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LEU CONVERSION ACTIVITIES AT THE MIT RESEARCH REACTOR: USE OF NEUTRONIC MODELS FOR SAFETY ANALYSES

T.H. Newton, Jr., N. Horelik, P. Romano, B. Forget, and E. E. Pilat
MIT Nuclear Reactor Laboratory and Nuclear Science and Engineering Department
138 Albany St, Cambridge, MA 02139 – USA

and

E. H. Wilson, B. Dionne, A. Bergeron, and J. Stevens
GTRI Program, Nuclear Engineering Division
Argonne National Laboratory
9700 South Cass Ave, Argonne, IL 60430 – USA

ABSTRACT

As part of the LEU conversion program at the MIT Nuclear Reactor Laboratory, existing neutronic models have been benchmarked and in some cases modified to improve dimensional representation for use in preparing safety analyses. Core models using fresh UAl_x HEU (MITR original core configurations operated in 1976) and fresh monolithic U-10Mo LEU were used to compare MCNP and REBUS-DIF3D calculations of reactivity, control blade worth and other neutronic parameters given in the MITR Safety Analysis Report and Startup report. Burnup models using the ORIGEN-MCNP coupling code MCODE as well as REBUS-DIF3D and REBUS-MCNP were used to compare parameters of recent core configurations using the actual operating history and refueling movements of 13 MITR core configurations in the period 2007-2009. Model development has included geometrical improvements to the 1976 and modern core representations, as well as conversion to ENDF-BVII libraries with isotopic definition for HEU and LEU cores. Significant progress has been made modeling the reactor physics parameters of the 1976 HEU cores, modern MITR HEU cores, and LEU cores.

1. Introduction

The MIT Reactor (MITR-II) core is a hexagonal design that contains twenty-seven fuel positions in three radial rings (A, B, and C), as shown in Figure 1. The reactor is currently licensed to operate at 5 MW, with an upgrade to 6 MW expected later this year. Typically at least three of these positions (two in the A-ring) are filled with either an in-core experimental facility or a solid aluminum dummy element to reduce power peaking. The remaining positions are filled with standard MITR-II fuel elements. Each rhomboid-shaped fuel element contains fifteen aluminum-clad fuel plates using HEU (93% enriched) in an aluminide cermet matrix with a fuel thickness of 0.76 mm (0.030 in.) and a length of 61 cm (24 inches). The cladding of each fuel plate is machined with 0.25 mm longitudinal fins to increase heat transfer to the coolant. The fuel has an overall density of 3.7 g/cm^3, with a total loading of 506 g ^{235}U in each element.

The core is light water moderated and cooled and is surrounded by a D_2O reflector. Boron impregnated stainless steel control blades are located at the periphery of the core on each of the sides of the hexagon. In addition, fixed absorbers can be installed in the upper axial region of the core in a hexagonal configuration between the A and B rings as well as in three radial arms extending to the edge of the core.

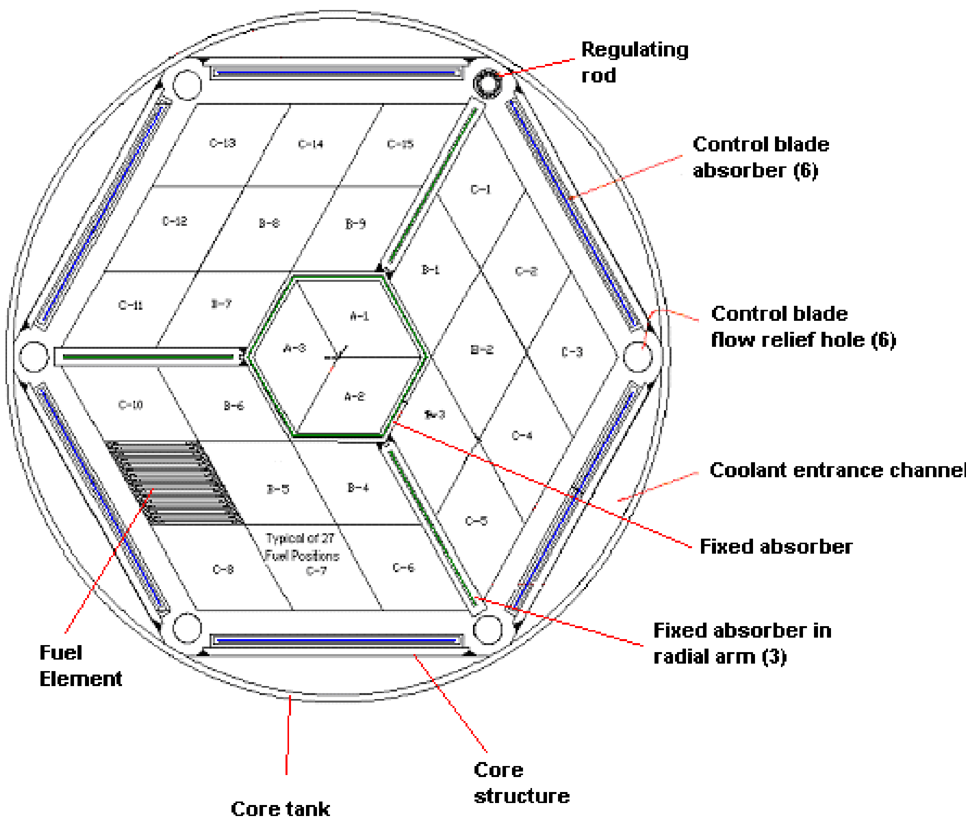

Figure 1. Layout of the MIT Reactor core.

2. LEU Fuel Design

LEU fuel optimization [1] performed for the MIT Reactor has shown that fuels with a density of at least 14 g/cm^3 would be required to reduce the enrichment to 19.75%. Because of this, monolithic U-10Mo LEU fuel with a density of 17.02 g/cm^3 was chosen for the design. This fuel is currently under evaluation by the RERTR program and is expected to be qualified for use within the next four years.

A feasibility study recently completed [2] has resulted in a design of a U-10Mo LEU fuel element containing 18 plates with 0.508 mm thick fuel with 0.25 mm finned aluminum cladding. This analysis shows that an equivalent 6 MW HEU experimental neutron flux can be generated at an LEU reactor power of 7 MW. Sufficient margins to onset of nucleate boiling are also met with the LEU core operating at this power level.

3. Neutronic Modeling

A number of neutronic models have been made for the MIT reactor. The Monte Carlo code MCNP has been used for many evaluations of HEU and LEU core and experiment design studies. The basic reactor design and fuel structure has also been input into the MCNP-ORIGEN linkage code MCODE for fuel management and burnup evaluations. A graphical user interface (GUI) has been designed and built into the MCODE model in a version called MCODE-FM. [3] A criticality search algorithm can also be utilized so that the control blade motion can be modelled. The number of axial nodes as well as fuel plate grouping can also be varied, depending on resolution and computational time needed. The GUI includes the ability to model all aspects of fuel management at the MIT reactor, including fuel flipping, rotating and storage for later use. The latter capability includes an ORIGEN calculation of decay and tracking of relevant radioisotopes, including actinides. Output of MCODE-FM now includes the ability to track and plot the number densities of any relevant isotope, as well as power peaking, on a per node, per plate, or per assembly basis.

A REBUS-DIF3D triangular-Z matrix model of the MITR has been developed at ANL. This model can use either a DIF-3D (diffusion theory) or a MCNP (Monte Carlo) routine to perform neutronic and burnup studies. [4] The REBUS model uses twelve axial nodes, including six in the fueled region. Cross-section generation for the DIF3D routine is made using a WIMS-ANL single dimension model of the MIT reactor and fuel.

Both the MCNP and REBUS-MC models have been reviewed and updated using ENDFB-VII cross-section libraries. Figure 2 shows a burnup calculation of fresh HEU fuel without fuel movement using the three models. ENDFB-VII was used for all cross-sections with the exception of the scattering kernel, which used ENDFB-VI, since ENDFB-VII was not available for REBUS. Six axial fuel nodes were used in the MCODE model to match the REBUS model.

The curves show excellent agreement, with the MCODE and REBUS-MC K_{eff}s being virtually indistinguishable, with any error between the two being primarily statistical. Although slight, the REBUS-DIF3D curve shows some divergence from the MCNP-based codes. The ENDFB-VI generated cross-sections are the likely source of the divergence since much better agreement is observed between REBUS-DIF3D and REBUS-MC when both use ENDFB-VI. Further comparison may be performed if ENDFB-VII libraries become available for use in REBUS-DIF3D calculations.

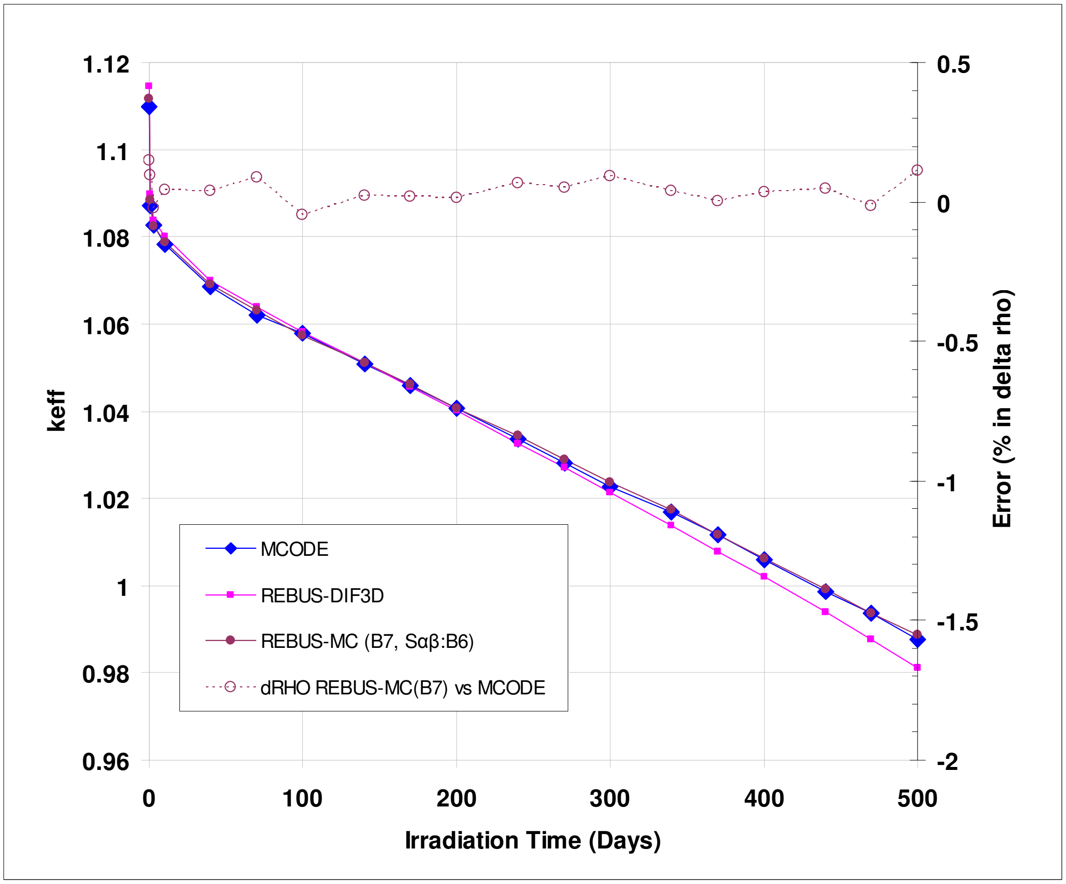

Figure 2. Burnup of fresh HEU fuel in the MIT Reactor.

4. Refueling Modeling

In order to better model actual and recent core conditions, MITR core configuration no. 178 was chosen as being a fairly typical representation. This core was operated for two months in early 2007. Starting with fresh HEU, a “rundown” burnup calculation was made for fuel assemblies in each core position, so that when U-235 masses matched the masses calculated for the end of core 178 by the existing MITR fuel management code CITATION, the fuel number densities were input into the model at that position. This was performed using both the MCODE and REBUS-DIF3D models.

Actual refueling movements and operating cycles were then modelled for the next twelve core configurations, ending with core no. 190, which ran through May of 2009. The movements throughout included nineteen fuel flips, the introduction of twelve new HEU fuel assemblies, and reintroduction of fourteen fuel assemblies from storage.

The MCODE model was run for these configurations both with the criticality search algorithm and with all control blades out. Figure 3 shows the MCODE calculated critical positions for each of the cores as compared with the actual measured values. The curves show the same trends, with about a 1 cm constant bias of the MCODE values being slightly lower. The cause of this is under investigation.

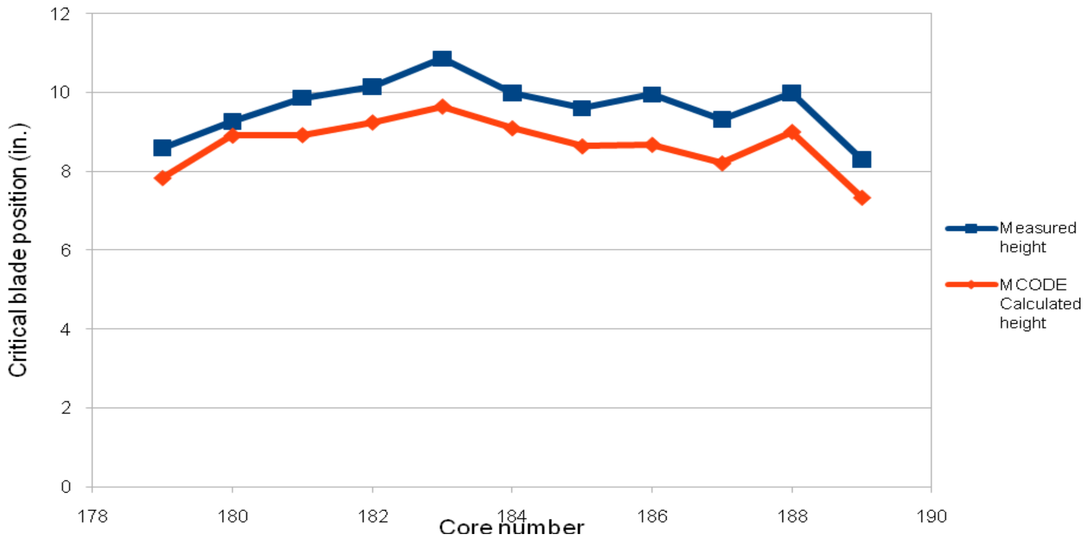

Figure 3. Critical control blade positions measured and calculated for core configs. 179-190.

Comparisons between the MCODE and REBUS-DIF3D models for the beginning of each core configuration with all blades out is shown in Figure 4. The K_{eff} of the two models are close, with the MCODE values being larger in a few cases. The trending of the two k_{eff} curves are generally the same. In addition, the beginning of cycle control blade position in the REBUS-DIF3D model were placed at the MCODE calculated position, K_{eff} calculated and is also shown in Figure 4. This shows the REBUS-DIF3D K_{eff} values in all cores to be about 1% lower that the MCODE values. The relatively constant bias, currently under investigation, is possibly due to diffusion theory modeling of the control blades which may be modified in future work to account for the actual flux shape in the vicinity of a black absorber. End of core calculations show similar trends.

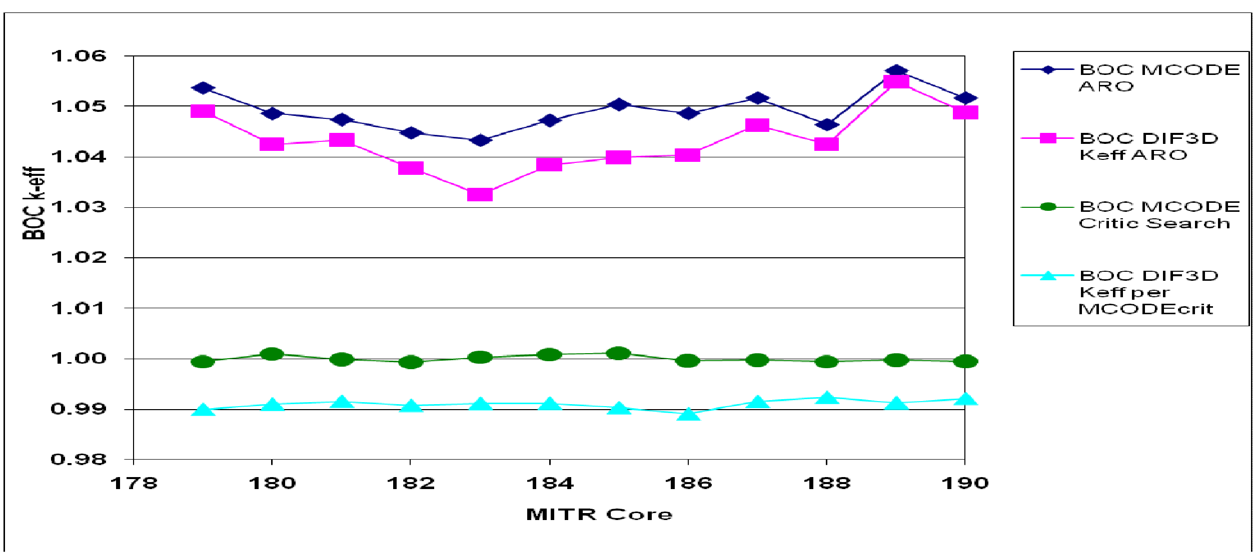

Figure 4. Beginning of core MCODE and REBUS-DIF3D models with all control blades out and at MCODE calculated critical positions

5. HEU Model Development and Start-up Test Comparisons

In preparation for comprehensive safety analysis calculations, the MCNP model used for feasibility and scoping studies of the MITR reactor was reviewed. Key areas of reactor model geometry were refined to represent the as-built dimensional specifications, including geometry and material representations of the fuel elements, dummy elements, and structural internals of the reactor. Table 1 lists key areas where the as-built specifications allowed improved model definition.

Table 1. Key areas of MCNP model refinement with as-built specifications.

I.	**Fuel Elements**
	- Refined element fuel and clad dimensions - Re-defined end fittings
II.	**Non-fuel Elements**
	- Non-fuel dummy element dimensions and inclusion of end-channel insets - Re-defined end fittings - In-Core Sample Assembly (ICSA) definition
III.	**Internal Core Geometry**
	- Fuel element pitch - Redefined core 'spider' internal structures, inserts and water gaps - Water gaps added between spider and elements as-built - Core housing definition including water holes and mating of spider to housing - Refined water regions zone definitions of light and heavy water

In conjunction with these refinements, an MCNP model was generated to revisit the comparison [5] to historical MITR-II reactor physics measurements [6-7]. Over the course of start-up in 1975-1976, MITR-II underwent four major core configurations and extensive internal modifications. In order to further benchmark the MITR MCNP model, core configurations were generated for the historical cores of interest. Table 2 summarizes major differences between the modern 2010 cores and the MCNP historical core models.

Table 2. Comparison of modern and historical core configurations modelled in MCNP.

MITR HEU Core Configurations	Core 1	Core 2	Core 4	2010
Fueled Elements	24	22	23	24
U-235 per element	445g	445g	445g	508g
Control Blades	Cd	Cd	Borated Steel	Borated Steel
Core Internals	Spider with Cd Inserts	Spider without Inserts	New Spider	New Spider with Inserts
BOC burnup (MWd/kg)	0	2	24	equilibrium

Agreement of the refined MCNP model shows excellent k_{eff} agreement with experimental criticalities over a wide range of critical control blade positions and modelled core configurations for fresh, or nearly fresh, historic cores 1 and 2 as summarized in Table 3.

Table 3. Representative results from k_{eff} validation of historical MITR cores.

Core Configuration	Critical bank & regulating rod (inches withdrawn)	Measured k_{eff}	Calculated k_{eff}	1-sigma (pcm)
Core 1	7.36	1	0.99700	4
Core 2	8.3	1	1.00206	4

Experimental measurements of control blade calibrations were compared to historical worth curves for core 2. Experimental records of control blade movements were modelled in MCNP in order to benchmark blade worth of individual control blades across a range of individual blade positions ranging from full-in to 21-inches withdrawn (full-out). Throughout the range of blade calibrations compared to experimental data, the MCNP model performed robustly. As an example, Figure 5 compares the calibration of core 2 cadmium control blade 3, where modelled results agree within 0.2% over the full 2% reactivity range of blade worth. Throughout the range of critical cases, the MITR MCNP model shows steady and low bias. In the case of blade 3 the standard deviation of the 0.2% bias is less than 30 pcm for the 10 criticals modelled across the 21-inch blade travel. Table 4 summarizes results for three experimentally measured core 2 blades.

Modeling of all historical cores has been performed assuming fresh fuel composition as has been done previously for cores 1, 2, and 4 [5]. Additional comparison of start-up measurements for cores 1 and 2 were performed including the modeling of core 2 reactivity coefficient of temperature. As shown in Table 5, the measurements of isothermal combined light and heavy water temperature reactivity agree with MCNP calculations within the somewhat large measurement uncertainties for comparable temperature ranges, such as seen in the core 4 experiments.

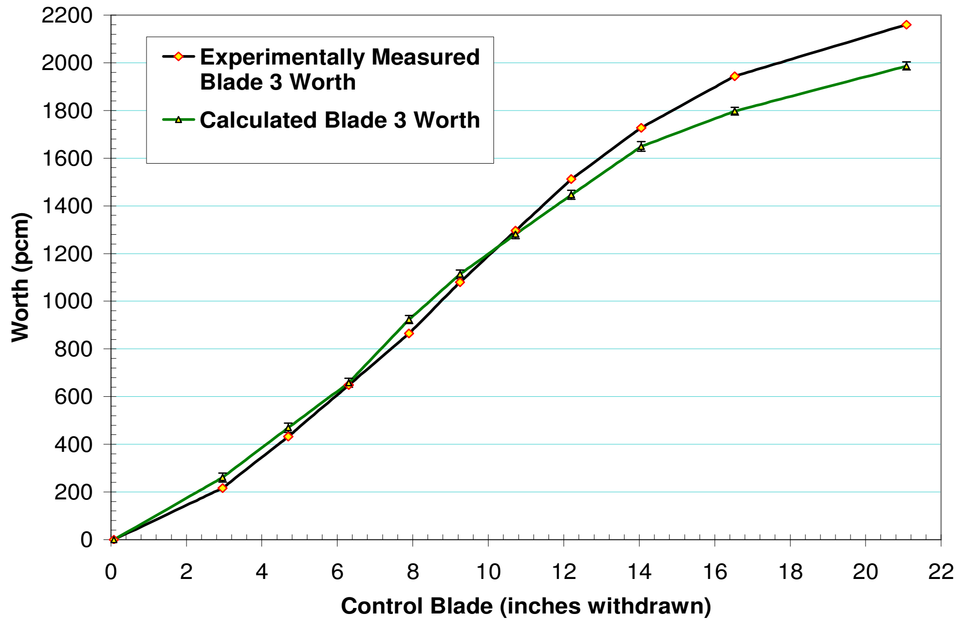

Figure 5. Comparison of core 2 Cd control blade 3 worth measurements to MCNP calculations.

Table 4. Representative results from k_{eff} validation of historical MITR cores.

Calibrated control blade	Number of critical cases	MCNP deviation from critical (pcm)	Std. Dev. 1-σ (pcm)	sum of calculated worth (pcm)	measured worth (pcm)	C-E (pcm)
1	4	170	54	2031	2222	-191
3	10	242	29	1993	2161	-168
5	11	244	44	2093	2377	-284
Summary	**25**	**228**	**48**	**2039**	**2253**	**-214**

Table 5. Comparison of experimentally measured isothermal temperature reactivity coefficients to calculations.

HEU Core	Temperature Reactivity Coefficient	Measured (pcm/ºC)	Experimental Uncertainty Range (pcm/ºC)	Calculated in MCNP (20ºC → 77ºC*) (pcm/ºC)	Uncertainty 1-sigma (pcm/ºC)
Core 1	35 → 45ºC	-5.1	n/a	-10.9	+/-0.2
Core 2	31 → 32 ºC	-2.0	n/a	-9.4	+/-0.2
	37 → 38ºC	-4.6			
Core 4	20 → 30ºC	-2.4	+/-2.4	-8.5	+/-0.1
	30 → 40ºC	-7.1	+/-2.4		
	40 → 50ºC	-11.8	+/-2.4		

* Note that at 5 MW, the core mixed mean coolant temperature is 48ºC.

6. LEU Modeling and Start-up Test Comparisons

MCNP models of the 18 plate LEU fuel design [2] have be used to compare the LEU fuel reactor physics parameters with the HEU values above. Tables 6 and 7 compare calculated HEU and LEU parameters to a wide variety of experimental data obtained with the historic HEU cores. Values show good agreement between experimental measurement and MCNP calculations. The calculated LEU values are also fairly close to HEU values. Because the LEU spectrum is harder than HEU, the temperature and void coefficients, as well as the calculated blade worths, are smaller.

The MIT Reactor will likely be the initial test case for the conversion to monolithic U-Mo LEU fuel. Instead of beginning with a complete fresh LEU core, MIT plans to introduce LEU fuel into the existing HEU core with a gradual conversion to all LEU fuel. In this manner, the LEU fuel can be evaluated within a known envelope and the fuel can be replaced with existing HEU fuel should problems occur. Neutronic and thermal-hydraulic evaluations have been made for this transition [8] and conclude that safety margins can be met throughout the conversion process.

Table 6. Comparison of HEU & LEU calculations to experimentally measured HEU core 2 data.

Comparison of HEU and LEU MCNP calculations to HEU core 2 experiments	**HEU core 2**		**LEU**
	Experiment	**MCNP**	**MCNP**
First critical shim bank position	8.3" (21.1cm)	8.3" (21.1cm)	8.17" (20.75cm)
k-effective	1	1.00206 ± 4pcm	1.00001 ± 3pcm
β_{eff} no photoneutrons (pcm) calc. σ < 6pcm	786	764	755
neutron lifetime (μs), calculation σ < 0.7 μs	100	78.1	60.5
Temperature coefficient (pcm/ºC) calculation σ < 0.1 pcm/ºC	-4.6 (37ºC)	-8.5 (20ºC→77ºC)	-6.1 (20ºC→77ºC)
Shutdown Margin (%Δk/k) assuming one of six blades & regulating rod are full-out (calculation σ < 0.04%)			
Control blade 1 and regulating rod out	-	-3.27	-2.84
Control blade 2 and regulating rod out	-	-3.41	-2.86
Control blade 3 and regulating rod out	-	-3.43	-2.98
Control blade 4 and regulating rod out	-	-3.52	-2.95
Control blade 5 and regulating rod out	-	-3.43	-2.94
Control blade 6 and regulating rod out	-	-3.33	-2.78
Control blade worth LEU calculated from critical moving one blade full in to out (calculation σ < 40pcm)			
Worth of control blade 1 (pcm)	2222	2031	1727
Worth of control blade 3 (pcm)	2161	1993	1634
Worth of control blade 5 (pcm)	2377	2093	1643
Dummy element worth Evaluated by inserting a dummy into unoccupied water-filled position (calculation σ < 15pcm)			
Dummy worth in A2-ring (pcm)	-	-	-531
Dummy worth in A3-ring (pcm)	-	-	-678
Dummy worth in B3-ring (pcm)	-600	-578	-958
Dummy worth in B6-ring (pcm)	-600	-519	-939
Dummy worth in B9-ring (pcm)	-600 **	-1110	-1512

** B9 agrees well with 0.6% experimental value if A-ring dummy is moved to non-adjacent location.

Table 7. Comparison of HEU & LEU calculations to experimentally measured HEU core 1 data.

Comparison of HEU and LEU MCNP calculations to HEU core 1 experiments	**HEU core 1**		**LEU**
	Experiment	**MCNP**	**MCNP**
Worth of the D2O reflector			
(Drained 22.6" = 57.4cm from overflow position)		(calculation σ < 38 pcm)	
Reflector dump worth (pcm)	-6995	-7454	-6927
Void coefficient (Full Channel)			
(Insertion of five full Al plates in fuel element water channel)		(calculation σ < 0.06 pcm/cm^3)	
A-ring void coefficient (pcm/cm^3)	-2.14	-1.76	-1.43
B-ring void coefficient (pcm/cm^3)	-2.13	-1.82	-1.42
C-ring void coefficient (pcm/cm^3)	-1.26	-1.04	-0.75
Void coefficient (Bottom 6 inches of Channel)			
(Insertion of 5 partial Al plates in fuel element water channel)		(calculation σ < 0.17 pcm/cm^3)	
A-ring void coefficient (pcm/cm^3)	-2.73	-2.45	-1.24
B-ring void coefficient (pcm/cm^3)	-2.68	-2.40	-1.42
C-ring void coefficient (pcm/cm^3)	-1.47	-1.23	-0.75

7. Conclusions

Neutronic and burnup models developed for the MIT Reactor have been upgraded and show consistency with each other as well as with measured reactivity values, both during initial start-up testing as well as in recent core configurations. It is important that, as we move forward with conversion using monolithic U-Mo fuel, confidence and consistency in the neutronics models remain high, particularly since no measured values for the LEU fuel will exist before conversion.

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