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A NEUTRONICS FEASIBILITY STUDY OF THE TRIGA LEU FUEL IN THE 20MWt NIST RESEARCH REACTOR

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ABSTRACT

The National Bureau of Standards reactor (NBSR) at the National Institute of Standards and Technology (NIST) is under conversion from high enriched uranium (HEU) to the low enriched uranium (LEU) schema under the Reduced Enrichment for Research and Test Reactors program (RERTR) as a part of the Global Threat Reduction Initiative (GTRI). The conversion of the high performance research reactors (HPRR) such as NBSR is a challenging task due to the high flux need (2.5×10^{14} n/cm^2-s for the NBSR), as well as other neutronics performance characteristics requirements without significant changes to the external geometrical configuration. One fuel candidate, the General Atomics (GA) UZrH LEU fuel, has showed particular promise in this regard. The TRIGA LEU fuel was initially developed in the 1980s with particular considerations for fuel conversion for high power regimes such as high density research and test reactors. This study performs a neutronics feasibility study of the UZrH LEU fuel schema for the NBSR, examining the accountability and sustainability of the TRIGA fuel when applying it to the NBSR conversion. To identify the best option to deploy the TRIGA fuel to NBSR in terms of key neutronic performance characteristic, the study is carried out with various considerations in the fuel dimensions, fuel rod layout configurations, and structure material selections. Monte Carlo based computational model is used to assist and facilitate the research procedure. The research findings in this study will determine the viability of the TRIGA fuel type for the NBSR conversion, and provide supporting data for future investigations on this subject.

INTRODUCTION

The Reduced Enrichment for Research And Test Reactors Program (RERTR) program started in 1978 by the United States, in order to reduce the amount of high enriched uranium in reactors across the country to prevent the threat of proliferation. By converting the High Enriched Uranium (HEU) fuel to Low Enriched Uranium (LEU) fuel, there is a significant reduction of risk in the HEU being used in weapons of mass destruction. The program has three main focuses; development of new LEU fuels, design and safety analysis for the conversion, as well as the production of the medical isotope Molybdenum-99 with LEU. The program requires the fuel contain below 20 wt.% U-235 in uranium[1]. Some reactors have already been converted, but some present greater challenges. The high performance research reactors (HPRR) are particularly challenging to convert due to the high flux requirements for running. One of these 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 MWt heavy water moderated research reactor currently operating using the material test reactor (MTR) plate type HEU fuel. The NBSR operates at an average thermal flux density of 2.0×10^{14} n/cm^2-sec, an extremely high flux requirement for operation and continued experimentation.

Presently the NBSR is fueled with a MTR curved plate type HEU fuel that is 93 wt.% enriched. The fuel is a U_3O_8 dispersion fuel clad in aluminum. The U-10Mo monolithic and U-7Mo/Al dispersion fuels, with 10 and 7 wt.% molybdenum in each, can be good LEU options for NBR and have shown some promising results[2, 3]. The fuel can match most of the high flux and energy requirements while still being very safe. The drawback with this is that the fuel is still years away from being able to be manufactured, making it less viable for a near-term conversion[4]. A solution to this problem could be to leverage a fuel already on the market that could satisfy the safety requirements of RERTR, and LEU conversion requirements for the NBSR's high performance needs.

The TRIGA (Training, Research, Isotopes, General Atomics) fuel developed by the General Atomics (GA) is such an available LEU fuel that is qualified under the RERTR program schema. TRIGA reactor was first commissioned in 1956 and has been identified as a "safe" fuel ever since its commencement. The TRIGA fuel, composited by the $UZrH_x$ material, is known for its prompt negative temperature coefficients and long core lifetimes, and was initially created specifically for HPRR development. TRIGA fuel is a cylindrical rod with stainless steel cladding while the NBSR fuel is a plate cladded with alumina. Thus one would raise the question that whether the TRIGA fuel can meet the NBSR's LEU conversion requirements if used for NBSR conversion without any alteration to the structure of the fuel element and the reactor?

The focus of this study is a performance and feasibility study of the TRIGA fuel in the NBSR with the aim to answer this question and determine if the LEU fuel can fulfill the heavy needs of the NBSR, which include maintaining the same level of flux, power, core lifetime, and so on.

The meet the goals, one have to meet three requirements to meet in the conversion. The first requirement is that the current core fuel holding must be maintained. Changes to the core will be restricted purely to the fuel elements inside in order to maintain the integrity of the NBSR. The second requirement for this conversion is to maintain the current irradiative testing capabilities of the NBSR. The neutron flux from the NBSR cannot vary greatly from its current operating level (2.5×10^{14} n/cm^2-s) as this may actively effect the testing capabilities of the site. The last requirement for a successful conversion of the NBSR is that the appropriate safety requirements be satisfied[5]. In this paper, we made a very preliminary study on this subject by assessing the performance of TRIGA fuel in meeting the first requirement mentioned above. Future efforts will be exercised focusing on the second and third requirement assessments.

In this study, we examined three different compositions of the TRIGA UZrH fuel for a wide range of ^{238}U concentrations. Particularly, we explored the effect of these slightly different fuels on the reactivity variations using the Monte Carlo code MCNP6 (Ref. 6). We also examine the effect of fuel rod configuration and cladding on the reactivity changes in order to obtain a wide insight of the fuel schema and determine its effectiveness for use in the NBSR. The results of the MCNP calculations are summarized and compared to the NBSR's current HEU fuel.

OVERVIEW OF THE NBSR

First went critical on December 7 1967. the NBSR is a 20MW thermal power research reactor. The NBSR is now evolved to be a major neutron beam experiment facility equipped with 28 neutron research instruments and hosts over 2000 guest researchers each year The NBSR design is unique in that it has a liquid hydrogen cold moderator[7] slowing neutrons down to less than 5meV to be used in 21 of the 28 instruments that require cold neutrons. The NBSR has been operated using HEU fuel since it went critical in 1967, and due to the challenges of conversion is still using HEU fuel to this day. The NBSR is composed of 30 rectangular fuel elements in three concentric rings as shown in Fig.1.

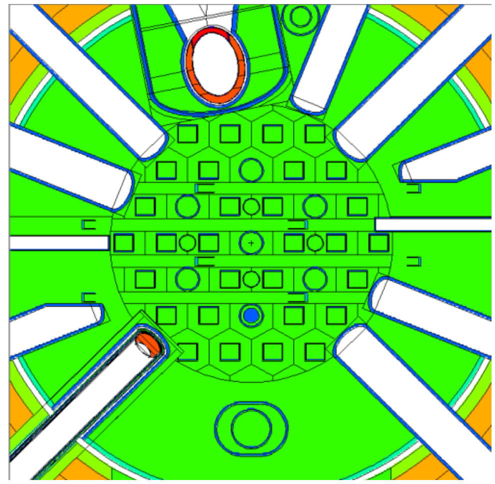

Figure 1: Cross section view of the mid plane of NBSR core.

The NBSR fuel element is composed of 17 plates of HEU fuel with a height of 27.94cm, width 0.051cm, and length of 6.25cm and has a curved MTR plate geometry as shown in Fig. 2. The elements are split into a top and bottom portion with a 7 inch (~17.8 cm) non-fueled gap in the middle of fuel element (see Fig. 2). The fuel element has an equivalent fuel volume of 296 cm^3 which indicates each element contains about 350 grams of ^{235}U for fission.

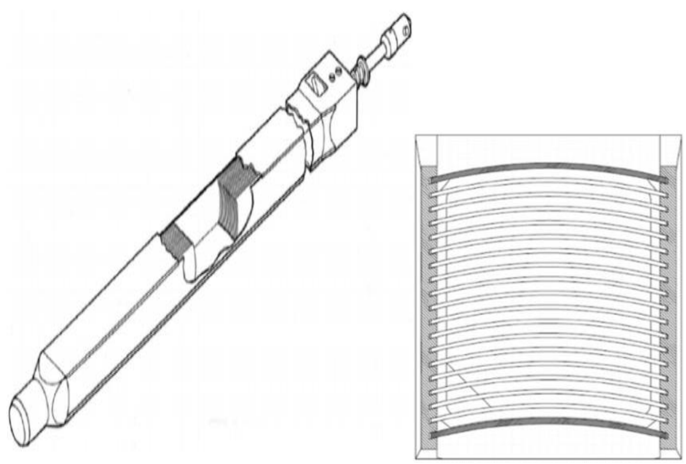

Figure 2: NBSR fuel element (left) and cross section view of the fuel plates in the element (right).

The 7 inch gap existing in middle of the NBSR fuel element and the core allows the beam tools to point directly to the middle of the core while having no direct line of sight with the fuel[2]. Since the NBSR is cooled by heavy water, it known to have a "loose" configuration because the distance between elements is relatively larger than the one in most of other light water cooled HPRRs. In the conversion from HEU to LEU it will be absolutely imperative to maintain the outer dimension of the fuel element for a pragmatic short term conversion.

GENERAL ATOMICS LEU FUEL

The TRIGA fuel mentioned earlier is an attractive candidate for the NBSR conversion due to its long credible history as a safe powerful fuel. Known for its long core life and prompt negative temperature coefficient the TRIGA fuel was originally designed by GA to be a safe substitute for the HEU fuel sitting at an average enrichment of 8.5% to 45% for higher density fuels. The TRIGA fuel is currently being studied in the advanced test reactor (ATR) with promising results delivered. In Lyons survey[8] she found the TRIGA fuel was able to maintain the cycle length, maintain the 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[9] found that 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, but with the drawback that at higher powers and using the equilibrium core the TRIGA fuel CHF did not meet the minimum requirements.

The NBSR creates some unique challenges compared to the previous studies if converted with TRIGA fuel; the NBSR is heavy water reactor. 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[5]. Furthermore, the TRIGA fuel is a cylindrical rod while the current NBSR fuel is plate type. The TRIGA fuel meat is wrapped with a 0.04 cm thickness cladding as seen in Fig. 3, and has no bonding space between the fuel meat and cladding. 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. More detailed explanation of the TRIGA fuel model will be provided in later sections.

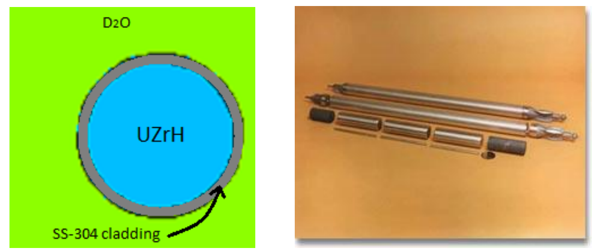

Figure 3: Cross sectional view of the TRIGA fuel (left) and the full rod (right)

Table 1: Different TRIGA Fuel Specification

Fuel type	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/cc)	3.16	10.36	11.04	11.71
Fuel Volume (cc)	296	489.80	402.47	337.22

RESEARCH PROCEDURE

In this study, we used MCNP, a Monte Carlo based neutron tracking code, to model the reactor and determine neutronics characteristics of the cores with varying cladding, fuel rod configuration, and fuel type. The original NBSR MCNP model has been modified to fit the TRIGA fuel schema. A wide view of the reactivity performance within these varying parameters will provide us some insights of the viability of the TRIGA fuel being used in the NBSR. With these initial testing-basis neutronics examination results, we can identify the most viable investigation direction to perform further study on the subject of reactor conversion with TRIGA fuel.

The TRIGA fuel used in this study is modeled to match the total amount of ^{235}U in the current HEU fuel of the NBSR with a trade-off consideration of performance and economy. Three different commercially available TRIGA fuel types were explored, referred to 35/20, 40/20, and 45/20 fuel respectively in Table 1. These fuels have 35%, 40%, and 45% uranium by weight with the maximum enrichment of 19.7% ^{235}U. A detailed description of the parameters in these three TRIGA fuels is given in Table 1, with a comparison to the existing NBR HEU fuel. The ratio for the Zr to H in the fuel varies from 1.58 to 1.65. The value of 1.60 was used in this study suggested by a sensitivity study performed in Ref. 11, as this ratio was found to be optimal and will be kept constant throughout the survey. In calculation of the densities of the individual fuel elements the weight percentages were used to determine the individual density contributions to the total density of each individual fuel. The $ZrH_{1.6}$ density was determined to be 5.66g/cc with the uranium density given as 19.1 g/cc[12]. At this moment, only fresh fuels are considered in calculating the keff value of a core as the key neutronics performance characteristic. This does not follow the reality since NBSR will never be loaded with all fresh fuels. The results obtained based on fresh fuel core study, however will render us many merits to understand the physics trend of core performance with varying parameters of our interest, and will substantially guide us to the next level of the research, in which a pragmatic multi-cycle equilibrium core will be created for the study.[13]

The Stainless Steel-304 (SS-304) and Incoloy-800, the two most commonly used cladding materials in the TRIGA fuel, are considered in this study. The density, element composition and weight fraction of these two materials are outlined in Fig. 3. Aluminum is generally used as a cladding for research reactors, but it is not considered here due to safety concerns as the long lifetime requirement of the TRIGA fuels can lead to significant corrosion and blistering, so it is precluded for this study[8].

Steel, Stainless 304		Incoloy-800	
Density (g/cm3) = 8.00		Density (g/cm3) = 7.94	
Element	Weight Fraction	Element	Weight Fraction
C	0.000400	C	0.000650
Si	0.005000	Al	0.003750
P	0.000230	Si	0.006500
S	0.000150	S	0.000100
Cr	0.190000	Ti	0.003750
Mn	0.010000	Cr	0.210000
Fe	0.701730	Mn	0.009750
Ni	0.092500	Fe	0.435630
		Ni	0.325000
		Cu	0.004880

Figure 3. Element and weight fraction in the claddings.

Fuel rod configuration is generally considered as a major factor that affects the neutronics performance in heterogeneous core, particularly for thermal reactors because different configuration leads to different neutron resonance escape probability during the neutron slowing down process. To examine these effects, four rod configurations are considered in this study as shown in Fig. 4. The fuel rods are arranged in a 3×3, 4×4, 5×5 array and 6×6 array on top and bottom portion of the fuel element with the constrain keeping a constant weight of ~350 grams of ^{235}U per element, which results in a varying radius of the rods in each configuration. With these settings, a higher reactivity is expected from the more homogenous configurations. In the regards of vertical considerations, the non-fueled gap distance in mid of the core remains identical to the current NBSR. The fuel is 33.2cm in length to match the length of the HEU fuel plate.

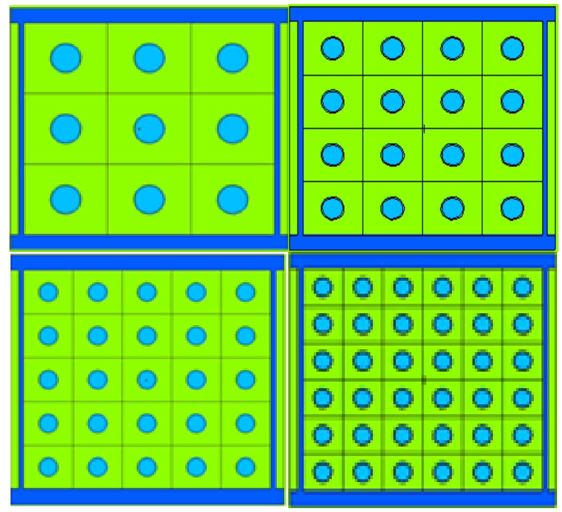

Figure 4. Fuel rod configuration in the fuel element: 3×3 case (top left), 4×4 case (top right) and 5×5 case (bottom left), 6×6 case (bottom right).

The material composition and density for the cladding were found in Ref. 14. For each reactor criticality calculation (i.e., the *kcode* calculation) in MCNP, we run totally 110 cycles with 10 inactive cycles skipped and 10,000 particle histories per cycle to ensure the standard error of the k-eff value is less than 0.001.

RESULTS

The keff values of the new cores with varying parameters considered are summarized in Table 2, Table 3, Table 4, and Table 5 shown in the next page, among them cores in Table 2 and 3 used Incolooy-800 cladding and the ones in Table 4 and 5 used SS-304 cladding. The composition difference existed in cladding between SS-304 and Incoloy-800 (see Fig. 3) should show a difference in reactivity due to the stainless steel containing 30% more Iron by weight, and Iron typically has a large neutron absorption cross section in thermal range. However, the results show that all cases with SS-304 cladding have a greater keff value than their companion ones. This is mainly due to the fact that Incoloy-800 have about 32.5 wt.% of Nickel which has a larger absorption neutron cress section than Iron.

By examining the tables, one can see that and the case with the highest keff value (1.08927) is the one that has 5×5 configuration, the 45/20 fuel in the with stainless steel cladding, and the case with the lowest keff value (1.06392) is the one having 3×3 rod configuration, the 45/20 fuel and Incoloy-800 cladding. The highest case can be intuitively interpreted because it is the case with more heterogeneous configuration (5×5) and the smallest size of fuel rod (45/20). With the ^{235}U mass set in a fixed value, the one with more uranium weight percentage will have a higher density and smaller volume (see Table 1). The rod with smaller size will somehow benefit reactivity because it helps restrain the geometric self-shielding effect which usually occurs at the surface area of the fuel rod. The smallest case, however appears slightly hard to explain as it also happens to a case with the smallest size of fuel rod.

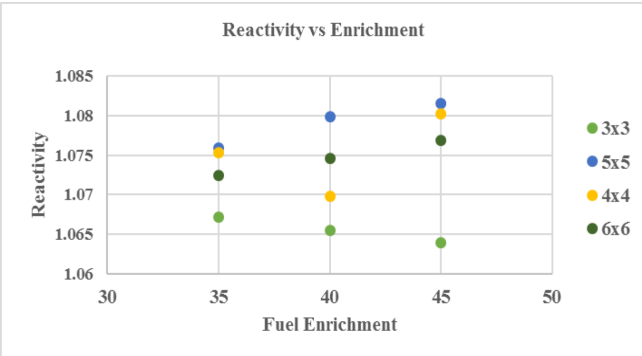

Figure 5. Reactivity of various fuel rod configurations using fuels with different enrichments.

Moreover, we observed the keff variation curve along with the fuel type exhibits completely opposite trend for different fuel rod configurations. Fig. 5 presents the keff (i.e., criticality) of all fuel rod configurations using fuels with different fuel types (the U-235 wt.% enrichments are different). For the 5×5 and 6×6 configuration, the reactivity increases with the fuel type shown in the tables, while for the 3×3, it decreases (see Fig. 5). This is likely due to the changes of the moderating condition from over-moderating scenarios (5×5 and 6×6 rod configurations) to under-moderating scenarios (3×3 rod configurations). However, the culprit of this phenomena remains unknown at this moment and a better explanation requires further investigation.

As a summary, the preliminary results presented in Table 2 through 4 inform that the level of heterogeneity in fuel rod configuration affect the criticality of the core with the same amount of fissionable materials. The higher level of heterogeneity generally results with a greater keff value, which again can be explained with the same physics reason as mentioned previously because the high level of heterogeneity ends with smaller size fuel rods (see Fig. 4), and a lower radii fuel rod is less likely to experience geometric "self-shielding" with a higher surface area/fissile material ratio. With the results shown in Fig. 5, it appears the configuration with a 5×5 fuel rod array gives the best performance in terms of fuel utilization economy.

Table 2: Results of the TRIGA fuel reactivity in the NBSR using the *Incoloy-800* cladding for 3×3 and 5×5 cases.

Fuel Type	35/20	35/20	40/20	40/20	45/20	45/20
Rod Configuration	3 x 3	5 x 5	3 x 3	5 x 5	3 x 3	5 x 5
Fuel density (g/cc)	10.36	10.36	11.04	11.04	11.71	11.71
Fuel rod radius (cm)	0.51	0.31	0.463	0.28	0.42	0.25
Cladding thickness (cm)	0.04	0.04	0.04	0.04	0.04	0.04
Fuel rod height (cm)	33.20	33.20	33.20	33.20	33.20	33.20
Total number of rods	18	50	18	50	18	50
Total U-235 mass (grams)	350	350	350	350	350	350
keff	1.0672	1.07594	1.06546	1.07989	**1.06392**	1.08157

Table 3: Results of the Triga fuel reactivity in the NBSR using the *Incoloy-800* cladding for 4×4 and 6×6 cases.

Fuel Type	35/20	35/20	40/20	40/20	45/20	45/20
Rod Configuration	4 x 4	6 x 6	4 x 4	6 x 6	4 x 4	6 x 6
Fuel density (g/cc)	10.36	10.36	11.04	11.04	11.71	11.71
Fuel rod radius (cm)	0.38	0.255	0.34	0.23	0.32	0.21
Cladding thickness (cm)	0.04	0.04	0.04	0.04	0.04	0.04
Fuel rod height (cm)	33.2	33.2	33.2	33.2	33.2	33.2
Total number of rods	32	72	32	72	32	72
Total U-235 mass (g)	350	350	350	350	350	350
keff	1.07528	1.07241	1.06985	1.07461	1.08023	1.07687

Table 4: Results of the TRIGA fuel reactivity in the NBSR using the *Stainless Steel-304* cladding for 3×3 and 5×5 cases.

Fuel Type	35/20	35/20	40/20	40/20	45/20	45/20
Rod Configuration	3 x 3	5 x 5	3 x 3	5 x 5	3 x 3	5 x 5
Fuel density (g/cc)	10.36	10.36	11.04	11.04	11.71	11.71
Fuel rod radius (cm)	0.51	0.31	0.463	0.28	0.42	0.25
Cladding thickness (cm)	0.04	0.04	0.04	0.04	0.04	0.04
Fuel rod height (cm)	33.20	33.20	33.20	33.20	33.20	33.20
Total number of rods	18	50	18	50	18	50
Total U-235 mass (g)	350	350	350	350	350	350
keff	1.07424	1.08322	1.07036	1.08752	1.06721	**1.08927**

Table 5: Results of the TRIGA fuel reactivity in the NBSR using the *Stainless Steel-304* cladding for 4×4 and 6×6 cases.

Fuel Type	35/20	35/20	40/20	40/20	45/20	45/20
Rod Configuration	4 x 4	6 x 6	4 x 4	6 x 6	4 x 4	6 x 6
Fuel density (g/cc)	10.36	10.36	11.04	11.04	11.71	11.71
Fuel rod radius (cm)	0.38	0.255	0.34	0.23	0.32	0.21
Cladding thickness (cm)	0.04	0.04	0.04	0.04	0.04	0.04
Fuel rod height (cm)	33.2	33.2	33.2	33.2	33.2	33.2
Total number of rods	32	72	32	72	32	72
Total U-235 mass (g)	350	350	350	350	350	350
keff	1.08083	1.07923	1.07338	1.08301	1.08508	1.08617

CONCLUSIONS AND FUTURE WORK

In this paper, a neutronics feasibility study on the TRIGA LEU fuel application to the NIST research reactor was performed. The neutronic performance characteristic of the resulting conceptually converted reactor is examined by varying several design parameters including the fuel rod configuration, fuel type and cladding material. The preliminary results indicate that the 5×5 rod configuration with the 45/20 type fuel and stainless steel cladding will create the most promising core, namely achieve the highest reactivity. The highest reactivity is desirable because most likely it will eventually produce the most economic model using the same amount of fuels. Moreover, it will ease the further studies in which a multi-cycle equilibrium core search will be carried out.

We are currently in a very early phase of the project, and have a great deal to dig into in the near future. But for future studies, an equilibrium core will give a more realistic examination of the feasibility of the TRIGA fuel on the NIST core conversion. Thus we will first produce TRIGA fuel inventories in a beginning, middle, and end of an equilibrium fuel cycle using the BURN feature in MCNP6 and calculate the reactivity at each state respectively. The fuel burnup analysis will also give some insights into the power output of the fuel to see if it can measure up to the current HEU schema. We will explore more options for cladding, particularly a zirconium cladding and aluminum if needed. We can also explore the possibility of a configuration less uniform, concentration of the rods in the outer or inner bounds of the inner fuel element to understand the reactivity change trend so as to find the best configuration option. Last but not least, we will have a more in-depth look into the feasibility, examining not just the reactivity but the power and flux distribution as well as some reactor safety coefficients.

ACKNOWLEDGMENTS

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