Nuclear Engineering and Design 360 (2020) 110469

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Reactor physics evaluation of the TRIGA LEU fuel in the 20 MW NIST research reactor☆

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ARTICLE INFO

Keywords:
Research reactor
Low enriched uranium
TRIGA
Monte Carlo

ABSTRACT

This paper performs a neutronics evaluation of the General Atomics $UZrH_x$ low enriched uranium fuel – TRIGA fuel – in the National Bureau of Standards Reactor (NBSR) at the National Institute of Standards and Technology (NIST). The objective of this study is to examine the accountability and sustainability of the TRIGA fuel on neutronics aspects when applying it to the NBSR conversion. A feasibility scoping study was previously undertaken with considerations on various fuel dimensions, fuel rod layout configurations, and structure material selections, identifying the best option of deploying the TRIGA fuel to NBSR. Continuing with these efforts, an equilibrium NBSR core using the identified fuel was generated, and a well-round physics assessment was carried out by examining key neutronics performance characteristics of the core. All calculations were completed with MCNP-6, a 3-D Monte Carlo neutron transport code. The same fuel management scheme and fuel cycle length as the existing NBSR was adopted in the equilibrium core generation adopts to retain performance consistencies. The effectiveness of the fuel was examined at four representative burnup states of the fuel cycle. Neutronics performances of the equilibrium core was characterized by 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 were also evaluated in the study. All results were compared against the current NBSR fueled with HEU for justifications. The findings in this research prove the viability of the TRIGA fuel for the NBSR conversion, and provide supporting data for future investigations on this subject.

1. Introduction

One of the primary security concerns in the nuclear reactor community is proliferation of nuclear material. Converting high enriched uranium (HEU) to low enriched uranium (LEU) is of the utmost importance in terms of material detection and accountability toward nuclear reactor safeguards. For this purpose, the United States launched the Reduced Enrichment for Research and Test Reactors (RERTR) program in order to decrease the amount of HEU being used specifically in research and test reactors. This program requires the fuel contains less than 20 wt% ^{235}U. The RERTR program focus on two main goals: 1) the production of the medical isotope molybdenum-99 with LEU, and 2) the development, design, and safety analysis of new LEU fuels to fit the needs of research and test reactors. This study carried out in this paper explores the latter goal of the two.

Conversion of high performance research reactors (HPRRs), such as the National Bureau of Standards Reactor (NBSR) located at the National Institute of Standards and Technology (NIST), poses a particular challenge due to their large flux level operation and abnormal geometries. The NBSR is a 20 MWth heavy water moderated research reactor fueled with a material test reactor (MTR) curved plate type HEU fuel. The fuel plate is consisted of a U_3O_8 dispersion fuel meat and aluminum claddings. The NBSR operates at an average thermal flux level of 2.0 × 10^{14}n/cm^2-s, required by scientific experiments undertaken at NIST. Several LEU fuels have been considered to convert the NBSR, including the U-10Mo monolithic and U-7Mo/Al dispersion fuels with 10 and 7 wt% molybdenum respectively. These two fuels have been shown to be safe, efficient, and reliable fuels, whereas 10% loss in flux capabilities has been observed over the course of their use (Hanson and Diamond, 2011, 2014). Furthermore, these fuels are not commercially available yet, and they are envisioned still years away from being manufactured at a production level (Wu, 2017). These fuels are not qualified under the RERTR either, meaning they will require certification before being able to be utilized in reality.

☆ Disclaimer: The work conducted in this paper is completely academic exercises 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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https://doi.org/10.1016/j.nucengdes.2019.110469
Received 18 August 2019; Received in revised form 2 December 2019; Accepted 5 December 2019
0029-5493/ Published by Elsevier B.V.

An alternative solution to this problem could be to leverage an already fully developed fuel on the market that could satisfy the safety requirements of RERTR and LEU conversion as well as the NBSR's high performance needs. One of the most attractive fuel candidates on these regards could be the Training Research Isotopes General Atomics (TRIGA) fuel, which was developed by General Atomics (GA) in the early 1950s and qualified under the RERTR in 1986. TRIGA fuel was composited by $UZrH_x$ material and designed with inherently safe features in research and test reactors. TRIGA fuel is well known for its prompt negative temperature coefficient and its long core lifetime. For a 250 kW TRIGA reactor operating 200 days a year, eight hours per day, and the ^{235}U consumption is approximately 20 g per year in a typical TRIGA MK-II reactor. Therefore, TRIGA fuel can be bought relatively easily on the current fuel market in comparison to the other options, and becomes a potential LEU fuel for the NBSR conversion.

The TRIGA LEU fuel must meet three requirements in order to be considered as a viable fuel for conversion of NBSR. First, the current core configuration must be maintained; major structural changes to the fuel element or reactor will cost significant time and money. Thus changes will be restricted only to the inside of fuel elements in order to maintain the external integrity of the NBSR. TRIGA fuel is a cylindrical rod with stainless cladding while the current NBSR fuel is a plate cladded with alumina. Thus one question would be raised whether the TRIGA fuel can meet the NBSR's LEU conversion requirements without any alteration to the structure of the fuel element and the reactor. Secondly, the neutron flux level must be retained in order to sustain the irradiation and other experimental capabilities at NIST. The neutron flux from the NBSR cannot vary greatly from its current operating level (2.5×10^{14} n/cm^2-s) as that may actively affect the testing capabilities of the site. Finally, the fuel must satisfy the necessary and relevant safety requirements. Neutron lifetimes, power peaking factors, and other safety parameters must be examined to ensure the safe operation of the NBSR.

To meet these requirements, a preliminary scoping study were previously undertaken on the neutronics feasibility of TRIGA fuel in NBSR conversion without any alteration to the structure of the fuel element and the reactor (Britton and Wu, 2018). This study determines the optimal composition, configuration, and cladding of fuel. Particularly, the effects on the reactivity variations of these three factors in the fuel have been explored using the Monte Carlo N-particle Transport MCNP-6 (LANL, 2017). Furthermore, the effects of fuel rod configuration and cladding on the reactivity have been examined in order to obtain a wide insight of the fuel and determine its effectiveness for use in the NBSR. Based on the outcome of the feasibility study, the feasibility study was continued with a well-round neutronics evaluation of the core design using the most efficient fuel configuration that was determined at the feasibility study stage. An equilibrium core based on the current NBSR fuel management scheme and the new fuel configuration was generated and intensively analyzed. Power peaking factors as well as the maximum fast and thermal flux at various burnup state are calculated to assess the reactor performance characteristics. These results are compared to the NBSR's current testing capability under its HEU fuel to confirm the viability of a conversion.

2. Overview of NBSR and TRIGA fuel

The NBSR is a heavy water moderated 20 MWth research reactor that first went critical in 1967. The primary use of the NBSR is for neutron scattering research and it is outfitted with 28 fine-tuned neutron instruments. The NBSR hosts more than 2000 guest researchers annually (Kopetka et al., 2008), and is a premier location on neutron research in the U.S. The NBSR has some interesting and unique features such as its "loose" configuration and cold neutron (CN) source. The "loose" configuration is related to the NBSR's moderation. The heavy water moderation allows the fuels to be farther spaced while still maintaining criticality. The additional liquid hydrogen moderator further slows thermal neutrons down to below 5 meV. Among the 28 instruments, 21 of them utilize the cold neutrons that can be transferred and utilized by multiple neutron scattering instruments tens of meters away from the reactor in the experimental hall.

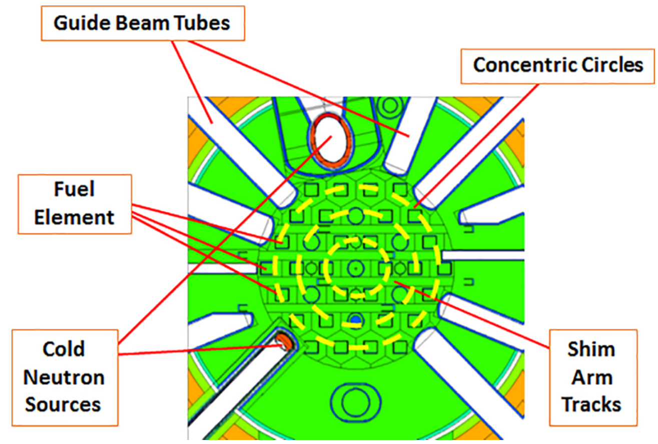

Fig. 1. A schematic top view of the NBSR. The concentric circles formed by the fuel elements are indicated with yellow dashed lines. (For interpretation of the references to colour in this figure legend, the reader is referred to the web version of this article.)

Fig. 1 presents a schematic top view of the mid-plane of the NBSR. The criticality of the NBSR is controlled by 4 cadmium shim arms that are inserted in the core horizontally. The four shim arm tracks are shown in the figure. The angle of the shim arms can be rotated within the tracks to control the criticality of the core during operation. The NBSR core contains 30 fuel elements that are arranged in three concentric circles (shown in Fig. 1). Among those fuel elements, 16 of them reside in the core for eight cycles and 14 reside in the core for seven cycles. At the end of each cycle, which is about 38 days, four fuel elements are removed and replaced with the fresh fuel elements, and other 26 are shuffled to new locations for the use of next cycle (Hanson and Diamond, 2014).

Fig. 2 illustrates the existing NBSR fuel element with a cross-sectional view of the fuel plates in the element. The current NBSR fuel element has an external size of 8.55 cm length, 7.6 cm width and approximately 175 cm in height. The fuel element is composed of 17 plates of HEU fuel with a height of 27.94 cm, width 0.051 cm, and length of 6.25 cm and possess a curved MTR plate geometry. This curved plate is made of U_3O_8 sintered with aluminum powder and clad in aluminum. The fuel element is 93 wt% enriched (HEU) and has an equivalent fuel volume of 296 cm^3 which indicates each element contains about 350 g of ^{235}U for fission. One unique feature of the NBSR fuel element is the element is split into a top and bottom portion with a ~17.8 cm non-fueled gap in the middle of fuel element (see Fig. 2). This gap, when the fuel is inserted into the core, allows the beam tools to point directly to the middle of the core while having no direct line of

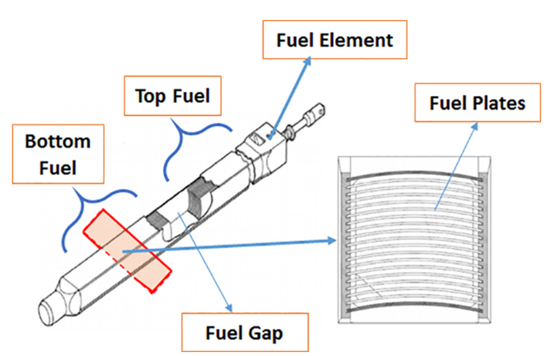

Fig. 2. The NBSR fuel element (left) and the fuel plates in the element (right).

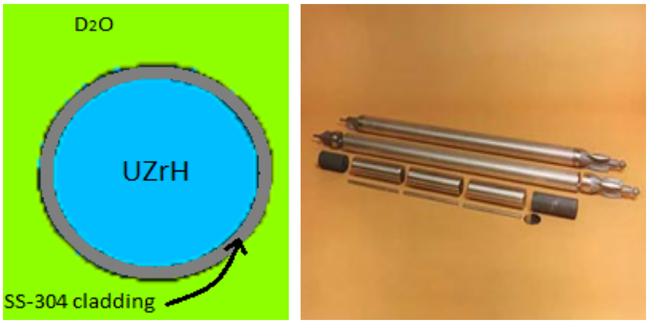

Fig. 3. A cross-sectional view of the TRIGA fuel (left) and the full rod (right).

sight with the fuel (Hanson and Diamond, 2011). In this way, the beams are accessible to the maximum thermal flux without significant contamination of fast neutrons.

Converting NBSR plate fuel to the cylindrical TRIGA fuel rod creates some unique challenges. The NBSR also has no grid flexibility, meaning the compact LEU core design would have trouble preserving the flexibility and range of beam science experiments currently conducted (Stevens, 2010). As illustrated in Fig. 3, the TRIGA fuel meat is wrapped with a 0.04 cm thickness cladding with no bonding space. The radii of individual rods is actually a design parameter and can be varied to meet the total fuel mass requirement in the core depending on the enrichment as well as number and placement of the rods in each element. Note in the right subfigure of Fig. 3, one fuel rod is shown in pieces: the fuel meats are the three metallic pieces in the middle, encapsulated by two graphite reflectors.

However, a couple of recent studies have shown the TRIGA fuel may be used as an alternative LEU fuel for HPRRs. TRIGA fuel has been considered as a possible LEU candidate for conversion of the Advanced Test Reactor (ATR) (Lyons, 2014), which indicates the TRIGA fuel was able to maintain the cycle length, minimum fission rate, and power density with only slight variations in the lobe power and fast to thermal flux ratios over the 56 day cycle. The TRIGA fuel is also being studied as a possible route of conversion in the MIT research reactor. Dunn et al. (2017) found that when operated at the minimum power for operation, the MIT reactor met the minimum critical heat flux (CHF) requirement for operation for the beginning of life cycle. The drawback of this is that at higher powers and using the equilibrium core the TRIGA fuel CHF did not meet the minimum requirements.

3. Feasibility scoping study

The optimal fuel element was determined through variation of three key parameters: fuel composition, cladding composition, and fuel rod placement pattern. TRIGA fuel has already commercially produced in three compositions, 35/20, 40/20, and 45/20, indicating fuels contain 35%, 40%, and 45% of uranium by weight, respectively, and the 20 notation represents the maximum enrichment dictated by the RERTR (19.75% in reality for the consideration of engineering manufacture uncertainties). In order to match the fuel cycle length of the current NBSR core, the tested TRIGA fuels was developed with similar amounts of ^{235}U as the HEU fuel. The exact fuel compositions of the TRIGA fuels are shown in Table 1.

The ratio for Zr to H in the fuel was chosen to be 1.60 as shown in a former study (Generic Procedures for Response to a Nuclear or Radiological emergency at Research Reactors-Training Material, 2011). This value was experimentally found to be the most stable ratio in the $UZrH_x$ compound because of its state of matter is unchanged with excessive heat. In our study, we determined fuel densities using the elemental makeup of each type. The $ZrH_{1.6}$ density was found to be 5.66 g/cc and the uranium density as 19.1 g/cm^3 (Bowman et al., 2010). The fuel densities, were calculated depending on the percentage of uranium

Table 1
Different TRIGA fuel specification.

Compositions	Fuel type			
	NBSR HEU	(35/20)	(40/20)	(45/20)
^{235}U (g)	350.00	350.00	350.00	350.00
^{238}U (g)	26.00	1426.65	1426.65	1426.65
O (g)	68.00	0.00	0.00	0.00
Al (g)	625.00	0.00	0.00	0.00
Zr (g)	0.00	3232.00	2619.23	2134.03
H (g)	0.00	67.39	45.75	37.43
Total mass (g)	1069.00	5076.00	4441.62	3948.11
Fuel Density (g/cm^3)	3.61	10.36	11.04	11.71
Fuel Volume (cm^3)	296	489.80	402.47	337.22

in the alloy, as shown in Table 1. Given this is a feasibility study only, the fuels tested at the initial feasibility study stage are fresh fuel only. A more rigourous study of the fuel over the whole fuel cycle is carried out in the equilibrium core generation section.

For the second parameter of interest, two different claddings commonly used in TRIGA fuels, Stainless Steel-304 and Incoloy-800, were examined at this stage. Both of these claddings are considered as highly resilient cladding materials in terms of safety and economy. The fractions of exact compositions in these two claddings are summarized in Table 2, where the data are obtained from Reference McConn (2011).

The last parameter of interest studied at the scoping survey stage was the fuel rod configuration. Due to the fact that heterogeneity in the fuel element is an important factor in evaluating neutronics performance, different configuration can affect many factors that lead to significant changes in reactivity. To examine the heterogeneity effect for the TRIGA fuel, the fuel rods were placed in grids of 3 × 3, 4 × 4, 5 × 5, and 6 × 6 on both the top and bottom portion of the fuel element (see Fig. 2). For consistency, the length of the fuel gap in the middle portion of the fuel element is retained unchanged. Fig. 4 illustrates the cross-sectional views of the four fuel rod arrays in the element. Each of these configurations keeps an equal amount of uranium mass in order to retain identical total fuel loading for different cores. To keep the same amount of fuel mass, it requires fuel rod radius be smaller for those elements with more rods inside the fuel element. The active fuel length for both top and bottom portion of the element was kept at a constant value 33.2 cm for all these tests, matching the HEU fuel plate length currently inside the NBSR.

The MCNP model of the NBSR received from NIST was modified to include the three parameters of study interests discussed above, and multiple simulations with different cases considered were performed to determine the best options for the fuel element. All calculations of the various cases selected above were performed on an eight core desktop computer with Python and MATLAB as assets for data processing tools.

Table 2
Element and Weight Fraction in the Claddings.

Stainless Steel-304		Incoloy-800	
Density (g/cm^3) = 8.00		Density (g/cm^3) = 7.94	
Element	Weight Fraction	Element	Weight Fraction
C	0.000400	C	0.000650
Si	0.005000	Si	0.006500
P	0.000230	Al	0.003750
S	0.000150	S	0.000100
Cr	0.190000	Cr	0.210000
Mn	0.010000	Mn	0.009750
Fe	0.701730	Fe	0.435630
Ni	0.092500	Ni	0.325000
		Ti	0.003750
		Cu	0.004880

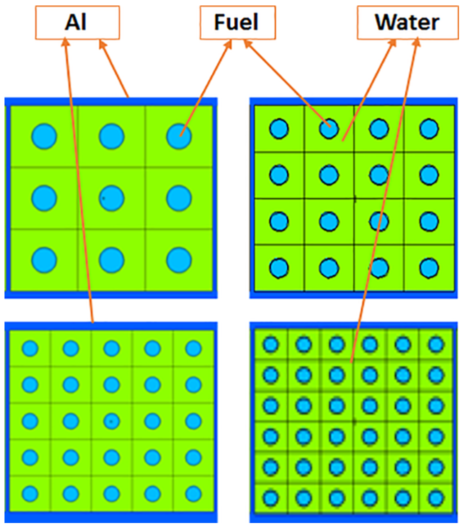

Fig. 4. Fuel rod configuration in the fuel element: 3 × 3 case (top left), 4 × 4 case (top right) and 5 × 5 case (bottom left), 6 × 6 case (bottom right).

For each *kcode* calculation, 110 total cycles with 10 inactive cycles skipped and 10,000 particles histories per cycle were executed to ensure the standard error of the k_{eff} less than 0.1% – the general acceptance criterion for criticality estimation. A detailed report on the scoping study can be found in Reference Britton and Wu (2018) and will not be repeated here. Main conclusions of this study regarding the three key parameter selection are summarized as follows:

- Fuels with the highest density of uranium were shown to have the highest reactivities. Fuels with a higher uranium composition have higher overall density and smaller volume due to the restriction on the amount of ^{235}U per fuel element.
- SS-304 cladding achieved a greater k_{eff} in comparison to Incoloy-800 largely due to the Incoloy-800's 32.5% nickel makeup. Iron is the largest neutron absorber in the SS cladding, making up over 70% of its mass. The 32.5% nickel has a higher neutron absorption cross section than the iron, making the SS-304 a more suitable candidate for our fuels cladding.
- It appears the 5 × 5 configuration gave the best result in terms of the fuel utilization economy. This is a trade-off outcome between the fuel element heterogeneity and fuel to moderation ratio effect (Britton and Wu, 2018).

As a result, the selected fuel rod configuration for further investigation is shown in Fig. 5 with a geometrical comparison of the HEU fuel. A detailed compositions of the two fuels are compared in Table 3. It actually can be seen that the TRIGA fuel needs a much larger amount of ^{235}U to be critical. This different amount of ^{235}U between the HEU and TRIGA fuels is somewhat reasonable because most LEU fuels studied require higher amounts of ^{235}U to stay critical (Turkoglu et al., 2019).

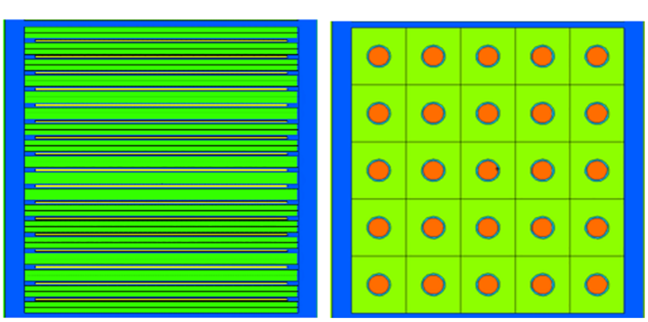

Fig. 5. The fuel element with plate-type HEU fuel (left) and rod-type TRIGA fuel (right).

Table 3
Fuel compositions in HEU and TRIGA fuel 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 (cm^3)	296	466.52
Fuel Height (cm)	66.4	66.4

The next part of the analysis is to examine the long term behavior of the NBSR with the TRIGA fuel using the fuel composition, cladding material and optimal rod configurations determined above. MCNP-6 was again used to generate an equilibrium core of the NBSR using the new fuel and predict key neutronics performance characteristics of the core.

4. Equilibrium core generation

The initial feasibility study suggested the optimal design for the rod-type TRIGA fuel element has a 5 × 5 layout of the fuel rods, with Stainless Steel 304 cladding and 45% uranium by weight fuel. Previous calculations however only consider fresh fuels, which gives an incomplete view of the fuel effectiveness. For a practical reactor analysis, an equilibrium core configuration is desired to demonstrate the fuel effectiveness over the whole fuel cycle length of the core, subject to being partially burned and shuffled of the fuel elements (Wu, 2017). In this study, in order to quickly achieve an equilibrium core with the TRIGA fuels, the same fuel shuffling scheme as the NBSR HEU fuel was adopted because this scheme has been well studied under the NBSR operation conditions. Fig. 6 shows the fuel management scheme used in this study with all thirty elements labeled exclusively. As described earlier in Section II, 16 of the 30 fuel elements in the scheme will be burned for eight cycles, and 14 elements burned for seven cycles.

In Fig. 6, each fuel is identified with two numbers and a letter: The first number denotes the total fuel cycles the fuel will reside in the core, the second number informs the current cycle of the fuel element, and the letter indicates the side of the reactor the fuel is located at – either the left side (L) or the right side (R) of the core. The shuffling strategy is applied to each side independently in a symmetric manner. At the beginning of a cycle, the 8.1 and 7.1 fuel elements are newly loaded fresh fuels, and the 8.8 and 7.7 are mostly burned fuels. At the end of the cycle, the 8.8 and 7.7 fuel elements will be discharged, and their positions will be filled by 8.7 and 8.6 fuels, respectively. All other fuels are thereby shifted accordingly. This shuffling scheme indicates 4 fresh elements will be loaded at the beginning of the cycle and 4 maximum burned elements will be removed at the end of the cycle. The full cycle length used for this study is designated as ~38.5 days, following the current NBSR operation practice.

Four representative burnup states are used to describe the full cycle: start-up (SU), beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC). The difference between SU and BOC is SU fuel has clean fuel and contains no Xe/Sm fission products while BOC fuel has an equilibrium Xe/Sm concentration. Fig. 7 illustrates the burnup time periods between two neighboring states, which also match the values used in the current NBSR LEU cycle analysis (Hanson and Diamond, 2011, 2014). Fig. 7 also briefly illustrates the iterative procedure applied to generate the 4-state equilibrium core based on the fuel shuffling scheme described in Fig. 6.

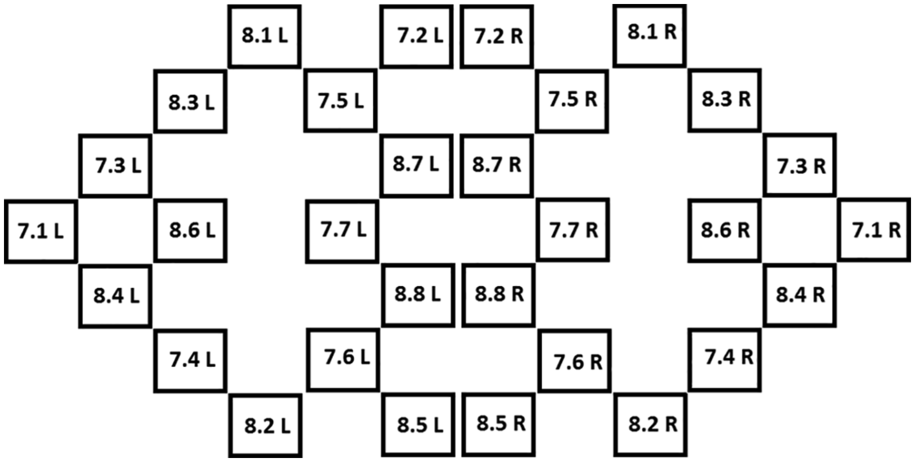

Fig. 6. The NBSR fuel shuffling scheme.

The control shim arm positions have to be adjusted at different states to achieve the criticality status of the core to compensate the fuel burnups. The research results indicate the shim arm critical positions of the equilibrium TRIGA fuel core at the four states are different to the ones of the equilibrium HEU core. This reflects the essential difference of the neutronics characteristics of the TRIGA LEU fuel compared to the HEU fuel. Table 4 summarizes the control shim arm positions for the equilibrium cores of different fuels.

The burnup calculation is enabled by the BURN card in MCNP-6, and fuel elements are replaced or shuffled after each cycle following the designated shuffling scheme given in Fig. 6. A fuel management code was developed to automate this process. After each state calculation, the burned fuel inventories in the previous state will be extracted and manipulated to form fuel materials for the next state. This procedure is completed by a Python script illustrated in Fig. 8. The equilibrium core search process begins with an automated creation of an MCNP input for the starting state. This input deck is then executed through Python and the burned fuel materials are extracted from the output after its completion. A new input for the following burnup state is then generated to continue the process. This process is repeated until equilibrium core inventories for each burnup state are achieved.

To account for axial burnup effects, each fuel element is divided into 6 axial zones. Considering the total fuel elements in the core, this requires a total of 180 different fuel materials in the core to properly represent all fuel compositions with different burnups. During the equilibrium core search, 110 cycles (10 skipped cycles) and 10,000 particles per cycle were used for each state calculation to ensure the standard deviation of k_{eff} less than 0.1%. With these computation parameters, the entire equilibrium core search procedure, which generally requires 12–14 cycles calculations, took approximately 500 h to complete on an eight-core processor. Fig. 9 shows the k_{eff} values of the four burnup states during the iterative search procedure. The search starts off with a fuel configuration containing all fresh fuels, with the k_{eff} curves for all burnup states gradually decreasing until they all reach

Table 4
The control shim arm positions of HEU and TRIGA fueled equilibrium cores.

State	HEU[a]	TRIGA
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. (Turkoglu et al., 2019).
[b] The position is shown in the angle of degrees the shim arms rotated, and 0.00 stands for the rods all out status.

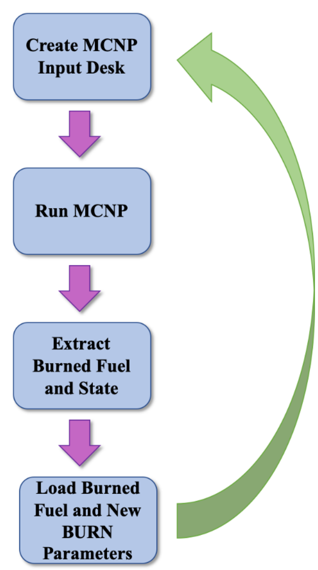

Fig. 8. Flow diagram of the Python script for the equilibrium core search.

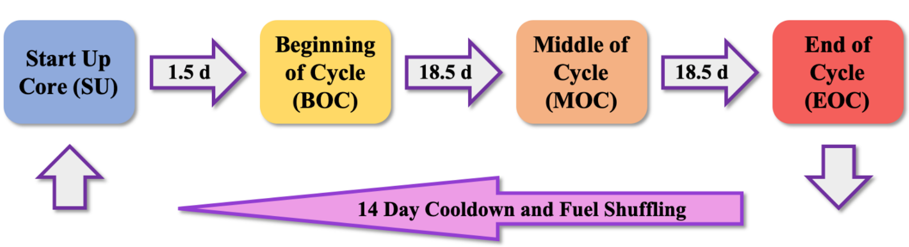

Fig. 7. An iterative procedure to generate the 4-state equilibrium core.

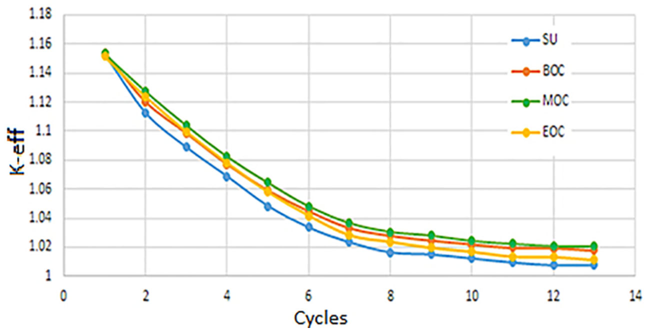

Fig. 9. The k_{eff} value changes with the iteration cycle number for all states.

a plateau at the 8th/9th iteration cycle. All four states were able to keep the k_{eff} value above one until reaching equilibrium status (the point of plateau in Fig. 9), indicating the fuel scheme used in the search will be able to sustain the NBSR with the designated fuel cycle length. Note the standard variances of the k_{eff} values are actually shown in Fig. 9, whereas they are too tiny (~0.001) to be visible in the plots. The equilibrium core configurations yielded from iteration Cycle 12 calculations were used for the rest of the studies in the paper.

5. Neutronics evaluation

5.1. Actinide burnup

The fuel burnup efficiency is primarily evaluated by the ^{235}U burn percentage. The amount of ^{235}U in the discharged fuel elements (elements 8.8L, 8.8R, 7.7L, and 7.7R in Fig. 6) have been subtracted from the amount of ^{235}U in the fresh fuel elements (elements 8.1L, 8.1R, 7.1L, and 7.1R in Fig. 6), and then divided by the original ^{235}U inventory. The calculated burnup and buildup of some key actinides at the end of equilibrium cycle (i.e., the Cycle 12) are summarized in Table 5.

In Table 5, the burnup results of the TRIGA fuel is compared the available counterpart data of the HEU fuel. As seen in the table, the TRIGA fuel does not burn as efficient as the HEU, but the ~60% ^{235}U burnup exceeds most typical LEU fuels such as U-10 M0, U-7Mo, or U_3Si_2 fuel. The closest competitor would be the U-10Mo fuel that achieved a 40% burnup in the new NBSR core, nearly 20% less burnup than the TRIGA fuel (Turkoglu et al., 2019). The results in Table 5 also indicates the TRIGA fuel produces on average 8.38 g of ^{239}Pu. As early studies revealed (Turkoglu et al., 2019; Wu and Williams, 2015), a larger ^{239}Pu production and ^{238}U depletion from LEU are under expected, which is most likely due to the higher ^{238}U content in the fresh LEU fuel.

5.2. Flux distribution

One of the most important neutronics parameters in the evaluation of the TRIGA fuel in the NBSR core is the flux distribution as flux is the key figure of merit in determination of the experimental capabilities of the NBSR. In order to calculate the physical flux, a fission rate 1.523×10^{18} neutrons/s was used as a normalization factor for the flux tallies produced in MCNP (Wu, 2017). This fission rate value reflects the NBSR's 20 MWth power rate with the assumptions of 200 MeV per fission and 2.44 neutrons per fission. In this study, the mesh tally card FMESH in MCNP was used to calculate the flux distribution in the core. The FMESH card allows the user to break the three dimensional (3-D) space into small geometric tally bins and tally the flux for each of those bins. For flux calculation, a right cylindrical geometry was adopted as it most closely approximates the shape of the reactor core. The core was divided into 50 radial, 80 height, and 50 azimuth angular segments for a total of 200,000 tally bins. The radius was from 0 to 56 cm, the height from −40.7 to 40.7 cm with axial mid-core elevation set as Z = 0, and the azimuth angular from 0 to 2π to cover the entire azimuthal range. For each calculation, 2000 active cycles and 36,000 particle histories per cycle were used to reduce the statistical error on the flux to an average of 1.3%.

Table 5
Burnup and buildup of key actinides at Cycle 12.

Actinides	^{235}U (%)	^{235}U (g)	^{238}U (g)	^{239}Pu (g)
TRIGA	59.50	207.10	21.00	8.38
HEU	71.00	250.00	0.23	0.09

The two dimensional (2-D) *radial* flux distribution of the EOC core is demonstrated in the first row of Fig. 10, where the fast and thermal flux are shown at axial elevation Z = -20 cm, or roughly at the middle plane of the bottom fuel region. The 2-D *axial* flux from the side of the core with the azimuth angle θ = π is illustrated in the second row of Fig. 10. In both radial and axial flux maps, one can clearly identify the fuel element positions, which indicate the locations where the majority of the fast flux is emitted. Since thermal flux is primarily from the moderating elements in the reactor, the thermal flux maps in turn indicate the regions where heavy water is concentrated – radially the largest amount of heavy water is situated in the center of the reactor, and axially it primarily resides between gaps the rods in the middle of the core. The flux distributions in other states have demonstrated similar characteristics. As shown in Fig. 10, the maximum thermal flux achieved at the EOC is ~2.3–2.4 × 10^{14} n/cm^2-s in the fuel gap region, while for other states the thermal flux is slightly higher (results not shown here). Compared to the NBSR, this flux has decreased by approximately 5%. It is not desirable for the thermal flux concentrated radially at the center of the core, but this is 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 (Hanson and Diamond, 2011).

5.3. Power distribution

The power distribution in the reactor core is a safety concern for any thorough reactor analysis. By examining the power distribution among the reactor, hot spots can be identified to determine the integrity of the fuel under various operational conditions. The radial power distribution can be obtained with tallies on each fuel rod. In order to obtain detailed axial power distribution, the fuel rod was divided into 12 axial sections. This means that with 50 rods per fuel element, and 30 fuel elements in the core, a total of 18,000 tally cells with a volume of 1.154 cm^3 for each cell were generated for the power distribution calculation. In this study, power distribution were estimated using the so-called "Table 128" method (Wu and Williams, 2015) rather than using the energy deposit tally approached typically used in MCNP because this method does not require additional tally computations. In the standard output of MCNP, Table 128 is a collection of the neutrons that enter, collide, and fission in each cell. The fission neutrons shown in the table are proportional to the fission rate in the cell containing fissional materials; thus these numbers can be used to infer the power information. This is the essence of the "Table 128" method for the power calculation (Wu and Williams, 2015).

The radial power distribution of the fuel element with the hottest power point at EOC are shown in Fig. 11, in which the power factor was used as an indicator representing the ratio of the power of the fuel rod to average power of the entire core.

Table 6 summarized the global power peaking factors (PPF) of the hot spot at each burnup state and their corresponding occurring locations in the fuel element. As seen in Table 6, the hottest PPF (2.80) occurs at SU, which is under expected because the most reactive fuel exists in SU during the cycle. The SU state also has the least amount of Xe/Sm, whose poisons can damp the PPF in certain level.

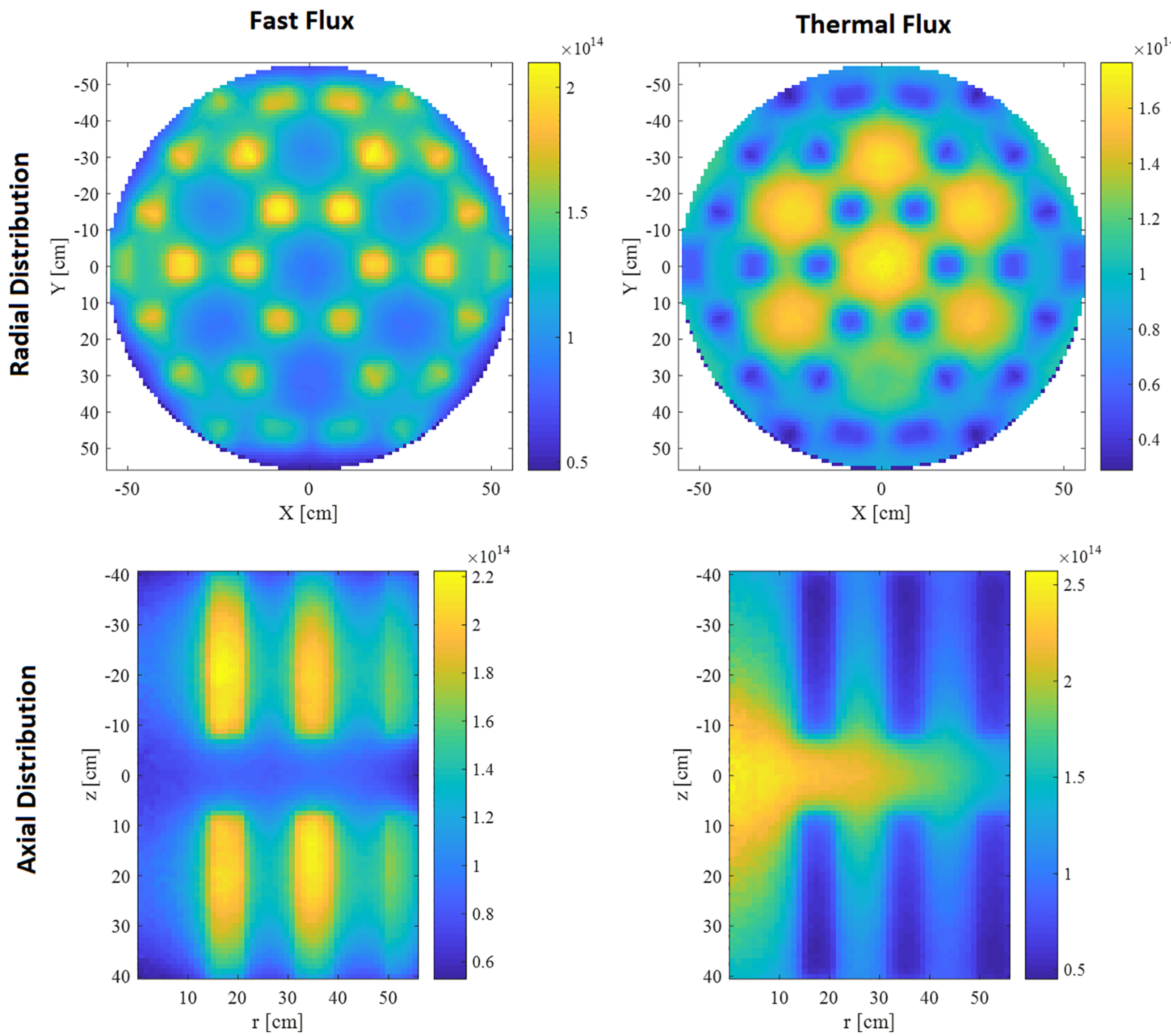

Fig. 10. The 2-D graphical view of the flux maps at EOC.

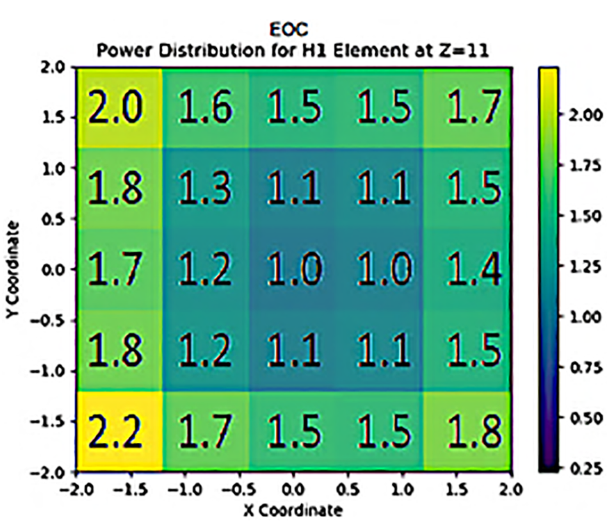

Fig. 11. Power factors at the hottest element in the EOC.

In Table 6, the horizontal location of the hot rod in the fuel element is presented in an X-Y coordinate system with a pair of indicating numbers (x, y), and the center rod is at the (0, 0) position. The axial position of the hot 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. The coordinate systems used in Table 6 are depicted in Fig. 12.

Table 6
PPF and occurring locations at different burnup 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 6, the hot spot locations in four states are nearly identical. The 7.2L and 7.2R fuel element are located at the north side of the core (see Fig. 6). This is mainly due to the location of the cold neutron source located closer to these areas (see Fig. 1), which may lead to higher moderations. In general, the power of LEU fuels such as U-10Mo and U-7Mo tends to be concentrated towards the center of the reactor (Hanson and Diamond, 2011, 2014). However, the TRIGA fuel power appears to be concentrated to the northwest edge of the reactor. Power concentrated in the center of the reactor decreases the amount of neutrons enter the beam tubes, whereas power concentrated at the edges of the reactor creates less of a hot spot, and allow the maximum amount of neutrons to enter the beams (Hanson and Diamond, 2011; Turkoglu et al., 2019).

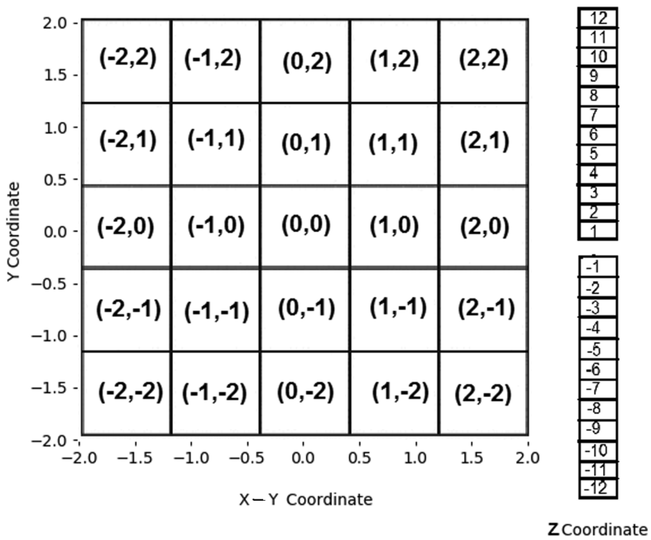

Fig. 12. The labeling system for a fuel element for power calculations.

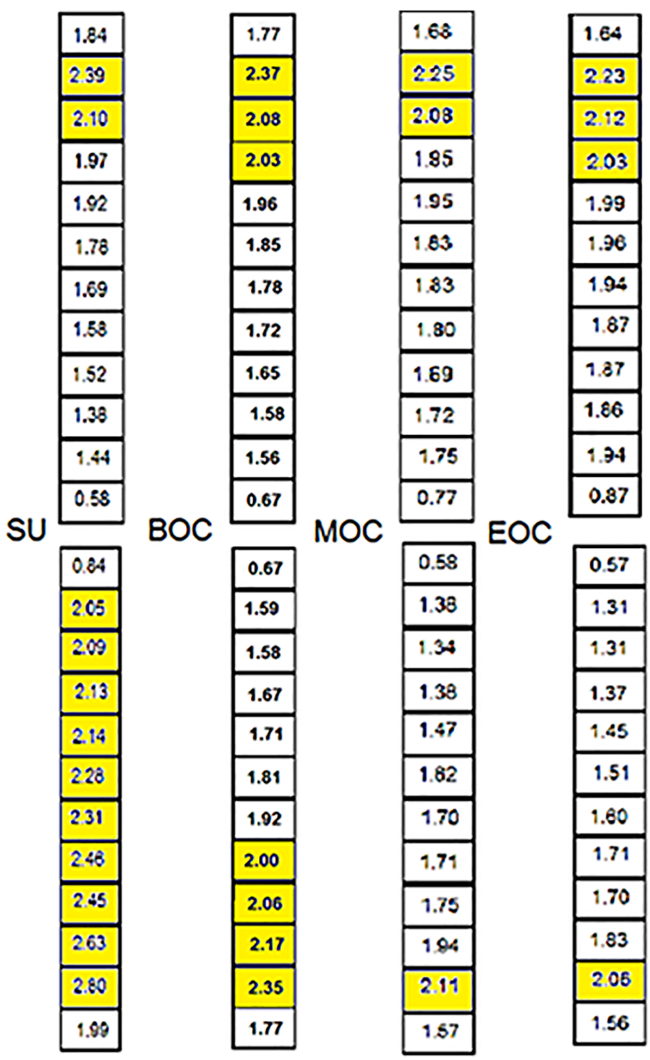

Fig. 13. Axial view of power distribution for all 4 states.

The axial power distribution of four states in the hottest rod are shown in Fig. 13, in which the PPF for each position is shown and values above 2.0 are highlighted. As seen in Fig. 13, axial power is shifted from bottom of the core to the top part at SU because of the control shim arm movements. As the control shim arms are retracted from the core, they pass from the bottom of the core to the top and cause changes in power concentration over the course of the cycle.

Table 7
Neutron Lifetime (microseconds).

State	SU	BOC	MOC	EOC
TRIGA	519 ± 15	476 ± 11	494 ± 12	538 ± 13
HEU	698	N/A	N/A	731

5.4. Neutron lifetime

The prompt neutron lifetime, which is defined as the average time elapsed between the birth of fission neutrons to absorption (Hanson et al., 2005), gives insights into the neutron multiplication process. If the neutron lifetime is high, there is more time between emission and absorption due to more scattering events. If the neutron lifetime is low, the reactor is generally more difficult to control. In this study, the KOPTS card was used to determine the lifetimes using the equilibrium material makeup for all four state (see Table 7).

As shown in Table 7, 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. With significantly shorter lifetimes, it will theoretically more difficulty to control the reactor with the TRIGA fueled reactor than the one with the HEU fueled reactor.

5.5. Control worth and shutdown margin

Another parameter of significance for reactor safety operation is the shim arm control worth, which provides data to directly estimate the control safety margins. The control worth and its corresponding shutdown margins for the TRIGA fuel are summarized in Table 8. Although the shim worth for the TRIGA fuel is appeared slightly lower than that of the HEU, it is still within the acceptable range for the NBSR (Hanson and Diamond, 2014). The shutdown margin was estimated with the one shim arm fully stuck out of the core.

6. Conclusions

This paper performed a neutronics evaluation on the TRIGA LEU fuel for the NIST research reactor. The evaluation was built upon a feasibility scoping study of the converted core options by varying several design parameters including the fuel rod configuration, fuel type, and cladding material. The scoping study determined that the 5 × 5 rod configuration with the 45/20 type fuel and stainless steel cladding would create the most reactive core design. The core with the highest reactivity is desirable for the study because it will be the most efficient one to hold its reactivity over the whole fuel cycle. Therefore the fuel element with these options were selected for further investigation.

The study continued to examine the steady-state neutronics performance characteristics of the hypothetically converted NBSR using the TRIGA LEU fuels and selected fuel rod configurations. An equilibrium core based on the existing NBSR fuel management scheme was generated through an iterative equilibrium core search procedure with the use of the MCNP BURN calculation and Python script processing.

Table 8
Control Worth and Shutdown Margin (%$\Delta k/k$).

State	SU	BOC	MOC	EOC
Control Worth	15.9	16.5	16.8	17.1
Shutdown Margin	4.0	7.4	9.4	11.4

The equilibrium cores at four representative burnup states were utilized for the neutronics evaluation. The burnup efficiency, power distribution, flux characteristics, neutron lifetime, and safety control worth and shutdown margins were calculated and examined. The burn up efficiency of the TRIGA fuel was shown to be of importance, outscoring its competitors by nearly 20 percent, it is an efficient alternative. Hot spots 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 fuel cycle, and is concentrated in areas near the upper cold source. Based on the flux calculations, the thermal flux was degraded ~5% of the NBSR's current operating flux in the center of the core, and flux was 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 concentrated in the fuel gap region. The neutron lifetimes generated were shown in reasonable bounds to other LEU fuels. The control worth and shutdown margin for the TRIGA fuel were determined to be acceptable in the context of the NBSR's needs. For future work, we will perform thermal hydraulics safety analyses of the TRIGA LEU core using system level safety transient code based on neutronics parameters from this study.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgments

We thank Robert E. William for providing us the original MCNP models of the NBSR. We also thank Danyal J. Turkoglu for providing the framework of the Python fuel management code.

Author contribution statement

Kyle Britton is a M.S. graduate and current PhD student at VCU, who performed most of the calculations in the paper and drafted the original manuscript.

Zeyun Wu is an assistant professor, who initiated and supervised this project. He provided advisory guidance to all the work performed in the paper. He also contributed significantly to the editorial part of the final manuscript.

References

Bowman, R., Craft, B., Cantrell, J., Venturinix, E., 2010. Effects of thermal treatments on the lattice properties and electronic structure of ZrHx. Phys. Rev. B 31 (9), 5604–5615.
Britton, K., Wu, Z., A neutronics feasibility study of the TRIGA LEU fuel in the 20mwt NIST research reactor. In: Proceedings of the 26th International Conference on Nuclear Engineering, London, England, July 22–26, 2018.
Dunn, F., et al., 2017. Evaluation of a Uranium Zirconium Hydride Fuel Rod Option for Conversion of the MIT Research Reactor (MITR) from Highly Enriched Uranium to Low Enriched Uranium, Laboratory Report. Argonne National Laboratory.
Generic Procedures for Response to a Nuclear or Radiological emergency at Research Reactors-Training Material, International Atomic Agency, 2011.
Hanson, A., Diamond, D., 2011. Calculation of Design Parameters for an Equilibrium LEU Core in the NBSR, Laboratory Report. Brookhaven National Laboratory.
Hanson, A., 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.
Hanson, A., Ludewig, H., Diamond, D., 2005. Calculation of the Prompt Neutron Lifetime in the NBSR. Brookhaven National Laboratory.
Kopetka, P., Williams, R.E., Rowe, J.M., NIST Liquid Hydrogen Cold Source, NIST Internal Report, NISTIR-7352, 2008.
LANL, MCNP-A General Monte Carlo N-Particle Transport Code, Version 6.1.1, LA-UR-03-1987, Los Alamos National Laboratory (August 2017).
Lyons, J., et al., 2014. Reactor physics scoping and characterization study on implementation of TRIGA fuel in the Advanced Test Reactor. Nucl. Technol. 189, 202–217.
McConn Jr., R.J., et al., 2011. Compendium of Material Compositions Data for Radiation Transport Modeling, PNNL-15870 Rev. 1. Pacific Northwest National Laboratory.
Stevens, J., 2010. Technical Challenges for Conversion of US High-Performance Research Reactors (USHPRR). Argonne National Laboratory.
Turkoglu, D.J., Wu, Z., Williams, R.E., Newton, T.H., 2019. Neutronic performance characteristics of different LEU fuels in a proposed NIST research reactor. Ann. Nucl. Energy 128, 209–215.
Wu, Z., et al., 2017. Neutronics and safety studies on a research reactor concept for an advanced neutron source. Nucl. Technol. 199 (1), 67–82.
Wu, Z., Williams, R.E., 2015. A Fast and Self-Consistent Approach for Multi-cycle Equilibrium Core Studies Using Monte Carlo Models, the Joint International Conference on Mathematics and Computation (M&C), Supercomputing in Nuclear Applications (SNA) and the Monte Carlo (MC) Method, Nashville, TN, April 19-23, on CD ROM.
Wu, Z., Williams, R.E., 2015. Reactor power distribution calculation in research reactors using MCNP. Trans. Am. Nucl. Soc. 113, 1171–1174.