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Modeling the MIT Reactor Neutronics for LEU Conversion Studies

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To investigate the possible use of high-density low enriched uranium (LEU) in the MIT Reactor, the core neutronic behavior has been modeled using the Monte Carlo based codes MCNP for statics and MCODE for burnup. These models have been validated using criticality, blade worth, and burnup reactivity measurement data from MITR-II Core #2. Preliminary studies indicate that the LEU novel Mo-U fuel U7Mo can provide the reactivity needed to operate the reactor for adequate life times. The LEU core design optimization to maximize the neutron flux is proceeding using these models.

KEYWORDS: MIT Reactor, LEU conversion, core modeling, MCODE, research reactors

1. Introduction

The Reduced Enrichment for Research and Test Reactors (RERTR) Program at Argonne National Laboratory is currently developing high-density low enriched fuel that may be suitable for use in the MIT Reactor. [1] This fuel is a monolithic U-Mo fuel with fuel density approaching 18 g/cm^3. In order to evaluate the possible use of this fuel in the MIT reactor, core models using the Monte-Carlo transport code MCNP have been developed. Depletion has also been modeled using MCODE, which couples MCNP with the ORIGEN point depletion code.

2. Reactor Description

The current MIT Reactor (MITR-II), which began operation in 1975, contains a hexagonal core that uses fuel elements in up to twenty-seven positions, as shown in Figure 1. Each rhomboid-shaped fuel element contains fifteen aluminum-clad fuel plates using 93% enriched uranium in an aluminide cermet matrix with a fuel thickness of 0.76 mm (0.030 in.) and a length of 610mm (24 inches). Each fuel plate contains 0.25 mm fins to increase heat transfer to the coolant. The fuel originally used had a fuel density of 3.4 g/cm^3, with a total loading of 445 g ^{235}U in each fuel element. Higher density fuel (3.7 g/cm^3) was used in cores after 1990 with a total fuel loading of 506 g ^{235}U per element.

The core is light water moderated and cooled and is surrounded by a D_2O reflector. Boron-stainless steel control blades are present at the periphery of the core at each of the sides of the hexagon. In addition, fixed absorbers of cadmium were originally installed in the upper twelve inches of the core in a hexagonal configuration between the inner and second fuel rings as well as in three radial arms extending to the edge of the core. These absorbers were removed after the first configuration because of

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swelling concerns. Fixed absorbers of boron-impregnated stainless steel were later installed in only the radial arms.

Several reentrant thimbles are installed inside the D_2O reflector, allowing a larger neutron flux to be delivered to the beam ports outside the core region. Beyond the D_2O reflector, a secondary reflector of graphite exists in which several horizontal and vertical facilities are present for thermal neutron irradiation facilities. In addition, the MITR Fission Converter Facility is installed outside the D_2O reflector. This facility contains ten or eleven partially spent MITR fuel elements for a delivery of a beam of primarily epithermal neutrons to the medical facility for use in Boron Neutron Capture Therapy. Figure 2 shows a view of the reactor and reflector regions.

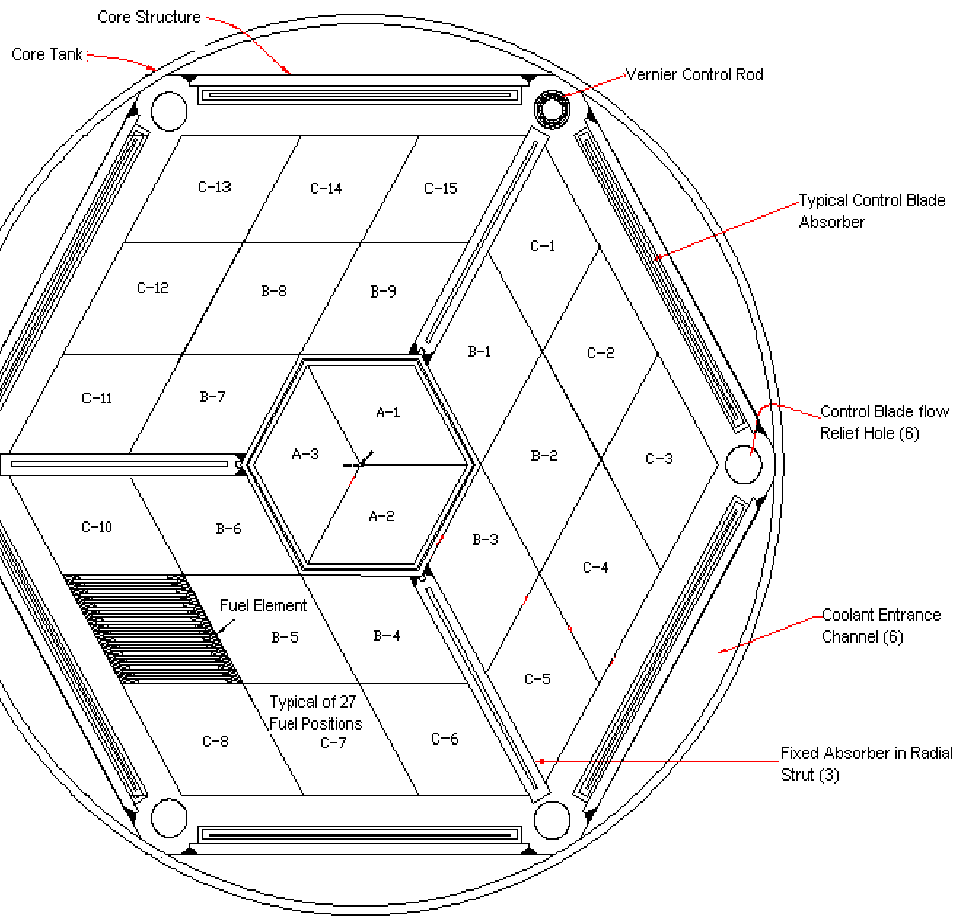

Fig. 1. The MITR-II core.

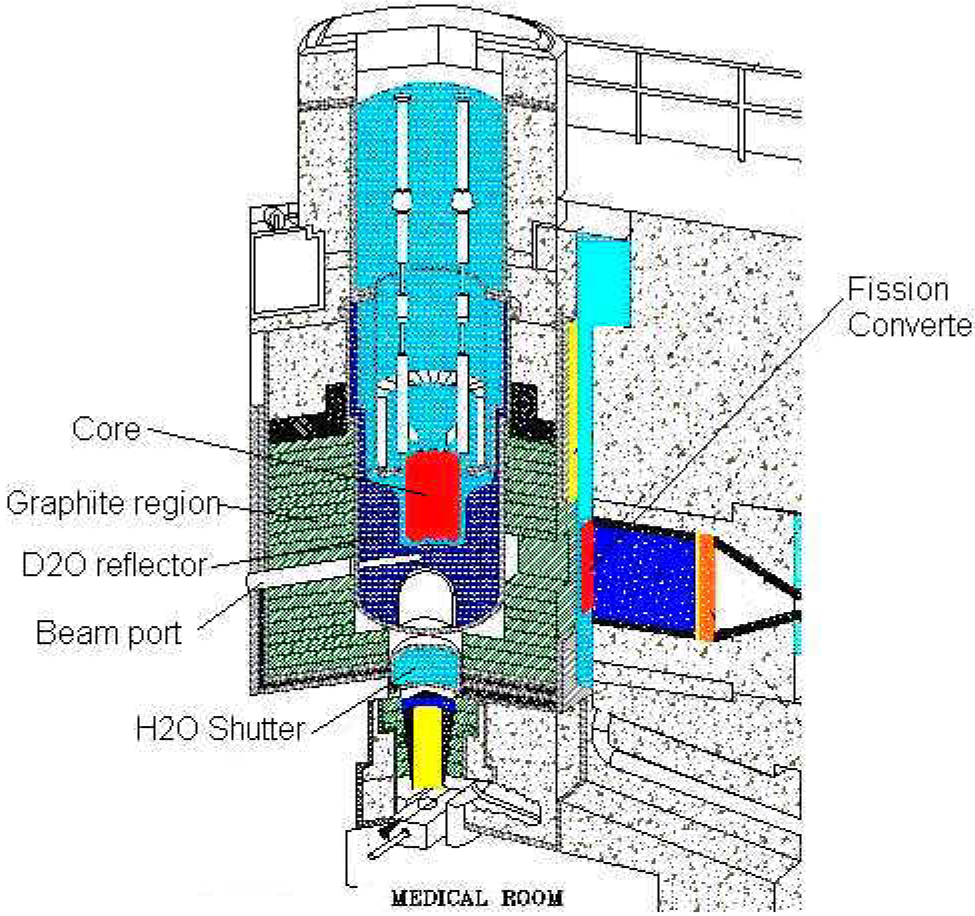

Fig. 2. The MIT Reactor

3. Models

3.1 MCNP Model

A model of the MITR-II using MCNP was first made by Redmond, Yanch and Harling[2]. This model was initially validated against measured values of K_{eff} and fast neutron flux. Each fuel plate is modeled discretely according to fuel specifications. However, manufacturing differences, such as the tapering of the ends of the fuel meat, are not modeled. All of the relevant reactor structures out to the outer edge of the graphite reflector are modeled explicitly. All cross-sections are evaluated at 300 K, since most reactor systems operate between 20^oC and 50^oC

For the LEU evaluation, the monolithic U-Mo fuel with a concentration of 7 w/o Mo (U7Mo) was chosen. [3] This fuel has a uranium density of 17.5 g/cm^3 and is currently under evaluation by the RERTR program at Argonne National Laboratory [1] for use in research reactors. For a fuel element of the exact dimensions of the HEU fuel, this would equate to a total ^{235}U loading of about 1200 g, over twice that of HEU. Therefore, it is possible to design the core with fewer plates of this type.

3.2 MCODE Model

For burnup calculations, the MCNP model was used with the MCNP-ORIGEN linkage code MCODE, developed by Xu, *et al*. [4]. Unlike other coupling codes, MCODE uses a predictor-corrector approach to burnup depletion, as opposed to single time step evaluations as in MONTEBURNS or MOCUP.

Because of the limitation on the number of points able to be tallied in MCNP (99), each individual plate could not be discretely depleted by the use of MCODE. Instead, each of the twenty-two fuel elements was initially modeled so that all fifteen plates in an element were assumed to have identical material compositions. A single axial zone was used for each plate. The materials specified in the MCNP model for cross-section and reaction rate generation included forty of the most reactive fission product isotopes and seventeen actinide isotopes, shown in Table 1. The remaining 520 fission product isotopes were generated using the ORIGEN libraries. The actinide isotopes generated by ORIGEN are also shown in Table 1.

MCNP depletion uncertainties come from the stochastic nature of the calculation. Even though at the beginning precise information about the core composition can be assumed, the statistical variations from calculating the reaction rates will be propagated into deterministic burnup calculations. However, MCNP depletion uncertainties from the statistical variations tend to cancel each other as burnup increases, in particular for a large number of neutron histories [5]. One possible reason might be that these statistical uncertainties are almost independent at different burnup points. As observed in practice, the burnup curve oscillates; and the magnitude of this oscillation implies the goodness of the MCNP reaction rates. In this calculation, the relative error (defined as one standard deviation divided by the average value) of U-235 fission reaction rate is typically less than 0.001. The k_{eff} oscillation is almost not visible.

Table 1 Isotopes used by MCNP and ORIGEN

MCNP Fission Products				MCNP Actinides		ORIGEN Actinides
Kr-83	Pd-105	Cs-135	Sm-147	Th-232	Np-237	Th-230
Zr-91	Pd-107	La-139	Sm-149	Pa-231	Np-238	U-237
Zr-93	Ag-109	Pr-141	Sm-150	Pa-233	Np-239	U-239
Mo-95	In-113	Nd-143	Sm-151	U-232	Pu-238	Np-240
Mo-97	I-129	Nd-144	Sm-152	U-233	Pu-239	Am-241
Mo-98	Xe-131	Nd-145	Eu-151	U-234	Pu-240	
Tc-99	Xe-133	Nd-148	Eu-153	U-235	Pu-241	
Ru-101	Xe-135	Pm-147	Eu-154	U-236	Pu-242	
Rh-103	Cs-133	Pm-148	Eu-155	U-238		
Rh-105	Cs-134	Pm-149	Gd-157			

4. Validation Methods

The core configuration chosen for validation studies is the second core configuration used after the reactor was redesigned in 1975. The reason this core configuration was chosen is because of low burnup in the initial core configuration (less than 2.5 MWd/kg), along with the absence of fixed absorbers in-core and a relatively large operating period (eight months) over which reactivity data was taken.

The MCNP model was validated by comparing the calculated K_{eff} at a blade height equivalent to the critical (K_{eff} =1) blade height initially used. As a further validation, the blade positions were varied in the model and the reactivity worths compared with measured blade worth curves. Initially, core #2 was operated with cadmium control blades. Because of swelling concerns, these blades were replaced with boron impregnated stainless steel blades. Both cadmium and boron blade worths were modeled and their reactivity worths compared. In order to keep uncertainties low, 3 x 10^6 to 10^7 particles were used in each MCNP run. This kept the K_{eff} standard error below 5 x 10^{-4} in all cases.

The MCODE model was run at a power density of 36 kW/l, which is equivalent to a reactor power level of 2.5 MW, the nominal power under which Core #2 was run. Burnup was calculated up to 20 MWD/kg, the point at which Core #2 ended. Burnup steps of 0.1, 2.7, 5, 10, 15 and 20 MWD/kg were used, with each step divided into five depletion substeps. Changes in K_{eff} (i.e. reactivity) with burnup were calculated and compared with measured reactivity changes (from blade position changes). In addition, in order to determine the effect of neutron spectra, the sensitivity of the burnup calculations to the choice of ORIGEN cross-section libraries was evaluated.

In order to optimize the design of the LEU core, the U7Mo fuel was modeled in place of the HEU fuel. However, the number and thickness of fuel plates as well as coolant material were varied to determine the K_{eff} and available neutron fluxes in each configuration.

5. Results

5.1 Validation Results

The initial critical blade height for core #2 was 21.6 cm (8.5 in.). At this blade height, the MCNP calculated K_{eff} was 0.990 +/- 0.00018. This value also matches values calculated by Redmond [1]. The slightly lower value is possibly due to the absence of fuel tapering in the MCNP model.

Reactivity values as compared with blades fully inserted were taken at blade positions at the operating ranges from the critical height to 34.3 cm (13.5 in.) and compared with measured reactivity worth curves. Results for both the cadmium and boron blades are shown in Figure 3. This shows good agreement between the model and actual measurements. In addition, very little reactivity difference can be seen between the two materials.

Burnup calculations from MCODE compared with measured reactivity data are shown in Figure 4. A normal operating schedule would consist of startup at the beginning of a week and run until the weekend. There were some cases in which the operating schedule did not allow for xenon poisoning to reach equilibrium. All values in Figure 4 were taken at the end of an operating week when xenon is most likely to be at or near equilibrium. This figure shows good agreement between measured and calculated values with non-equilibrium reactivity values obviously being smaller. The slight offset at the zero reactivity point is due to the 2.3 MWd/kg generated in Core #1. A burnup of 20 MWD/kg is equivalent to about a 4% burnup of ^{235}U, a point at which fuel reshuffling typically takes place.

The results of two vastly different ORIGEN cross-section libraries, that of a PWR library (PWRU) and that of an LMFBR (FFTFC) in the use of a theoretical LEU MITR-II core with 12 plate elements is shown in Figure 5. This shows no significant difference in reactivity values in library use for the fission product isotopes and actinide isotopes used by ORIGEN. Note however that the effective one-group cross sections used in the ORIGEN depletion for the most reactive isotopes are obtained directly from the MCNP calculations, and therefore reflect the local neutron spectra in the MITR-II core.

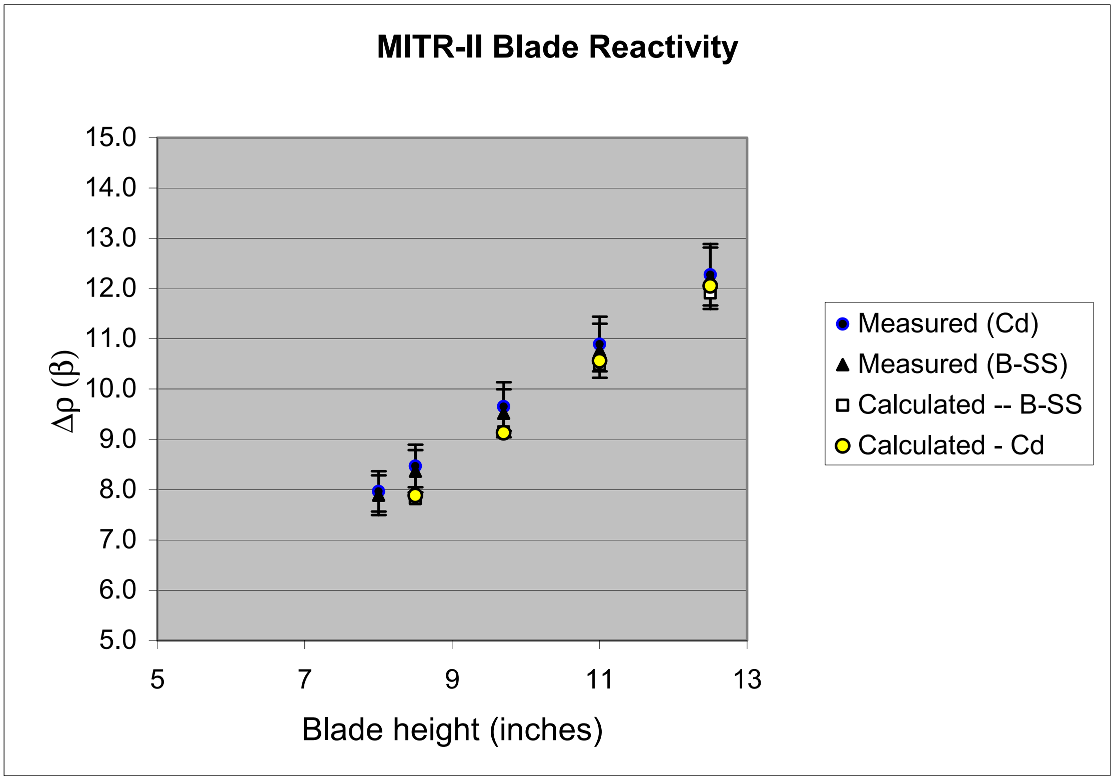

Fig. 3. Blade reactivity worths

5.2 LEU Fuel Studies

In optimizing a preliminary LEU core design, it is clear that increased moderation is needed to give an equivalent K_{eff} to that of the fifteen plate per element HEU core. This could be done by decreasing the fuel and cladding thickness or by reducing the number of plates in a fuel element. This, among other things, must also be done to increase the available thermal neutron flux for experiments using an LEU core. However, heat transfer concerns limit the possible plate number reductions. Preliminary studies indicate that an LEU core containing fifteen plate elements with a fuel thickness of 0.56 mm and a cladding thickness of 0.25 mm will give an equivalent end of cycle (20 MWD/kg) K_{eff} to that of the HEU core. However, the thermal flux available at experimental facilities for this core is about 20% less than that of the HEU core.

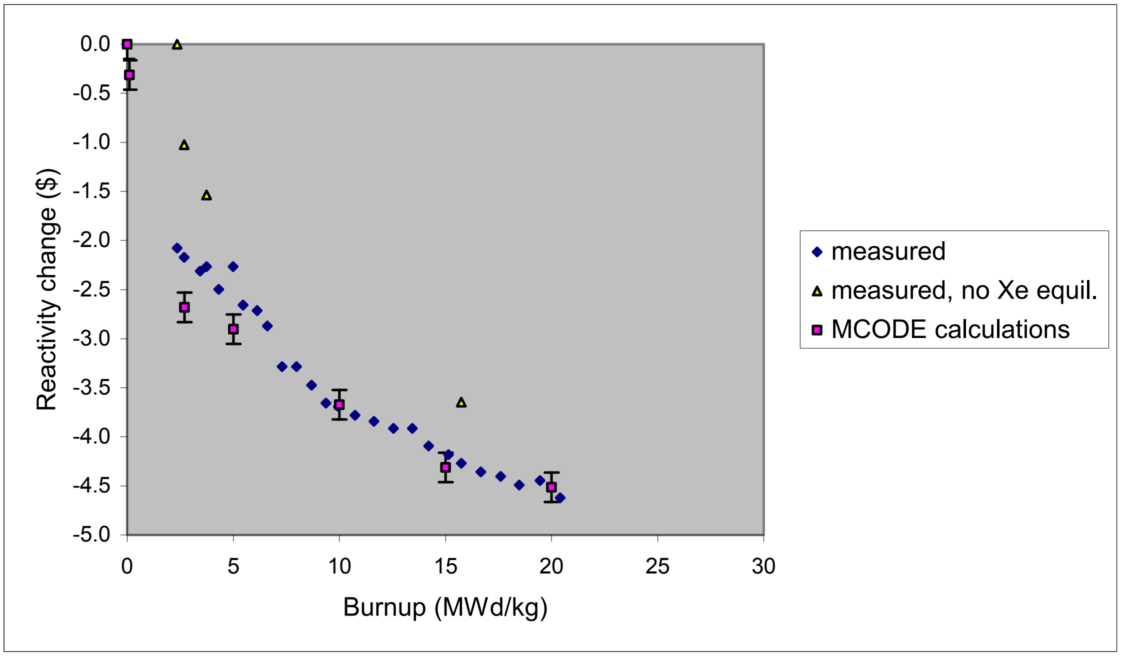

Fig. 4. Reactivity values of MITR-II Core #2

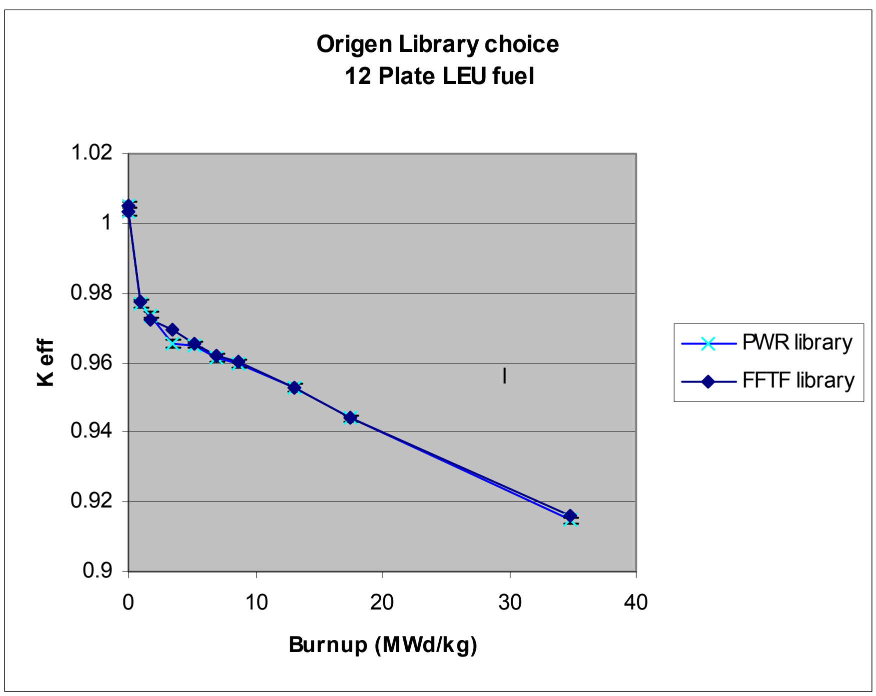

Fig. 5. Effect of ORIGEN cross-section library on K_{eff}

Various options are being explored to increase the neutron flux available to experiments. Table 2 shows a few selected options for fifteen plate LEU fuel elements. Although an HEU equivalent thermal flux in core can be obtained with a fewer number of plates per element, higher fluxes in the thermal irradiation facilities using an LEU core require a larger epithermal component in the core. Preliminary results indicate that this can be done by varying the plate thickness and changing the moderating characteristics of the primary coolant. The use of about 20% D_2O in the primary coolant appears to be optimum in balancing experimental facility flux increases with reactivity loss. Flux increases in fast neutron facilities will be made by individual facility design optimization.

Table 2. Comparison of selected fifteen plate LEU options

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	HEU	LEU (0.25 mm clad)		
Fuel thickness (mm)	0.762	0.381	0.330	0.330
Primary coolant % D_2O in H_2O	0	0	0	20
BOC K_{eff}	0.996	1.040	1.036	0.991
Beam port thermal flux (n/cm^2 s)	1.20E+13	9.74E+12	9.62E+12	1.13E+13

6. Conclusion

The MCNP and MCODE models for the MITR-II core have been developed and used to compare initial criticality values, control blade worths, and burnup for HEU core #2 against available measured data. These calculational models show good agreement with measurements, indicating their validity in use of data verification and proposed core modifications.

These models are also being used to develop a core design using monolithic LEU fuel. Preliminary studies indicate that LEU fuel can be used that will give equivalent K_{eff} values to an HEU core. Designs to increase the available neutron flux are being evaluated.

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