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Document Control Desk
US Nuclear Regulatory Commission
11555 Rockville Pike, Rockville, MD 20852
ATTN: Cindy Montgomery

RE: University of Maryland (License # R-070, Docket # 50-166) Response to Requests for Additional Information Issued on November 22, 2019

Enclosed please find the University of Maryland's response to the Second Request for Additional Information Regarding the License Amendment Request for the Use of 16 Additional Fuel Elements.

I declare under penalty of perjury that the foregoing response is true and correct.

Sincerely,

Amber S. Johnson

Amber S. Johnson

Director, Nuclear Reactor and Radiation Facilities

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NRR

RAI-1 The regulations in 10 CFR 50.34, "Contents of applications; technical information," paragraph (b)(2) require that the safety analysis report (SAR) include the evaluations required to show that safety functions of the structures, systems, and components (SSCs) listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Section 4.6, "Thermal-Hydraulic Design," states that licensees should present the information and analyses necessary to show that sufficient cooling capacity exists to prevent fuel overheating and loss of integrity for all anticipated reactor operating conditions. The licensee should address the coolant flow conditions for which the reactor is designed and licensed; a detailed description of the methods used in the thermal-hydraulic analysis should be provided.

The guidance in NUREG-1537, Part 2, Section 4.5.2, "Reactor Core Physics Parameters," states that: "The calculational assumptions and methods should be justified and traceable to their development and validation, and the results should be compared with calculations of other similar facilities and previous experimental measurements."
Additional information is needed for the NRC staff to ensure that sufficient cooling exists for the proposed core geometry to prevent the fuel from overheating. The NRC staff needs more information to understand the heat removal conditions (such as fuel surface saturation temperature, onset of nucleate boiling, departure from nucleate boiling, and/or flow instability) that provide for adequate fuel channel cooling.

In response to RAI No. 2, UMD stated that a new thermal-hydraulic analysis is unnecessary. UMD further stated that average power per element will be reduced from 2.00 kilowatts (kW) to 1.95 kW.

1. Provide the methodology used to determine average power per element for both the current and proposed core configurations.

   Power per element was determined by dividing the total power of the core (300kW) by the total number of elements in the core. The response to RAI No.2 was an error and the correct power per element for the old core was 3.23kW, while the power per element for the new core is 2.75kW.

2. Justify why the core neutronics analysis for the proposed core was performed at 250 kW even though the Reactor Power Level scram maximum setpoint is 120 percent of full power (300 kW).

   Oregon State University has reperformed the neutronics analysis at 300kW. Please see Attachment 1 for the updated neutronics report.

3. Justify by using specific parameters (i.e., peaking factor, departure from nucleate boiling ratio, core power level, average fuel element power etc.) why the current thermal-hydraulic analysis bounds the proposed core configuration.

   The thermal hydraulic analysis for the current core bounds the thermal hydraulics for the proposed core configuration. The new core will have a similar power peaking factor (1.65 vs. 1.6), and a lower highest power element (4.23kW vs 4.63 kW, see Attachment 1). The same power will be distributed across a larger core

resulting in a lower power per element (2.75kW vs 3.23kW at 300kW) and more conservative fuel temperatures. Lower fuel temperatures will result in a larger DNBR. For the current core, General Atomics has simulated the DNBR at power of 300kW and a 92° C pool temperature to be 5.92; the proposed core will have a greater DNBR. The proposed core is more conservative thermal-hydraulically than the current core thus the current thermal-hydraulic analysis is bounding.

4. Explain and justify the influence, if any, to the neutronic and the thermal hydraulic analysis for the proposed core configuration based on UMD's recent identification of several slightly misaligned fuel bundles in the existing core. In addition, explain how UMD would verify that the proposed addition of 16 fuel elements (i.e., four fuel bundles) are fully aligned and seated properly in the grid plate.

   University of Maryland and Oregon State University separately analyzed the misalignment of fuel bundles using MCNP and found that it has a negligible contribution to reactivity. UMD found that the observed off shifting of the bundles contributes -\$.0.03 to the excess reactivity of the core. OSU found that if all of the fuel was offset (raised) by 2 cm from normal, with the control rods remaining at the same axial locations, that the offset bundles had a k_{eff} of 1.03273, vs a k_{eff} of 1.03481, which amounts to a reactivity worth of approximately \$0.03. Given the minimal reactivity worth of the offshifting, and that it occurs in the vertical direction and does not change coolant channel widths, the misalignment does not significantly perturb the thermal-hydraulics of the analyzed core.

   When new fuel bundles are added to the core they will be verified to be aligned and seated using an underwater camera. See Attachment 2, Revised Start-up Plan.

RAI-2 The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the SSCs listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 2, Section 4.5.2, states that "The calculational assumptions and methods should be justified and traceable to their development and validation, and the results should be compared with calculations of other similar facilities and previous experimental measurements. The ranges of validity and accuracy should be stated and justified."

In response to RAI No. 4, UMD performed a different method (i.e., rod drop) for measuring rod reactivity worth and concluded that these measurements better agree with the simulated values of rod worth.

Additional information is needed for the NRC staff to understand which method(s) UMD will use to measure rod reactivity worth to ensure the reactor can be shutdown with sufficient margin from any operating condition.

1. Explain which method(s) UMD plans to use to measure rod reactivity worth for initial startup of the reactor in the Startup Plan – Additional Reactor Fuel included in the LAR and during its required surveillance in technical specification (TS) 4.2.1.

   Rod worths will be measured using the asymptotic period method in accordance with SP204: Control Rod Calibration, the approved procedure for annually measuring rod worths. Rod drop worth measurements may be performed at the same time to validate this method, however these will not replace the method in the approved procedure at this time.

2. Explain when UMD will perform the control rod reactivity worth measurements considering the fuel loading process. For example, does UMD plan to conduct multiple control rod reactivity worth measurements as fuel is loaded into the core or perform a single measurement after all 16 elements are loaded into the core.

   Control rod worth measurements will be performed after each individual bundle is added. This is the more conservative approach to adding fuel and will help to ensure that reactivity limits will not be violated. See Attachment 2, Revised Startup Plan, for more information.

3. Provide a current graph and quantitative data of measured control rod reactivity worth from 2010 to 2019 (Figure 14 – Historic MUTR Control Rod Data).

   See Attachment 3, Control Rod Worth Data.

RAI-3 The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the SSCs listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Section 13.1.2, "Insertion of Excess Reactivity," states, in part, that an insertion-of-excess-reactivity event is a ramp insertion of reactivity by drive motion of the most reactive control rod or shim rod, or ganged rods, if possible. (This event could occur during reactor startup procedures or when the reactor is at power.)

The current Maryland University Training Reactor (MUTR) TS 3.1, "Reactor Core Parameters," Specification 1, states that: "The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place, shall not be greater than $1.12."

The LAR proposes a limiting condition for operation (LCO) for excess reactivity of not greater than $3.50.

In response to RAI No. 5, UMD stated that the rod withdrawal analysis of December 18, 2006, does not provide a bounding condition for the proposed excess reactivity LCO of $3.50. The NRC staff needs more information to ensure that peak fuel temperature remains below temperature safety limit of 1,000 degrees Celsius in TS 2.1, "Safety Limit," and that thermal-hydraulic conditions in the fuel channel remain subcooled.

1. Provide a ramp reactivity analyses, taking into consideration the proposed excess reactivity limit of \$3.50, starting at both high and low power conditions for the proposed core configuration. Provide the values of maximum fuel centerline temperature and maximum power achieved for all scenarios evaluated.

   See response to 3.2 below.

2. Explain the methodology, inputs, and assumptions used in the control rod withdrawal analysis.

   A ramp insertion of reactivity analysis was performed by considering the maximum power that ramp insertion of reactivity would result in, and using the RELAP analysis performed by General Atomics in 2011 (ML110350175) to determine the maximum fuel temperature. This analysis is a bounding case for the proposed core as described in the response to RAI #1.3. 3 scenarios were considered, (1) a ramp as quickly as possible from low power, (2) a ramp from full power just below the scram setpoint, and (3) a ramp with no functioning Reactor Safety System.

   In the first case, the reactor power is initially taken to be 1W and the fuel temperature to be 20°C. The control rods withdraw such that a 5s period, just below the period scram setpoint, is maintained during the ramp. The reactor will be scrammed by the 2 independent power scram channels when it reaches 300kW. Assuming a 1s delay between reaching the scram setpoint and the control rod release terminating the power rise to account for channel response time, the reactor power will increase to 366 kW. According to the 2011 Thermal Hydraulic Analysis (Figure 3-3) by GA the peak fuel temperature at 366kW is approximately 270°C, below the maximum operational, 350°C, fuel temperature allowed by the TS, and far below the 1000°C Safety Limit.

   In the second case, the MUTR is operating just below the 300kW power scram setpoint when uncontrolled rod withdrawal begins inserting \$0.30 per second. It would immediately reach the scram setpoint monitored by 2 independent scram channels. Assuming a delay of 1s from reaching the scram setpoint to the rods being released, \$0.30 of reactivity would be inserted. This would cause the reactor to prompt jump to 429kW before control rod insertion could terminate the power rise. According to the GA analysis, 429 kW corresponds to a peak fuel temperature of 285°C, still well below the maximum allowed operating temperature and Safety Limit.

   In the third case, it is assumed that the Reactor Safety System is inoperable when the ramp begins. In this case, the MUTR would increase in power until all excess reactivity was inserted. The MUTR's power coefficient of reactivity is \$0.0053/kW as described in the SAR. Assuming \$3.50 of excess reactivity, the peak power reached would be 661 kW. Using the GA analysis, this corresponds to a peak temperature of about 340°C, still below the maximum allowed operating fuel temperature and with a 660°C margin to the Safety Limit. This is not a plausible scenario as there are 2 independent scram channels that will terminate the reactivity insertion at 300kW as well as a period scram channel that will terminate reactivity insertion if the period exceeds 5s. Any 1 of these 3 channels, if operational, would prevent an uncontrolled rod withdraw from reaching this power level, and all undergo a channel test prior to each operation.

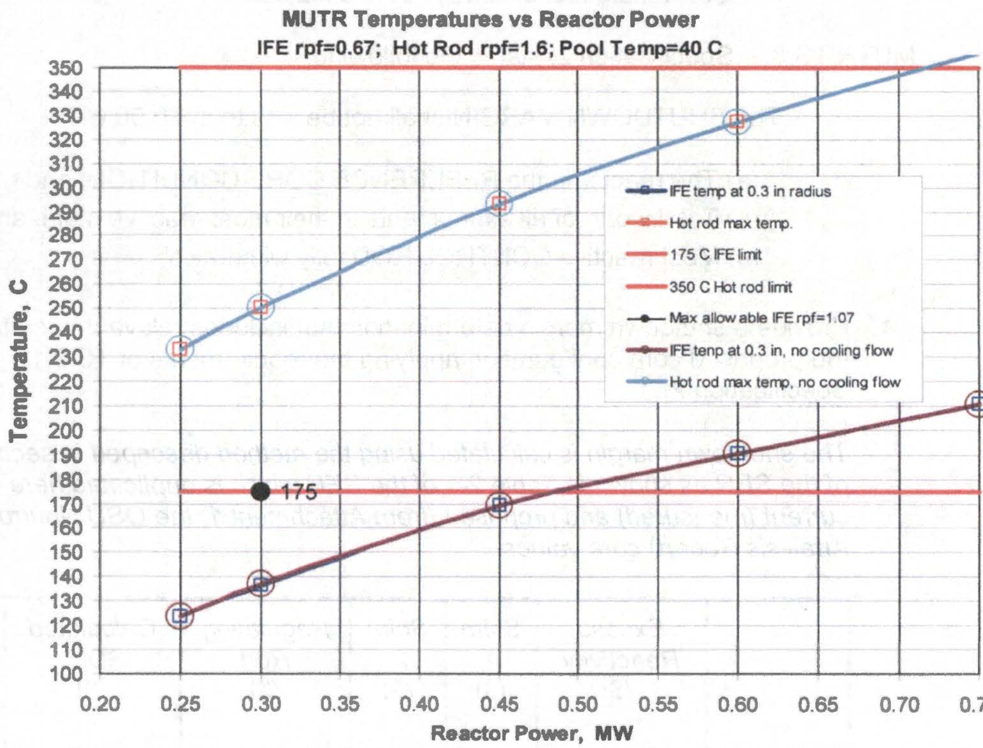

Figure 1: Reactor Power vs Fuel Temperature as modeled by General Atomics using RELAP in 2011.

RAI-4 The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the SSCs listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Section 4.5.3, "Operating Limits," states that the licensee should present information on the amount of negative reactivity that must be available by control rod action to ensure that the reactor can be shut down safely from any operating condition and maintained in a safe shutdown state. The analyses should assume that (1) the most reactive control rod is fully withdrawn (one stuck rod), (2) non-scrammable control rods are at their most reactive position, and (3) normal electrical power is unavailable to the reactor. The licensee should discuss how shutdown margin will be verified. The analyses should include all relevant uncertainties and error limits.
Additional information is needed for the NRC staff to ensure the reactor can be shutdown with adequate margin as stated in TS 3.1 from any operating condition for the proposed core configuration.

The LAR states the following:

> Using the control rod values from Table 3, the Shutdown Margin is calculated from the total rod worth minus the most reactive rod minus the excess reactivity. An upper limit of $0.50 on the Shutdown Margin is defined in technical specification 3.1.2.

Allowing for an excess reactivity of $3.50, guarantees that the shutdown margin shall always be maintained.

MUTR TS 3.1, Specification 2, states the following:

The SHUTDOWN MARGIN shall not be less than $0.50 with:

(a) The reactor in the REFERENCE CORE CONDITION; and
(b) Total worth of all experiments in their most reactive state; and
(c) Most reactive CONTROL ROD fully withdrawn.

1. Provide a shutdown margin determination that includes relevant uncertainties for the proposed core configuration applying the requirements of TS 3.1 Specification 2.

   The shutdown margin is calculated using the method described in section 2.5.1.2 of the SER as shown in Table 2-1 of the SER which is duplicated here with the current (measured) and proposed (from Attachment 1, the OSU Neutronics Analysis Report) core values.

	Excess Reactivity ($)	*Shim 1 ($)*	*Shim 2 ($)*	*Regulating Rod ($)*	*Calculated SDM ($)*	*TS 3.1.2 SDM Value ($)*
Current Core (2017)	*+1.12*	*-2.08*	*-2.75*	*-2.20*	*-3.16*	*-0.50*
Proposed Core	*+3.50*	*-3.00*	*-3.09*	*-3.26*	*-2.59*	*-0.50*

2. Explain the methodology used to obtain the data supporting the shutdown margin determination for the proposed core. Additionally, state any inputs or assumptions used in the shutdown margin determination.

   The data used to calculate the SDM are the control rod worths for the proposed core as reported by OSU in the Neutronics Analysis Report, while the excess reactivity is assumed to be $3.50, the maximum allowed by the proposed TS.

RAI-5 The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the SSCs listed in 10 CFR 50.34(b)(2)(i) will be accomplished

The guidance in NUREG-1537, Part 1, Section 13.1.2, states, in part, that an insertion-of-excess-reactivity event can be used to show why LCO on reactivity are justified.

In the LAR, UMD applies the Fuchs-Nordheim technique to analyze a $4.00 insertion of reactivity in the proposed core configuration. The NRC staff needs more information

to understand the input values to the analysis to ensure the peak fuel temperature remains below temperature safety limit of 1,000 degrees Celsius in TS 2.1.

Explain and justify each value used in the Fuchs-Nordheim technique that is used to analyze a $4.00 insertion of reactivity in the proposed core configuration.

UMD analyzed the $4.00 pulse using the constant heat capacity Fuchs-Nordheim equations.

The Fuchs-Nordheim equations are:

$$\text{Average fuel temperature: } \Delta T = \frac{2\Delta k_p}{\alpha}$$

$$\text{Total Energy Released: } E = 2C\frac{\Delta k_p}{\alpha}$$

$$\text{Peak Power: } P_{max} = \frac{C(\Delta k_p^2)}{2l\alpha} + P_0$$

$$\text{Peak Temperature: } T_{max} = F_p\Delta T + T_0$$

Where:

Prompt neutron lifetime = $l = 1*10^{-4}$ *seconds*

Prompt temperature coefficient of reactivity = $\alpha = 1.25*10^{-4}\ \rho/°C$

Core heat capacity = $C = 11.9*10^{4}\ J/°C$

Portion of the reactivity insertion above prompt critical = $\Delta_{k_p} = (\$\text{ inserted} - 1)*\beta_{eff}$

Power peaking factor = $F_p = 1.65$

The prompt neutron lifetime was chosen as $1*10^{-4}$ *seconds since this is the standard accepted value for light water reactors. The prompt negative temperature coefficient of reactivity was taken from the SAR. The core heat capacity is determined by multiplying the heat capacity of a single TRIGA Fuel element (1088 J/°C) the total number of elements in the core (109). This number can be derived from the equation for fuel element heat capacity in GA-7882 given the approximate average temperature of a fuel element during a pulse. The power peaking factor was determined by dividing the power of the highest power element in the 109 element core by the average power per element in the 109 element core as analyzed in the OSU Neutronics Behavior Report.*

These inputs lead to the following results for a $4.00 reactivity insertion in a critical reactor:

Initial Power (kW)	Initial Temp (°C)	Average Fuel Temp (°C)	Peak Fuel Temp (°C)	Energy Added Per Element (MJ)
.01	25	361	579	0.36
250	190	526	744	0.36

In all cases the fuel temperature remains below the safety limit. Since these estimates were determined using the constant heat capacity Fuchs-Nordheim equations they are more conservative than the estimates in the SAR using the variable heat capacity Fuchs-Nordheim equations as described in the IAEA TRIGA Reactor Characteristics Manual.[1]

[1] https://ansn.iaea.org/Common/documents/Training/TRIGA%20Reactors%20(Safety%20and%20Technology)/pdf/chapter1.pdf

RAI-6 The regulations in 10 CFR 50.36, "Technical specifications," paragraph (c)(3) require TSs to contain surveillance requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCO will be met.

NUREG-1537, Part 1, Appendix 14.1, Section 4.1(6) "Fuel Parameters," states that: "For non-pulsing TRIGA reactors, the fuel should be inspected and measured on at least a 5-year cycle. Approximately 20 percent of the fuel could be inspected and measured annually."

The NRC staff needs more information to ensure the integrity of the fuel cladding is maintained to minimize the possibility of an inadvertent release of radioactive fission products.

In response to RAI No. 5, UMD proposes a change to MUTR TS 4.1, "Reactor Core Parameters," Specification 4, to modify the fuel bundles that are inspected annually as follows:

4.1 Reactor Core Parameters

> *4. A visual inspection of 2 fuel bundles from rows B and C shall be performed annually at intervals not to exceed 15 months. The bundles inspected shall change each year so that in a 5 year period the entire group will be inspected. If any are found to be damaged, an inspection of the entire MUTR core shall be performed.*

UMD proposes that only 2 out of 28 fuel bundles which represents only 7 percent of the core be visually inspected on an annual basis.

Provide a basis for visually inspecting 2 fuel bundles on an annual basis considering that UMD is proposing to increase the total number of fuel elements from 93 to 109 with slightly irradiated fuel that has been in storage for a considerably long time.

UMD proposes that Technical Specification 4.1.4 be updated to read "A visual inspection of a representative group of at least 4 FOUR ELEMENT FUEL BUNDLEs from rows B and C shall be performed annually at intervals not to exceed 15 months. The bundles inspected shall rotate such that in a 2-year period all accessible 4 FOUR ELEMENT FUEL BUNDLEs in rows B and C are inspected. If any are found to be damaged, an inspection of the entire MUTR core shall be performed."

Only the bundles in rows B and C are accessible due to the CRDM support plate which blocks access to most of the core. In the SER, the NRC states that because the MUTR facility does not pulse, does not use a forced circulation coolant system, has relatively low fuel burn up given the operating history, uses stainless steel fuel elements, the low risk of damage to instrumentation, and its current licensed power of only 250 kWt, that visually inspecting the fuel in grid plate locations listed in TS 4.1, Specification 4 would provide an adequate representative profile of all other fuel elements in the core.

An inspection of approximately 20% of the core fuel on an annual basis was judged to be sufficient to meet the requirements of NUREG 1537 so long as the entire core would be inspected if damaged fuel was identified. Only 9 fuel elements are readily accessible without excessive risk of damage from moving them. These 9 elements are in a group of 4 and a group of 5. Alternating between the group of 4 and the group of 5 allows for

inspecting nearly 20% of the core in accordance with NUREG 1537 and rotating among the 9 accessible bundles allows for 30% of the core to be inspected biennially.

See attachment 4 for updated Technical Specification pages. The change to TS 4.1.4 is on page 19. However, the increased length of the revised specification has caused the locations of other specifications to shift on pages 18 and 20. All 3 revised pages are presented, however, no revisions to the text have taken place other than TS 4.1.4.

RAI-7 The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the SSCs listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Section 13.1.2, states that an insertion-of-excess-reactivity event can be used to show limiting conditions for operation on reactivity are justified.

Additional information is needed for the NRC staff to ensure the reactor can be shutdown with adequate margin as required by TS 3.1 from any operating condition for the proposed core configuration.

TS 4.1, Specifications 1 and 2, state the following:

1. The EXCESS REACTIVITY shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed, these changes include any removal or replacement of CONTROL RODS.

2. The SHUTDOWN MARGIN shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed, these changes include any removal or replacement of CONTROL RODS.

UMD proposes to change TS 3.1, Specification 1, to the following:

The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place shall not be greater than $3.50.

1. Explain how UMD ensures that the excess reactivity of $3.50 proposed in TS 3.1, Specification 1, is not exceeded when experiments are placed in the core.

   All experiments must be approved by the Reactor Safety Committee before they can be performed as per TS 6.5. Reactivity estimates are made as part of the experimental review process. The estimate is used to ensure that the specifications for excess reactivity, shutdown margin, and reactivity worth of experiments are still met with the experiment is installed before the experiment is approved by the committee. Experiments do not typically approach any limits on reactivity.

2. Explain how UMD ensures that the shutdown margin requirement of $0.50 is not exceeded when experiments are placed in the core.

   All experiments must be approved by the Reactor Safety Committee before they can be performed as per TS 6.5. Reactivity estimates are made as part of the experimental review process. The estimate is used to ensure that the specifications for excess reactivity, shutdown margin, and reactivity worth of experiments are still met with the experiment is installed before the experiment is approved by the committee. Experiments do not typically approach any limits on reactivity.

Attachment 1

ANALYSIS OF THE NEUTRONIC BEHAVIOR OF THE MARYLAND UNIVERSITY TRAINING REACTOR, REVISION 4

ANALYSIS OF THE NEUTRONIC BEHAVIOR
OF THE
MARYLAND UNIVERSITY TRAINING REACTOR

Submitted By:

Radiation Center
Oregon State University
Corvallis, Oregon

June 2017

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1. Introduction

This report contains the results of investigation into the neutronic behavior of the Maryland University Training Reactor (MUTR). The objectives of this study were to: 1) create a model of the MUTR to study the neutronic characteristics, 2) demonstrate acceptable reactor performance and safety margins for the MUTR core under normal conditions, and 3) suggest ways to improve performance of the MUTR.

2. Summary and Conclusions of Principal Safety Considerations

The conclusion of this investigation is that the MCNP model does an acceptable job of predicting behavior of the MUTR core. As such, the results suggest that the current MUTR core can be safely operated within the parameters set forth in the technical specifications; however, the MUTR as currently loaded is unable to operate at 250 kW due to depleted fuel, which the MCNP model confirmed. Discussion and specifics of the analysis are located in the following sections. The final sections of this analysis provide suggestions for a new core configuration and suggested changes to the technical specifications to accommodate the new core configuration.

3. Reactor Fuel

The fuel utilized in the MUTR is standard TRIGA® fuel manufactured by General Atomics. The use of low-enriched uranium/zirconium hydride fuels in TRIGA® reactors has been previously addressed in NUREG-1282 [1]. This document reviews the characteristics such as size, shape, material composition, dissociation pressure, hydrogen migration, hydrogen retention, density, thermal conductivity, volumetric specific heat, chemical reactivity, irradiation effects, prompt-temperature coefficient of reactivity and fission product retention. The conclusion of NUREG-1282 is that TRIGA® fuel, including the fuel utilized in the MUTR, is acceptable for use in reactors designed for such fuel.

The design of standard stainless steel clad fuel utilized in the MUTR is shown in Figure 1 [2]. Stainless steel clad elements used at MUTR all have fuel alloy length of 38.1 cm. The characteristics of standard fuel elements are shown in Table 1.

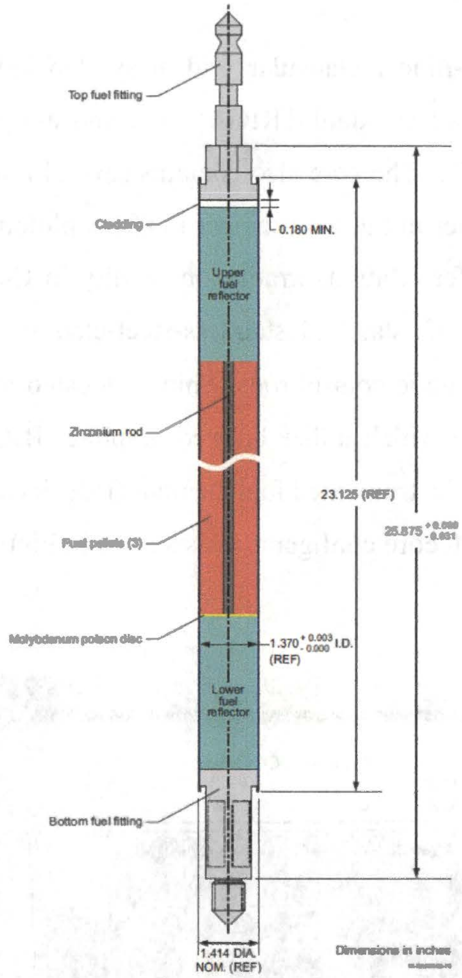

Figure 1 – TRIGA® Stainless Steel Clad Fuel Element Design used in the MUTR Core [2]

Table 1 – Characteristics of Stainless Steel Clad Fuel Elements

Uranium content [mass %]	8.5
BOL ^{235}U enrichment [mass % U]	19.75
Original uranium mass [gm]	37
Zirconium rod diameter [in]	0.225
Fuel meat inner diameter [in]	0.25
Fuel meat outer diameter [in]	1.374
Cladding outer diameter [in]	1.414
Cladding material	Type 304 SS
Cladding thickness [in]	0.020
Fuel meat length [in]	15
Graphite slug outer diameter [in]	1.291
Graphite slug length [in]	3.47
Molybdenum disc thickness [mm]	0.8

4. Reactor Core

The MUTR core is a five-by-nine rectangular grid array (labeled B through F, 1 through 9) composed of stainless-steel-clad standard TRIGA® fuel and aluminum-clad graphite reflector clusters (located in E2 and D2). The core also contains several non-fueled locations that house instrumentation (fission chamber in F9, ion chamber in F2), a plutonium-beryllium startup source in B6, and a pneumatic transfer (Rabbit) irradiation facility in C4. The current configuration established in 1974 includes 93 standard stainless-steel-clad fuel elements. The reactor is controlled by three electromagnetic control rods (Shim I, located in E4; Shim II, located in E7; and Regulating, located in C6) which utilize borated graphite (B_4C) as a neutron poison. Fuel temperature is measured by an instrumented fuel element (IFE) located in the southeast corner of the D8 fuel cluster. The current core configuration is shown in Figure 2.

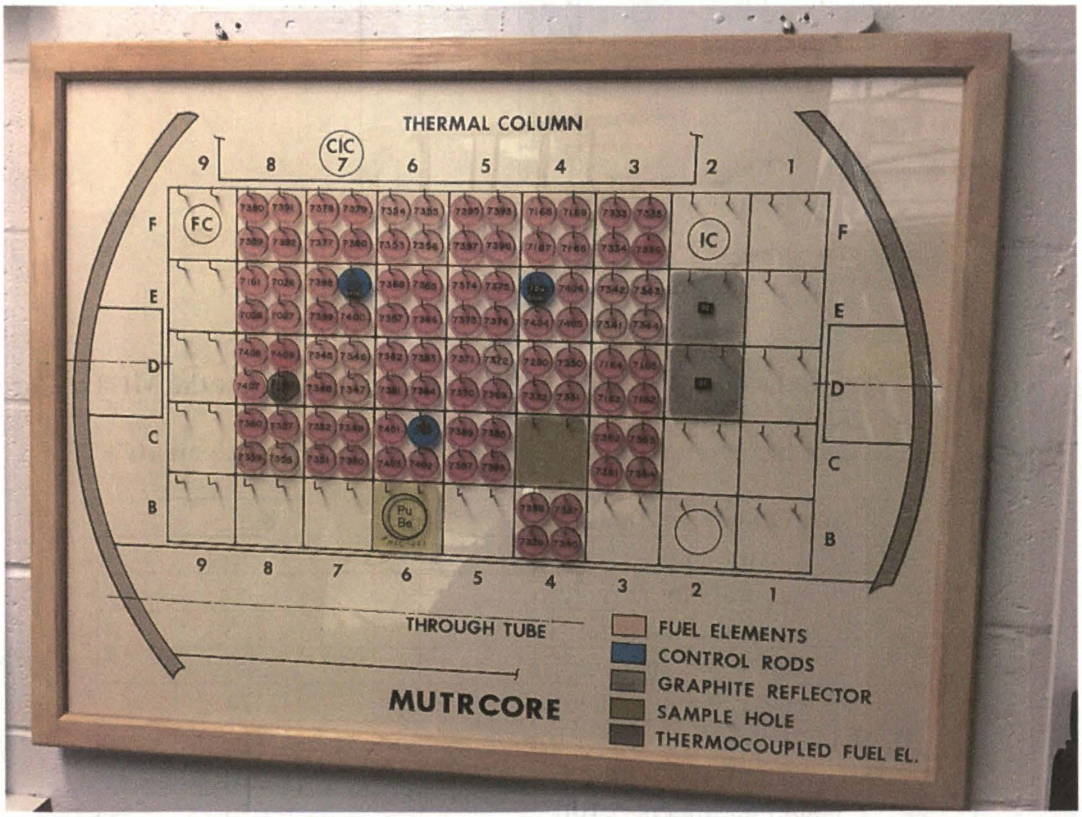

Figure 2 – Schematic Illustration of the MUTR Showing the Current Core Configuration

Detailed neutronic analyses of the MUTR core were undertaken using MCNP6.1.1 [3]. MCNP6.1.1 is a general purpose Monte Carlo transport code which permits detailed neutronic calculations of complex 3-dimensional systems. It is well suited to explicitly handle the material and geometric heterogeneities present in the MUTR core. The original input deck for the MUTR model was developed by Dr. Ali Mohamed [4]. Facility drawings provided by the manufacturer at the time of construction of the facility were used to define the geometry of the core and surrounding structures. The geometry of the stainless steel clad fuel elements and control rods were based upon the manufacturing drawings. Representative cross-sectional views of the MCNP model are shown in Figure 3 and Figure 4.

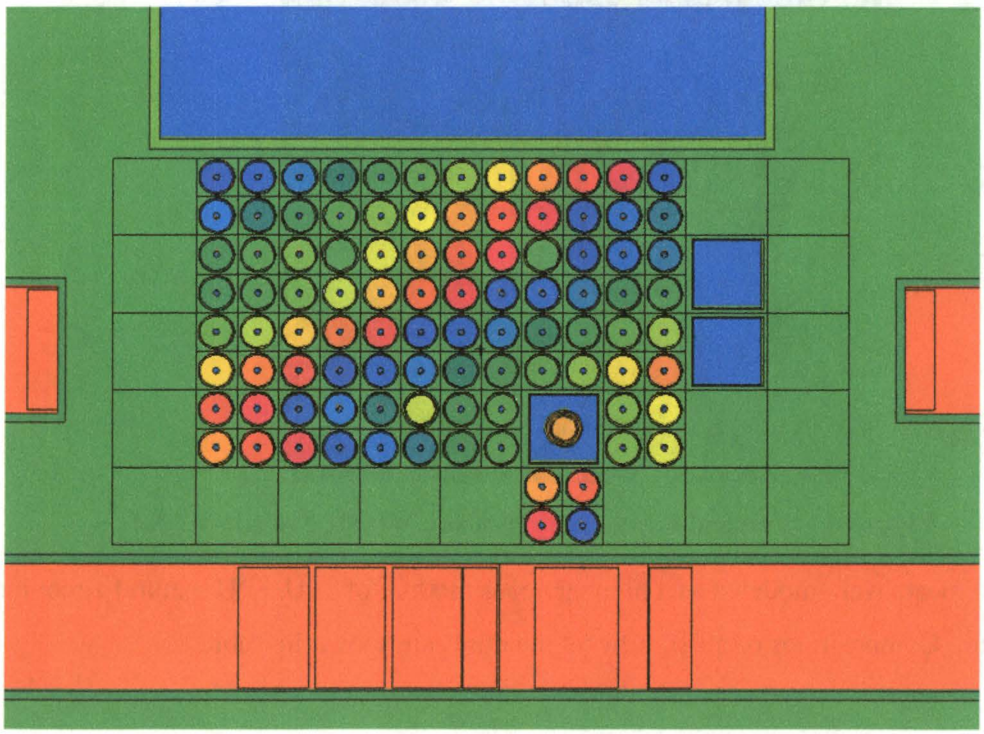

Figure 3 – Horizontal Cross-section of the MUTR MCNP Model

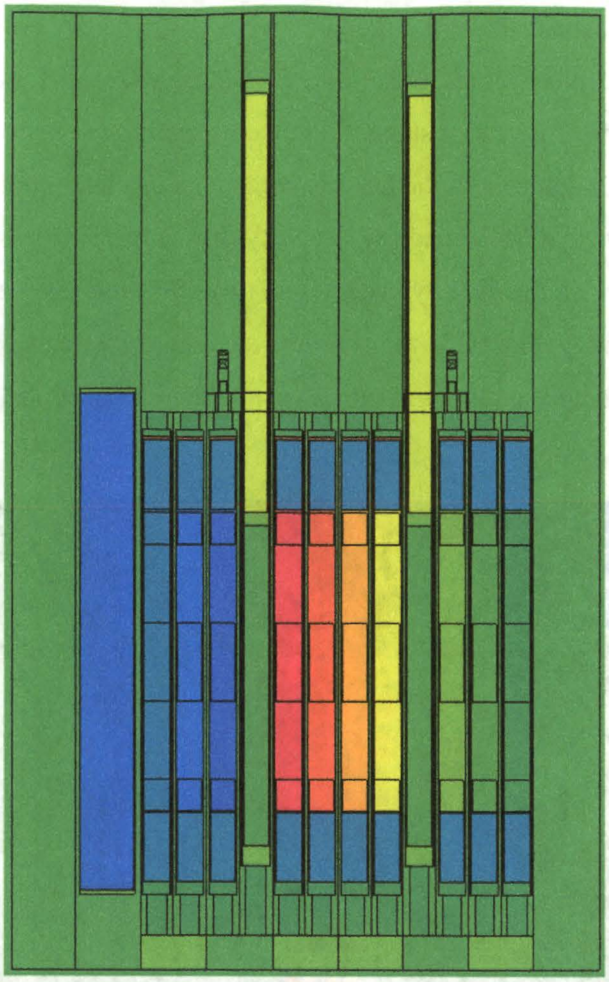

Figure 4 – Vertical Cross-section of the MUTR MCNP Model

Fuel element meats were modeled as a homogeneous mixture of ^{235}U, ^{238}U, natural zirconium and hydrogen. Compositions of all significant materials are shown in Table 2.

Table 2 – Physical Densities and Mass Fractions for Selected Core Components in the MCNP Model of the MUTR

Material	Physical Density [g/cm^3]	Nuclide	Mass Fraction
Standard 8.5 wt% U-ZrH fuel	5.95	^{90}Zr	0.462087
		^{91}Zr	0.100770
		^{92}Zr	0.154029
		^{94}Zr	0.156095
		^{96}Zr	0.025148
		^{1}H	0.016871
		^{235}U	0.017000
		^{238}U	0.068000
Type 304 SS (Fuel Clad)	7.92	Natural Mn	0.020
		Natural Cr	0.190
		Natural Ni	0.095
		Natural Fe	0.695
Graphite	1.6	^{12}C	1.0000
Zirconium Rod	6.5	Natural Zr	1.0000
Pure Aluminum	2.700	^{27}Al	1.0000
Aluminum 6061-T6	2.700	^{27}Al	0.975114
		Natural Cr	0.003837
		Natural Cu	0.005861
		Natural Mg	0.008970
		Natural Si	0.006218
Water	1.000	^{1}H	0.11
		^{16}O	0.89
Air	1.29E-03	^{14}N	0.752308
		^{15}N	0.002960
		^{16}O	0.231687
		^{17}O	0.000094
		^{12}C	0.000124
		^{40}Ar	0.012827

5. Model Bias

Using critical rod height data from control rod calibrations performed at initial fuel loading, a series of MCNP analyses based upon various core configurations were performed to determine the bias of the model. This bias represents such things as differences in material properties that are difficult to determine or unknown (i.e., lack of manufacturer mass spectroscopy data on the exact composition of individual fuel meats and trace elements contained therein) or applicability of cross section data sets used to model the reactor (i.e., interpolation between temperatures). As a result, the validation of the model was based upon the ability of the code to accurately predict criticality as compared with measurements made on the reactor in the summer of 1974.

Kcode calculations were performed using eleven different critical rod height configurations from control rod calibrations performed on June 20th, 1974. These MCNP calculations utilized 100,000 neutrons per cycle for 200 total cycles (150 active cycles). The final k-effectives were compiled and averaged, yielding a model bias of \$1.97 ± \$0.05 (2-sigma error). While this model bias appears to be high, it is similar to bias observed at the NRAD facility [2]. MCNP appears to over-estimate criticality in small MTR-conversion-type TRIGA® reactors, which may also be attributed to the zirconium cross sectional data [6]. This bias will be used to determine reactivity values in the following sections.

6. Beginning-of-Life (BOL) Core Configuration

<u>Effective Delayed Neutron Fraction</u>

The effective delayed neutron fraction for the MUTR core was calculated with MCNP6.1.1 by utilizing the expression:

$$\beta_{eff} = 1 - \frac{k_p}{k_{p+d}}$$

where k_p is the system eigenvalue assuming fission neutrons are born with the energy spectrum for prompt neutrons, and k_{p+d} is the system eigenvalue assuming fission neutrons are born with the appropriately weighted energy spectra for both prompt and delayed neutrons. Using the "totnu" card and running two identical cases, the effective delayed neutron fraction β_{eff} was calculated to be 0.007035 ± 0.000165. This is in reasonable agreement with values predicted in other LEU

TRIGA® cores (i.e., Oregon State University $\beta_{eff} = 0.0076$ [5], Washington State University $\beta_{eff} = 0.0075$) and also the value historically used for the MUTR of $\beta_{eff} = 0.007$. The value $\beta_{eff} = 0.007$ will be used to express all dollar values of reactivities in this report.

Core Excess, Control Rod Worth and Shutdown Margin

Five different MCNP calculations were performed: (1) All control rods in, (2) Shim I rod out, Shim II and Reg rods in, (3) Shim II rod out, Shim I and Reg rods in, (4) Reg rod out, Shim I and Shim II rods in, and (5) All control rods out. Core excess, shutdown margin, and individual rod worths were calculated from these outputs and the reactivity values of these five MCNP calculations (with the bias taken into account) are shown in Table 3.

Table 3 – BOL Rod Worth Calculations

Case	MCNP k-effective	Standard Deviation	Reactivity	Error (2-sigma)
All Rods In	0.97373	0.00018	-$5.82	$0.05
Shim I fully out	0.99922	0.00020	-$2.08	$0.06
Shim II fully out	0.99919	0.00019	-$2.09	$0.05
Reg fully out	0.99231	0.00020	-$3.08	$0.06
All Rods Out	1.03453	0.00019	$2.80	$0.05

These calculations show a core excess of $2.80 ± $0.05. This is well-below the original technical specification limit of $3.50 but higher than the current technical specification limit of $1.12.

Individual rod worths are simply the absolute value of "all rods in" minus the absolute value of the worth of the individual rod. Thus the control rod worths of the rods are Shim I: $3.74 ± $0.06, Shim II: $3.74 ± $0.05, Reg: $2.75 ± $0.06 and the total rod worth of $10.23 ± $0.10. Experimental control rod worth calculations were unavailable for comparison, however the console log book indicates a "shutdown margin" (total rod worth) calculation was performed on 6/20/1974 using a prompt drop approximation. The three control rods were dropped from heights of 99.6%, 26.7% and 100%, respectively. Power promptly dropped from 570 mW to 60 mW and an approximate "shutdown margin" of $8.50 was calculated. Although this is labeled "shutdown margin", we interpret the data to mean rod worth. Additionally, the accuracy of this datum is unknown. In comparison, the MCNP-predicted rod worths in this configuration, adding the full value of Shim I and Reg ($3.74 + $2.75 = $6.49) and approximating the value of Shim II at 26.7% withdrawn (approximately $0.75), yields a total shutdown margin of $7.24.

The technical specification definition of shutdown margin is "the minimum reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible OPERATING condition and with the most reactive rod in its most reactive position." The most reactive rod is the Shim II rod. Total rod worth minus the Shim II rod is $6.49 ± $0.10. NRC shutdown margin is this value minus the core excess, which would be $3.69 ± $0.10, which is far above the technical specification limit of $0.50. It should be noted that the implication of the results is that the MUTR was initially loaded with more than enough fuel to achieve 250 kW.

Prompt Fuel Temperature Coefficient

The prompt-temperature coefficient associated with the MUTR fuel, α_F, was calculated by varying the fuel meat temperature while leaving other core parameters fixed. The MCNP model was used to simulate the reactor with all rods out at 293, 600, 900, 1200 and 2500 K. The prompt-temperature coefficient for the fuel was calculated at the mid-point of the four temperature intervals. The results are shown in Figure 5 and tabulated in Table 4. Results from GA were added to show similarity [7]. The prompt-temperature coefficient is observed to be negative for all evaluated temperature ranges with decreasing magnitude as temperature increases. The coefficient has a value of -1.6¢/°C at 446.8 K, which is similar to the value of -1.2¢/°C stated in the original SAR.

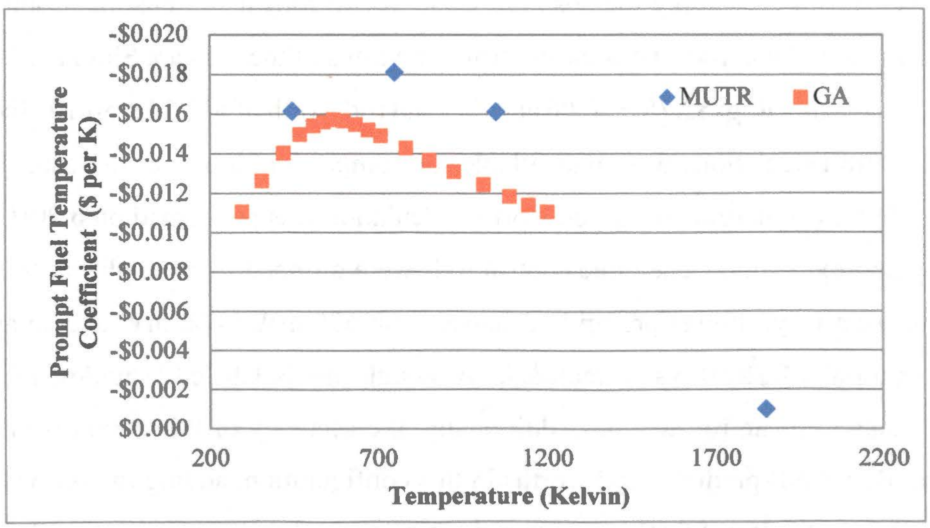

Figure 5 – BOL Prompt Temperature Coefficient, α_F, as a Function of Temperature

Table 4 – BOL Prompt Temperature Coefficient

Fuel Temperature [K]	Prompt Temperature Coefficient [$/°C]
446.8	-$0.0162
750	-$0.0181
1050	-$0.0161
1850	-$0.0010

Moderator Void Coefficient

The moderator void coefficient of reactivity was also determined using the MCNP model. The voiding of the core was introduced by uniformly reducing the density of the liquid moderator in the entire core. The calculation was performed from 0% to 100% voiding at 10% intervals. The void coefficient was negative for every interval and steadily decreased, as can be seen in Figure 6.

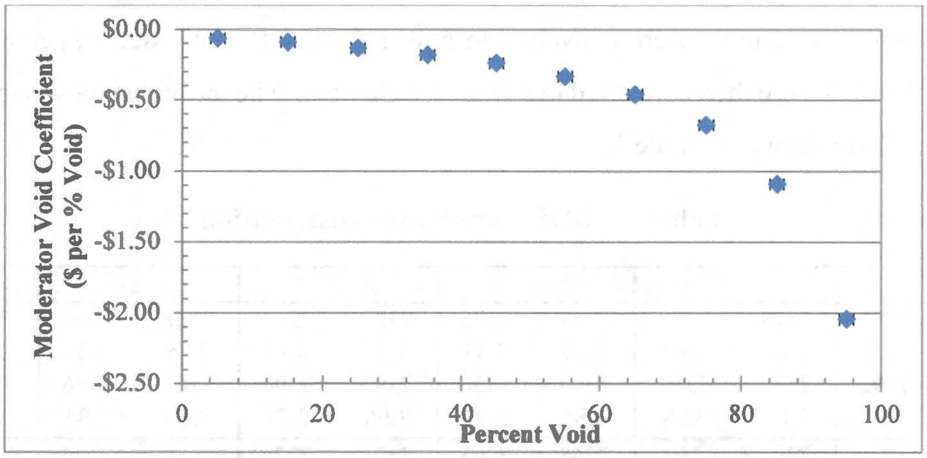

Figure 6 – BOL Moderator Void Coefficient

Moderator Temperature Coefficient

The moderator temperature coefficient of reactivity, α_M, was determined by varying the moderator density with respect to temperature within the MCNP model MUTR core from the expected operating temperature range of 20°C to 50°C (using Engineering Toolbox [8] to determine water density). The results are shown in Figure 7. The moderator temperature coefficient is calculated to slightly increase from 20˚C to 25 ˚C and from 45 ˚C to 50 ˚C, but these changes are on the order of $0.01/˚C and all points (with 2-sigma error) are bounded around zero. The moderator temperature coefficient appears to be negligible.

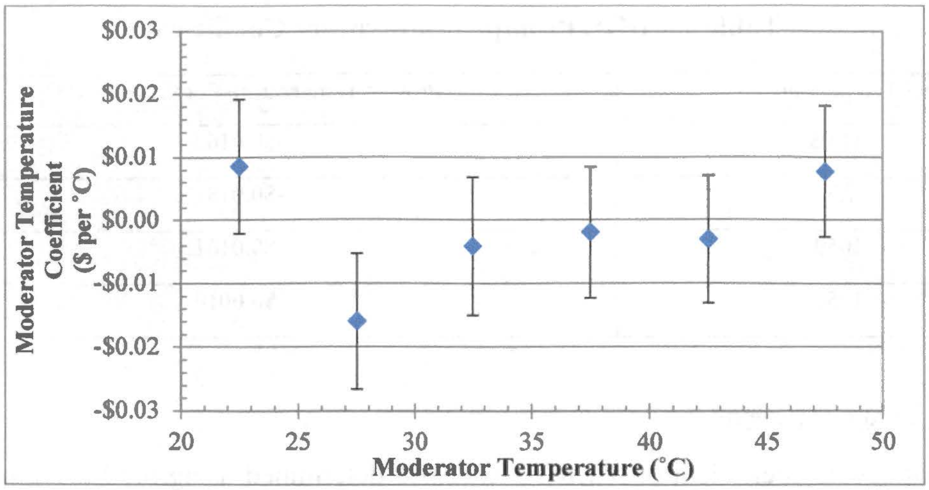

Figure 7 – BOL Moderator Temperature Coefficient

<u>Core Power Distribution</u>

F4 flux tallies were used to determine the power-per-element. The tallies output as a fluence per fission neutron. These units were converted to power density (W/cm^3, see Appendix for more information) which were then converted to power-per-element. The individual power-per-element values (in kW) are shown in Table 5.

Table 5 – BOL Core Power Distribution

	8		7		6		5		4		3	
F	7390	7391	7378	7379	7354	7355	7395	7393	7168	7169	7333	7335
	1.60	2.01	2.49	2.89	3.19	3.33	3.31	3.15	2.85	2.49	2.02	1.64
	7389	7392	7377	7380	7353	7356	7397	7396	7167	7166	7334	7336
	1.84	2.33	3.01	3.78	3.85	3.97	3.96	3.77	3.73	2.94	2.33	1.84
E	7161	7026	7398	259	7368	7365	7374	7375	304	7406	7342	7343
	2.07	2.66	3.67	0.00	4.63	4.54	4.52	4.53	0.00	3.59	2.64	2.01
	7028	7027	7399	7400	7367	7366	7373	7376	7404	7405	7341	7344
	2.18	2.84	3.67	4.54	4.66	4.78	4.74	4.52	4.37	3.50	2.77	2.09
D	7408	7409	7345	7346	7382	7383	7371	7372	7290	7330	7164	7165
	2.15	2.85	3.58	4.10	4.50	4.66	4.62	4.31	3.88	3.31	2.67	2.04
	7407	7160	7348	7347	7381	7384	7370	7369	7332	7331	7163	7162
	1.96	2.56	3.22	3.72	4.08	4.46	4.14	3.81	3.45	3.01	2.40	1.85
C	7360	7357	7352	7349	7401	260	7388	7385			7362	7363
	1.59	2.09	2.65	3.05	3.53	0.00	3.55	3.17			2.09	1.57
	7359	7358	7351	7350	7403	7402	7387	7386			7361	7364
	1.28	1.69	2.15	2.43	2.53	2.84	2.72	2.66			1.85	1.33
B									7338	7337		
									1.88	1.68		
									7339	7340		
									1.13	0.98		

The orange bundles indicate control rod clusters, yellow indicates the IFE, and red indicates the hottest fuel element location. The hottest fuel element in the core is located in the southeast corner of the bundle in E6, with a maximum power of 4.78 kW (at a total core power of 300 kW). The IFE produces greater than 50% of the hottest fuel element so it is sufficient in its current location.

Burnup

The MUTR reports burnup each year (in MW-hr and grams ^{235}U) as part of their annual report [9]. Summing the annually-reported burnup yields a total burnup of 557.272 MW-hr, or 23.22 MW-days, with a reported burnup of 21.93 grams of ^{235}U. Three annual reports were unable to be found (1996, 2001, 2003) but based upon operating trends around those years, the expected unaccounted-for burnup should not be significant. Thus, for a burnup calculation, a total burnup of 25 MW-days was assumed. The MCNP BURN option was then utilized to perform a 25 MW-day burnup calculation.

The results of the burnup calculation indicate very minimal burnup. The average burnup of fuel was 0.92% of original fuel loading (3441 grams of ^{235}U), or 31.76 grams of ^{235}U. This is slightly higher than the reported total of 21.93 grams, but is still far below the technical specification limit of 50% ^{235}U burnup (1720.5 grams of ^{235}U).

7. Current Core Configuration

After performing the burnup calculation, the burned-up fuel isotopics were parsed from the burnup outputs and re-inserted into the MCNP deck. The previous safety analyses were performed again in the current core configuration.

Core Excess, Control Rod Worth and Shutdown Margin

The five configurations were performed again and the results are shown in Table 6.

Table 6 – Current Rod Worth Calculations

Case	MCNP k-effective	Standard Deviation	Reactivity	Error (2-sigma)
All Rods In	0.95703	0.00019	-$8.38	$0.05
Shim I fully out	0.98201	0.00020	-$4.59	$0.06
Shim II fully out	0.98248	0.00020	-$4.52	$0.06
Reg fully out	0.97544	0.00020	-$5.57	$0.06
All Rods Out	1.01637	0.00017	$0.33	$0.05

These calculations show a core excess of \$0.33 ± \$0.05. This would appear to show agreement with current reactor status, as the MUTR is currently unable to reach full licensed power of 250 kW. Thirty-three cents of core excess would not be sufficient to overcome the negative temperature coefficient at the point of adding heat.

Individual rod worths are simply the absolute value of "all rods in" minus the absolute value of the worth of the individual rod. Thus the control rod worths of the rods are Shim I: \$3.80 ± \$0.05, Shim II: \$3.87 ± \$0.06, Reg: \$2.82 ± \$0.06 and the total rod worth of \$10.48 ± \$0.10. These values appear to be greater than values calculated during rod calibrations. The last available rod calibration data yielded rod worths of \$1.81 for Shim I, \$2.58 for Shim II and \$2.54 for Reg, for a total rod worth of \$6.93.

The technical specification definition of shutdown margin is "the minimum reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible OPERATING condition and with the most reactive rod in its most reactive position." The most reactive rod is the Shim II rod. Total rod worth minus the Shim II rod is \$6.61 ± \$0.10. NRC shutdown margin is this value minus the core excess, which would be \$6.28 ± \$0.10, which is far above the technical specification limit of \$0.50.

Prompt Fuel Temperature Coefficient

The prompt-temperature coefficient calculation results are shown in Figure 8 and tabulated in Table 7 with the GA results added to show similarity [7]. The prompt-temperature coefficient is observed to be negative for all evaluated temperature ranges with decreasing magnitude as temperature increases. The coefficient once again has a value of -1.6¢/°C at 446.8 K, which is similar to the value of -1.2¢/°C stated in the original SAR.

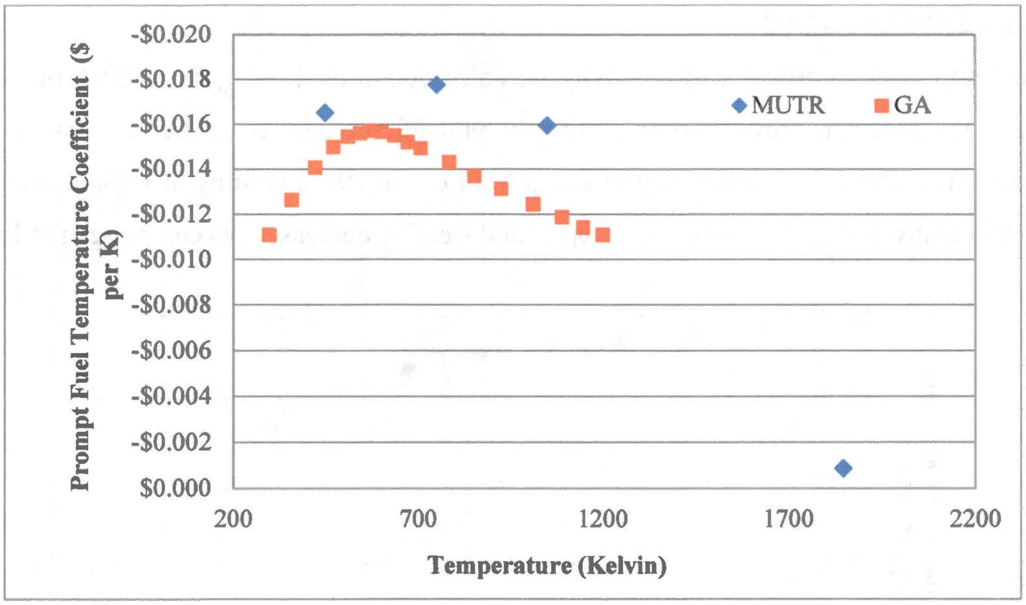

Figure 8 – Current Prompt Temperature Coefficient, α_F, as a Function of Temperature

Table 7 – Current Prompt Temperature Coefficient

Fuel Temperature [K]	Prompt Temperature Coefficient [$/°C]
446.8	-$0.01651
750	-$0.01776
1050	-$0.01595
1850	-$0.00086

Moderator Void Coefficient

The moderator void coefficient of reactivity was also determined using the MCNP model. The voiding of the core was introduced by uniformly reducing the density of the liquid moderator in the entire core. The calculation was performed from 0% to 100% voiding at 10% intervals. The void coefficient was negative for every interval and steadily decreased, as can be seen in Figure 9.

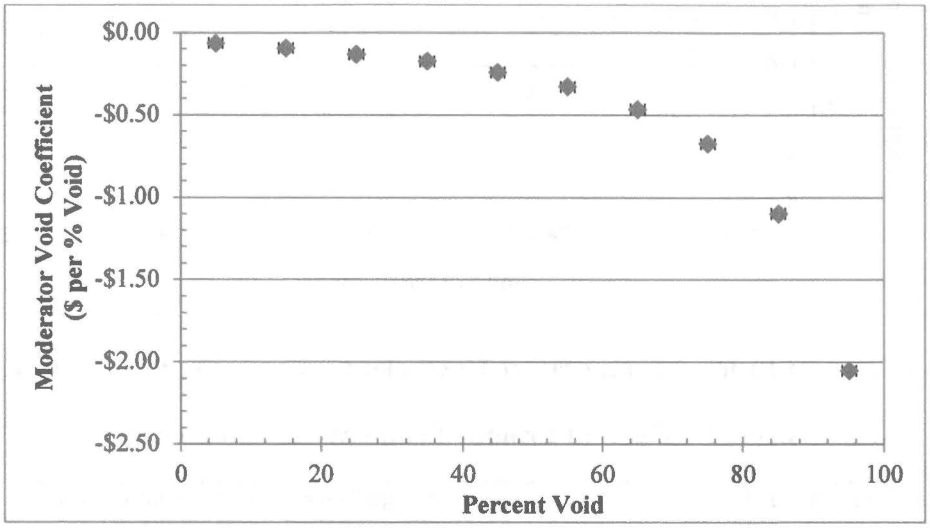

Figure 9 – Current Moderator Void Coefficient

Moderator Temperature Coefficient

The moderator temperature coefficient of reactivity, α_M, was determined by varying the moderator density with respect to temperature within the MCNP model MUTR core from the expected operating temperature range of 20°C to 50°C (using Engineering Toolbox [8] to determine water density). The results are shown in Figure 10. The moderator temperature coefficient is calculated to slightly increase from 25°C to 30 °C and from 40 °C to 45 °C, but these changes are on the order of $0.01/°C and all points (with 2-sigma error) are bounded around zero. The moderator temperature coefficient appears to be negligible.

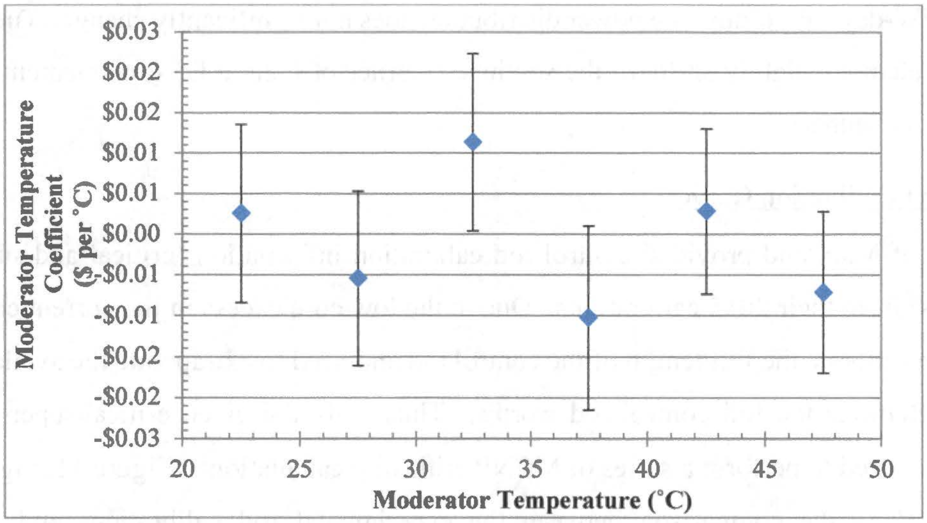

Figure 10 – Current Moderator Temperature Coefficient

Core Power Distribution

The power-per-element calculations were once again performed and the values are seen in Table 8.

Table 8 – Current Core Power Distribution

	8		7		6		5		4		3	
F	7390	7391	7378	7379	7354	7355	7395	7393	7168	7169	7333	7335
	1.56	1.98	2.44	2.82	3.08	3.24	3.24	3.09	2.82	2.47	2.01	1.64
	7389	7392	7377	7380	7353	7356	7397	7396	7167	7166	7334	7336
	1.78	2.27	2.90	3.68	3.71	3.84	3.83	3.70	3.68	2.94	2.32	1.86
E	7161	7026	7398	259	7368	7365	7374	7375	304	7406	7342	7343
	2.01	2.59	3.56	0.00	4.48	4.39	4.39	4.45	0.00	3.57	2.63	2.03
	7028	7027	7399	7400	7367	7366	7373	7376	7404	7405	7341	7344
	2.13	2.75	3.55	4.38	4.49	4.63	4.63	4.40	4.30	3.47	2.75	2.10
D	7408	7409	7345	7346	7382	7383	7371	7372	7290	7330	7164	7165
	2.11	2.72	3.47	3.99	4.36	4.53	4.48	4.22	3.79	3.28	2.65	2.04
	7407	7160	7348	7347	7381	7384	7370	7369	7332	7331	7163	7162
	1.92	2.51	3.15	3.63	4.00	4.36	4.03	3.75	3.40	2.97	2.40	1.86
C	7360	7357	7352	7349	7401	260	7388	7385			7362	7363
	1.57	2.05	2.59	2.97	3.46	0.00	3.47	3.11			2.10	1.59
	7359	7358	7351	7350	7403	7402	7387	7386			7361	7364
	1.27	1.66	2.09	2.38	2.50	2.81	2.65	2.62			1.86	1.35
B									7338	7337		
									1.89	1.70		
									7339	7340		
									1.12	0.98		

After 25 MW-days of burnup, the power distribution does not significantly change. The maximum power per element slightly shifts to the southwest corner of bundle E5 (one element east of the previous hot channel).

Control Rod Calibration Curves

University of Maryland provided control rod calibration information (critical and super-critical rod heights) from their 2015 calibrations. Due to the low core excess in the current core, MUTR is unable to calibrate the full length of the control rod and need to extrapolate the available data in order to determine the full control rod worths. Thus, only the given critical/super-critical rod heights were used to perform a series of MCNP criticality calculations. Figure 11, Figure 12, and Figure 13 show the comparison between the experimental rod calibrations and the MCNP prediction.

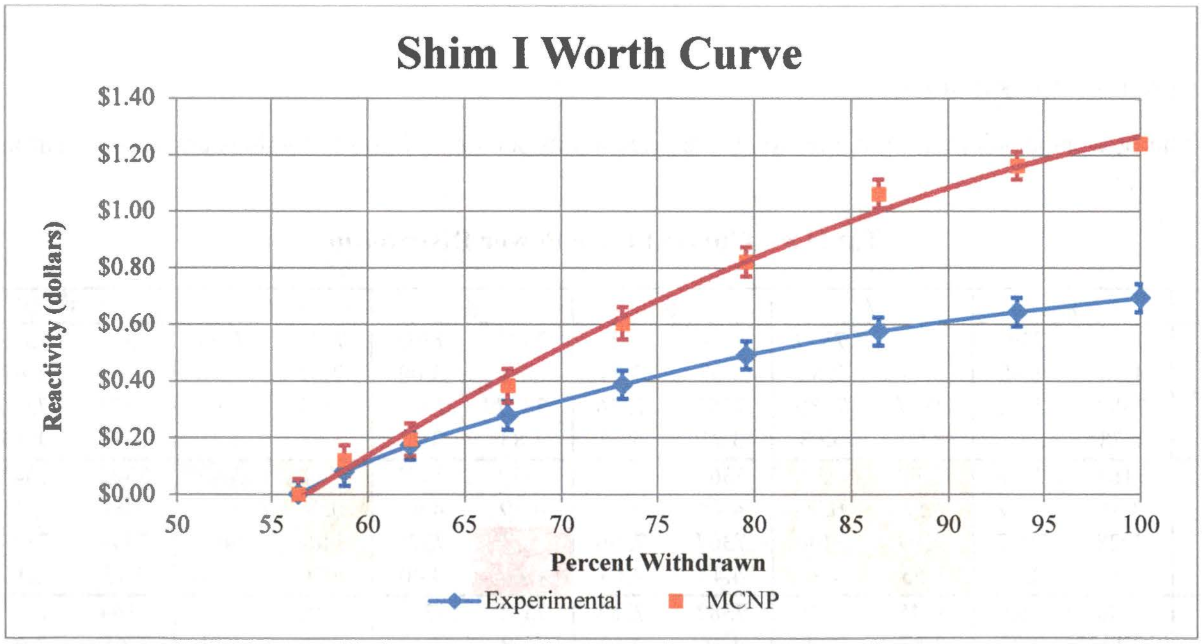

Figure 11 – Shim I Rod Calibration Curves (Experimental vs. MCNP Prediction)

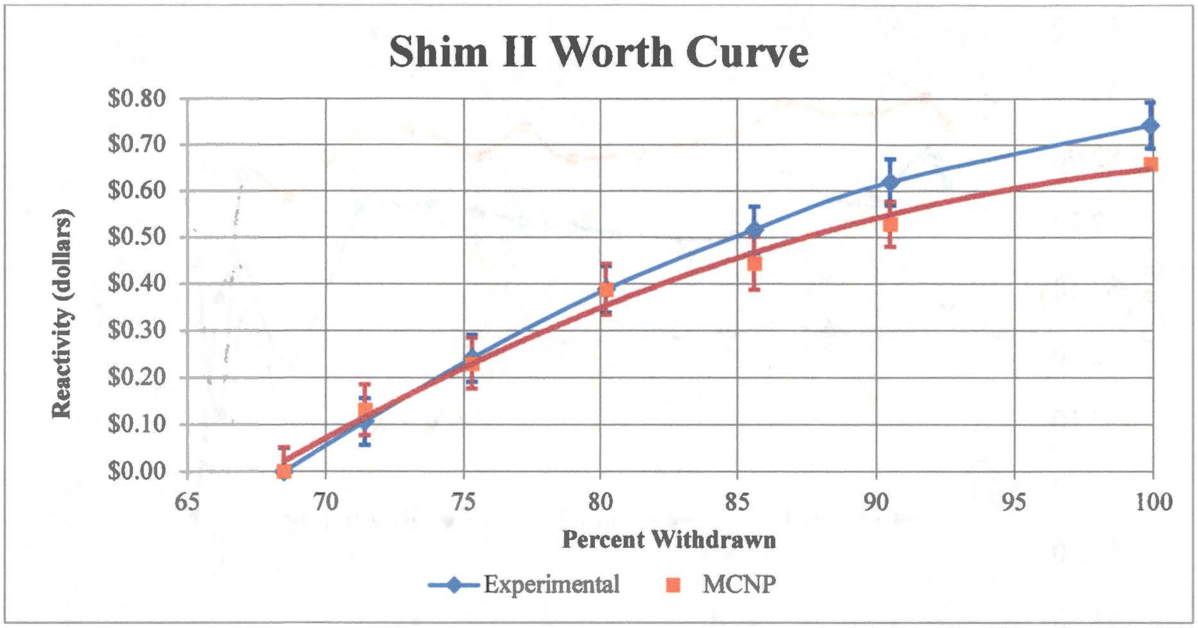

Figure 12 – Shim II Rod Calibration Curves (Experimental vs. MCNP Prediction)

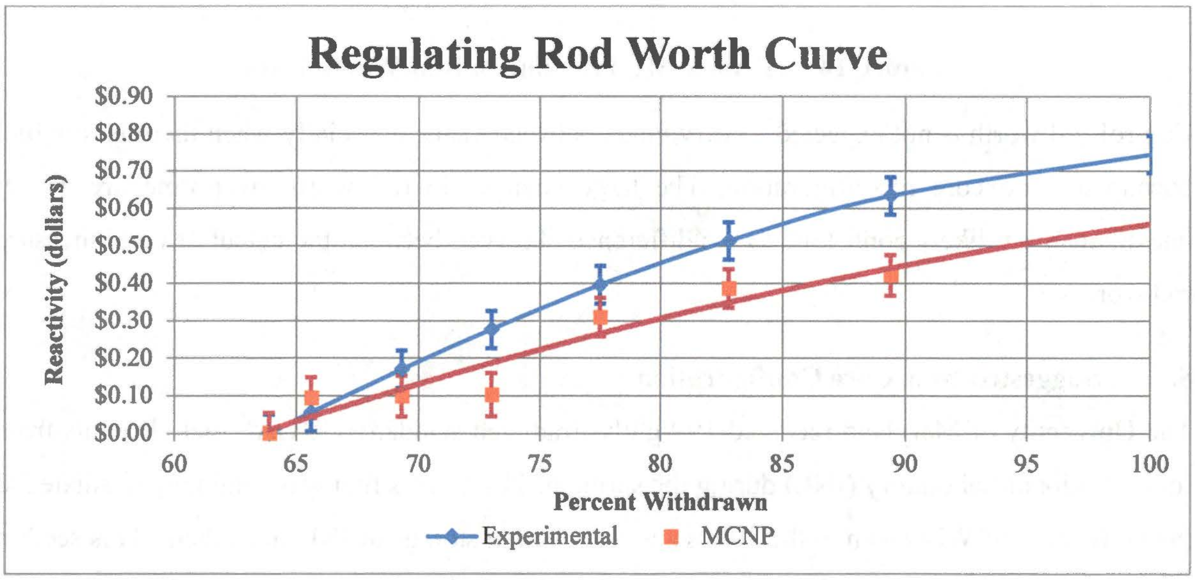

Figure 13 – Regulating Rod Calibration Curves (Experimental vs. MCNP Prediction)

There appears to be a disagreement between the MCNP compared to the experimental control rod data, but there may be some uncertainty associated with the experimental control rod data. Figure 14 shows historic MUTR rod worth data compiled over three decades.

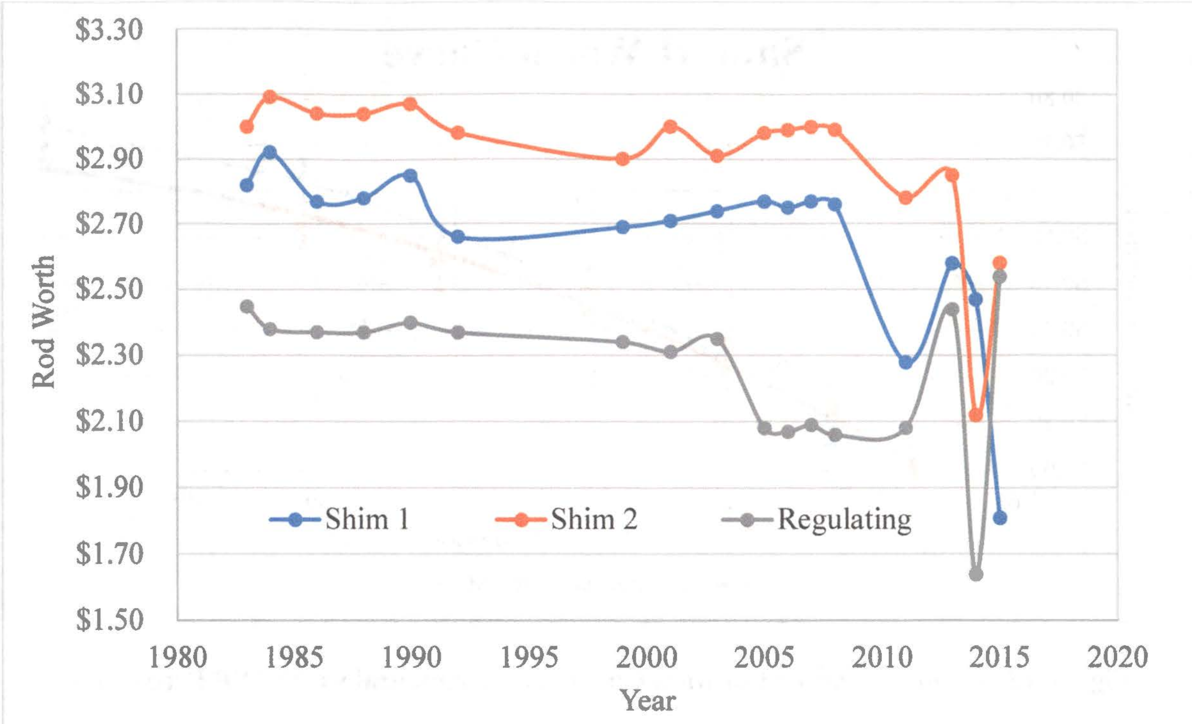

Figure 14 – Historic MUTR Control Rod Worth Data

Control rod worth is not expected to vary much between years, especially when there is low fuel burnup and no core reconfiguration. The large changes in rod worth over time are as yet unexplained but likely contribute to the difference observed between the calculated and measure rod worths.

8. Suggested New Core Configuration

The University of Maryland received 19 lightly-irradiated standard TRIGA® fuel elements from Idaho National Laboratory (INL) during the spring of 2017. This fuel was originally irradiated at the University of Wisconsin in the 1970s and had been in storage at INL since then. This section will suggest a new core configuration that utilizes this fuel to improve reactor efficiency while maintaining proper safety margins. Idaho National Laboratory provided burnup information from Wisconsin and this data was incorporated into the MCNP decks.

Table 9 shows the suggested new core configuration. For this analysis, it is suggested that sixteen fuel elements be added to the core to increase the core inventory to 109 total fuel elements. The

green highlighted clusters are the new fuel additions. Column 2 (not shown) maintains graphite reflectors in locations E2 and D2.

Table 9 – Suggested New Core Configuration (109 Element Core)

	8		7		6		5		4		3	
F	7390	7391	7378	7379	7354	7355	7395	7393	7168	7169	7333	7335
	7389	7392	7377	7380	7353	7356	7397	7396	7167	7166	7334	7336
E	7161	7026	7398	259	7368	7365	7374	7375	304	7406	7342	7343
	7028	7027	7399	7400	7367	7366	7373	7376	7404	7405	7341	7344
D	7408	7409	7345	7346	7382	7383	7371	7372	7290	7330	7164	7165
	7407	7160	7348	7347	7381	7384	7370	7369	7332	7331	7163	7162
C	7360	7357	7352	7349	7401	260	7388	7385	Rabbit		7362	7363
	7359	7358	7351	7350	7403	7402	7387	7386			7361	7364
B	6286	6284	5861	6281	PuBe Source		6287	6289	7338	7337	6282	6277
	6283	6285	5862	5864			6279	6290	7339	7340	6288	6268

Effective Delayed Neutron Fraction

Once again using the "totnu" card and running two identical cases, the effective delayed neutron fraction β_{eff} was calculated to be 0.007244 ± 0.000160. There is a slight increase in β_{eff} compared to the beginning-of-life, but 0.007 will continue to be used to express all dollar values of reactivities in this report.

Core Excess, Control Rod Worth, and Shutdown Margin

The same five MCNP rod worth calculations were performed again for the new core configuration: (1) All control rods in, (2) Shim I rod out, Shim II and Reg rods in, (3) Shim II rod out, Shim I and Reg rods in, (4) Reg rod out, Shim I and Shim II rods in, and (5) All control rods out. Core excess, shutdown margin, and individual rod worths were calculated from these outputs and the reactivity values (with the bias taken into account) of each of these five calculations are shown in Table 10.

Table 10 – New Core Configuration Rod Worth Calculations

Case	MCNP k-effective	Standard Deviation	Reactivity	Error (2-sigma)
All Rods In	0.97872	0.00021	-$5.08	$0.06
Shim I fully out	0.99926	0.00019	-$2.08	$0.05
Shim II fully out	0.99992	0.00018	-$1.98	$0.05
Reg fully out	1.00106	0.00021	-$1.82	$0.06
All Rods Out	1.03513	0.00016	$2.88	$0.05

These calculations show a core excess of $2.88 ± $0.05. This is below the original technical specification limit of $3.50 but higher than the current technical specification limit of $1.12.

Individual rod worths are simply the absolute value of "all rods in" minus the worth of the individual rod. Thus the control rod worths of the rods are Shim I: $3.00 ± $0.05, Shim II: $3.09 ± $0.05, Reg: $3.26 ± $0.06 and the total rod worth of $9.35 ± $0.08.

Now the most reactive rod is the Regulating Rod, due to having more fuel near the through tube, making the Regulating Rod more valuable. Total rod worth minus the Regulating Rod is $6.09 ± $0.09. NRC shutdown margin is this value minus the core excess, which would be $3.22 ± $0.10, which is still far above the technical specification limit of $0.50.

Prompt Fuel Temperature Coefficient

The results of the new core configuration prompt fuel temperature coefficient calculations are shown in Figure 15 and tabulated in Table 11.

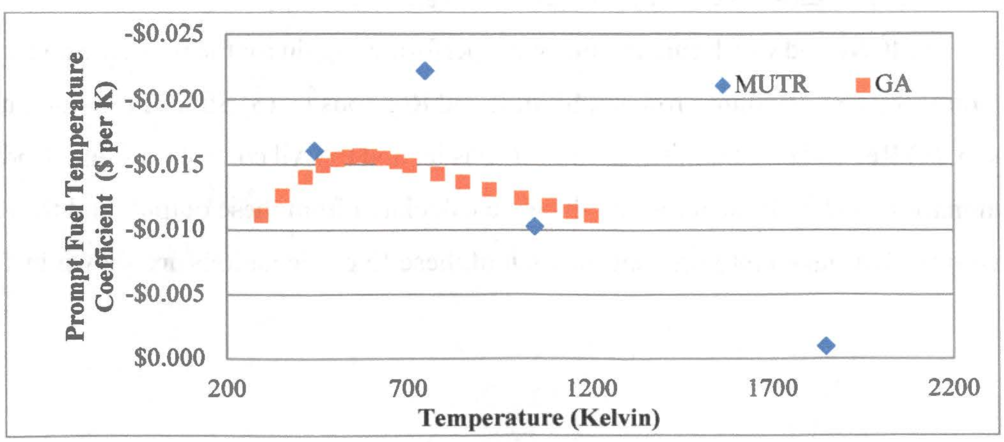

Figure 15 – New Core Prompt Temperature Coefficient, α_F, as a Function of Temperature

Table 11 – New Core Prompt Temperature Coefficient

Fuel Temperature [K]	Prompt Temperature Coefficient [$/°C]
446.8	-$0.01605
750	-$0.01739
1050	-$0.01504
1850	-$0.00095

These values are similar to the original beginning-of-life coefficients.

Moderator Void Coefficient

Figure 16 shows the moderator void coefficient in the suggested new core configuration.

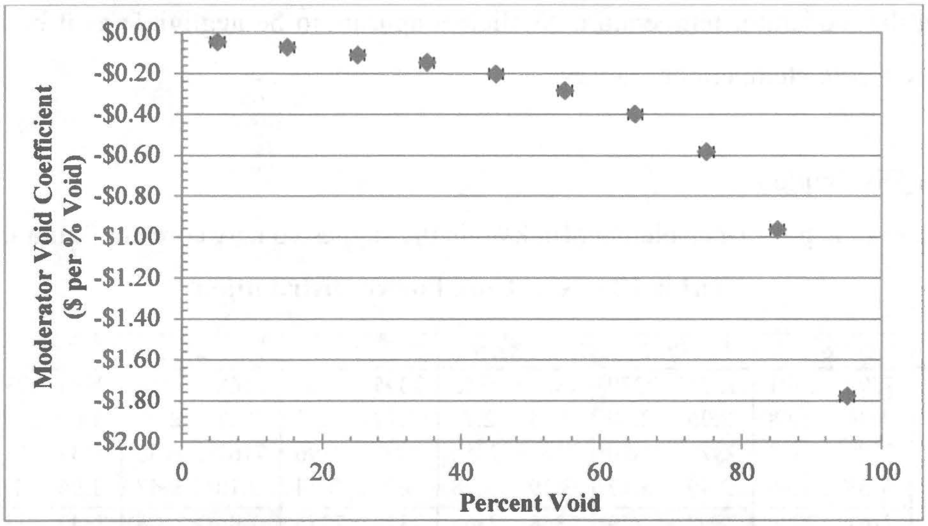

Figure 16 – New Core Configuration Moderator Void Coefficient

The void coefficient was negative for every interval and steadily decreased, similar to the beginning-of-life configuration. The void coefficient is slightly less negative in this core configuration but is still negative overall.

Moderator Temperature Coefficient

Figure 17 shows the power-per-element (in kW) in the suggested new core configuration.

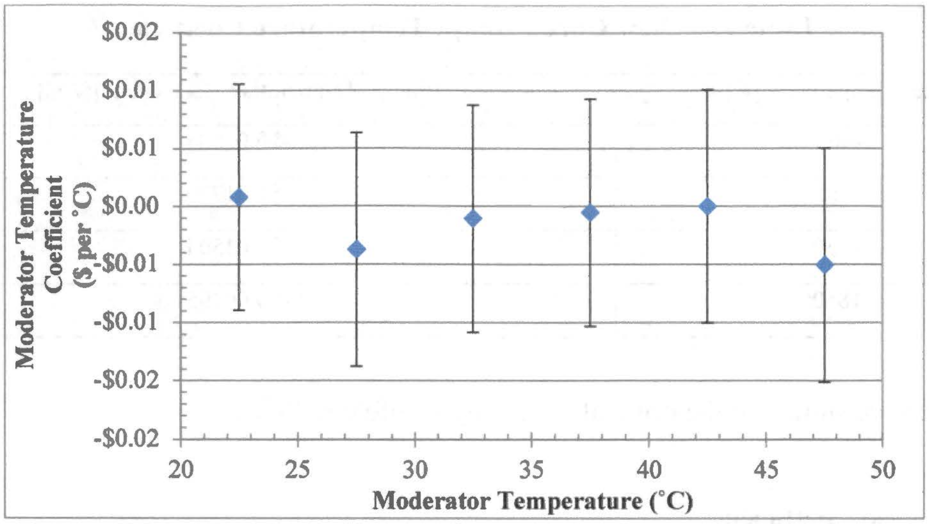

Figure 17 – New Core Configuration Moderator Temperature Coefficient

Once again the moderator temperature coefficient appears to be negligible as it bounds around $0.00 at all observed temperature ranges.

<u>Core Power Distribution</u>

Table 12 shows the power-per-element (in kW) in the suggested new core configuration.

Table 12 – New Core Power Distribution

	8		7		6		5		4		3	
F	7390	7391	7378	7379	7354	7355	7395	7393	7168	7169	7333	7335
	1.34	1.68	2.06	2.39	2.63	2.73	2.72	2.59	2.38	2.07	1.69	1.39
	7389	7392	7377	7380	7353	7356	7397	7396	7167	7166	7334	7336
	1.53	1.96	2.49	3.17	3.19	3.28	3.25	3.14	3.10	2.47	1.96	1.59
E	7161	7026	7398	259	7368	7365	7374	7375	304	7406	7342	7343
	1.78	2.28	3.12	0.00	3.88	3.80	3.78	3.82	0.00	3.03	2.23	1.75
	7028	7027	7399	7400	7367	7366	7373	7376	7404	7405	7341	7344
	1.96	2.50	3.20	3.95	4.01	4.13	4.09	3.90	3.79	3.05	2.43	1.86
D	7408	7409	7345	7346	7382	7383	7371	7372	7290	7330	7164	7165
	2.01	2.58	3.25	3.73	4.04	4.16	4.10	3.86	3.49	3.01	2.43	1.87
	7407	7160	7348	7347	7381	7384	7370	7369	7332	7331	7163	7162
	1.96	2.52	3.14	3.58	3.87	4.23	3.88	3.61	3.31	2.91	2.32	1.79
C	7360	7357	7352	7349	7401	260	7388	7385			7362	7362
	1.79	2.28	2.85	3.19	3.62	0.00	3.62	3.25			1.82	2.19
	7359	7358	7351	7350	7403	7402	7387	7386			7361	7361
	1.53	1.95	2.40	2.65	2.84	3.12	2.80	2.76			1.61	1.93
B	6286	6284	5861	6281			6287	6289	7338	7337	6282	6277
	1.21	1.54	1.86	2.13			2.23	2.12	2.12	1.91	1.45	1.11
	6283	6285	5862	5864			6279	6290	7339	7340	6288	6268
	0.87	1.11	1.36	1.63			1.72	1.53	1.43	1.27	1.03	0.81

The hottest fuel element in now FE 7384 in the southeast corner of location D6. This makes sense as the core geometry is closer to a square, which would better centralize the location of the maximum power production. Also, the hottest power-per-element is 4.23 kW, which is far lower than the previous high of 4.78 from BOL, due to a higher fuel loading spreading out the power.

9. Suggested Changes to Technical Specifications

This section contains suggested changes to technical specifications, using the previous sections' analyses, in order to improve MUTR operation and efficiency.

<u>Change to Section 3.1</u>

3.1 Reactor Core Parameters

1. *The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place shall not be greater than $1:12.*

This report shows that the MUTR can be safely operated with a core excess of $2.88, as the shutdown margin would exceed $2.00. It is suggested that this specification be rewritten as such:

1. *The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place shall not be greater than $3.50.*

This would return the core excess specification to the previous technical specification limit of $3.50. This value allows the MUTR the ability to add new fuel in order to improve operational efficiency.

<u>Change to Section 3.3</u>

3.3 Primary Coolant System

<u>*Objectives*</u>

4. The pool water temperature shall not exceed 90°C, as measured by thermocouples located in the pool.

While the reactor could safely operate at this temperature, there is potential for tank degradation. It is suggested that this specification be rewritten as such:

4. The pool water temperature shall not exceed 49°C (120°F), as measured by thermocouples located in the pool.

Change to Section 5.3

5.3 Reactor Core and Fuel

1. *The core shall consist of 93 TRIGA fuel elements assembled into 24 fuel bundles, 21 bundles shall contain four fuel elements and 3 bundles shall contain three fuel elements and a CONTROL ROD guide tube*
2. *The fuel bundles shall be arranged in a rectangular 4 x 6 configuration, with one bundle displaced for the in-core pneumatic experimental system.*
3. *The reactor shall not be operated at power levels exceeding 250 kW.*
4. *The reflector shall be a combination of two graphite reflectors.*

This report shows that the MUTR can accommodate more fuel than was originally loaded. The addition of fuel can allow the MUTR to return to 250kW operation as well as improve the flux in the rabbit and beam port facilities. It is suggested that this specification be rewritten as such:

1. *The core shall consist of TRIGA® fuel elements assembled into three or four element fuel bundles.*
2. *The fuel bundles shall be arranged in a close-packed rectangular 5 x 9 configuration, with bundles displaced for the in-core pneumatic experimental system, PuBe source, neutron detectors, and graphite reflector elements.*
3. *The reactor shall not be operated at power levels exceeding 250 kW.*
4. *The reflector shall be a combination of graphite reflectors and water.*

This technical specification change allows the MUTR to incorporate all of the fuel received from INL. This analysis shows that the MUTR can safely operate with all newly-received fuel elements added to the current core configuration. Due to the short core life of standard TRIGA® fuel elements (approximately 100 MW-days [1]), the core needs to be over-loaded to compensate for reactivity loss due to fuel depletion and poison buildup. A similar over-loading was performed at Texas A&M in the 1960s [10]. When Texas A&M increased maximum power from 100 kW to 1 MW, they increased their fuel element inventory to a 126-element (completely full) core.

A close-packed "5 X 9" configuration allows MUTR the freedom to place fuel and graphite reflectors in any available position within columns 1 through 9 and rows B through F, as long as other limits (such as shutdown margin) are maintained. Finally, "the reflector shall be a

combination of graphite reflectors and water" allows MUTR the freedom to place future reflector elements on the periphery of the core to increase core efficiency as long as other limits (such as shutdown margin) are maintained.

It is also suggested that the reactor power limit be increased to 300 kW so that a regular operating power of 250 kW can be produced without fear of technical specification violation during full-power operations or power calibrations, but the thermal hydraulic calculations to support this increase are beyond the scope of this report.

10. Summary

MCNP6.1.1 was used to calculate fundamental and operational parameters for the Maryland University Training Reactor to demonstrate the reactor's adherence to safety margins in the technical specifications. Values of fundamental parameters agree well with theoretical values. Values of operational parameters agree well with measured values. The results of this study indicate that the MUTR can be operated safely within the Technical Specification bounding envelope. These results also confirms that the MUTR currently cannot operate at full power of 250 kW.

Further analysis was performed to determine a more efficient core configuration, incorporating 16 lightly-irradiated fuel elements delivered from Idaho National Laboratory. This analysis indicates that the MUTR can safely operate with 16 extra fuel elements installed, though technical specifications will need to be revised to allow this operation. This new core configuration increases the flux in irradiation facilities (beam ports, pneumatic transfer facility) and provides more core excess reactivity, which will prolong operating lifetime.

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APPENDIX - Explanation of Power-Per-Element Calculations

Power-per-elements are calculated by using an F4 tally (which outputs in neutrons per cm^2-fission neutron) and then using an FM multiplier card to convert the F4 tally to W/cm^3.

The FM4 multiplier is as follows:

FM4:n (-3705.6 7374 -6 -8)

FM4 begins with fluence per fission neutron (neutron per cm^2-fission)

-3705.6 is the multiplier (negative sign uses units of atoms/barn-cm)

7374 is fuel material (material does not need to be same as defined in cell)

-6 calls the microscopic fission cross-section (units of barn-fission/neutron-atoms)

-8 uses the fission energy (units of MeV/fission)

Units work out to MeV/cm^2-fission neutron

This is the factor-label method to convert these units:

$$\frac{n}{cm^2 - fisneutron} \cdot \frac{atoms}{b-cm} \cdot \frac{b-fis}{n-atoms} \cdot \frac{MeV}{fis} \cdot \frac{3.0E5W}{} \cdot \frac{2.442 fisneutrons}{fis} \cdot \frac{fis}{197.7MeV}$$

Notice that the first four terms are multipliers from the FM card as explained above.

3.0E5W is the power level of the reactor (300 kW). 2.442 fission neutrons per fission was determined by MCNP. 197.7 MeV/fission is a standard energy used in fission energy calculations. Multiplying these three values gives 3705.6, which is the multiplier used in the first part of the FM card.

Attachment 2

REVISED START-UP PLAN

Startup Plan-Additional Reactor Fuel
Within 6 months following the completion of the loading of additional reactor fuel into the core, the following information will be summarized and submitted to the NRC.

1. **Initial Assembly of Bundles**
Before being assembled into bundles, the new fuel elements will be visually inspected using an underwater camera. Once the bundles are assembled, the fit of the parts will be verified with an underwater camera.

2. **Adding Bundles to the Core**
Bundles will be added to the core 1 at a time. An underwater camera will be used to verify that the bundles are fully seated in the grid plate.

3. **Measurements to be Made After Achieving Criticality**

3.1. Control Rod Calibrations
The MUTR is equipped with 3 control rods that are routinely calibrated using the positive asymptotic method. Current measurements and simulation results compiled in Table 3.

Table 3: Rod worth measurements and calculations.

	BOL(MCNP)	Current (MCNP)	Current(Measured)	New Configuration(MCNP)
Reg Rod	\$2.75	\$2.82	\$2.20	\$3.26
Shim 1	\$3.74	\$3.80	\$2.08	\$3.00
Shim 2	\$3.75	\$3.87	\$2.75	\$3.09

3.2. Excess Reactivity and Shutdown Margin
Following the addition of each bundle, the reactor will be brought to criticality and control rod worth measurements will be made by the asymptotic period method to ensure that Excess Reactivity and Shutdown Margin limits will be met when the next bundle is added.

3.3. Calorimetric Power Calibration
The calorimetric power calibration takes advantage of the fact that natural convection provides adequate cooling for a TRIGA core operating at power levels up to and including 2.0 MW. In the so-called "slope" method of calibration, the rate of temperature rise will be determined for the reactor pool water [dT/dt (°C/hr)] while the reactor is operating at power P and the tank water is stirred. Combined with the measured time rate of pool water temperature rise, the actual reactor power can be calculated from:

$$P(kW) = \left[\frac{{dT}/{dt}\left({^{\circ}C}/{hr}\right)}{TankConstant\left({^{\circ}C}/{kW}\right)}\right]$$

A power calibration will be completed following the addition of all 4 fuel bundles.

3.4. Primary Coolant Measurements

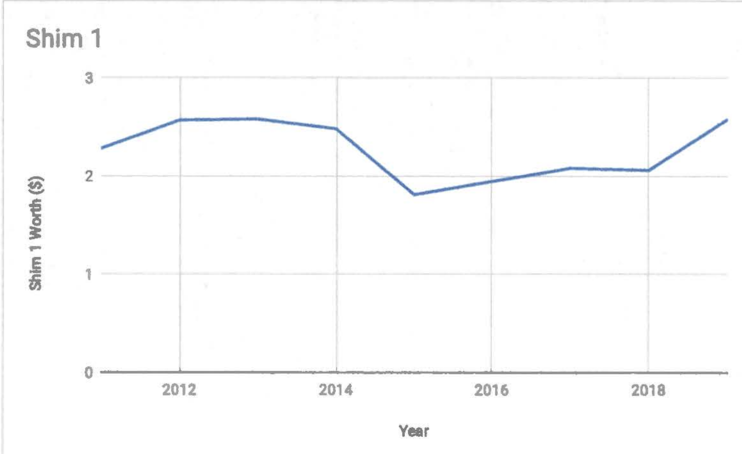
A primary coolant water sample taken within 30 days of the core additions will be analyzed for fission product activity.

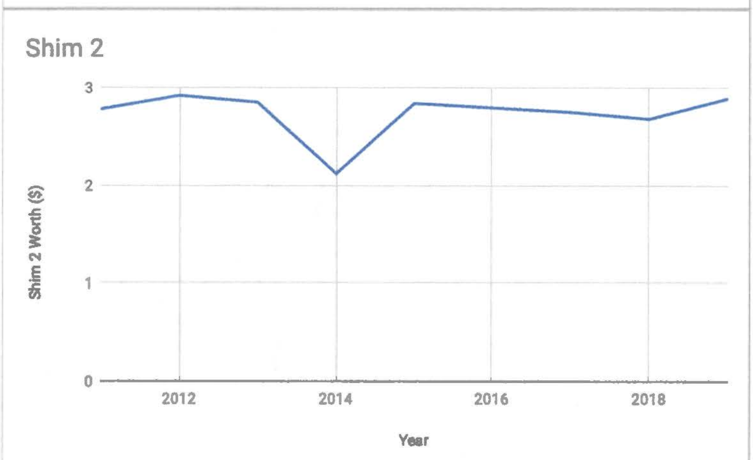
Attachment 3

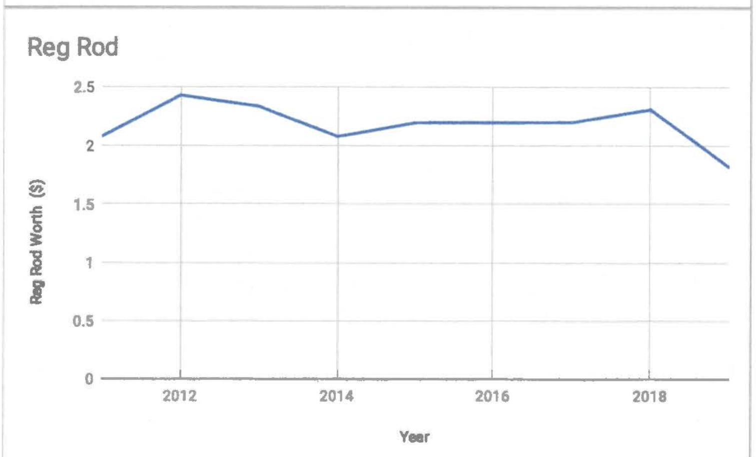
MUTR CONTROL ROD WORTH DATA

Shim 1		Shim 2		Reg Rod	
Year	Worth ($)	Year	Worth ($)	Year	Worth ($)
2011	2.28	2011	2.78	2011	2.08
2012	2.57	2012	2.92	2012	2.43
2013	2.58	2013	2.85	2013	2.34
2014	2.48	2014	2.12	2014	2.08
2015	1.81	2015	2.84	2015	2.20
2017	2.08	2017	2.75	2017	2.20
2018	2.06	2018	2.68	2018	2.31
2019	2.58	2019	2.89	2019	1.82

*Rod worth data from 2010 and 2016 is not available







Attachment 4

UPDATED PAGES FOR TECHNICAL SPECIFICATION 4.1.4

4 Surveillance Requirements

Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety.

Specification

Surveillances shall be performed on a timely basis. In the event that the reactor is not in an OPERABLE condition, such as during periods of refueling, or replacement or repair of safety equipment, surveillances may be postponed, see Table 4.1, until such time that the reactor is OPERABLE. In such case that any surveillance must be postponed, a written directive signed by the Director, shall be placed in the records indicating the reason why and the expected completion date of the required surveillance. This directive shall be written before the date that the surveillance is due. Under no circumstance shall the reactor perform routine operations until such time that all surveillances are current and up to date. Any system or component that is modified, replaced, or had maintenance performed shall undergo testing to ensure that the system/component continues to meet performance requirements.

Technical Specification	Defer during shutdown?	Required prior to operations?
4.1 Reactor Core Parameters	Yes	Yes
4.2 Reactor Control and Safety Systems	Yes	Yes
4.3 Primary Coolant System	No	N/A
4.4 Confinement	Yes	Yes
4.5 Radiation Monitoring and Effluents	No	N/A
4.6 Experiments	Yes	Yes

Table 4.1: Surveillance Requirements

4.1 Reactor Core Parameters

Applicability

These specifications apply to the surveillance requirements for the reactor core.

Objective

The objective of these specifications is to ensure that the specifications of Section 3.1 are satisfied.

Specifications

1. The EXCESS REACTIVITY shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed, these changes include any removal or replacement of CONTROL RODS.

2. The SHUTDOWN MARGIN shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed, these changes include any removal or replacement of CONTROL RODS.
3. CORE CONFIGURATION shall be verified prior to the first startup of the day.
4. A visual inspection of a representative group of at least 4 FOUR ELEMENT FUEL BUNDLES from rows B and C shall be performed annually at intervals not to exceed 15 months. The bundles inspected shall rotate such that in a 2-year period all accessible FOUR ELEMENT FUEL BUNDLES in rows B and C are inspected. If any are found to be damaged, an inspection of the entire MUTR core shall be performed.
5. Burnup shall be determined annually, not to exceed 15 months.

Bases

Experience has shown that the identified frequencies ensure performance and operability for each of these systems or components. For EXCESS REACTIVITY and SHUTDOWN MARGIN, long-term changes are slow to develop. For fuel inspection, visually inspecting the bundles annually will identify any developing fuel integrity issues throughout the core.

4.2 Reactor Control and Safety Systems

Applicability

These specifications apply to the surveillance requirements for reactor control and safety systems.

Objective

The objective of these specifications is to ensure that the specifications of Section 3.2 are satisfied.

Specifications

1. The reactivity worth of each CONTROL ROD shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed or a CONTROL ROD is inspected.
2. The CONTROL ROD withdrawal and insertion speeds shall be determined annually, at intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect rod travel times.
3. CONTROL ROD DROP TIMES shall be measured annually, at intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect their DROP TIME.
4. All scram channels listed in Table 3.1 shall have a CHANNEL TEST, including trip actions with CONTROL ROD release and specified interlocks as listed in Table 3.2 performed after each SECURED SHUTDOWN, before the first operation of the day, or prior to any operation scheduled to last more than 24 hours, or quarterly, with intervals not to exceed 4 months. Scram channels and interlocks shall be calibrated annually, at intervals not to exceed 15 months.
5. CHANNEL TESTS shall be performed on all affected safety and control systems after any maintenance is performed.
6. A CHANNEL CALIBRATION shall be made of the linear power level monitoring channels annually, at intervals not to exceed 15 months.
7. A visual inspection of one of the CONTROL ROD poison sections shall be made annually, at intervals not to exceed 15 months. In a 3 year period, all sections shall be inspected.

8. A visual inspection of the CONTROL ROD drive shall be made annually, at intervals not to exceed 15 months.

Bases

1. The reactivity worth of the CONTROL RODS, specification 4.2.1, is measured to assure that the required SHUTDOWN MARGIN is available and to provide a means to measure the REACTIVITY WORTH OF EXPERIMENTS. Long term effects of TRIGA reactor operation are such that measurements of the reactivity worths on an annual basis are adequate to insure that no significant changes in SHUTDOWN MARGIN have occurred.
2. The CONTROL ROD withdrawal and insertion rates, specification 4.2.2, are measured to insure that the limits on maximum reactivity insertion rates are not exceeded.
3. Measurement of the CONTROL ROD DROP TIME, specification 4.2.3, ensures that the rods can perform their safety function properly.
4. The surveillance requirement specified in specification 4.2.4 for the reactor safety scram channels ensures that the CHANNELS are OPERABLE.
5. The surveillance test performed after maintenance or repairs to the REACTOR SAFETY SYSTEM as required by specification 4.2.5 to ensure that the affected CHANNEL will be OPERABLE.
6. The linear power level CHANNEL CALIBRATION specified in specification 4.2.6 will assure that the reactor will be operated at the licensed power levels.
7. Specification 4.2.7 assures that a visual inspection of CONTROL ROD poison sections is made to evaluate corrosion and wear characteristics and any damage caused by operation in the reactor.
8. Specification 4.2.8 assures that a visual inspection of control drive mechanisms is made to evaluate corrosion and wear characteristics and any damage caused by operation in the reactor.

4.3 Primary Coolant System

Applicability

These specifications apply to the surveillance requirements of the reactor primary coolant system.

Objective

The objective of these specifications is to ensure the reactor primary coolant system is OPERABLE as described in Section 3.3.

Specifications

1. The primary coolant level shall be verified before each reactor startup or daily during operations exceeding 24 hours.
2. Pool water conductivity shall be determined prior to the first startup of the day.
3. Pool water gross gamma activity shall be determined monthly, at intervals not to exceed six weeks. If gross gamma activity is high (greater than twice historical data), gamma spectroscopy shall be performed. Gamma spectroscopy shall be performed quarterly, not to exceed 4 months.
4. Pool water temperature shall be measured prior to the reactor startup and shall be monitored during reactor operation.