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AN EQUILIBRIUM CORE ASSESSMENT OF THE NIST REACTOR WITH GA LEU FUEL ELEMENTS

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ABSTRACT

As part of the Reduced Enrichment for Research and Test Reactors program (RERTR) under the Global Threat Reduction Initiative (GTRI), the National Bureau of Standards Reactor (NBSR) at the National Institute of Standards and Technology (NIST) is in the process of converting from highly enriched uranium (HEU) to low enriched uranium (LEU) fuel. The fuel conversion in high performance research reactors such as the NBSR presents a challenge in meeting the flux requirements (2.5×10^{14} n/cm^2-s for the NBSR) and other neutronics performance characteristics without significant modification to the external geometrical configuration of the fuel element. One promising fuel candidate in meeting this challenge is the General Atomics (GA) UZrH LEU fuel. GA LEU fuel was initially developed in the 1980s to provide fuels for fuel conversion in high power regimes such as are found in high density research and test reactors. This study performs a physics assessment of an equilibrium core of the NBSR using the GA LEU fuel by examining the neutronics performance characteristics of the core employing MCNP An equilibrium core configuration employing the designated fuel management scheme and fuel cycle length is generated, and the effectiveness of the fuel at four key burnup states of the fuel cycle lifetime is examined. The neutronics performance of the equilibrium core is assessed by examining the fast and thermal neutron flux level as well as power distribution in the core. Reactor safety related parameters such as kinetics parameters and power peaking factors are also calculated and assessed. The neutronics performance is compared against the current NBSR performance that is achieved with HEU fuels. The results of this study demonstrate the viability and provide constructive recommendations for the use of the GA fuels in the NBSR.

1. INTRODUCTION

The Global Threat Initiative (GTRI) initiated in the late 1950s was enacted by the US in an effort to reduce the amount of High Enriched Uranium (HEU) around the world as a nonproliferation anti-nuclear terrorism initiative. Under this umbrella, **t**he Reduced Enrichment for Research and Test Reactors Program (RERTR) program began in 1978 and still is in effect today. The US goal in this program is to reduce the amount of weapons grade uranium being used in research and test reactors. This is primarily for nonproliferation purposes. The conversion from high enriched uranium (HEU) to low enriched uranium (LEU) will significantly reduce the risk of highly enriched uranium being stolen and used for the production of weapons of mass destruction.

The RERTR has three main goals; the production of the medical isotope Molybdenum-99 with LEU, the development of new LEU fuels that can fit the needs of research and test reactors, and lastly the design and safety analysis of these fuels. The program requires the fuel contain below 20 wt.% U-235 in uranium. Some of these reactors are not easy to convert due to their high requirements for operation. These are also known as high performance research reactors (HPRRs). HPRRs usually operate at a large flux level and have abnormal geometries. One of these reactors is the National Bureau of Standards Reactor (NBSR) located at the National Institute of Standards and Technology (NIST) in Gaithersburg, Maryland USA. The NBSR is a 20 MW thermal research reactor that is heavy water cooled, and currently operating at an average thermal flux density of 2.5×10^{14} n/cm^2-s. The fuel has been shown to be extremely efficient and on all fronts powerful. This high flux requirement

*__Disclaimer:__ The work conducted in this paper is completely academic exercise performed solely at the VCU side. Neither NIST nor DOE has asked for, endorsed or supported this research. Currently NIST plans to use U-10Mo monolithic fuel and has already submitted a PSAR to NRC for conversion.

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is required for the continued research and operation of the campus.

Several LEU fuels have been conceptually studied in the NBSR. U-10Mo monolithic and U-7Mo/Al dispersion fuel have both been shown to be safe, efficient, and reliable fuels suffering only a 10% loss in flux capabilities over the course of their use [1]. These fuels unfortunately are not commercially available, as they are still years away from being manufactured at a production level. This makes these fuels not viable for a quick and easy conversion of the NBSR. To fix this, we can use fuels that have already been well developed as alternatives. There are a variety of well tested fuels currently qualified under the RERTR, but General Atomics (GA) Training Research Isotopes General Atomics (TRIGA) fuel is the most attractive candidate for the NBSR.

The TRIGA fuel was developed in the early 1950s, and qualified in 1956 for use under the RERTR. The fuel was created by Edward Teller to be a fuel "a high school student could play with without fear of getting hurt". Well known for its prompt negative temperature coefficients, the TRIGA fuel was created to be inherently safe. The fuel was originally designed to be used in research and test reactors, making it an even more attractive fuel for the NBSR conversion. The TRIGA fuel is also well known for its long core lifetime. For a 250kW TRIGA reactor operating 200 days a year, 8 hours per day, the U-235 consumption is approximately 20 grams per year. Lastly, the TRIGA fuel is commercially available. It can be bought relatively easy on the current fuel market in comparison to the other options.

In order to convert the NBSR using the TRIGA LEU fuel without making significant structural changes to the fuel element and the reactor, one would have three requirements to meet owing to the NBSR's specialized needs. The first is that the current core fuel holdings must be maintained. Any construction or core structural changes are out of the question given the cost and time. Secondly, the neutron flux level must be maintained in order to sustain the irradiation experiments performed at NIST. Any loss of integrity in the current flux (2.5×10^{14} n/cm^2-s) can result in loss of experimental capabilities. Finally, the fuel must satisfy the necessary and relevant safety requirements. Neutron lifetimes, peaking power factors, and other safety parameters must be examined to ensure the safe operation of the NBSR.

In this paper, a followup of a prior feasibility and scoping study [2], we continue the analysis of a steady state solution for the long term behavior of the TRIGA fuel in the NBSR based on the most efficient fuel configuration determined in the prior study. With this long-term behavior, we examined the key neutronics performance characteristics of the core. Peaking power factors as well as the maximum fast and thermal flux were determined globally in the core. Integral safety parameters such as control worth and neutron lifetime were also calculated. The Monte Carlo code MCNP6 [10] was used to generate an equilibrium core of the NBSR using the new fuel as well as to predict the relevant flux and power distribution over the entire core. MATLAB and Python programs were used for associated data management. These results are compared to the NBSR's current testing capability under its HEU fuel schema to better give an understanding of the conversion.

2. OVERVIEW OF THE NBSR

The NBSR, a heavy water moderated 20 MWth research reactor, first went critical in December of 1967. The NBSR is a large neutron research facility, hosting more than 2000 guest researchers annually. It is outfitted with 28 fine-tuned neutron instruments. One interesting and unique feature of the NBSR is its "loose" configuration. The heavy water moderation allows the fuels to be farther spaced than that with other HPRRs while still maintaining criticality. Another particularly unique feature of the NBSR is its cold neutron source. The liquid hydrogen moderator slows neutrons to below 5 meV. These cold neutrons can be transferred and utilized by multiple neutron scattering instruments tens of meters away from the reactor in the experimental hall. Among the 28 instruments, 21 of them utilize the cold neutron source. The criticality of the NBSR is controlled by 4 cadmium shim arms that are inserted in the core horizontally. The angle of the arms can be adjusted to control the criticality of the core during operation. A schematic overview of the main components in the NBSR is illustrated in Fig. 1.

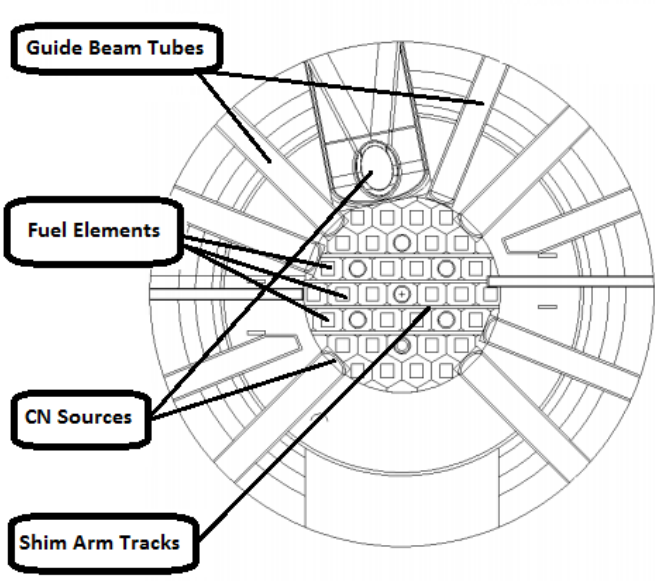

Fig. 1. A schematic top view of the NBSR.

The NBSR core contains 30 fuel elements that are arranged in three concentric circles (see Fig. 1). 16 fuel elements reside in the core for eight cycles and 14 reside in the core for seven cycles. The fuel cycle length is about 38 days. At the end of each cycle four fuel elements are removed, the other 26 are moved to new locations, and four fresh fuel elements are inserted [1].

3. NBSR CURRENT FUEL AND GA FUEL ELEMENT

The current NBSR fuel element has an external size of 8.55 cm long, 7.6 cm wide and approximately 175 cm in height. The material test reactor (MTR) type curved plate in the fuel element is made of U_3O_8 sintered with aluminum powder and clad in aluminum. The fuel plate is 93 wt.% enriched (HEU) and has a total volume of 296 cm^3 per element. The fuel has 350 grams of uranium U-235 per element. There are 17 plates and two empty plates per fuel element. The fuel element is separated into an

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upper and lower section with a gap of 15.2 cm in the middle region. Fig. 2 illustrate the side and top view of the current NBSR plate-type fuel element.

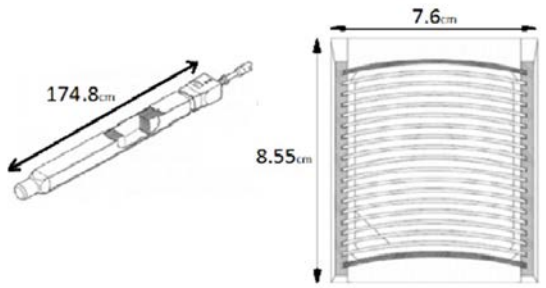

Fig. 2. The current NBSR fuel element.

The configuration the GA TRIGA fuel element for the NBSR was determined in the previous feasibility study [2]. The optimal design for the rod-type TRIGA fuel element has a 5×5 layout of the fuel rods, which are cladded with stainless steel 304 and contain 45% uranium by weight fuel. The enrichment of U-235 is about 19.7 wt.% (LEU). The rods are 33 cm long and have a diameter of 0.7 cm with a pitch of 1 cm. The high surface area to volume ratio of the rods reduces self-shielding, and produces a relatively high reactivity. The stainless steel 304 cladding is strong enough to prevent blistering and corrosion. Fig. 3 compares the MCNP model of the rod-type TRIGA fuel element to the simplified NBSR plate-type fuel element model.

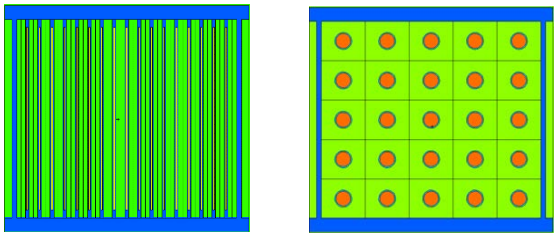

Fig. 3. The fuel element with plate-type HEU fuel (left) and rod-type TRIGA fuel (right).

Table 1. Fuel Compositions in Equilibrium Core per Element

Fuel type	HEU	TRIGA
^{235}U (g)	350.00	483.88
^{238}U (g)	26.00	1972.38
O (g)	68.00	0.00
Al (g)	625.00	0.00
Zr (g)	0.00	2950.35
H (g)	0.00	51.74
Total mass (g)	1069.00	5458.36
Fuel Density (g/cc)	3.16	11.71
Fuel Volume (cc)	296	466.52
Fuel Height (cm)	68.4	80.0

Table 1 summarizes the compositions for the HEU and TRIGA fuel. Note the difference in U-235 mass, which may seem contrary to the goal of a cheap conversion. This difference is however normal [3], and important for maintaining core criticality.

4. AN EQUILIBRIUM CORE GENERATION

One primary task for any reactor analysis is to generate an equilibrium core configuration. The equilibrium core illustrates the fuels effectiveness over the entire lifetime in the core, subject to being partially burned and shuffled. Determining an equilibrium core is necessary for any practical core study, as an analysis of a core with all fresh fuels gives an incomplete view of the fuel effectiveness. The TRIGA fuel element configuration determined to be most effective in the previous study [2] will serve as the basis for our equilibrium core search. In this study, the TRIGA fuel follows the same fuel shuffling scheme as the NBSR HEU fuel, as it has been well tested and validated under the NBSR operation conditions.

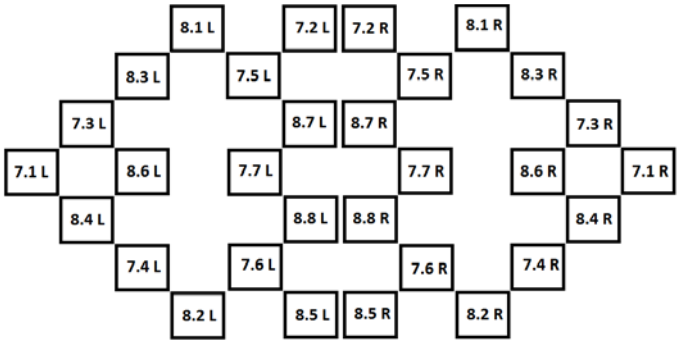

Fig. 4. The NBSR fuel shuffling scheme.

As described earlier, the NBSR contains 30 fuel elements. Under the current shuffling scheme, 16 of the elements will be burned for eight cycles, and 14 elements burned for seven cycles. The fuel management scheme for the NBSR is shown in Fig. 4, with all thirty elements labeled. Each fuel position is identified with two numbers and a letter, the first number denoting how many cycles the fuel will be in the core, with the second designating the current cycle for the fuel cell. The letter designates the side of the reactor, as the shuffling is symmetric. This means that at the beginning of a cycle, the 8.1 and 7.1 fuel elements are the fresh fuel newly loaded, and the 8.8 and 7.7 fuel elements have been burned for their full lifetime. At the end of the cycle the 8.8 L and 7.7 L fuel elements will be discharged, the 8.7 L and 7.6 L fuel elements will be moved into the 8.8 L and 7.7 L locations respectively, and so on with the 8.1 L and 7.1 L locations being filled with fresh fuels. This shuffling scheme indicates 4 fresh elements will be loaded at the beginning of each cycle and 4 maximum burned elements will be removed at the end of each cycle. Currently the NBSR operates on a 38.8 day cycle.

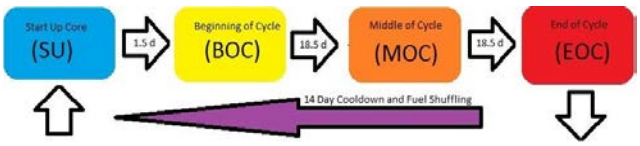

Fig. 5. A multi-step equilibrium core generation diagram.

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To describe the full fuel cycle, four representative cycle burnups are selected: start-up (SU), beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC). The burnup time length of each state is indicated in Fig. 5, which briefly illustrates the iterative procedure we implemented to generate the equilibrium core status. The burnup length of each state for the TRIGA fuel matches the current HEU fuel cycle. At each state, a processing script was developed in Python that extracts the burned fuel inventories and uses these fuel inventories to create fuel materials for the next burnup state. In between these states, the control shim positions are adjusted to maintain the criticality of the core. However, differences of control shim positions at each state were found between the TRIGA and HEU fuel. The control shim positions of these two fuels are summarized in Table 2.

Table 2. The Control Shim Positions of HEU and LEU Core at Equilibrium States.

	HEU[a]	LEU
SU	19.7[b]	23.0
BOC	14.6	14.0
MOC	9.20	7.00
EOC	0.00	0.00

[a]The results for HEU were obtained from Ref. [3].
[b]The position is shown in the unit of angle of degrees.

The fuel inventories at each burnup state of the equilibrium core were calculated using the MCNP6 BURN card. A fuel management code in Python was developed to automate this process. As shown in Fig. 6, the equilibrium core search process begins with the automated creation of an MCNP input deck for one state. This input deck is then run remotely. After its completion, the fuel is extracted from the output and a new input deck for the next burnup state is generated for the next step in the cycle. This process is repeated until an equilibrium status for each burnup state is achieved.

During this research, an error was identified in the MCNP BURN card. When MCNP finds an isotope that cannot be recognized by its burnup module, this isotope is ignored in the burnup cycle. This results in a nonphysical loss of mass in the core. To compensate for this deficit, the fuel management code determines this loss of mass, and then adds a certain amount of Bismuth to compensate for the loss and conserve the total mass of the fuel. Bismuth was chosen to be an optimal element for this purpose because it has little effect on the reactivity [2, 4].

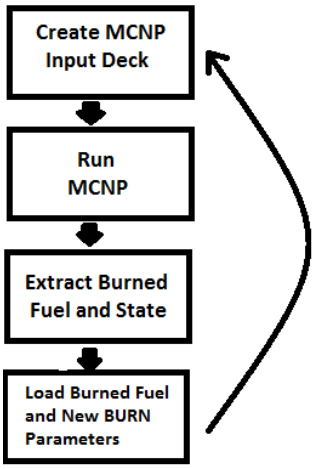

Fig. 6. Python flow diagram during the equilibrium core search.

To efficiently count the spatial effects in the burnup, the fuel elements were split into 180 different inventories to properly assay the fuel compositions. As shown in Fig. 7, the fuel element was divided axially into 6 separate sections (3 on top and 3 on bottom). During the equilibrium core search process, 100 cycles (10 skipped cycles) and 10,000 particles per cycle were used in the MCNP criticality calculation, which minimizes the standard deviation of k_{eff} below 0.001. These jobs were executed on a personal computer system using 8 processor cores.

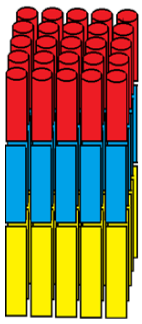

Fig. 7. Fuel is divided for the top and bottom section of the element. Different color indicates unique burnup effect.

5. RESULTS

The equilibrium core search process took approximately 500 hours on the 8-core processor, using the parameters specified earlier in the process. Fig. 8 shows the k_{eff} results during the iterative search procedure for the four burnup states. Since the search starts off with a fuel configuration with all fresh fuels, it is clearly seen that the k_{eff} curves for all states are gradually decreasing until they all reach a plateau at the 8th or 9th cycle. However, all four states were able to keep criticality above one until reaching equilibrium status, which indicates the fuel scheme used in the search will be able to sustain the NBSR long-term. Please noted the variances of the k_{eff} values are actually

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shown in Fig. 8, but they are too tiny (~0.001) to be visible in the plots.

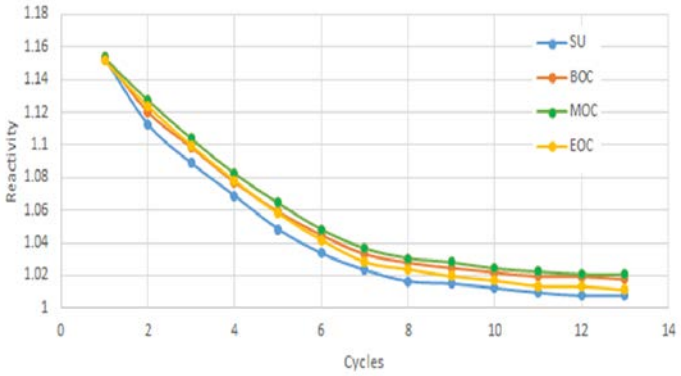

Fig. 8. The k_{eff} changes along the iteration cycle number for all states in the equilibrium core search procedure.

5.1 Actinide Consumption and Buildup

The burnup and buildup of some key actinides at the end of equilibrium cycle (that is, cycle 12 results) are shown in Table 3.

Table 3. Burnup and Buildup of Key Actinides at Cycle 12.

Fuel	^{235}U (%)	^{235}U (g)	^{238}U (g)	^{239}Pu (g)
TRIGA	59.50	207.10	21.00	8.38
U_3O_8	75.00	N/A	N/A	N/A

The burnup percent was calculated by taking the amount of uranium in the fresh fuel elements (fuel 8.1 and 7.1 in Fig. 4) and subtracting the amount of uranium in discharged fuel elements (fuel 8.8 and 7.7 in Fig. 4). The difference was then divided by the original uranium inventory to determine the percent of uranium burned. One interesting point to note from Table 3 is the comparison of the percent burnup between the HEU and TRIGA fuel. The TRIGA fuel does not burn as efficient as the HEU, but the ~60% burnup rate exceeds most typical LEU fuels. The closest competitor would be the U-10Mo fuel that achieved a 40% burnup in the new core, nearly 20% less burnup than the TRIGA fuel [3]. The TRIGA fuel produces on average 8.38 grams of ^{239}Pu as shown in Table 3. This was determined by the same method as that used for the ^{235}U. Having a larger ^{239}Pu production and ^{238}U depletion than that in HEU fuel is very typical of LEU fuels as seen in earlier studies [3, 4], most likely due to the higher U^{238} content.

MCNP does not normalize the flux it generates, so in order to calculate the physical flux, the generated flux must be normalized. By using the NBSR's power, and assuming 200 MeV per fission and 2.44 neutrons per fission, we have determined a fission rate 1.523×10^{18} neutrons/s, which can be used as a normalization factor for the flux tallies in MCNP [7] By incorporating the material inventories generated from the multicycle equilibrium core, key neutronics characteristics can be calculated using MCNP. Fast and thermal flux, power, neutron lifetimes, and shim worth are all calculated using these inventories.

5.2 Power Distribution

The reactors power distribution in the core is a safety concern for any thorough reactor analysis. By examining the cores total power distribution, hot spots can be identified to determine the integrity of the fuel under operational conditions. In order to examine the axial power distribution of the power, the fuel rod was divided into 12 axial sections in each half portion in the MCNP model, which means that with 12 axial sections per rod, 5 × 5 × 2 = 50 rods per fuel element, and 30 fuel elements in the core, thus a total of 18,000 tally cells were needed in the power distribution calculation. Each cell has a volume of 1.154 cm^3. Instead of using the mesh tally approach typically used in MCNP to calculate power distributions, the power in this study was calculated using the Table 128 method [5]. In the standard output file of MCNP, Table 128 is a collection of the neutrons that enter, collide, and fission in each cell. The fission number shown in the table is proportional to the fission rate in the cell; thus these numbers can be used to infer the power information. In this calculation, 36,000 particle histories per cycle and 2000 cycles were used in the calculations to make the standard errors of calculated results all less than 1%. The equilibrium core obtained in the 13th iterative search cycle was used to evaluate the power distributions and the material compositions at each of the four states. Python was used to extract the contents of the results, and to normalize and calculate the power. Python was also used to sort and graph the power distribution. The power factors (normalized to the core averaged power) of the fuel element with hottest power at EOC are shown in Fig.9. The global power peaking factors (PPF) at each burnup state and their corresponding locations are summarized in Table 4.

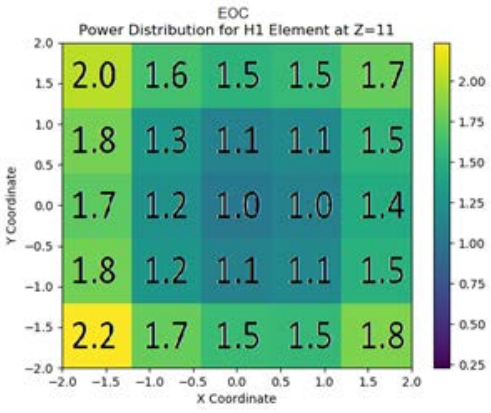

Fig. 9. Power factors at the hottest element in the EOC state.

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Table 4. PPF and Occurring Locations at Each State.

Core State	SU	BOC	MOC	EOC
PPF	2.80	2.37	2.26	2.24
Z-position	-11	11	11	11
Fuel Rod (*x*, *y*)	(-2, 2)	(-2, -2)	(-2, -2)	(-2, -2)
Fuel Element	7.2L	7.2R	7.2R	7.2R

As seen in Table 4, the hottest PPF (2.80) occurred at SU, which is expected because the SU state contains the fuel with the highest k-infinity during the cycle. The location notation used in Table 4 is described in Fig. 10 and in the following paragraph.

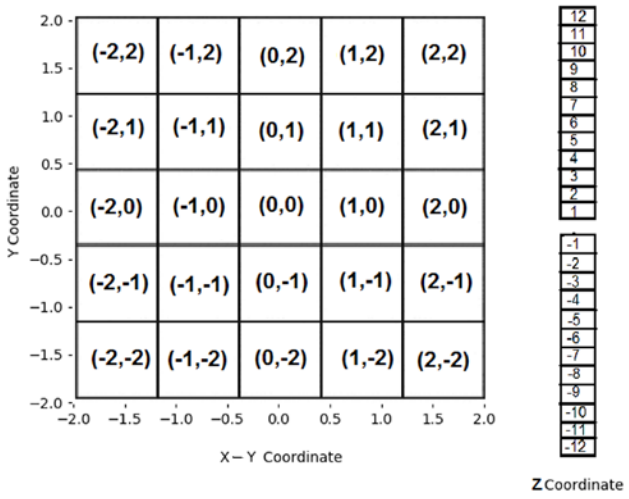

Fig. 10. Labeling system for a fuel element for power calculations.

The rod position in a fuel element is presented in an X-Y coordinate plane with x and y numbers, with the center rod at the (0, 0) position. The axial position of the rod is presented in a Z coordinate system in reference to the mid-core elevation, in which a negative number represents the bottom half of the core and positive the top half. As may be seen in Table 4, the hotspot locations for the four states are similar. The 7.2L and 7.2R fuel element positions (see Fig. 4 for the fuel element labeling) are positioned at the top most north section of the core, and the rod position for the BOC, MOC, and EOC states are all identical. This concentration of power and location of the hotspots in the most northern region of the core is likely due to the location of the UNIT-2 cold neutron source located directly above (see Fig. 1), that is, this is explained by the high moderation occurring near the cold source. Unlike any other LEU fuel, the TRIGA fuel power appears to be concentrated to the top edges of the reactor. Generally, in LEU fuels, such as U-10Mo and U-7Mo/Al, the power tends to be concentrated towards the center of the reactor [1, 4]. This is a problem because the central based power decreases the amount of neutrons that enter the beam tubes [2, 4]. Having the power concentrated at the edges of the reactor creates less of a hotspot and a safer reactor, as well as allowing the maximum amount of neutrons to enter the beam tubes.

Fig. 11 shows the axial distribution of the power in the hottest rod for each of the states. The PPF for each position with respect to height is shown, with any above 2.0 highlighted. From a very clearly hotter SU state, the power can be seen to shift from the bottom of the core to the top. This is likely due to the position of the control shims. As the control shims are retracted from the core, they pass from the bottom of the core to the top. This shift of moderation could be the cause of the power concentration changing from the bottom to the top of the core over the course of the cycle.

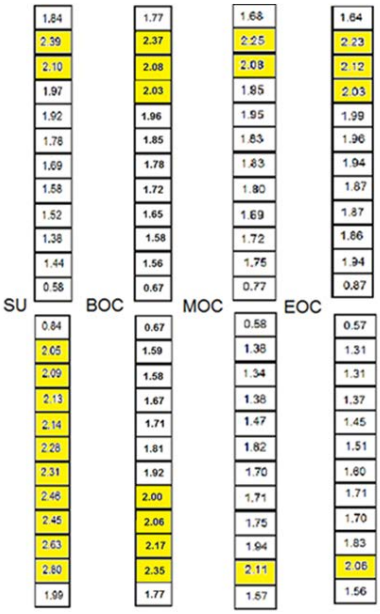

Fig. 11. Axial view of power distribution for all 4 states.

5.3 Flux Distribution

Flux is arguably the most important neutronics parameter in the evaluation of the TRIGA fuel in the NBSR core, because flux is the key figure of merit in determination of the experimental capabilities of neutron source center at the NIST campus. The FMESH card in MCNP was used to calculate the flux. The FMESH card allows the user to break the 3D space into small geometric tally bins, and tally the flux for each of those bins. A right cylindrical geometry setting was adopted for the flux calculation as it most closely approximates the reactor shape. The reactor was split into 50 radial, 80 height, and 50 angular segments for a total of 200,000 bins. The radius was from 0 to 56 cm, the angular from 0 to 2π, and the height from -40.7 to 40.7 cm, respectively. The axial mid-core elevation of the reactor is at Z = 0. The output results were processed by MATLAB and plotted on a mesh plot to graphically show the flux distributions. For each calculation, 36,000 particle histories and 2000 cycles were used to reduce the error on the flux to an average of 1.3%. The results of the flux calculations are shown in Fig. 12 and Fig. 13, demonstrating the 2D and 1D flux distribution in the core, respectively.

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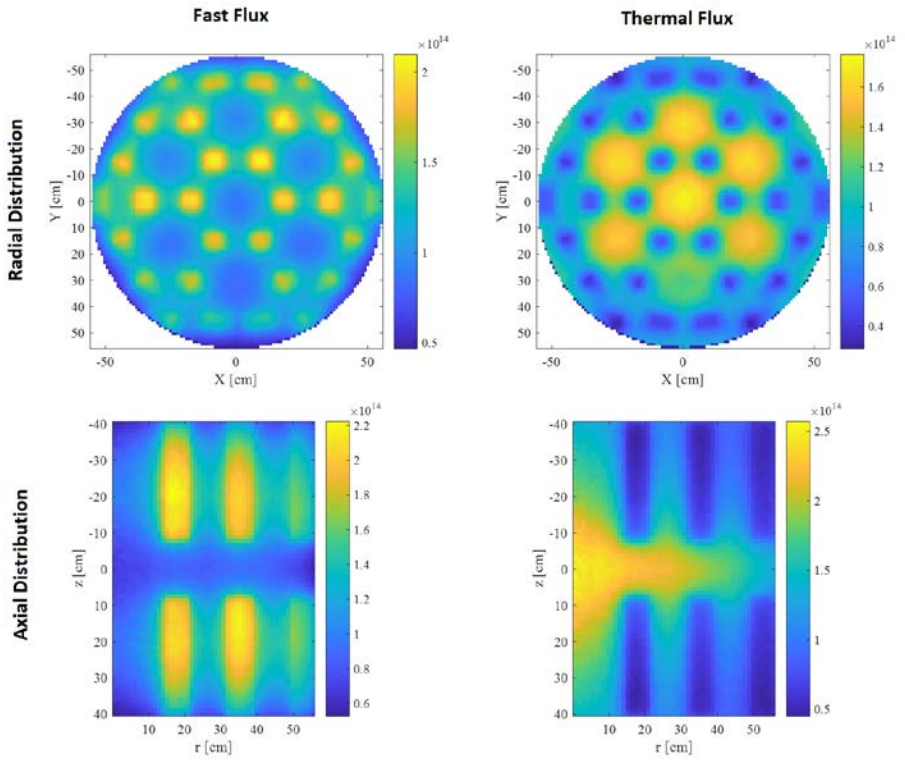

Fig. 12. Two-dimension graphical view of the flux distribution on a heat map at EOC.

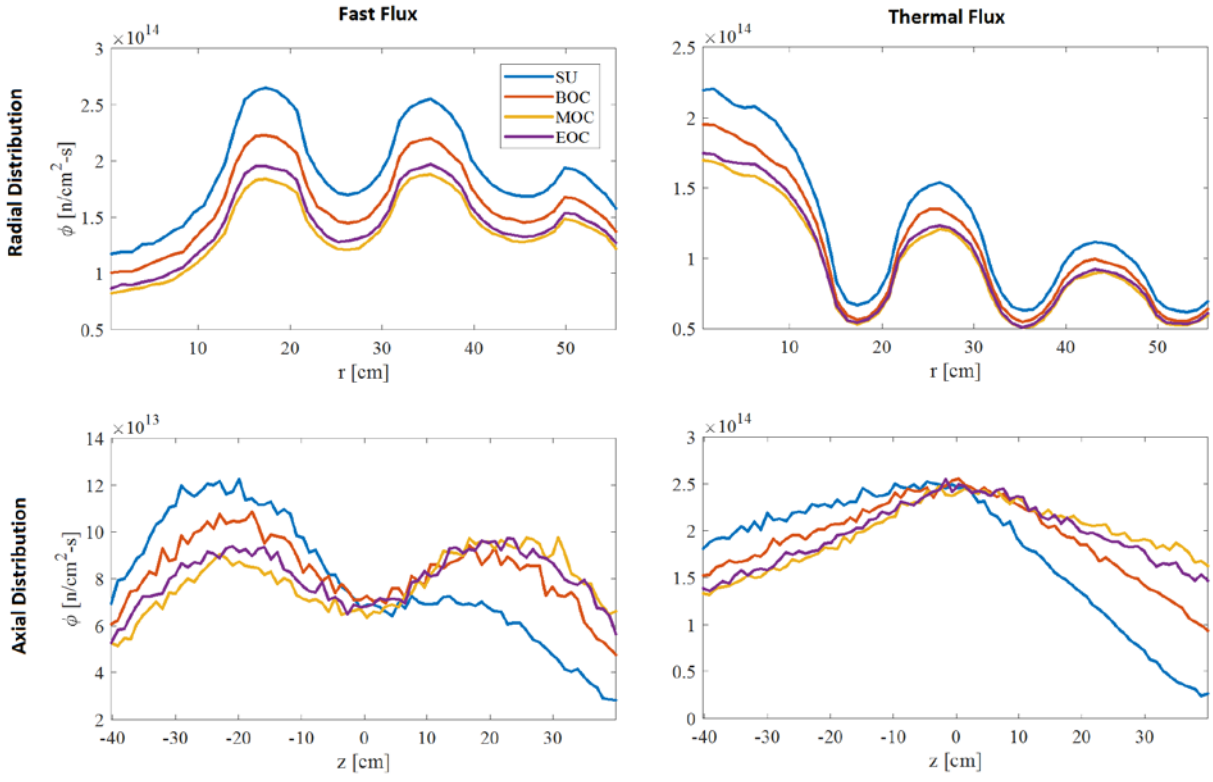

Fig. 13. One-dimension flux distribution for all four states.

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In Fig. 12, the 2D radial flux distribution is shown in the first row, where the fast and thermal flux are shown at Z = -20 cm, or roughly at the middle plane of the fuel region. The 2D axial flux in shown in the second row is plotted for entire height of the core and with the azimuth angle $\theta = \pi$. As fission is the primary source of fast flux, this can be the explanation for the coloring of the fast flux plots. In both the radial and axial plots, the fuel elements are highlighted showing where the majority of the fast flux is emitted. Since thermal flux is primarily from the moderating elements in the reactor, the above plots of the thermal flux in turn highlight the regions in the reactor where heavy water is concentrated. Radially the largest amount of heavy water is situated in the center of the reactor, and axially it can be seen between the rods and in the center. The maximum thermal flux at the EOC state can reach about 2.3 to 2.4×10^{14} n/cm^2-s in the areas of interest, while for other states the thermal flux is slightly higher. In the context of the NBSR this is a decrease of integrity by approximately 5% from the current thermal flux operating level of 2.5×10^{14} n/cm^2-s. The thermal flux concentration at the center of the core is not ideal, but very typical of LEU fuels. In previous studies, U-10Mo and U-7Mo/Al performed similarly with flux peaking near the center, making it more difficult to harvest the neutrons [4]. In Fig. 13, the 1D flux distribution for all four states are plotted for clarity and for providing a more quantitative look at the results. All of the 1D plots are with maximum thermal fluxes. The radial fast flux peaks at radial dimensions of 17.5, 35.0, and 50 cm, indicating the locations of fuel elements in the core. The radial thermal flux peaks at the center of the core, with a slow drop off in all four states, again corresponding to the location of the largest volume of heavy water in the core. The axial fast flux peaks at about Z = -20 cm and Z = 20 cm for all four states, most likely because these are the locations of the center of the fuel rods. The minima of the axial fast and the maxima of the axial thermal flux coincidently occurs at Z = 0 cm for all four states. This is likely because of the gap between the fuel rods that is filled with heavy water.

5.4 Neutron Lifetime

The prompt neutron lifetime is defined as the average time elapsed between the generation of fission neutrons and when they are absorbed [8]. The neutron lifetime is particularly important when examining a fuel because it gives insight into the neutron multiplication process. A higher neutron lifetime means there is more time between emission and absorption giving the reactor operator more control over the reactor. Similarly, the lower the lifetime the less effective control the operator has over the reactor. The MCNP KOPTS card was used to determine the lifetimes using the equilibrium material makeup for all four states. The results are summarized in Table 5.

Table 5. Neutron Lifetime (microseconds).

Fuel type	SU	BOC	MOC	EOC
TRIGA	519 ± 15	476 ± 11	494 ± 12	538 ± 13
U_3O_8	698	NA	NA	731

As shown in the table, the neutron lifetime for the TRIGA LEU fuel ranges from 550 to 650 microseconds, while a typical value of neutron lifetime for the heavy water reactor with HEU is ~700 microseconds.

5.5 Control Shim Worth

Another parameter of significance for reactor operation and safety management is the control shim worth. The variation of shim worth with cycle burnup indicates what percent of the shim was depleted over the course of the cycle. The shim worth results for the TRIGA fuel is summarized in Table 6.

Table 6. Shim Arm Worth.

Fuel	SU	BOC	MOC	EOC
TRIGA	15.9	16.5	16.8	17.1
U_3O_8	24.9	NA	NA	27.2

As shown in the table, the control worth for the TRIGA fuel is lower than the HEU. This is as expected, as LEU fuel historically has been shown to have lower shim worth [1]. Although the shim worth is lower, it is still within the acceptable margin of use in the NBSR.

6. CONCLUSIONS

This study examines steady-state neutronics performance characteristics of the hypothetically converted NBSR using the commercially available TRIGA LEU fuels. An equilibrium core based on the NBSR regular fuel management scheme was generated through an iterative search procedure with the use of the MCNP BURN calculation and Python processing. The fuel inventory generated from this equilibrium core search was utilized for the neutronics study. The burnup efficiency, power distribution, flux information, neutron lifetime, and control shim worth were calculated. Hotspots in the core were identified based on the power calculation results, and examined following standard safety criteria. The power appeared to transition from the bottom to the top of the core over the cycle's course, and is concentrated near the BT-2 cold source. The flux calculations showed the thermal flux was degraded about 5 % of the NBSR's current operating flux in the center of the core, with thermal flux heavily concentrated at the center of the reactor. This is due to the radial concentration of heavy water in the core center, but also axially from the axial gap in the fuel. The neutron lifetimes generated were shown to be of merit for other LEU fuels. The control shim worth for the TRIGA fuel were determined to be acceptable in the context of the NBSR's needs. For future work, we will consider an analysis of the cold neutron performance under this hypothetical conversion.

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REFERENCES

[1]. Hanson A., and Diamond D., 2014. "Calculation of Design Parameters for an Equilibrium LEU core in the NBSR using a U7Mo Dispersion Fuel," Laboratory Report, Brookhaven National Laboratory.

[2]. Britton K., and Wu Z., 2018. "A neutronics feasibility study of the TRIGA LEU fuel in the 20mwt NIST research reactor", *Proceedings of the 26th International Conference on Nuclear Engineering*, London, England, July 22–26.

[3]. Turkoglu D., Wu Z., Williams R., and Newton T., 2018. "Comparison of neutronics performance characteristics of the proposed NIST reactor with different LEU fuels", *PHYSOR 2018 – Reactor Physics Paving the Way Towards More Efficient Systems*, Cancun, Mexico, April 22-26

[4]. Hanson A., and Diamond D., 2011. "Calculation of Design Parameters for an equilibrium LEU core in the NBSR," Laboratory report, Brookhaven National Laboratory, September 30.

[5]. Wu Z., and Williams R., 2015. "Reactor Power Distribution Calculation in Research Reactors Using MCNP," *Transactions of the American Nuclear Society*, **113**, 1171-1174.

[6]. Hanson A., and Diamond D., 2010. "Calculation of inventories, power distributions and neutronic parameters for the NBSR using MCNPX," TRTR/IGORR Joint meeting, Knoxville, Tennessee, Sept 19-23.

[7]. Wu Z., Williams R., Rowe J., Newton T., and O'Kelly S., 2017. "Neutronics and Safety Studies on a Research Reactor Concept for an Advanced Neutron Source," *Nuclear Technology*, **199**(1), 67-82.

[8]. Hanson A., Ludewig H., and Diamond D., 2005. "Calculation of the prompt neutron lifetime in the NBSR," Brookhaven National Laboratory, September 2005.

[9]. Hanson A., and Diamond D., 2011. "Planning the HEU to LEU transition for the NBSR", Laboratory report, Brookhaven National Laboratory.

[10]. LANL, 2017. "MCNP- A General Monte Carlo N-Particle Transport Code, Version 6.1.1," LA-UR-03-1987, Los Alamos National Laboratory, August 1.

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