Modeling and Simulation Efforts to Support Conversion of Research Reactors to Low-Enriched Uranium Fuel: II

The 2023 Status Update on the LEU Conversion of the NBSR

Abdullah G. Weiss*, Osman S. Celikten*, Dagistan Sahin*, Anil Gurgen*†, Thomas H. Newton*

NIST Center for Neutron Research, 100 Research Drive, Gaithersburg, MD 20899, USA
†Department of Materials Science and Engineering, University of Maryland, College Park, MD 20742, USA
abdullah.weiss@nist.gov; dagistan.sahin@nist.gov; anil.gurgen@nist.gov; osman.celikten@nist.gov;
thomas.newton@nist.gov

INTRODUCTION

The National Bureau of Standards Reactor (NBSR) is located at the National Institute of Standards and Technology (NIST). NBSR is one of the five U.S. high-performance research reactors. It is a 20 MW, heavy water-cooled and moderated tank-type research reactor operational since 1967. The NBSR is one of six U.S. High Performance Research Reactors (USHPRRs) that are planned to convert from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel in cooperation with the National Nuclear Security Administration (NNSA) Office of Material Management and Minimization (M³) reactor conversion program.

The NBSR is planning on converting from the existing U_3O_8-Al dispersed HEU fuel plates to monolithic U-10Mo LEU fuel plates. The fuel plates adopt the curved material test reactor (MTR) type fuel plates with aluminum cladding encasing the fuel meat. Although an earlier version of a preliminary safety analysis report (PSAR)[1] for the NBSR conversion was completed in 2014, an updated version is expected to be delivered to the U.S. Nuclear Regulatory Commission (NRC) after the completion of the irradiation experiments for the demonstration LEU elements at Belgian Reactor-2 (BR-2).

This paper seeks to describe the milestones that have been reached and the future planned events, as well as the ongoing modeling efforts the NIST Center for Neutron Research (NCNR) in support of the LEU conversion. It is important to note that this conversion effort is being actively supported by multiple US DOE labs including Argonne National Laboratory, Brookhaven National Laboratory, and Idaho National Laboratory. Works from these other labs are being continuously published in regards to supporting the LEU conversion of the NBSR.

LEU FUEL ELEMENT DESIGN & MILESTONES

Throughout the previous fiscal year, three major milestones have been reached; (1) the completion and approval of the NBSR's LEU fuel specifications document, (2) the completion and approval of the NBSR's LEU fuel element drawings, and (3) the completion of the drawings and specifications for the NBSR design demonstration elements (DDEs). The DDE milestone was reached with efforts by the staff at Idaho National Laboratory, while the actual fuel element's specifications and drawings were completed by

NCNR staff with support from the team at Argonne National Laboratory.

The NBSR Fuel element consists of a hollow aluminum rectangular structure that houses 2 sets of 17 fuel plates, namely the upper and lower plates separated by a 7" thermal flux trap. The fuel element geometry is shown in Figure 1. The conversion seeks to replace the existing U_3O_8-Al disepersed HEU fuel plates (~93% enriched) with monolithic U-10Mo high assay LEU fuel plates (19.75% enriched).
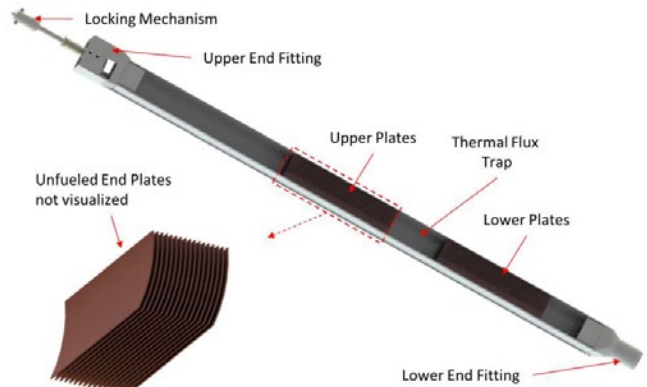

Figure 1. An illustration of an NBSR fuel element.

The LEU fuel element and plates maintain the same exact dimensions as the existing HEU fuel elements and plates, but it has a thinner nominal fuel thickness (per Figure 2). The HEU fuel meat spans 0.02" nominal thickness, whereas the LEU fuel meat spans only 0.0085" thickness with a zirconium interlay that is 0.001" on each side of the fuel meat. The increased uranium density that the monolithic U-10Mo offers allows for this decrement in nominal fuel meat thickness while maintaining performance that is adequate for normal operation in the NBSR.
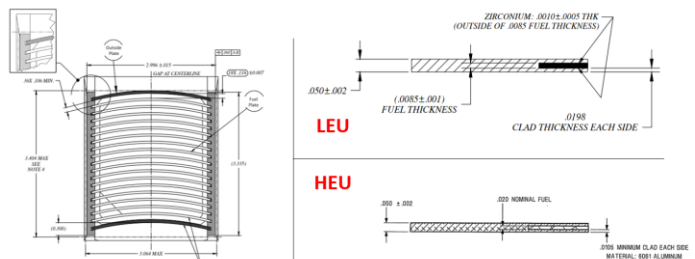

Figure 2. The NBSR fuel element drawing with LEU and HEU fuel meat thickness comparisons.

The fuel specification document was developed in agreeance with various industry standards, where modern standards were preferred. In particular, the document follows ASME NQA-1 (2008 with 2009 addenda) standards [2], while the drawings follow ASME Y14.5-2018 standards [3]. Note that the Y-12 impurities [4] found via downblending HEU to LEU are accounted for in the fuel specifications document. Table 1 presents a comparison between the HEU and LEU fuel plates characteristics.

Table 1. A comparison between HEU and LEU NBSR fuel plates.

Characteristic	HEU	LEU
Composition	Dispersed U_3O_8-Al	Monolithic U-10Mo
Fuel dimensions (cm)	27.94 × 6.134 × 0.0508	27.94 × 6.134 × 0.0216
U-235 (g)	10.31±0.02	11.31±1.13 [a]
Total Uranium (g)	11.07±0.22	57.28±5.73 [a]
Alloy material (g)	2.00	6.36
Dispersing material (g)	18.38	N/A
Total fuel core (g)	31.45	63.64
Cladding thickness (cm)	0.0381	0.0502
Cladding weight (g)	56.82	67.98
Interlayer material	N/A	Zr
Interlayer thickness (cm)	N/A	0.00254 cm
Interlayer weight (g)	N/A	2.83 g (×2)
Plate dimensions (cm) [b]	33.02 × 7.087 × 0.127	
Total plate weight (g)	88.26 g	137.29 g

[a] Bounding overload value calculated for LEU plates based on 10% overload for each plate.
[b] For curved plates, flat plate dimension (before forming) is provided.

ONGOING MODELING EFFORTS AT NIST

Monte Carlo N-Particle code (MCNP), version 6.2 is used for neutronics assessments of the LEU fuel in the NBSR core, and to estimate the spent future spent fuel shipment needs. Equilirbium core conditions are assumed for the analyses, and a 60-material model (for all fuel meats) are assumed for the analyses. A typical NBSR fuel cycle consists of a nominal 38.5-days of operation with at least 11.5-days outage, allowing for up to seven cycles per year. The cycle is descritized as shown in Table 2, which is used to track the behavior of the LEU fuel throughout the cycle in MCNP.

Table 2. The discretization of NBSR cycle states.

State	Start time (days)	Length (days)	Fraction of cycle
Startup (SU)	0.0000	1.5	0.039
Beginning of cycle (BOC)	1.5000	8.125	0.211
Quarter 2 (Q2)	9.6250	9.625	0.250
Middle of Cycle (MOC)	19.250	9.625	0.250
Quarter 4 (Q4)	28.875	9.625	0.250
End of cycle (EOC)	38.500	0.0	0.000
Total		**38.5**	**1.000**

Analyses have been conducted to provide conservative results for the decay heat and radioactive source term, which serve as bounding cases for spent LEU fuel shipment activities. Consider case 1, which involves an 8th cycle lower fuel section (which typically has the highest burnup) in the current HEU core, which was analyzed using best-estimate power distributions at each cycle state in an equilibrium cycle. Case 2 considers the nominal U-235 and U loadings in the U-10Mo fuel, while case 3 considers 10% higher U-235 and U loadings in the U-10Mo fuel. The fuel plate uranium masses for each of the three cases are presented in Table 3.

Table 3. Fuel plate uranium masses for each of the three cases analyzed.

Mass (g)	Case 1: HEU (best estimate)	Case 2: LEU (conservative)	Case 3: LEU (bounding)
U-232	N/A	1.95E-06	2.14E-06
U-234	1.84	2.53	2.78
U-235	175.30	192.32	211.55
U-236	0.85	4.48	4.93
U-238	10.19	774.42	851.87

Case 1 used the best estimate for the power distribution in each cycle state (Table 2), while case 2 assumed uniform power throughout the core (i.e., ~333.33 kW per fuel plate), which conservatively overpredicts decay heat and activity in discharged fuel sections. Case 3 assumes that the fuel plates produce 25% more power than the uniform power distribution for all 8 cycles, or ~416.67 kW per fuel plate. The basis for the 25% higher power in case 3 is to meet the anticipated fission density limit of the U-10Mo fuel [5]. Figure 3 shows the fuel plate power for each case over the course eight 38.5-days cycles with 11.5-days decay (outage) period between each two cycles.
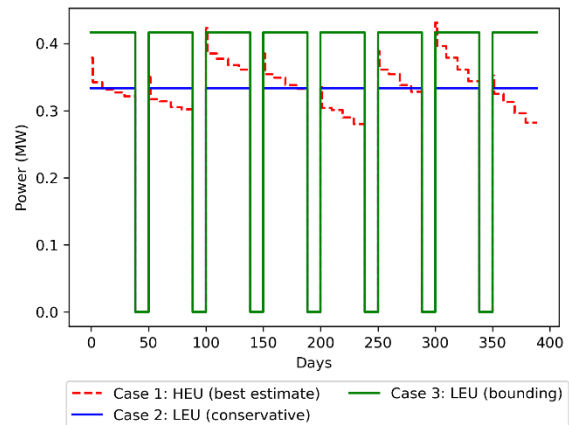

Figure 3. The fuel plate power throughout 8 full cycles for each of the three investigated cases.

The averaged neutron spectra for the LEU and HEU plates are presented in Figure 4 for multiple cycle sates. The neutron spectrum is nearly unchanged throughout the cycle for the LEU fuel, which is very similar to the HEU spectrum

if ever-so-slightly harder than than HEU. With the averaged LEU mean neutron spectrum obtained with MCNP (Figure 4), the decay heat, and isotopics of the spent U-10Mo fuel was computed for supporting the shipment analyses.
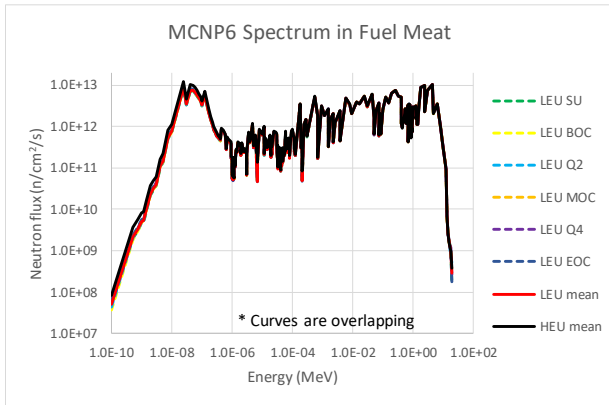

Figure 4. MCNP results for the neutron spectrum in the fuel meat of LEU and HEU plates.

The Oak Ridge Isotope GENeration code (ORIGEN) [6], part of the Standardized Computer Analyses for Licensing Evaluation (SCALE) code package [7], is used to compute the power of the spent fuel plates as well as their radionuclides production, depletion, and decay. The results for the decay heat and total activity for the fuel plates for 50-years following their removal from the core are shown in Figure 5 and Figure 6, respectively.
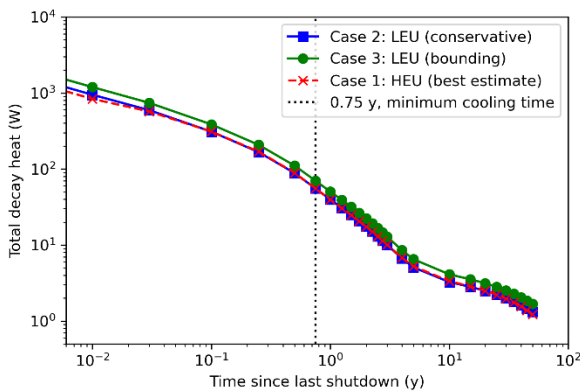

Figure 5. Decay heat (W) produced by the fuel plates for 50-years following removal from the NBSR.

In both figures, note the presence of a dotted vertical line at 0.75 years, which represents the minimum cooling time for any of the spent fuel elements. This is based on the fact that typical NBSR fuel has a minimum cooling time of ~280 days. Note how the power decays to a fraction of a kW within the 0.75 minimum cooling time, making it more appropriate for shipment with minimal heat removal needs. On the other hand, the activity profile (Figure 6) shows activities in excess of 10,000 Ci at around the minimum cooling time. This indicates the need for notable shielding efforts for shipping

the spent fuel elements, but not considerably more than the existing needs for the spent HEU fuel elements. Table 4 shows pre-and-post-irradiation data for the HEU and LEU fuel plates at minimum cooling time.
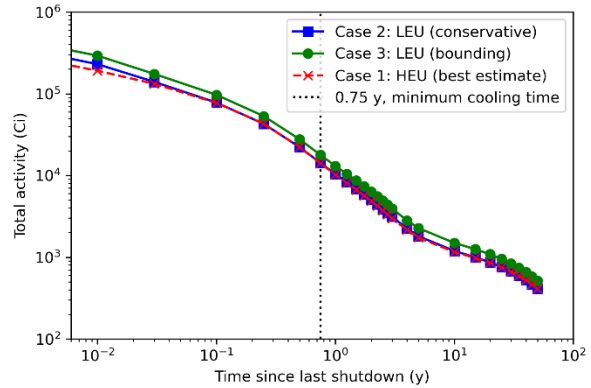

Figure 6. The total activity (Ci) of the fuel plates for 50-years following removal from the NBSR.

Table 4. The pre-and-post-irradiation characteristics for the fuel plates after the minimum cooling time.

Characteristics		HEU	LEU
Pre-irradiation mass (g)	U	188.2	1,071.1
	U-235	175.3	211.5
Post-irradiation mass (g)	U	76.78	916.8
	U-235	45.47	68.64
	U-236	20.36	26.47
	Np-237	0.76	1.07
	Pu	0.43	17.42
	Pu-239	0.22	12.10
	Pu-241	0.03	1.55
Full-power days in core		308	
Burnup (MWd)		103.1	128.3
Burnup (%)		74.61	67.55
Cooling Time (days)		274	274
Total Activity (Ci)		14,060	17,790
Total Decay Heat (W)		55.61	70.1

CONCLUSIONS

A status update on the NBSR LEU conversion is provided in this work alongside some of the ongoing computational work at the NCNR. MCNP and ORIGEN were used in-tandem to get an understanding of the spent fuel characteristics, which can help in planning the fuel shipment schedule when the conversion begins. It is anticipated that no spent LEU fuel element will need to be shipped out of the facility until 3.5-years after the first LEU fuel elements are inserted into the core. In the meantime, the remaining spent HEU fuel elements are anticipated to be shipped out every 5 quarters from the beginning of the LEU conversion (i.e., at quarter 5, then quarter 10). Future modeling works will begin developing coupled multiphysics models for both HEU and

LEU NBSR fuel elements as a means of providing in-house engineering tools for the staff to predict the behavior of the fuel elements throughout the conversion and beyond. Such efforts will also be used to accelerate future multiphysics modeling activities for the planned replacement reactor, the NIST Neutron Source (NNS) [8].

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DISCLAIMER

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REFERENCES

1. D. J. DIAMOND et al., "Conversion Preliminary Safety Analysis Report for the NIST Research Reactor," BNL-- 107265-2015-IR, 1169032, p. BNL--107265-2015-IR, 1169032 (2015); https://doi.org/10.2172/1169032.
2. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, "Quality Assurance Requirements for Nuclear Facility Applications: An American National Standard," ASME NQA-1, American Society of Mechanical Engineers (2013).
3. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, "Dimensioning and Tolerancing," ASME Y14.5, American Society of Mechanical Engineers (2013).
4. O. S. CELIKTEN and D. SAHIN, "The Effects of Impurities in Down-Blending Highly Enriched Uranium on the Reactor Neutronics and Cycle Length," presented at 2021 ANS Winter Meeting and Technology Expo, 2 December 2021, Washington, DC, ANS.
5. D. DIAMOND, *Irradiation Experiment Design Parameters for NBSR Fuel Conversion*, BNL-211807-2019-IR-R1, Brookhaven National Laboratory, Upton, NY (2020).
6. M. J. BELL, "ORIGEN: the ORNL isotope generation and depletion code," United States (1973); https://doi.org/10.2172/4480214.
7. W. WIESELQUIST and R. A. LEFEBVRE, "SCALE 6.3.1 User Manual," United States (2023); https://doi.org/10.2172/1959594.
8. D. SAHIN et al., "Pre-Conceptual Design of the NIST Neutron Source," 16 April 2023, The European Research Reactor Conference, Antwerp, BE.