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Neutronic Analyses for HEU to LEU Fuel Conversion of the Massachusetts Institute of Technology MITR Reactor

Nuclear Engineering Division

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Neutronic Analyses for HEU to LEU Fuel Conversion of the Massachusetts Institute of Technology MITR Reactor

by
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Abstract

The Massachusetts Institute of Technology (MIT) reactor (MITR-II), based in Cambridge, Massachusetts, is a research reactor designed primarily for experiments using neutron beam and in-core irradiation facilities. It delivers a neutron flux comparable to current LWR power reactors in a compact 6 MW core using Highly Enriched Uranium (HEU) fuel.

In the framework of its non-proliferation policies, the international community presently aims to minimize the amount of nuclear material available that could be used for nuclear weapons. In this geopolitical context, most research and test reactors both domestic and international have started a program of conversion to the use of Low Enriched Uranium (LEU) fuel. A new type of LEU fuel based on a mixture of uranium and molybdenum (U-Mo) is expected to allow the conversion of compact high performance reactors like the MITR-II .

This report presents the results of steady state neutronic safety analyses for conversion of MITR-II from the use of HEU fuel to the use of U-Mo LEU fuel. The objective of this work was to demonstrate that the safety analyses meet current requirements for an LEU core replacement of MITR-II.

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I Introduction

This report describes the neutronic performance and safety analyses performed to study the MITR-II reactor conversion from the use of Highly Enriched Uranium fuel (HEU) to the use of a Low Enriched Uranium (LEU) fuel. Due to its compact core design and high power density, the MITR conversion has been studied using high-density LEU fuel. Currently, development of uranium – molybdenum (U-Mo) alloy monolithic fuel is under way which would provide a high density (17-18 g/cm^3) fuel to allow MITR conversion.

ANL and MITR have collaborated to perform neutronic analyses using LEU fuel according to the section 5 of chapter 4 of the MITR-II Safety Analysis Report [1]. This report discusses the results of steady state neutronic safety analyses for conversion of MITR-II from the use of HEU fuel to the use of U-Mo LEU fuel. The objective of this work was to demonstrate that the safety analyses meet current requirements for an LEU core replacement of MITR-II. Calculations presented in this report are the results of the neutronic model of MITR-II using MCNP5, with ENDF/B-VII cross section libraries used where possible. LEU calculations will be discussed and compared to HEU reference values from HEU modeling and experimentation. Details of HEU model validation and LEU model development are presented in reference [2].

II SAR Chapter 4.5: Nuclear Design

There are no foreseen impacts of an LEU fueled core which would alter this section.

Many research reactors function by defining a limiting core configuration as that core which yields the highest power density for the specified fuel. Other configurations are then allowed, provided that they are within the envelope of this limiting one. For the MITR, the limiting core would be the one with the maximum number of non-fueled positions because the power density in the fuel elements would be at its highest. Criticality considerations restrict MITR operation to a maximum of five non-fueled positions. Such a core was installed and operated following the initial startup of the MITR-II. A more routine configuration is three non-fueled positions. However, cores have also been installed that had either two or four non-fueled positions. It is anticipated that two to four non-fueled positions will be the standard configurations using LEU as well.

The power peaking factors used for the thermal-hydraulic analysis given in Section 4.6 of the HEU MITR SAR [1] are for the core with five non-fueled positions and hence are worst case. Limiting thermal-hydraulic conditions (e.g., SAR equations 4-26 and 4-37 for forced convection flow) are then developed as a function of the maximum power deposited in a fuel channel. Cores other than the limiting one are evaluated by verifying that these limiting core conditions are met. An analysis is then done of every fuel channel in a proposed core to be certain that these limiting conditions are not exceeded. If so, the proposed core is acceptable. The principal steps in the analysis are:

a) A three-dimensional model of the core is developed. The model specifies the physical layout of the core components (fuel, control devices, moderator, housing, reflectors, etc.), the material composition of each component (uranium, aluminum, water, boron, etc.), and the number density of each composition. Provisions exist to include xenon and other fission products.

b) A numerical code is used to obtain the k-effective, flux distribution, and power density distribution of the modeled core. The codes used for this purpose are ones obtained from the Radiation Safety Information Computational Center (RSICC at Oak Ridge). CITATION, which is based on diffusion theory, has historically been used at the MITR [3]. MCNP, a Monte Carlo code, is now being used [4]. Number densities of the partially depleted fuel assemblies are now based upon Monte Carlo flux solutions coupled with ORIGEN [5] or REBUS [6] transmutation models.

c) The proposed core thermal-hydraulic limits are evaluated using the power distribution for every fuel channel. If all channels pass, then the proposed core is acceptable.

The above approach is computationally intensive. However, it offers flexibility in terms of both in-core experiments and fuel utilization.

4.5.1 Normal Operating Conditions

4.5.1.1 Core Components

Core components are described in Section 4.2 of the MITR HEU SAR [1].

4.5.1.2 Planned Core Configurations

There are no foreseen impacts of an LEU fueled core which would alter this section.

There is no set of pre-planned core configurations for the MITR. Rather, any configuration is acceptable provided that certain criteria are met. These include:

a) Certain thermal-hydraulic parameters must not be exceeded for any channel in the core.
b) Each of the twenty-seven positions within the core must contain a fuel element, or a solid aluminum dummy, or an approved in-core sample assembly (ICSA). (Note: Also acceptable is a neutron source tube as discussed in Section 4.2.4 of the MITR HEU SAR [1].)
c) The shutdown margin requirement is met.
d) No fuel element, or any portion thereof, exceeds the fission density limit.

4.5.1.3 Reactor Operating Characteristics

There are no foreseen impacts of an LEU fueled core which would alter this section.

MITR fuel management policy is, in general, to place fresh fuel in the B-ring where peaking factors are lowest and to place partially-spent fuel in the A- and C-rings where peaking tends to be greater. (Refer to Figure 1.) The net result is that power is concentrated in the core interior when a core is newly installed. As the fuel depletes the average withdrawal of the shim bank increases. All six shim blades are located in the core periphery. Hence, as the bank rises, power shifts from the A- and B-rings to the C-ring. Detailed analysis of the power distributions of selected cores are available [3].

These studies have shown that as shim blades and/or fixed absorbers are raised, the following occurs:

a) The axial spatial location of the point of maximum flux will be higher.
b) The magnitude of this maximum will be decreased.
c) The magnitude of the thermal neutron flux available at the beam port reentrant thimble tips decreases.
d) Power density decreases in the A- and B-rings but rises in the C-ring.

These same studies have established that insertion of fresh fuel in a given position increases the power density in that position at the expense of neighboring positions, and that replacement of a solid dummy and/or experimental facility with fuel increases the power density in the neighboring elements.

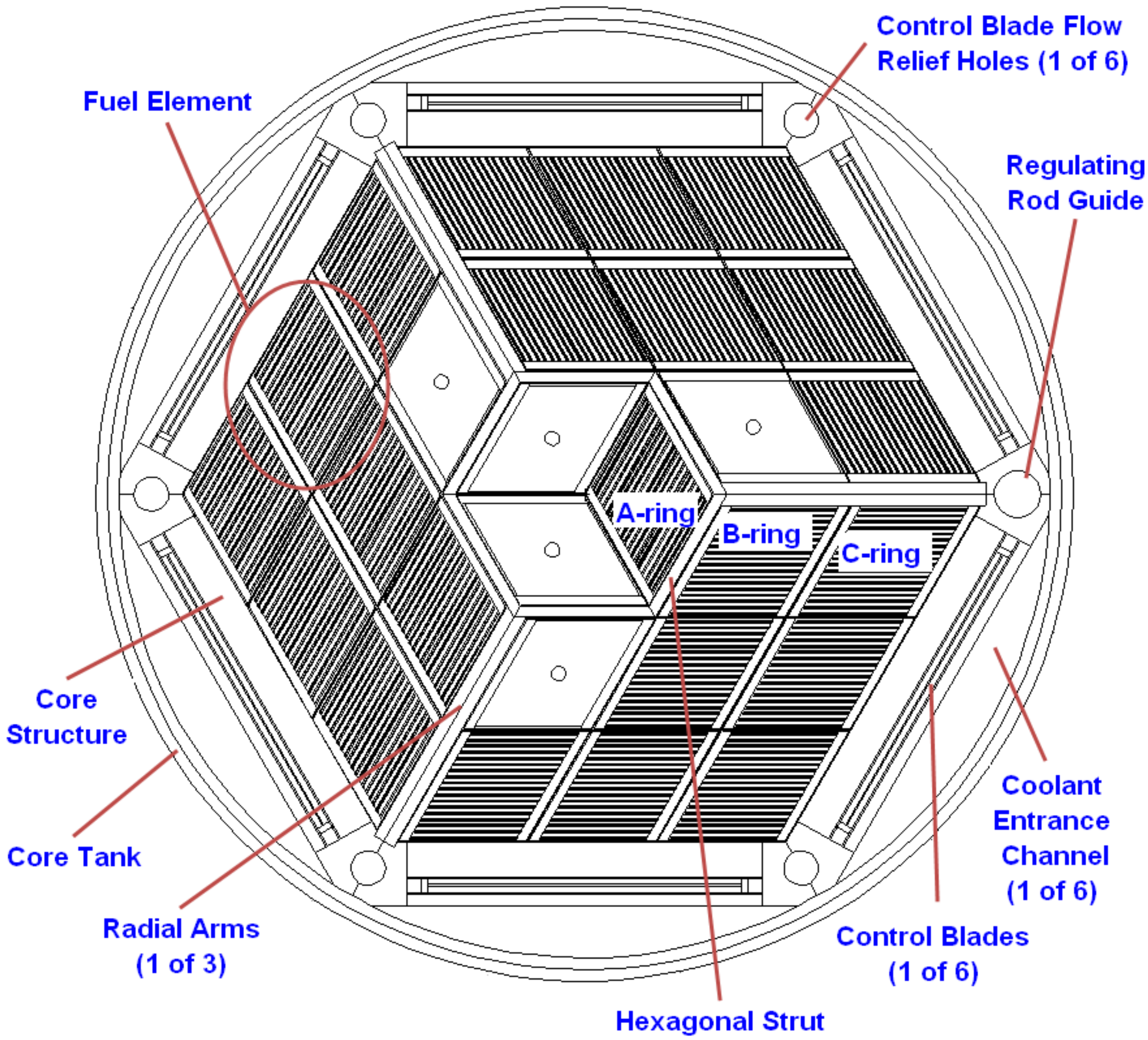

Figure 1. MITR-II core cross section.

4.5.1.4 Effect of Fuel Burnup

The MITR operates 24 hours per day, seven days per week. Hence, refuelings are frequent. These may involve any or all of the following:

- replacement of spent fuel with fresh or partially used fuel
- shuffling of fuel to achieve better overall burnup
- rotation (in the horizontal plane) of individual elements to offset the effect of radial flux gradients
- axial inversions (flipping) of individual elements to negate the effect of axial flux gradients.

Detailed analyses of these refueling strategies are available, and the effect of fuel depletion on reactivity has been quantified [3]. The change in HEU core reactivity with core energy production is -0.25 mbeta/MWH. LEU depletion effects will be discussed in more detail in a forthcoming Transition Core report.

4.5.1.5 Kinetic Behavior/Requirements and Features of Control Devices

There are six control blades which are identical in design and located at the same radial distance from the core center. The blades are distinguished from one another in the calculations made for this report as follows: as viewed from the top of the reactor, control blade #1 is the first blade clockwise from the regulating rod, and blade #2 is the next blade clockwise, so that blade #6 is opposite the regulating rod from control blade#1. Figures 2 and 3 show the integral reactivity worth of the MITR shim bank (six shim blades) and the regulating rod, respectively. These curves are similar in shape to the HEU control mechanisms found in reference [1].

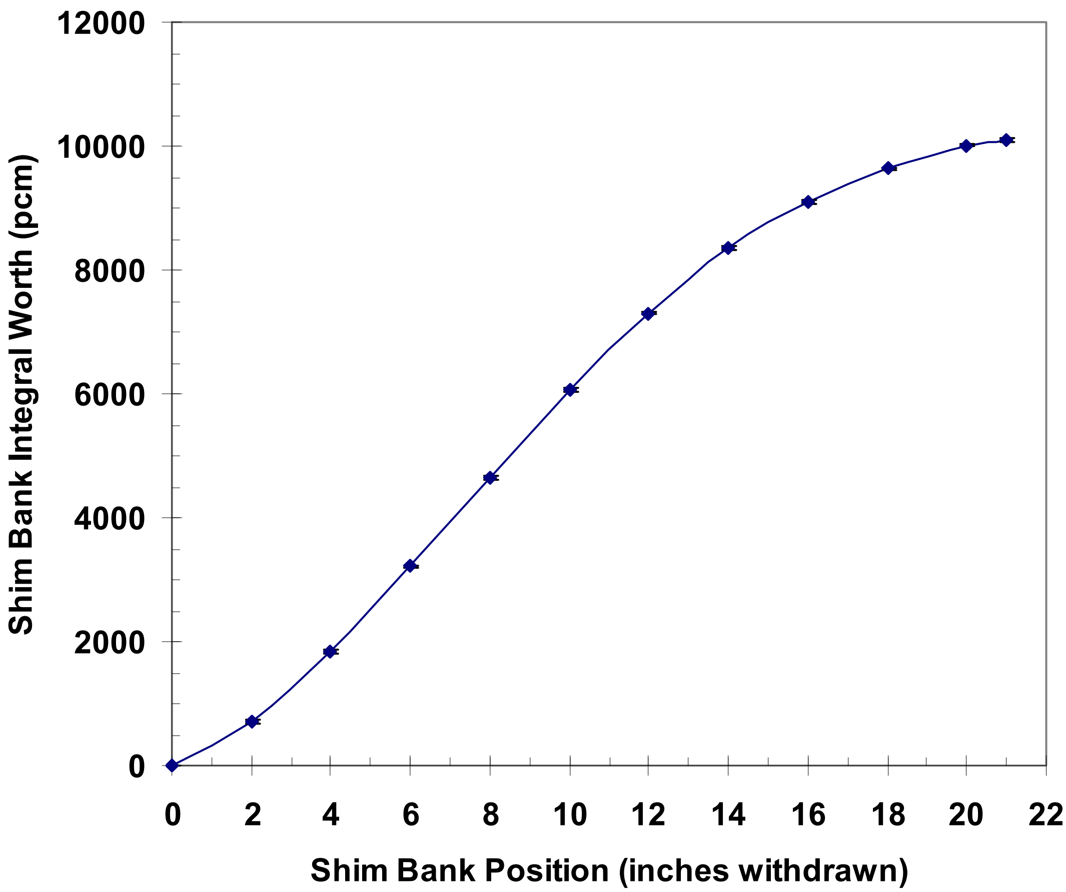

Figure 2. MITR LEU core shim bank integral curve.

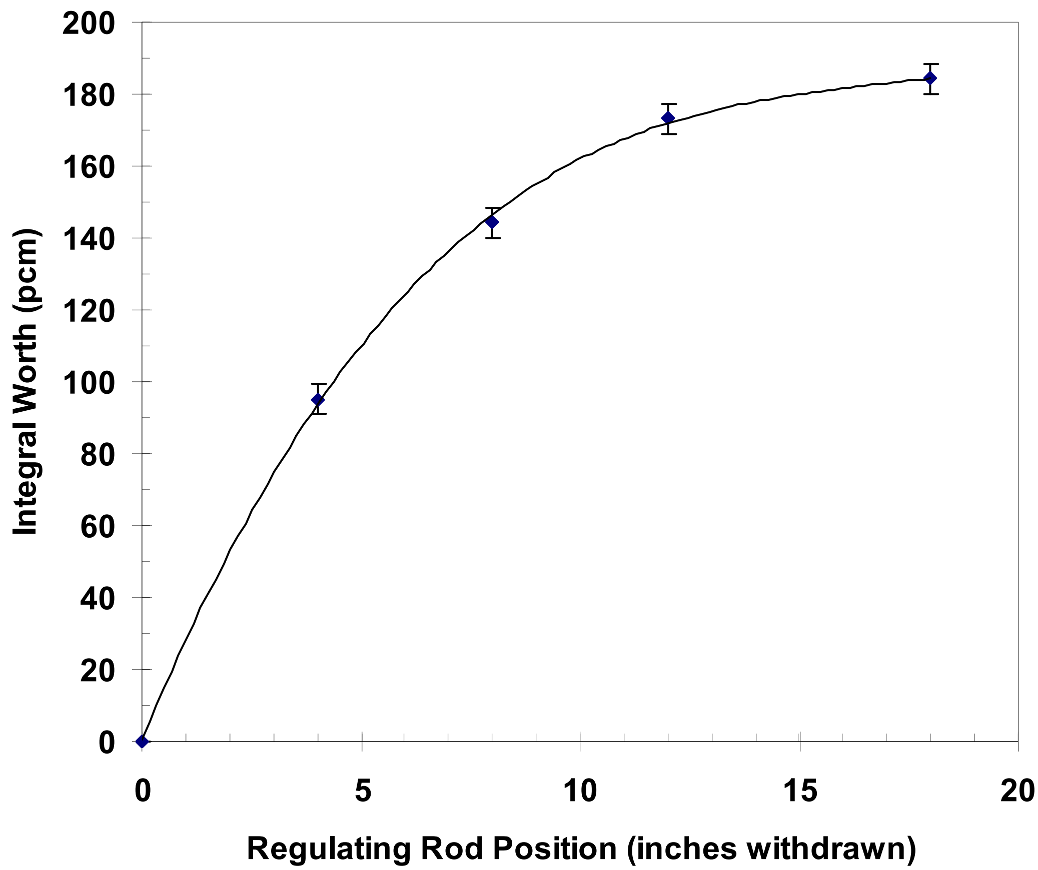

Figure 3. LEU core regulating rod worth curve for 8.17 inch withdrawn shim bank.

Table 1 summarizes parameters related to the six control blade control devices and regulating rod. The HEU values are taken from the HEU SAR [1], while the LEU values are calculated using MCNP [2]. When values appear in the MITR HEU SAR [1] or MITR-II Start-up Report [7] in units derived from the effective delayed neutron fraction, beta, these HEU reported values have been converted to Δk/k using beta = 0.00786 as was applied in those references.

Table 1. Parameters associated with kinetic behavior of control devices.

Parameter Worth	HEU *	LEU Calculated *	LEU 1-σ Uncertainty
Shim bank integral worth (pcm)	9927	9948	11
Shim blade integral worth of blade#5 (pcm)	1776	1643	13
Regulating rod integral worth (pcm)	124	146 (Bank in)	22
		301 (Bank out)	10

*Blade and rod worth figures shown vary with fuel loading.

The MITR HEU SAR [1] describes the integral worth of the HEU shim bank and regulating rod as typically 12 beta (9432 pcm), and variation with fuel loading as typically in the range 11.5 - 13.0 beta (9039 - 10218 pcm). The MITR HEU SAR [1] refers (in Table 4-4 of that report) to a shim bank integral worth of 12.63 beta (9927 pcm), a shim blade number five integral worth of 2.26 beta (1776 pcm), and regulating rod integral worth of 158 mbeta (124 pcm). Note that where the unit 'pcm' is used in this report it refers to %millirho of reactivity ($\Delta k/(k_0 \cdot k_1)$).

LEU integral worths have been calculated for a fresh LEU fuel configured with dummies in A2 A3 B3 B6 B9. This configuration was chosen as the fresh fuel LEU configuration since the shutdown margin requirements were met with five dummy elements, as discussed in section 4.5.3.3 of this report, and the A2 A3 B3 B6 B9 locations [8] are those used in the HEU fuel configuration run during the nearly fresh HEU core 2 [7]. This LEU configuration yielded a shim bank worth of 9948 pcm ±11 pcm. For this LEU fuel configuration, the calculated LEU shim bank worth is nearly equal to the maximum of the HEU typical range cited in the MITR HEU SAR [1].

A comparison of experimentally measured and calculated HEU control worths, as well as calculated LEU worths can be seen in reference [2].

Figure 2 shows the shape of the integral shim bank reactivity worth curve. These shapes are a function of the energy and magnitude of the flux to which the blades are exposed. The high differential worth that exists in the center of travel corresponds to a maximum in axial flux shape. Figure 3 provides the corresponding information for the regulating rod. The peak in the differential regulating rod worth occurs at low rod height because the full-in position for the regulating rod is six inches above the bottom of the fuel elements, and because once the regulating rod is withdrawn any appreciable amount, it is heavily shadowed by the adjacent shim blades. MITR refuelings are normally designed so that the new core will go critical at a shim bank height of 7 to 9 inches withdrawn.

Considerations that limit the shim bank height at which criticality is attained include the subcritical limit interlock and the shutdown margin. The shim bank height will gradually increase as the fuel depletes. Refuelings are normally performed when it is no longer possible to override xenon if the reactor is restarted several hours after a shutdown.

Increases of reactor power are accomplished subject to the following administrative limits:

a) A reactor scram will occur at a period between 10 and 11 seconds.

b) The minimum allowed dynamic period is 30 seconds.

c) If reactor power is less than 80% of demanded, the steady period shall be longer than 50 seconds.

d) If the reactor power is within 80% of demanded, the steady period shall be longer than 100 seconds.

Either a shim blade or the regulating rod may be used to accomplish a power adjustment. The choice is at the discretion of the licensed console operator. (Note: The term "steady period" implies non-zero reactivity and the absence of control device movement. The term "dynamic period" implies the presence of control device movement with or without non-zero reactivity.)

4.5.1.6 Interactions of Fuel/Moderator/Reflector/Control Devices

The criticality of any given core configuration is a function of certain interactions between some of the core components. These include:

a) Moderator/Reflector Temperature

The MITR is intentionally under-moderated so that there will be a negative coefficient of reactivity associated with the temperature of both the moderator (coolant) and the reflector. This temperature coefficient of reactivity encompasses two distinct phenomena. The first is the temperature rise of the light water because of an increase in the thermal power output of the reactor core. Any such temperature rise will insert negative reactivity by causing a hardening in the neutron spectrum. The second phenomenon is the heating of the heavy water reflector. Temperature rises of this type add negative reactivity by allowing neutron leakage to increase. This second process lags the temperature rise of the light water in the core proper.

Tables 2 and 3 summarize MITR temperature coefficients of reactivity. The MITR HEU SAR [1] cited a temperature coefficient of reactivity associated with the entire reactor (H_2O and D_2O) heat-up that varied from -6 mbeta/°C to -15 mbeta/°C (-4.7 pcm/°C to -11.8 pcm/°C) over the normal band of operating temperatures from 25°C to 50°C. LEU calculations were performed based on changing all water, H_2O and D_2O in the entire reactor, from 20.5°C to 77°C, in a single delta in order to properly account for the effect of $S(\alpha,\beta)$ cross section libraries of light and heavy water, which are available in ENDF/B-

VII at these temperatures. Whereas the experimental measurements of temperature reactivity were reported in the temperature range up to 50°C [7], the range of interest extends well above this since the outlet H_2O temperature under normal LEU full-power operation of 7 MW would be in the range of 55°C. These calculations show that the LEU temperature coefficient remains negative, and that the calculated value falls within the range reported in the MITR HEU SAR [1].

Table 2. Summary of temperature coefficients of reactivity for HEU and LEU cores.

Fuel	**Fuel Loading Pattern**	**Temperature Coefficient (pcm/ºC)**
HEU measured	Range of HEU patterns	-4.7 to -11.8
LEU calculated	Non-fuel dummy elements A2 A3 B3 B6 B9	-6.2 +/- 0.2 (1-σ)

Calculated HEU and LEU temperature coefficients are compared for identical fresh core configurations in Table 3. These calculated values indicate that the LEU has a modestly lower response to temperature than HEU; however, the temperature coefficient remains negative and within the range given in the MITR HEU SAR [1]. Calculations also indicate these temperature coefficients are relatively stable across a variety of fuel loading patterns. Calculations compared to experimental values and for fresh cores demonstrate a fairly weak dependence on core loading configuration [2].

Table 3. Summary of calculated temperature coefficients of reactivity for HEU and LEU fueled cores.

Core Configuration	**HEU Core 2**	**LEU**
Non-fuel dummy element position	A2 A3 B3 B6 B9	A2 A3 B3 B6 B9
Temperature coefficient (pcm/°C)	-8.1	-6.2
Uncertainty 1-σ (pcm/°C)	0.1	0.2

b) Heavy-Water Reflector Dump

The portion of the MITR's heavy water approximately above the bottom plane of the fuel elements is the volume of heavy water that may be dumped as a secondary shutdown mechanism. The height of this region is 22.6 inches. Because the shim blades also operate in the region between the core and the radial heavy water reflector, the reactivity worth of dumping this radial reflector is dependent on the position of the shim bank. This effect is the result of the shadowing influence that the blade bank exerts on the reflector. The reactivity worths of dumping the radial reflector both when the shim blades are at various heights from fully-inserted to fully-withdrawn are given in Table 4 and Figure 4.

Table 4. Calculated reflector dump worth for LEU and HEU cores for various shim bank positions.

D_2O Dump Worth (pcm) (1-σ < 25 pcm)	**HEU Core 1 Calculated**	**LEU Calculated**
Shim Bank Height	Non-fuel elements B2 B8 (ICSA A1)	Non-fuel elements A2 A3 B3 B6 B9
Fully withdrawn	-7461	-6927
12 inches withdrawn	-7097	-6267
Initial critical height	-6093 (7.36 inches)	-5323 (8.17 inches)
Fully inserted	-4050	-3047

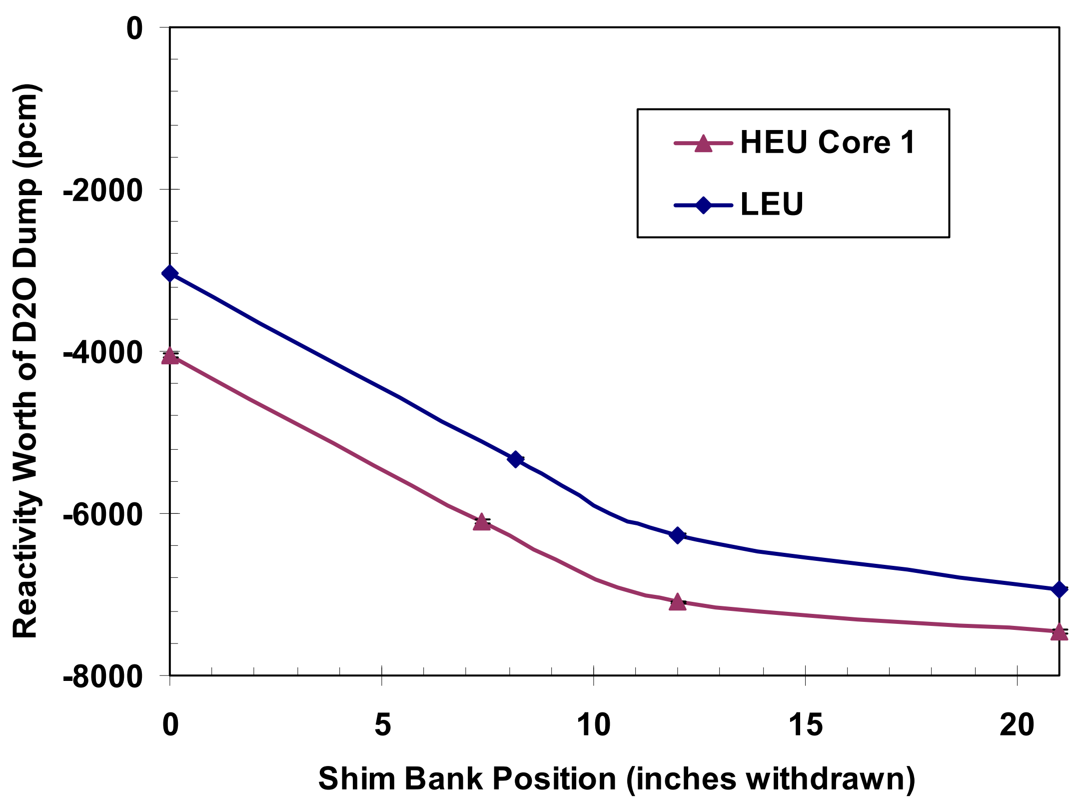

Figure 4. HEU and LEU calculated D2O reflector dump worth for various shim bank positions.

For both HEU and LEU calculations, the reactivity worth of dumping the radial heavy water reflector when the shim bank is at the modeled initial critical height is about 80% that of the corresponding value when the bank is fully withdrawn. HEU experimental data of the 22.6 inch reflector dump indicates 8.9 beta of reactivity loss (-6995 pcm) [2]. The experimental -6995 pcm falls between the calculated worths at a shim bank of 7.36 inches (-6093 pcm) and fully withdrawn (-7461 pcm). Since during the experimental dump, the shim height varied from 7.36 inches to the fully withdrawn 21 inches, HEU experimental and calculated values agree well.

Safety considerations dictate that the radial heavy-water reflector be pumped up with the shim bank in the fully inserted position. This ensures that the reactivity insertion for this process will not occur when the reactor is or could go critical.

c) Effects of Coolant and Reflector Leakage

This section deals with accident and transient related phenomena, and is not within the scope of steady-state analysis.

d) In-Core Facilities

This section deals with transient related phenomena, and is not within the scope of steady-state analysis.

There are no current plans for any in-core experimental facilities that would be moved, or in which samples would be moved, during reactor operation. If such a facility or samples were planned, they would be designed to comply with the "movable" reactivity limit as discussed in Section 10.3.2.6 of the MITR HEU SAR [1].

4.5.1.7 Safety Considerations for Different Core Configurations

The approach utilized to ensure the safety of every MITR core configuration is discussed in Section 4.5 of this report.

4.5.1.8 Reactivity Worths

Reactivity data was calculated for an MITR LEU core in a manner similar to the experimental method used for measuring HEU fuel and void coefficients [2]. Reactivity data on HEU MITR fuel, void coefficients, and in-core facilities was estimated by calculation prior to the startup of the MITR-II in 1975. This data was subsequently confirmed by measurement. Much of it is documented in the MITR-II Startup Report [7]. MITR HEU SAR [1] Table 4-5 listed some of the more important items [1]. Calculations for a fresh LEU core are presented alongside these HEU measurements. Fuel reactivity worth for various rings is shown in Table 5, where calculation uncertainties were ≤0.05 pcm/g ^{235}U. HEU fuel reactivity values were experimentally measured [7] for a single plate replacement with aluminum as well as for a single element. The MITR HEU SAR [1] cites the plate replacement; however, which plate was replaced is not now known. Since sufficient information on element replacement is available, element replacement calculations were performed on the same element as in the HEU experiment. Worth follows a similar decreasing trend with radial distance from core center, and element values are worth more per mass of ^{235}U than the plate measurements, and hence conservative.

Reactivity worths of void coefficients are summarized in Tables 6, 7 and 8. Differences between calculated and experimental values are likely within measurement and statistical errors.

Table 4-5 of the MITR HEU SAR [1] gives a -70 mbeta (-55 pcm) worth for draining the D2O in the blister tank completely. Table 9 compares HEU and LEU calculations to this reference HEU value.

Table 5. LEU and HEU reactivity worth of additional of ^{235}U fuel in various rings.

Location	HEU Measured Element Worth (pcm/g ^{235}U)	HEU Calculated Element Worth (pcm/g ^{235}U)	HEU (C-E)/E	LEU Calculated Element Worth (pcm/g ^{235}U)
A1 (A2 LEU)	7.19	5.83	-19%	2.91
B4	4.75	4.39	-8%	1.76
B5	3.87	3.27	-15%	1.20
C6	3.74	3.34	-11%	1.31
C7	3.46	3.14	-9%	1.16
C8	3.89	3.63	-7%	1.46
A-ring average	**7.19**	**5.83**	**-19%**	**2.91**
B-ring average	**4.31**	**3.83**	**-11%**	**1.48**
C-ring average	**3.70**	**3.37**	**-9%**	**1.31**

Table 6. Experimental and calculated HEU void coefficients for voided water channels in various rings.

HEU Full Channel Void Coefficient (pcm/cm^3)	Experiment	Calculated	Calc. 1-σ	(C-E)/E
A-ring	-2.14	-1.80	0.04	-16%
B-ring	-2.13	-1.92	0.06	-10%
C-ring	-1.26	-1.18	0.07	-7%
24 Element core	***-1.59***	***-1.44***	***0.11***	**-9%**

Table 7. Experimental and calculated HEU void coefficients for voided bottom six inches of channel in various rings.

HEU Bottom 6 inch Void Coefficient (pcm/cm^3)	Experiment	Calculated	Calc. 1-σ	(C-E)/E
A-ring	-2.73	-2.62	0.11	-4%
B-ring	-2.68	-2.77	0.16	3%
C-ring	-1.47	-1.59	0.22	8%
24 Element core	***-1.93***	***-2.02***	***0.31***	**5%**

Table 8. Calculated LEU void coefficients in various element rings for full channel void or bottom six inches void.

LEU Calculated Coefficient (pcm/cm^3)	Full Channel Void	Calc. 1-σ	Bottom 6-inch Void	Calc. 1-σ
A-ring	-1.427	0.037	-1.239	0.117
B-ring	-1.416	0.049	-1.417	0.152
C-ring	-0.750	0.065	-0.751	0.182
22 Element core	***-0.96***	***0.09***	***-0.95***	***0.27***

Table 9. Comparison of calculated HEU and LEU reactivity worths to the HEU reference value for draining the D_2O in the blister tank completely.

D2O Blister Tank Worth (drained)	HEU Reference Value	HEU Core 1 Calculated	LEU Calculated
Non-fuel dummy element position	-	B2 B8 ICSA in A1	A2 A3 B3 B6 B9
Worth of D2O in blister tank (pcm)	-55.0	-53.3	-42.0
Uncertainty 1-σ (pcm)	-	5.7	5.0
Deviation from reference value (%)	-	-3%	-24%

Analysis of experimental facilities will be treated during transient and accident analyses.

4.5.1.9 Core Reactivities

There are no foreseen impacts of an LEU fueled core which would alter this section.

Core reactivities are not established in advance because there is no set of pre-defined core configurations. Instead, the following analysis is performed and documented for every planned core in order to be certain that the core is appropriately configured:

a) The expected reactivity change for the proposed refueling is calculated. This can be done by comparing the k-effective values from the neutronic calculations of the core before and after the refueling is done in the model. Alternatively, the expected reactivity change may be derived by combining estimates of the change in grams of ^{235}U with the reactivity coefficients given in Table 5.

b) The shutdown margin for the refueled core is calculated and verified to be acceptable.

c) Items (a) and (b) above are documented and reviewed by someone other than the individual who did the calculations.

d) The refueling is performed.

e) The reactivity change is measured and compared to the predicted estimate.

f) The shutdown margin is recalculated using the measured reactivity change. It is again verified to be acceptable.

g) Items (e) and (f) are documented and reviewed.

4.5.1.10 Administrative and Physical Constraints

There are no foreseen impacts of an LEU fueled core which would alter this section.

Both administrative and physical constraints preclude the inadvertent addition of positive reactivity.

a) Administrative: Movement of fuel is not permitted in the core, fuel storage pool, or fission converter without the prior written approval of the Reactor Superintendent or his designate. All such approvals include a schedule of the authorized moves.

b) Physical: The k-effective of all areas where fuel may be stored (except the core itself) is less than 0.9. The principal concern is therefore the core. Changes to the core configuration are not possible when the reactor is operating because of the grid-latch mechanical interlock. In order to obtain a "reactor start" condition, the upper grid plate must be in the latched position. When the grid is latched, no fuel element positions are accessible. That is, fuel can neither be inserted nor removed. This eliminates movement of fuel as a means of inserting positive reactivity during reactor operation. That leaves the possibility of manipulating a sample in an in-core sample assembly (ICSA). All manipulations are scheduled in writing by the Superintendent of Reactor Operations and Maintenance or by the Superintendent's designate. Also, ICSAs are not accessible unless either a reactor top lid penetration or other equivalent shielding is removed. This requires both use of the overhead crane and access to the reactor top. Use of the former is under the control of the licensed console operator. Entry to the latter triggers an alarm in the control room.

4.5.2 Reactor Core Physics Parameters

4.5.2.1 Neutron Lifetime and Effective Delayed Neutron Fraction

LEU values for both the neutron lifetime and the effective delayed neutron fraction have been calculated and compared to HEU calculations. HEU calculations are compared to values cited in the MITR-II Start-up Report which were estimated prior to the modification in 1974-1975 and confirmed by both the dropped rod method and noise analysis as part of the startup testing [7]. Tables 10 and 11 compare the HEU and LEU effective delayed neutron fraction and prompt neutron lifetime, respectively.

Table 10. Calculated effective delayed neutron fraction for LEU and HEU cores.

Core Configuration	HEU Reference Value	HEU Core 2 Calculated	LEU Calculated
Non-fuel dummy element position	-	A2 A3 B3 B6 B9	A2 A3 B3 B6 B9
Effective delayed neutron fraction *	0.00786	0.00769	0.00761
Uncertainty 1-σ	-	0.00006	0.00004
Deviation from HEU reference value	-	-2%	-3%

* Calculated results do not include photoneutrons.

Table 11. Calculated prompt neutron lifetime for LEU and HEU cores.

Core Configuration	HEU Core 2 Calculated	LEU Calculated
Non-fuel dummy element position	A2 A3 B3 B6 B9	A2 A3 B3 B6 B9
Prompt neutron lifetime (μs)	77.3	60.5
Uncertainty 1-σ (μs)	0.7	0.7

In the MITR HEU SAR [1], the effective delayed neutron fraction cited is 0.00786, and the measured prompt lifetime 1.0E-4 sec (100 μs). The effective delayed neutron fraction calculation does not include photoneutrons, and for the studied HEU fresh cores is consistently between 2%-4% below the reference value. A study of photoneutrons which included calculations of fresh MITR reactor core cited a contribution to the effective delayed neutron fraction on the order of +1-2% for HEU and LEU MITR cores [9]. Although the calculated LEU effective delayed neutron fraction, 0.00761, is 1% lower than the calculated HEU value, 0.00769, all are in reasonable agreement given the uncertainties in the values, and the additional photoneutron contribution not included in this calculation.

Prompt neutron lifetime was calculated by the 1/v insertion method [10]. The calculated prompt neutron lifetime of an LEU fueled fresh core is given in Table B, along with the calculated HEU neutron lifetime. Both are significantly lower than the experimentally measured value of 1E-4 sec. The LEU lifetime is significantly lower than the HEU calculation. However, the calculated HEU value is closer to the LEU lifetime than it is to the HEU measurement. A large uncertainty may exist in the experimental value as several measurement methods were attempted before a noise analysis was deemed successful [7].

In the MITR HEU SAR [1], the prompt lifetime had been taken as 100 μs and the effective delayed neutron fraction as 0.00786 (786 pcm). The former is typical of a reactor with a thermal neutron spectrum. The latter is higher than that of most other reactors. This is the result of two factors. First, the heavy-water reflector is the source of a significant photoneutron flux that adds to the delayed neutron population that originates from precursor decay. Second, the geometry of the MITR core results in relatively little attenuation of photons within the core. Hence, the photon fraction that interacts with deuterium nuclei in the heavy water is greater than in most other heavy-water reflected cores. A second factor that may cause the effective delayed neutron fraction to change is the buildup of Pu-239. This also has not been observed for the MITR because the fuel is highly enriched.

Specific effects of LEU depletion, including potential changes in photoneutron impact and higher actinide build-up, will be studied in the future as part of the transition core evaluations.

4.5.2.2 Coefficients of Reactivity

There are no foreseen impacts of an LEU fueled core which would alter this section.

Values and signs for coefficients of reactivity were given in Table 4-5 of the MITR HEU SAR [1]. Most of these coefficients were obtained during the startup testing of the MITR-II. Others have been measured as part of special test programs that are required prior to experiment installation.

The test method used is as follows:

a) Reactor operation is restricted to low power so that there is no xenon production and/or temperature-dependent reactivity feedback.
b) The reactor is taken critical and critical data (shim blade and regulating rod height, temperature) recorded.
c) The reactor is shut down.

d) The change that is to be investigated is made. For example, a shutter might be opened or a portion of an in-core sample facility voided.

e) The reactor is again taken critical and critical data is recorded.

f) The reactivity worth of the change is determined by comparison of the critical data.

LEU calculated values for fuel and void in the fuel region were given in Tables 5 through 9. Changes with fuel burnup will be studied after depletion of an LEU core. Other coefficients (such as Cd in an experimental device liner) will be evaluated as a precursor to specific transition core and transient analyses.

4.5.2.3 Flux Distributions

The MITR-II's axial and radial neutron flux densities were measured as part of the startup testing in 1975 [7]. This was done by use of a special element from which the fuel plates were removable. The method used was to irradiate this special element as part of the core configuration that was to be studied. Individual plates would then be removed and placed in a shielded counter so that a specific portion of the plate, such as a particular axial segment, could be counted.

The axial neutron flux densities were re-measured about a year after these plate scans in order to gain experimental confirmation of the effect of shim bank height on the flux profile. These measurements were performed using copper wires with an appropriate correction for epithermal neutron activation [7]. These wire scans confirmed the earlier plate scans.

The axial and radial neutron flux density profiles are independent of power level. They depend on the position of the control devices, and hence to the extent that core configuration and burnup change the height of the shim bank, they will also change the flux profiles. The principal findings are listed in Section 4.5.1.3 of the MITR HEU SAR [1].

Both the plate-scan and the copper wire data have been used to normalize calculations of the flux profiles that were obtained using various codes. This process is described in the introduction to Section 4.5 of the MITR HEU SAR [1]. At present, the Monte-Carlo code MCNP [4] is being evaluated for use in the fuel management of the MITR.

Flux and power distributions are being developed as part of thermal hydraulic analysis since the level of detail needed for safety is determined by thermal safety margins.

4.5.3 Operating Limits

There are no foreseen impacts of an LEU fueled core which would alter this section.

Safety is ensured by observance of a substantial shutdown margin requirement as well as by specifying a minimum shim bank height (the subcritical limit interlock) below which the reactor is not to be operated in a critical condition.

4.5.3.1 Reactivity Conditions

This section deals with the transition to depleted LEU cores which will be included in future study of the impacts of depletion on core reactivity.

4.5.3.2 Excess Reactivity

A typical value for the MITR's HEU excess reactivity is 6 to 7 beta (4716 – 5502 pcm) [1]. Table 12 compares typical HEU reference values of excess reactivity, or reactivity available for insertion, to values calculated for LEU. The LEU fresh core configuration with five un-fueled dummy elements in A2 A3 B3 B6 B9 has an excess reactivity of 5327 pcm, which falls within the range of cited typical excess HEU reactivities.

Table 12. Typical HEU values of excess reactivity and reactivity available for insertion compared to an LEU fresh core calculations.

Excess Reactivity	**HEU Typical Reference Values (beta)**	**(pcm)**	**LEU Calculation (pcm)**	**1-σ Uncertainty (pcm)**
Core configuration	*Typical*		*Non-fuel elements in A2 A3 B3 B6 B9*	
Excess reactivity	6 - 7	4,716 - 5,502	5,327	9
Bank worth	12.8	10,061	10,249	13
Bank worth 5" to full out	9.4	7,388	7,704	13
Single blade from 5" to full out	1.6	1,258	1,708	2

However, the excess reactivity could be larger because the integral worth of the control devices (shim blades and regulating rod) is much larger, as discussed in Section 4.5.1.5 of this report. Two provisions in the MITR's design and operation limit the excess reactivity that is potentially available. First, the subcritical interlock, as described in Section 7.3.1.2 of the MITR HEU SAR [1], is observed during startups. Specifically, if the reactor should attain criticality at a shim bank height of less than 5.0 inches, it is to be shut down. The requirement has the effect of reducing, in the HEU case, the maximum possible excess reactivity from about 12.8 beta (10,061 pcm) to about 9.4 beta (7,388 pcm) [1]. In the LEU case, the maximum possible excess reactivity is reduced from 10,249 pcm to

7,704 pcm. Second, only one shim blade can be withdrawn at a time. This fact, coupled with the identical nature of the six blades and the requirement that they be operated as a bank when above the subcritical interlock, limited the excess reactivity per blade in the HEU core to about 1.6 beta (1,258 pcm) [1] and for the LEU core to 1,708 pcm. These are typical amounts that could, in a worst case scenario, be inserted.

4.5.3.3 Shutdown Margin

The shutdown margin requirement for the MITR is that it be possible to shut the reactor down by at least 1% Δk/k using shim blades from the cold (10° C), xenon-free condition with both the most reactive blade and the regulating rod fully withdrawn and all movable samples in their most reactive state.

Shutdown margin (in terms of %Δk/k) was calculated at 10°C for the LEU core configuration with five un-fueled dummy elements in A2 A3 B3 B6 B9, and 22 fresh, xenon-free, fuel elements each loaded with 831.4 g ^{235}U. There are no movable samples present in the calculation. As calculated in Table 13, the most reactive blade for this LEU configuration is blade six. The shutdown margin calculated for this LEU configuration is therefore 2.69% Δk/k (1-σ 0.01%) which exceeds the requirement.

Table 13. Calculated shutdown margin for HEU and LEU cores with the same fuel loading configuration.

Shutdown Margin (%) with 5 of 6 Shim Blades Fully Inserted	HEU Core 2 Calculated	LEU Calculated
Non-fuel dummy element position	A2 A3 B3 B6 B9	A2 A3 B3 B6 B9
Control blade #1 & reg. rod out	3.35	2.73
Control blade #2 & reg. rod out	3.49	2.80
Control blade #3 & reg. rod out	3.56	2.87
Control blade #4 & reg. rod out	3.55	2.84
Control blade #5 & reg. rod out	3.50	2.84
Control blade #6 & reg. rod out	3.37	2.69

The availability of normal electric power (or for that matter emergency electric power) has no effect on the capability of the MITR to meet the shutdown margin as the blades drop into the core under the influence of gravity upon loss of power to electromagnets.

As noted in Section 4.5.1.9 of this report, the shutdown margin is determined prior to every refueling. It is verified following every refueling using the measured worth of the refueling. The error bar of the shutdown margin calculation is the result of uncertainties associated with the measurement of the individual reactivities that are used in the computation. These in turn reflect the accuracy with which the control device, temperature, and xenon reactivity worth curves were determined as well as the correctness of measurements of the blade and rod height, coolant and reflector temperature, and the time since shutdown. An error analysis of these factors was performed as part of the fuel management studies for the MITR [3]. The MITR uses the power doubling time method once asymptotic conditions have been established to measure the reactivity of the control devices. This method is believed to be accurate within ± 10%.

4.5.3.4 Limiting Core Configuration for Thermal-Hydraulic Analysis

Thermal-hydraulic analyses will not be performed as a part of this report.

4.5.3.5 Transient Analysis

This section deals with accident and transient related phenomena, and is not within the scope of steady-state analysis.

4.5.3.6 Redundancy and Diversity of Reactor Shutdown Mechanisms

There are no foreseen impacts of an LEU fueled core which would alter this section.

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