A COMPARISON OF NEUTRON COINCIDENCE RATES DETECTED FROM SIMULATED MATERIALS TEST REACTOR FUEL ASSEMBLIES

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ABSTRACT
The Advanced Experimental Fuel Counter (AEFC) is a nondestructive assay (NDA) instrument used for active neutron coincidence measurements of spent research reactor fuel. The instrument performs active and passive scans of spent fuel assemblies to establish a relationship between the measured net active Doubles rate and the fissile mass (^{235}U, ^{239}Pu, and ^{241}Pu) present in the assembly. A calibration curve was previously established between net active Doubles and assembly fissile mass in highly enriched uranium (HEU) materials test reactor (MTR) fuel assemblies. However, many MTR-type reactors use low enriched uranium (LEU) instead of HEU. Simulations using the Monte Carlo N-Particle (MCNP) code were performed to determine if the calibration curve relationship for these HEU assemblies could be applied to LEU MTR assemblies. Fresh HEU and LEU MTR fuel assemblies were modeled in the fuel funnel of a benchmarked MCNP model of the AEFC. The results indicate that the HEU calibration curve will require a correction to accurately measure the fissile mass in the LEU assemblies. This result arises from changes in the system moderation due to the different geometries of the HEU and LEU assemblies as well as increased self-shielding in the LEU fuel.

INTRODUCTION
Civilian nuclear research reactors have been an integral part of developments in nuclear engineering around the world for many decades. Many of these reactors were initially designed for use with high enriched uranium (HEU) fuel. Concerns about the widespread use of HEU for civilian nuclear applications grew due to its potential use in nuclear weapons. These concerns resulted in a global effort to eliminate the use of HEU in research reactors. In 1978, the United States Department of Energy established the Reduced Enrichment for Research and Test Reactors Program (RERTR) in collaboration with the International Atomic Energy Agency (IAEA) with the goal of promoting and facilitating the conversion of research reactors to the use of low enriched uranium (LEU) fuel [1,2]. Under the RERTR program, research reactors around the world have successfully converted to the use of LEU fuel [1-3].

Spent research reactor fuel contains residual ^{235}U as well as Pu isotopes that build up during irradiation. Countries under Comprehensive Safeguards Agreements (CSAs) with the IAEA are required to declare this special fissionable material for verification purposes; however, spent nuclear fuel poses a significant challenge to safeguards measurements due to the high radioactivity and complex radiation profile of the irradiated fuel [4]. The Advanced Experimental Fuel Counter (AEFC) is a nondestructive assay (NDA) measurement tool developed for the quantification of residual fissile mass in spent research reactor fuel assemblies. The AEFC uses active neutron coincidence counting measurements to quantify the amount of residual fissile mass (^{235}U, ^{239}Pu, and ^{241}Pu) present in spent fuel assemblies. The instrument is used to perform active and passive neutron measurements of spent research reactor fuel assemblies at three different axial positions. During each measurement, the AEFC uses six ^{3}He tubes to record time-correlated neutron count rates, which are referred to as Doubles. The Doubles serve as a signature of nuclear fission, as the time-correlated nature of the neutron captures is indicative of neutrons that were born from the same fission chain. During active neutron measurements, a ^{252}Cf neutron source is placed in the AEFC to interrogate the spent fuel assembly and induce fission in the residual fissile material. However, the Doubles rates recorded during active measurements will also contain time-correlated neutron counts resulting from the spontaneous fission of higher-level actinides built up in the spent fuel during irradiation. To isolate the Doubles produced by induced fissions from those produced by spontaneous fission, passive neutron measurements are taken at each axial position. Measurements are also taken of a dummy assembly (with no fissile material) with the neutron source in the interrogation slot to account for time-correlated neutron counts produced by the interrogation source. The passive Doubles rate, the dummy Doubles rate, and the background Doubles rate are subtracted from the active Doubles rate to produce a net active Doubles rate at each axial position [5,6]. The net active Doubles rate represents Doubles produced only by induced fissions in residual fissile material.

Previous work performed using the AEFC has established a calibration curve between the net active Doubles rate and assembly fissile mass in HEU Materials Test Reactor (MTR)-type fuel assemblies [6]. This works seeks to use simulation to determine if the calibration curve developed for HEU MTR assemblies can also be used to quantify residual fissile mass in LEU MTR assemblies.

METHOD

MTR Fuel Assembly Models

Models of a fresh HEU MTR fuel assembly and a fresh LEU MTR fuel assembly were produced using the Monte Carlo N-Particle (MCNP) radiation transport code [7]. The assemblies have similar plate-type geometries but have slight variations such as the number of fuel plates and the fuel plate thickness. Relevant geometry and material parameters for each assembly are listed in Table 1.

Table 1. Parameters for MTR Fuel Assembly Models

Parameter	HEU	LEU
Fuel Form	UAl_x-Al	U_3O_8-Al
Number of Fuel Plates	23	19
Fuel Plate Thickness	0.51 cm	0.70 cm
Fuel Active Length	59.65 cm	80 cm
^{235}U Enrichment	93 w%	19.75 w%
^{235}U Mass	282 g	404.7 g
Fuel Meat Density	3.0 g cm^{-3}	4.8 g cm^{-3}

Simulation

The assembly models were placed inside of an MCNP model of the AEFC, and simulations were performed of active neutron measurements at three axial locations (TOP, MID, BOTTOM) with a ^{252}Cf spontaneous fission neutron source placed in the AEFC interrogation slot. At the MID position, the center of the fuel assembly was in line with the center of the AEFC's ^{3}He tubes. At the TOP position, the fuel assembly was shifted 18 cm down from the MID position so that the top portion of the fuel assembly could be measured. At the BOTTOM position, the fuel assembly was shifted 18 cm up from the MID position. This study used fresh fuel assembly models, so the axial profile of the fuel material was uniform; however, the three-point scan of the assemblies was conducted to mimic the standard measurement procedure of the AEFC [6]. Passive neutron simulations were not conducted for the fuel assemblies because no spontaneous fission was modeled in the fuel; since the purpose of passive measurements with the AEFC is to remove Doubles resulting from spontaneous fission, active neutron simulations that do not model spontaneous fission in the fuel can be used to represent the net active Doubles rate without performing passive simulations. The active neutron measurement simulations include time-correlated neutron counts resulting from the 252Cf spontaneous fission interrogation source. During physical measurements, these Doubles are accounted for by performing three-point axial measurements of a dummy assembly. No dummy assembly was simulated in this study, so the active simulations did not produce a true net active Doubles rate, which introduced bias into the simulation results.

A modified F8 tally was used to simulate the detection of time-correlated neutron captures on the ^{3}He in the AEFC's ^{3}He tubes. The F8 tally was modified using the FT8 CAP option, which modifies an F8 tally to track the number of captures in specific nuclides and is often used in the simulation of neutron coincidence detectors in MCNP [7]. The modified F8 capture tally reports the number of time-correlated neutron counts per source particle. The results of the modified F8 capture tally were recorded at each axial position for both fuel assemblies. To account for the different ^{235}U mass content in the different fuel assemblies (Table 1), the modified F8 capture tally results were divided by their respective assembly's ^{235}U mass. These values were compared between the HEU and LEU fuel assemblies to determine if the HEU calibration curve could be directly applied to an LEU MTR fuel assembly to accurately predict the residual fissile mass from a net active Doubles measurement with the AEFC.

RESULTS

The results of the modified F8 capture tallies are recorded at each axial measurement position for both fuel assemblies. These results are normalized by the ^{235}U mass of their respective assemblies to produce values of net active Doubles per gram of ^{235}U per source particle simulated. Normalizing the tally results by the ^{235}U mass allows for direct comparison of assemblies with different ^{235}U mass content. Assemblies that have a similar value of Doubles per gram of ^{235}U per source particle can be analyzed using the same calibration curve for residual fissile mass calculations. The normalized F8 tally results are compared by measuring the relative decrease in Doubles per gram of ^{235}U per source particle simulated observed in the LEU simulation with respect to the HEU simulation results.

Direct Comparison

Table 2. Comparison of Modified F8 Capture Tally Results

Position	HEU Capture Tally Result (Doubles g^{-1} $particle^{-1}$)	LEU Capture Tally Result (Doubles g^{-1} $particle^{-1}$)	Relative Decrease
TOP	$2.81 \cdot 10^{-6}$	$2.01 \cdot 10^{-6}$	28.34%
MID	$2.91 \cdot 10^{-6}$	$2.05 \cdot 10^{-6}$	29.30%
BOTTOM	$2.76 \cdot 10^{-6}$	$1.99 \cdot 10^{-6}$	27.96%
Total	$8.48 \cdot 10^{-6}$	$6.06 \cdot 10^{-6}$	28.55%

The results of the normalized F8 capture tally at each axial position for both fuel assembly models are recorded in Table 2. After summing the results of the TOP, MID, and BOTTOM measurements for both fuel assemblies, the LEU MTR assembly was observed to have a relative decrease of 28.55% in Doubles per gram of ^{235}U when compared with the HEU MTR assembly. These results indicate that the HEU calibration curve relating the measured net active Doubles rate to residual fissile mass cannot be directly applied to the LEU MTR assembly.

The difference in the Doubles per gram of ^{235}U values between the HEU and LEU assemblies can be explained by variations in the assembly geometries and fuel materials. The LEU fuel assembly has a longer active length than the HEU fuel assembly (Table 1), but it was measured at the same axial positions. The AEFC does not evenly interrogate and measure the entire length of a fuel assembly at a single measurement position; instead, most of the measured Doubles come from the length of the assembly directly adjacent to the ^{3}He tubes. The three-point measurement technique employed by the AEFC helps account for axial variations in residual fissile mass content, but it also establishes a set measurement length for all assemblies. Measuring the longer LEU fuel assembly at the same number of positions as the HEU fuel assembly prevents the AEFC from measuring the full active length of the LEU fuel. As a result, the value of the Doubles per gram of ^{235}U decreases.

Other factors that may be influencing the measured Doubles per gram of ^{235}U include the change in the cross-sectional geometry of the LEU fuel assembly and the changes in the fuel material. The LEU assembly has thicker fuel plates and fewer cooling channels than the HEU assembly, which could influence the neutron moderation in the system and affect the detection of neutrons. The LEU assembly could also demonstrate increased self-shielding due to its higher ^{238}U content.

Effect of Active Length

To account for the effect of variations in fuel assembly active length on the AEFC measurement technique, a correction factor was applied to the LEU MCNP simulation results. This correction factor was a ratio of the active length of the LEU fuel assembly to the active length of the HEU fuel assembly (Table 1). The results of the normalized F8 capture tally at each axial position for both fuel assembly models are recorded in Table 3.

Table 3. Comparison of MTR Fuel Assemblies Corrected for Active Length

Position	HEU Capture Tally Result (Doubles g^{-1} particle^{-1})	LEU Capture Tally Result (Doubles g^{-1} particle^{-1})	Relative Decrease
TOP	$2.81 \cdot 10^{-6}$	$2.70 \cdot 10^{-6}$	3.89%
MID	$2.91 \cdot 10^{-6}$	$2.75 \cdot 10^{-6}$	5.19%
BOTTOM	$2.76 \cdot 10^{-6}$	$2.67 \cdot 10^{-6}$	3.39%
Total	$8.48 \cdot 10^{-6}$	$8.12 \cdot 10^{-6}$	4.17%

After applying the active fuel length correction factor, the LEU MTR assembly was observed to have a relative decrease of 4.17% in Doubles per gram of ^{235}U when compared with the HEU MTR assembly. These results demonstrate much closer agreement in Doubles per gram of ^{235}U between the HEU and LEU assemblies, indicating that correcting AEFC measurements for variations in fuel assembly active length could allow for application of the HEU calibration curve to LEU net active Doubles measurements.

The simulations performed for this study use fresh fuel assemblies with no axial variation in fissile content, so the simple active length ratio correction factor can be applied. However, the simple correction factor would not be valid for irradiated fuel measurements because it does not account for the axial variation in residual fissile mass that is observed in spent fuel assemblies. Instead of using a correction factor, AEFC measurements could account for variations in assembly active length by performing measurements at a sufficient number of axial positions to interrogate and measure the full active length of the fuel material. Determining the necessary number of axial measurement positions requires an analysis of the effective fuel length measured by the AEFC at a single measurement position and careful selection of measurement positions such that the full length of the assembly is interrogated. Alternatively, separate calibration curves for different fuel assembly types could be used to accurately account for different active fuel lengths.

Effect of Cross-Sectional Geometry

The impact of the cross-sectional geometry on the MCNP simulation results was examined by swapping the HEU fuel material into the LEU fuel assembly geometry and performing the three-point measurement MCNP simulation. The impact of the fuel material type was examined by swapping the LEU fuel material into the HEU fuel assembly geometry and performing the same simulations.

Table 4. Modified F8 Tally Results for HEU Fuel Material in LEU Assembly Geometry

Position	LEU Geometry Capture Tally Result (Doubles g^{-1} particle^{-1})	Relative Decrease Compared to Standard HEU
TOP	$2.82 \cdot 10^{-6}$	0.48%
MID	$2.85 \cdot 10^{-6}$	1.94%
BOTTOM	$2.77 \cdot 10^{-6}$	0.11%
Total	$8.44 \cdot 10^{-6}$	0.47%

Table 4 contains the modified F8 capture tally results for MCNP simulations that swapped the HEU fuel material into the LEU assembly geometry. This simulation was performed to examine the effect of cross-sectional geometry variations on the value of measured Doubles per gram of ^{235}U. Because the simulation used the LEU fuel assembly geometry with the longer active fuel length, the active fuel length correction factor was applied to the modified F8 capture tally results. When compared with the results of the standard HEU fuel assembly simulations, the effect of the cross-sectional geometry was found to be 0.47%, which is much smaller than the effect of active fuel length.

Effect of Fuel Material Changes

Table 5. Modified F8 Tally Results for LEU Fuel Material in HEU Assembly Geometry

Position	LEU Fuel Material Capture Tally Result (Doubles g^{-1} $particle^{-1}$)	Relative Decrease Compared to Standard HEU
TOP	$2.73 \cdot 10^{-6}$	2.78%
MID	$2.87 \cdot 10^{-6}$	1.28%
BOTTOM	$2.73 \cdot 10^{-6}$	1.09%
Total	$8.33 \cdot 10^{-6}$	1.71%

Table 5 contains the modified F8 capture tally results for MCNP simulations that swapped the LEU fuel material into the HEU assembly geometry. This simulation was performed to examine the effect of fuel material changes on the value of measured Doubles per gram of ^{235}U. When compared with the tally results of the standard HEU fuel assembly simulations, swapping the fuel material was found to decrease the tally results by 1.71%. This decrease can be attributed to the increased ^{238}U content of the LEU fuel, which results in self-shielding due to resonance absorption of neutrons in the fuel. While the fuel material properties were found to have some effect on the measured tally results, the impact was much smaller than the effect of the active fuel lengths.

CONCLUSIONS

The calibration curve between the net active Doubles rate and an assembly's residual fissile mass cannot be directly translated from HEU MTR assemblies to LEU MTR assemblies. The LEU fuel assembly simulation results indicate that applying the HEU assembly measurement technique and calibration curve to an LEU assembly would significantly underpredict the residual fissile mass in the LEU fuel. The most significant cause of this underprediction was found to be the difference in active fuel length between the HEU and LEU fuel assembly models. Applying a simple correction factor for the fuel active length was found to reduce the relative difference between the fresh HEU and fresh LEU Doubles per gram of ^{235}U values from 28.55% to 4.17%. Other potential causes for the difference in the HEU and LEU simulation results were shown to have a much smaller effect compared to the difference resulting from the active fuel length. If MTR fuel assembly measurements with the AEFC can be accurately corrected for differences in active fuel length, it may be possible to apply the same calibration curve to assemblies with different initial ^{235}U enrichments. However, performing this correction would require additional work for irradiated fuel assemblies with axial variations in residual fissile mass.

Future work will consider methods for modifying the AEFC measurement technique for fuel assemblies with different active fuel lengths. The effective length of fuel interrogated and measured by the AEFC at a single measurement position will be characterized. This will allow for optimization of fuel measurement positions in the AEFC to ensure that the full active fuel length is interrogated and measured for any assembly.

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