A NEUTRONIC FEASIBILITY STUDY FOR LEU CONVERSION OF THE BROOKHAVEN MEDICAL RESEARCH REACTOR (BMRR)*

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ABSTRACT

A neutronic feasibility study for converting the Brookhaven Medical Research Reactor from HEU to LEU fuel was performed at Argonne National Laboratory in cooperation with Brookhaven National Laboratory. Two possible LEU cores were identified that would provide nearly the same neutron flux and spectrum as the present HEU core at irradiation facilities that are used for Boron Neutron Capture Therapy and for animal research. One core has 17 and the other has 18 LEU MTR-type fuel assemblies with uranium densities of 2.5g U/cm^3 or less in the fuel meat. This LEU fuel is fully-qualified for routine use. Thermal hydraulics and safety analyses need to be performed to complete the feasibility study.

INTRODUCTION

A study was conducted to determine the feasibility of converting the Brookhaven Medical Research Reactor (BMRR) from HEU fuel to LEU fuel. The BMRR has two key irradiation facilities: the Epithermal Neutron Irradiation Facility (ENIF) which is used for Boron Neutron Capture Therapy (BNCT) and the Thermal Neutron Irradiation Facility (TNIF) which is used for animal research. In this phase of the study, the design objectives for the LEU core were to match the HEU neutronic performance (flux magnitude and spectra) at the two irradiation facilities, and to match the fuel usage of about one fuel assembly per year.

BMRR DESCRIPTION

Reactor Model

The BMRR is a 3 MW light-water cooled, graphite reflected reactor that is used for medical and research purposes. A plan view of the BMRR is shown in Fig. 1. The reactor has two irradiation facilities that have tailored neutron spectra. A thermal neutron irradiation facility (TNIF) is used for animal research and an epithermal neutron irradiation facility (ENIF) is used for boron neutron capture therapy (BNCT). The materials between the reactor core and the irradiation facilities were modeled in detail in order to provide an accurate comparison of the neutron fluxes at the irradiation facilities with HEU and LEU fuel in the core.

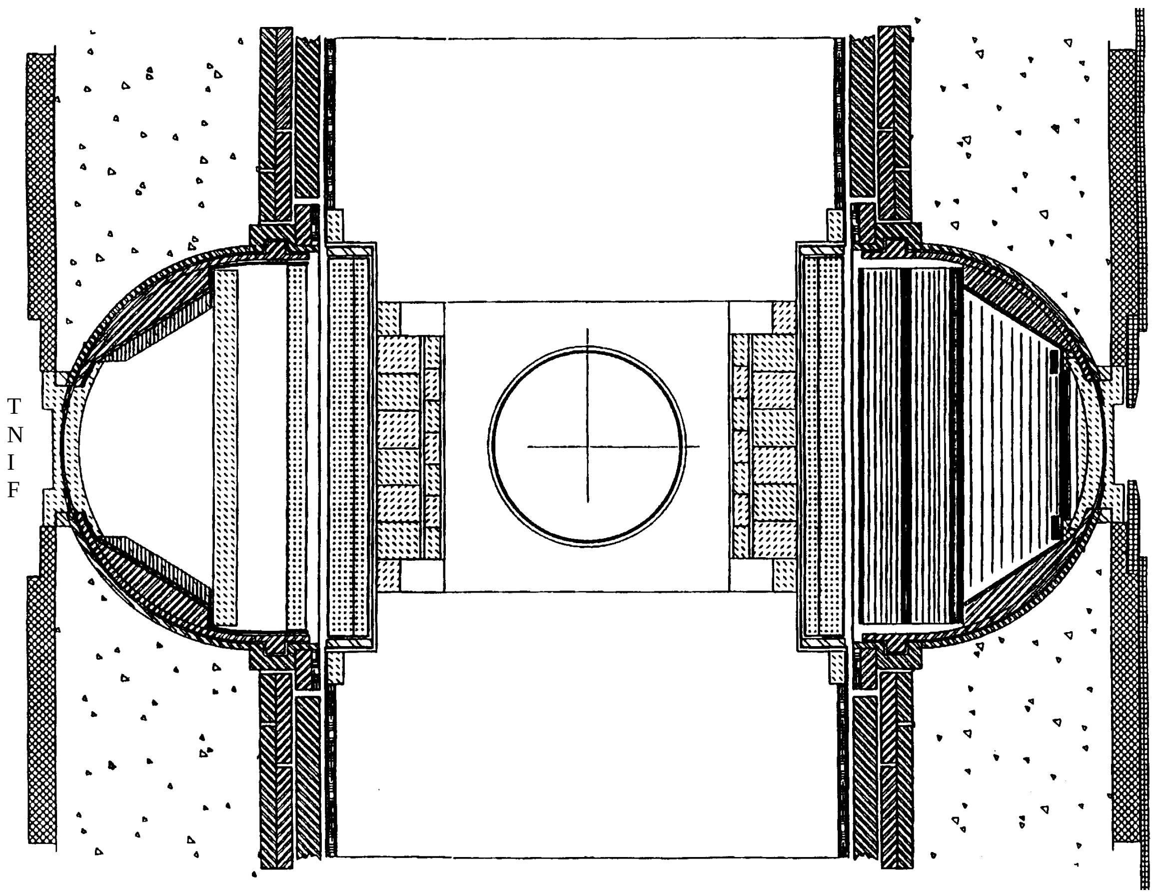

Figure 1. Plan View of the BMRR.

In Fig. 1, the TNIF is located at the end of the west port and the ENIF is located at the end of the east port. The many sections shown between the reactor and the irradiation facilities represent materials (e.g., layers of Al, Al_2O_3, Bi, Cd and D_2O) used to tailor the neutron spectra. The reactor vessel containing the reactor core is located between the ports and is a vertical aluminum cylinder with a 60.96 cm outside diameter and a 6.35 mm-thick wall. A 1.27 cm-thick air gap region separates the reactor vessel from the surrounding graphite reflector which is 173 cm high by 257 cm in the N-S direction and 161 cm in the E-W direction. A 91.44 cm square by 165.1 cm high portion of the graphite reflector immediately surrounding the core has air coolant channels which reduce (5.7%) the graphite density in this region. Forced air cooling is provided to remove heat from the gap region, the inner low-density graphite region and from a small coolant area distributed in the outer portion of the graphite reflector.

The BMRR core is light-water cooled and moderated, and contains four borated stainless steel control rods and 32 possible fuel assembly locations. When a location is not used for a fuel assembly, it is filled with a graphite assembly. The irregular-shaped region inside the reactor vessel is filled with graphite and aluminum. The active core height is about 60 cm with light-water reflectors above and below the core. A thermal shield of laminated steel surrounds most of the reflector and high-density concrete biological shields/shutters are provided on four sides. Within the reactor vessel, the water shielding above the core is more than 4.5 m deep in addition to the concrete plugs covering the top of the reactor vessel. The high-density concrete floor upon which the reactor rests is about 91 cm thick.

HEU Fuel Assembly

The reactor uses HEU (93% enriched) MTR-type fuel assemblies that have 18 fuel plates with U-Al alloy fuel meat. Most of the fuel assemblies had an initial loading of 140g ^{235}U. (Presently, four of the fuel assemblies had an initial loading of 190g ^{235}U in 19 fuel plates). The fuel meat is 0.508 mm thick by 6.350 cm wide by 60.01 cm long and the uranium density is 0.43 gU/cm^3. The Al clad and H_2O coolant channel thicknesses are 0.508 mm and 2.845 mm, respectively; the coolant channel on the outside of the outer fuel plates is 2.350 mm. Each aluminum side plate of the fuel assembly is 4.763 mm by 8.049 cm. The fuel assembly unit-cell dimensions are 7.610 cm by 8.049 cm, spaced on a reactor lattice pitch of 7.709 cm by 8.100 cm.

Reactor Operation

The BMRR began operation in 1959 with an initial core that contained 17 fresh HEU fuel assemblies[1]. The core was expanded over many years of reactor operation to one with 32 partially burned fuel assemblies[2]. The reactor currently operates for about 1000 MWh per year at a maximum power of 3 MW or approximately 14 full-power-days (FPD) per year. Approximately one fuel assembly is replaced each year.

The BMRR operates as a medical research reactor with particular emphasis on the neutron flux spectra present in the two irradiation facilities. Of importance in the TNIF is the thermal neutron flux below 0.4 eV and in the ENIF, the epithermal flux between 0.4 eV and 10 keV. The reactor primarily acts as a source of leakage neutrons from the core with neutron spectra tailored by the various materials located between the core and the thermal and epithermal beam ports. An effort is also made by material selection to reduce the gamma-ray fluxes at both irradiation facilities.

EFFECT OF FUEL ENRICHMENT ON NEUTRON SPECTRA AT IRRADIATION FACILITIES

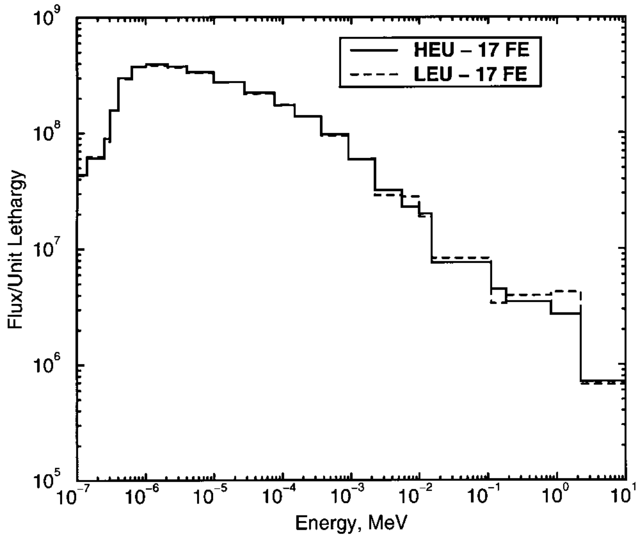

Figure 2. Neutron Spectra at ENIF Beam Port.

The first step in this conversion study was to determine the impact of replacing HEU fuel with LEU fuel in the same core configuration. Because of the distance and materials between the core and the epithermal irradiation facility, it is possible that the enrichment of the fuel in the core has little effect on the magnitude and spectrum of the neutrons reaching this irradiation facility. The original BMRR core configuration (17 fresh fuel assemblies) was selected for this purpose. The results of the analysis show that the magnitude and spectrum of the neutron flux at the ENIF is indeed nearly the same for both HEU and LEU fuels. Figure 2 shows the flux per unit lethargy spectra at the ENIF beam port, that is used for BNCT.

LEU CONVERSION FEASIBILITY STUDY

Core Configurations

These results and analyses of other configurations led to the identification of two possible LEU core configurations that have reactor performance similar to the current HEU core. The LEU core configuration starts with a small (17 or 18 assembly) core of fresh LEU fuel and periodically introduces additional fresh fuel to increase the reactor core size. This is the same pattern of core size development that was used initially with HEU fuel. Over the first few years of reactor operation, less than one fresh LEU fuel assembly, on average, would be introduced per year. Results presented below show that reactor operation and neutronic performance with LEU fuel is essentially the same as that with HEU fuel. Models of the current 32-fuel assembly HEU core and the initial 17-fuel assembly LEU core are shown in Fig. 3. The initial 18-assembly LEU core is similar to the 17-assembly LEU core with the addition of one fuel assembly in position E5 in Fig. 3.

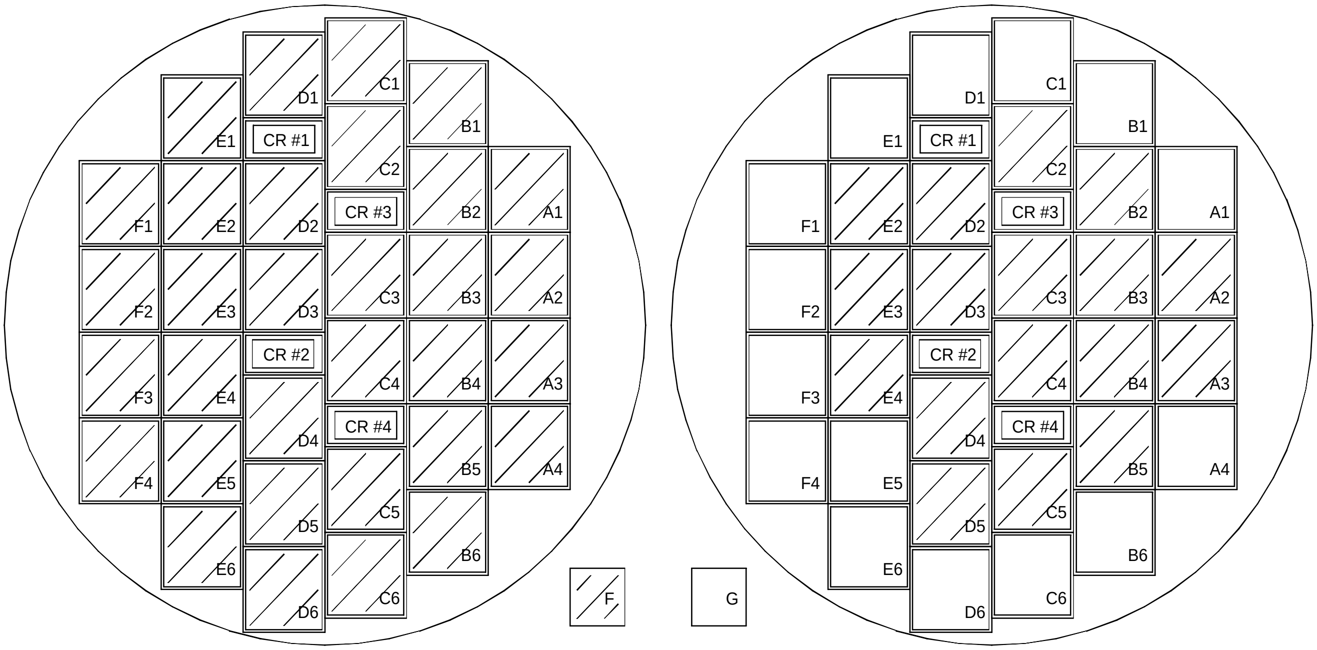

Figure 3. BMRR: HEU (32 Fuel Assemblies) and LEU (17 Fuel Assemblies).

LEU Fuel Assembly

The LEU fuel assembly proposed for both LEU cores is an MTR-type fuel assembly that uses a standardized fuel plate design. A comparison of the LEU (HEU) fuel assembly specifications follows: fuel meat thickness– 0.508 mm (0.508 mm); width– 6.08 cm (6.35 cm); and length– 59.06 cm (60.01 cm); Al clad– 0.381 mm (0.508 mm); and coolant channel thickness– 3.202 mm (2.845 mm). It is noted that with the 25% reduction in clad thickness, the coolant channel thickness increases 12.5%. All other fuel assembly dimension specifications remain the same. The LEU fuel assemblies contain 18 fuel plates and U_3Si_2-Al fuel with 19.75% enriched uranium. A 162.1 g ^{235}U load (2.50 gU/cm^3) is used for the 17-assembly LEU core configuration and a 154.3 g ^{235}U load (2.38 gU/cm^3) is used for the 18-assembly core configuration.

Reactor Models And Computational Methods

Calculations for the LEU and HEU reactor models were performed using the DIF3D diffusion theory code[2], the TWODANT transport theory code[3] and the MCNP Monte Carlo code[4]. Each code was used to investigate specific aspects of the reactor design. Non-equilibrium burnup calculations were performed for the LEU cores using the REBUS code[5]. A ten-group neutron cross section set for the diffusion (DIF3D and REBUS) and transport (TWODANT) codes was generated using the WIMS-D4M code[6]. The energy group structure is shown in Table 1.

Table 1. BMRR Energy Group Structure
(Group – Lower Energy)

1 – 0.821 MeV	2 – 0.183	3 – 9.118 keV	4 – 5.530	5 – 2.10 eV
6 – 0.625	7 – 0.400	8 – 0.250	9 – 0.058	10 – 1.0×10^{-5}

The reactor was modeled in detail, including the many material regions in the beam ports that lead to the irradiation facilities. Control rods were modeled only in MCNP for purposes of control rod worth calculations; they were otherwise considered withdrawn in all other reactor calculations.

All calculations performed with DIF3D were compared with MCNP results and the comparisons were very good (differences in k-effective of less than $0.5\%\Delta k/k^2$). For the LEU cores, end-of-cycle k-effectives were also calculated using MCNP with fuel assembly burnup and fission product data generated from REBUS, and 69-group lumped fission product cross sections generated by WIMS-D4M. The end-of-cycle k-effective was taken to be equal to 1.01.

Flux Performance Results

As previously discussed, the BMRR is designed to provide tailored neutron fluxes at the two irradiation facilities: ENIF and TNIF. Table 2 presents a comparison of the neutron flux intensity at the ENIF beam port and the TNIF reflector/shutter interface for both the current HEU core and the two all-fresh fuel LEU core configurations. Both of the LEU core designs have basically the same thermal flux in the TNIF and the same epithermal flux in the ENIF as the current HEU core. Figure 4 is a plot of the ENIF flux spectra for the current HEU core and the 17-assembly LEU core configuration. It can be seen from Fig. 4 that, at the ENIF, the neutron spectrum for both the HEU and the LEU cores are nearly identical. Similar irradiation facility fluxes are present at different LEU fuel burnup stages. The MCNP fluxes in Table 2 and Fig. 4 have uncertainties of less than 2% and 8%, respectively.

Table 2. Neutron Fluxes at TNIF Reflector/Shutter Interface and ENIF Beam Port with Existing Shutter/Reflector (n/cm^2-s)

Configuration	Thermal E <0.4eV	Epithermal 0.4eV< E <10keV	Fast Flux E >10keV
TNIF:			
Current HEU	5.7×10^{11}	1.9×10^{11}	4.3×10^{10}
17-Assembly LEU	5.9×10^{11}	1.3×10^{11}	2.5×10^{10}
ENIF:			
Current HEU	2.0×10^{8}	2.6×10^{9}	5.1×10^{7}
17-Assembly LEU	2.3×10^{8}	3.0×10^{9}	5.0×10^{7}
18-Assembly LEU	1.9×10^{8}	2.7×10^{9}	4.6×10^{7}

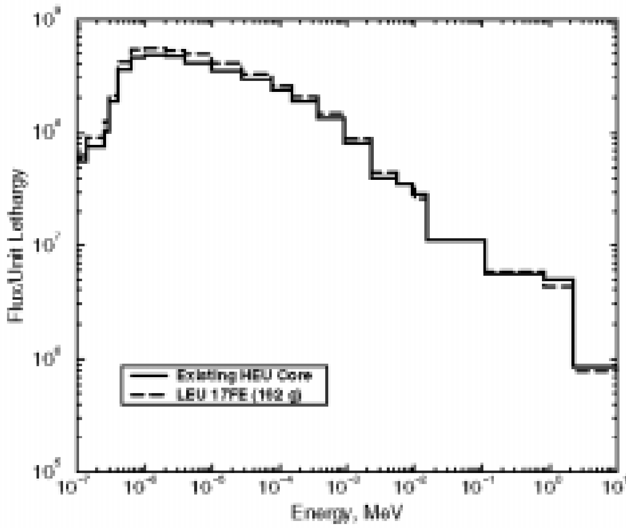

Figure 4. Neutron Spectra at ENIF Beam Port: Existing Reflector/Shutter.

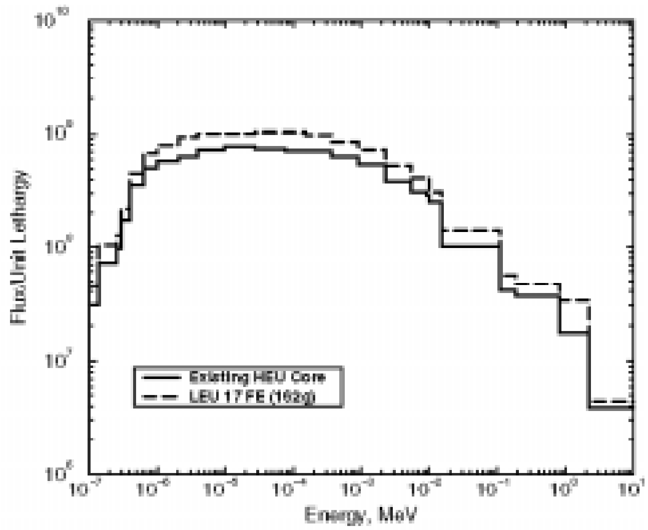

Figure 5. Neutron Spectra at ENIF Beam Port: BNL-Proposed Reflector/Shutter.

The results discussed above are for the existing configuration of the BMRR reflector/shutter that consists of Al, Al_2O_3 and Bi. In Ref. 8, BNL has proposed a modified reflector/shutter design using ^{235}U fission plates. With this modified shutter design, calculations were also performed to compare the ENIF performance for both the current HEU and 17-assembly LEU core configurations. The results are shown in Table 3 for the ENIF and indicate that the LEU core would have about 30% more flux in the important epithermal range than would the current HEU core. Figure 5 shows that the shape of the neutron spectrum in the epithermal range (0.4 eV to 10 keV) would be very similar in the HEU and LEU cores.

Table 3. ENIF Neutron Fluxes at End of Collimator:
BNL-Proposed Reflector/Shutter (n/cm^2-s)

Configuration	Thermal E <0.4eV	Epithermal 0.4eV< E <10keV	Fast Flux E >10keV
ENIF:			
Current HEU	1.5×10^8	6.2×10^9	4.1×10^7
17-Assembly LEU	2.0×10^8	8.2×10^9	5.4×10^7

Fuel Cycle Length Results

Approximately one fuel assembly is replaced per year in the current HEU core. Burnup calculations were performed for the first six years of operation for the 17-assembly LEU core and for the first four years for the 18-assembly LEU core. The results show that in both LEU core designs the fuel consumption would be about one assembly every two years.

CONCLUSIONS

The results of this study show that conversion of the BMRR from HEU fuel to LEU fuel is feasible as far as neutronic performance at the two key irradiation facilities is concerned. The proposed LEU U_3Si_2-Al fuel with a uranium density no greater than 2.5 g/cm^3 is fully-qualified for routine use. Due to the large distance and materials between the core and the epithermal irradiation facility (ENIF), the enrichment of the fuel in the core has almost no effect on the magnitude and spectrum of the neutron flux at the ENIF. Two LEU cores were identified that essentially match the neutron flux performance of the HEU core at both the thermal (TNIF) and the epithermal neutron irradiation facilities. The initial LEU core would start with either 17 MTR-type fuel assemblies and 162 g ^{235}U per assembly or with 18 assemblies and fuel with 154 g ^{235}U per assembly. Both LEU cores would use an average of one fuel assembly every two years. The current HEU core uses approximately one assembly per year. A new reflector/shutter design proposed by BNL would have about 30% more flux in the epithermal range for boron neutron capture therapy (BNCT) using the 17-assembly LEU core rather than the current HEU core. Reactor safety and thermal hydraulic analyses need to be performed in order to complete the fuel conversion study.

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