.	DIF3D	MCNP[a]	Loc.	DIF3D	MCNP[a]
OIC01	6.09	6.70	HBT1	2.84	3.55	HBT6	3.11	3.72
OIC05	5.02	5.75	HBT2	3.68	3.75	HBT7	4.19	3.95
OIC17	3.29	3.83	HBT3	3.26	4.14	HBT8	4.35	4.54
OIC18	3.03	3.14	HBT4	4.00	4.45	TAN	4.46	4.33
OIC19	2.85	2.91	HBT5	4.97	4.89	CNS	1.83	1.83
IIC29	7.42	7.76						

[a] MCNP uncertainties are of the order of 1 to 2%.

Shown in Table 7 are the BOEC thermal neutron fluxes calculated with DIF3D for a 228-assembly equilibrium core with either HEU (36%) UAl alloy fuel (40.2 g^{235}U), HEU (36%) UO_2-Al fuel (44.3 g^{235}U) or LEU (19.75%) UO_2-Al fuel (52.3 g^{235}U). The experiment location fluxes for the two 36% enriched HEU fuels are nearly the same as indicated in the Table 7, footnote (a). The fluxes with the 19.75% enriched LEU fuel are consistently 7 to 10% less than the fluxes with HEU fuel; see Table 7, footnote (b).

Table 7. BOEC Thermal Neutron Flux Comparisons in the BRR Experiment Locations for Three Fuel Assembly Types.

228-Assembly BOEC Core, $k\Phi_{th}$ (10^{13} n/cm^2-s), $\Phi_{th} < 0.625$ eV								
Loc.	HEU 40.2 UAl alloy 36% Enr.	LEU 52.3 UO_2-Al 19.75%	Loc.	HEU 40.2 UAl alloy 36% Enr.	LEU 52.3 UO_2-Al 19.75%	Loc.	HEU 40.2 UAl alloy 36% Enr.	LEU 52.3 UO_2-Al 19.75%
OIC01	4.99	4.66	IIC41	7.15	6.57	FT1	14.1	12.9
OIC02	3.15	2.94	IIC39	7.48	6.76	FT2	11.7	10.6
OIC03	4.86	4.50	IIC37	7.90	7.14	FT3	11.5	10.4
OIC04	3.48	3.24	IIC35	7.72	6.99	FT4	13.5	12.4
OIC05	4.33	4.03	IIC33	7.38	6.74	FT5	11.7	10.6
OIC06	4.64	4.33	IIC29	7.17	6.51	FT6	12.4	11.2
OIC07	3.09	2.90	IIC27	7.93	7.20	HBT1	2.31	2.17
OIC08	4.64	4.34	IIC25	7.99	7.23	HBT2	3.04	2.81
OIC09	4.14	3.89	IIC23	7.26	6.54	HBT3	2.70	2.50
OIC10	4.12	3.88	IIC21	6.56	6.00	HBT4	3.43	3.18
OIC11	4.70	4.43	IIC19	7.07	6.34	HBT5	4.49	4.15
OIC12	5.12	4.80	IIC17	7.76	6.95	HBT6	2.65	2.48
OIC34	4.68	4.38	IIC15	8.03	7.20	HBT7	3.51	3.28
OIC15	3.67	3.43	IIC13	7.60	6.82	HBT8	3.69	3.45
OIC16	2.75	2.61	IIC11	6.97	6.37	TAN	3.68	3.40
OIC17	2.82	2.67	IIC07	8.36	7.55	CNS	1.51	1.41
OIC18	2.57	2.42	IIC05	8.45	7.64	RAB	5.83	5.44
OIC19	2.39	2.25	IIC03	7.65	6.86			
			IIC01	6.94	6.35			

[a] The BOEC flux ratios of HEU UO_2-Al fuel to HEU UAl alloy fuel for the experiment locations are: OIC ≈ 1.000; IIC ≈ 1.000; FT ≈ 0.983; HBT ≈ 1.000; TAN ≈ 1.000; CNS ≈ 1.000; RAB ≈ 1.000.
[b] The BOEC flux ratios of LEU UO_2-Al fuel to HEU UAl alloy fuel for the experiment locations are: OIC ≈ 0.937; IIC ≈ 0.905; FT ≈ 0.909; HBT ≈ 0.931; TAN ≈ 0.924; CNS ≈ 0.934; RAB ≈ 0.933.

CONCLUSIONS

The neutronic results of this study show that conversion of the BRR from HEU fuel (36% enriched) to LEU fuel (19.75% enriched) is feasible if a qualified LEU fuel is available. The LEU UO_2-Al type fuel used in this study is currently being irradiation tested in the VVR-M reactor at the Petersburg Nuclear Physics Institute.

The equilibrium fuel cycle length is 127.8 days with LEU UO_2-Al fuel which is 30 and 50% longer than the 98.8- and 84.5-day cycle lengths with the HEU UO_2-Al and HEU UAl alloy fuels that are currently in use. This longer cycle length translates into 33 and 22% fewer LEU fuel assemblies (50) used per year compared to the number of HEU fuel assemblies (64 and 75) that are currently used per year.

Control- and safety-rod reactivity worths are smaller with LEU fuel. However, the shutdown margin criteria of 2% $\Delta k/k^2$ or more, is still easily satisfied. With the safety rods withdrawn, the shutdown margin is about 8% $\Delta k/k^2$ with HEU fuel and about 6% $\Delta k/k^2$ with LEU fuel. The thermal neutron flux in the BRR experiment locations are 7-10% smaller with LEU fuel than with HEU fuel.

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