SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 10

RENEWED FACILITY OPERATING LICENSE R-70

UNIVERSITY OF MARYLAND

MARYLAND UNIVERSITY TRAINING REACTOR

DOCKET NO. 50-166

1.0 INTRODUCTION

By letter dated January 29, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18032A096), as supplemented by letters dated March 26, 2018, June 6, 2019, January 13, 2020, February 24, 2020, March 10, 2020, and May 12, 2020 (ADAMS Accession Nos. ML18092A086, ML19165A021, ML20016A314, ML20062A169, and ML20076C347, and ML20133K121, respectively), the University of Maryland (UMD or the licensee) requested a license amendment that would revise the technical specifications (TSs) in Appendix A to Renewed Facility Operating License No. R-70 for the Maryland University Training Reactor (MUTR).

Specifically, this amendment would do the following:

- Revise TS 1.3, “Definitions” to define core configuration to accurately describe the MUTR core grid with the inclusion of an additional 16 TRIGA fuel elements configured as 4 fuel bundles.

- Revise TS 3.1, “Reactor Core Parameters,” to change the limiting condition for operations by increasing the excess reactivity limit from $1.12 to $3.50.

- Revise TS 4.1, “Reactor Core Parameters,” to change the surveillance requirement to require visual inspection of at least 4 Four-Element Fuel Bundles from rows B and C annually, and to rotate the inspections such that all accessible Four-Element Fuel Bundles will have been visually inspected every two years.

- Revise TS 5.3, “Reactor Core and Fuel,” to update the normal core configuration to include the additional TRIGA fuel elements.

Additionally, UMD proposes to make the following changes to Renewed Facility Operating License R-70:

- Remove License Condition (LC) 2.B.2.b, which authorizes the licensee to receive and possess 1,060 grams of contained U-235 enriched to less than 20 percent in the form of “Alternate Reactor Fuel” TRIGA-type reactor fuel;

- Revise LC 2.B.2.f, to authorize UMD to receive, possess, and use, but not separate, special nuclear material as may be produced by the operation of other facilities in the form of TRIGA-type reactor fuel;

- Revise LC 2.B.3.b, to authorize UMD to receive, possess, and use, but not separate, such byproduct materials as may be produced by the operation of other facilities in the form of TRIGA-type reactor fuel;

- Revise LC 2.B.2.a. to increase the authorized limit for receipt, possession and use, but not separation, in connection with the operation of the facility, of U-235 from 3,441 grams to 4,501 grams of contained uranium-235 (U-235) enriched to less than 20 percent in the form of TRIGA-type reactor fuel; and

- Make minor editorial changes, such as relabeling and reformatting certain LCs.

By letters dated April 16, 2019, and November 22, 2019 (ADAMS Accession Nos. ML19024A291 and ML19312C376, respectively), the U.S. Nuclear Regulatory Commission (NRC) staff requested additional information from UMD. By letters dated June 6, 2019, and January 13, 2020, UMD responded to the NRC staff's requests for additional information (RAIs).

2.0 REGULATORY EVALUATION

The MUTR is a pool-type reactor that uses Training, Research, Isotopes, General Atomics (TRIGA) fuel elements contained in bundles. The reactor is licensed to operate at a steady-state power level up to 250 kilowatts thermal (kWt). The MUTR is not licensed to perform pulsing operations. The MUTR was originally designed for fuel assemblies of the plate materials testing reactor (MTR) type but was converted to the use of TRIGA fuel elements contained in bundles that can hold up to four TRIGA-type fuel elements and are designed to fit in an MTR-type grid plate. The low-enriched uranium TRIGA fuel is in the form of a uranium-zirconium hydride (U-ZrH). The MUTR is presently fueled with TRIGA-type U-ZrH solid fuel-moderator elements clad in stainless steel that contains 8.5 weight percent uranium (U) enriched to less than 20 percent isotopic enrichment of U-235.

The licensee stated in the license amendment request (LAR) that the requested amendment would revise the TSs to reflect changes required for the addition of 16 slightly used TRIGA fuel elements into the current reactor core. These fuel elements are of the same design, weight percent, and enrichment as the fuel elements currently in the MUTR core grid plate. UMD stated that the MUTR is currently constrained in operations and is unable to achieve the full licensed power level of 250 kWt because of uranium burnup and fission product poison buildup after 40 years of operation. The insertion of the 16 TRIGA fuel elements into the current core plate structure would allow the MUTR to achieve full licensed power operation. UMD does not anticipate moving any existing fuel bundles or any other component in the core with the addition of the 16 used TRIGA fuel elements into vacant positions with rows B and C of the core grid plate.

The NRC staff reviewed UMD's LAR. The NRC staff evaluated the proposed changes based on the following regulations and guidance:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.36, "Technical specifications," which requires the TSs to be incorporated in facility operating licenses, including research reactor licenses. Paragraph (a)(1) in 10 CFR 50.36 requires that a summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TSs. Per 10 CFR 50.36(c)(2), limiting conditions for operation must represent the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(3), TSs must include requirements to test, calibrate, or inspect to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits (SLs), and that the limiting conditions for operation will be met. Section 50.36(c)(4) requires that TSs include design features, which are those features of the facility such as materials of construction and geometric arrangements that, if altered or modified, would have a significant effect on safety.

- Section 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," of 10 CFR, identifies licensing, regulatory, and administrative actions eligible for categorical exclusion from the requirement to prepare an environmental assessment or environmental impact statement.

- NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," Section 9.5, "Possession and Use of Byproduct, Source, and Special Nuclear Material," Section 13.1.2, "Insertion of Excess Reactivity," and Appendix 14.1, "Format and Content of Technical Specifications for Non-Power Reactors" (ADAMS Accession No. ML042430055), provide guidance to licensees preparing research reactor applications and TSs.

- NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," Section 4.5.3, "Operating Limits," Section 4.6, "Thermal-Hydraulic Design," and Chapter 14, "Technical Specification" (ADAMS Accession No. ML042430048), provide guidance to the NRC staff for reviewing LARs.

- American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors," Section 1.3, "Definitions," Section 3.1, "Reactor core parameters," and Section 5.3, "Reactor core and fuel," provides guidance used by the NRC staff and licensees, including the parameters and operating characteristics that should be included in the TSs. The 2007 version is a revision of the ANSI/ANS-15.1-1990 standard cited in NUREG-1537 that was issued in 1996, and adds the following applicable information:

    - The Section 1.3 definition of core configuration.

        Core configuration: The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and regulating/control/transient rods occupying the core grid.

- The Section 3.1 guidance that limits shall be established for fuel burnup, as appropriate, and fuel inspection, if appropriate.

- The following Section 5.3 guidance on fuel inspection and fuel burnup.

  > Any requirements for periodic inspection of the fuel or fuel burnup shall be included as appropriate. Conditions for operation of the reactor with damage or leaking fuel elements should be described.

3.0 TECHNICAL EVALUATION

3.1 TS 3.1, "Reactor Core Parameters"

UMD proposes to increase the excess reactivity limit from $1.12 to $3.50 in TS 3.1, Specification 1, to characterize the core parameters with the proposed inclusion of 16 TRIGA fuel elements into grid plate rows B and C of the current core configuration.

The current TS 3.1, Specification 1, states:

> The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place shall not be greater than $1.12.

The proposed TS 3.1, Specification 1, states:

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

The NRC staff evaluated proposed TS 3.1, Specification 1, using the guidance in ANSI/ANS-15.1-2007, Section 3.1, and NUREG-1537, Part 1, Section 14. TS 3.1, Specification 1, establishes a limit on excess reactivity that allows for operational flexibility while limiting the reactivity available for reactivity addition accidents. The maximum excess reactivity helps establish a basis for ensuring that adequate shutdown reactivity is available by control rod insertion.

TS 1.3 defines excess reactivity as follows:

> EXCESS REACTIVITY—Excess Reactivity is that amount of reactivity that would exist if all CONTROL RODS were moved to the maximum reactive condition from the point where the reactor is exactly critical (keff = 1).

The NRC staff reviewed UMD's neutronic methodology, shutdown reactivity determination, instrumented fuel element (IFE) location, thermal-hydraulic parameters, and accident analysis of reactivity insertion events, to evaluate the acceptability of the excess reactivity limit of $3.50 proposed in TS 3.1, Specification 1, as discussed below.

3.1.1 Neutronic Analysis Methodology

In its response to RAI-1, by letter dated January 13, 2020, UMD submitted the calculated MUTR core performance for the proposed core configuration. The proposed core configuration adds 16 used TRIGA fuel elements into grid plate rows B and C. The licensee simulated the performance of the MUTR proposed core configuration using the Monte Carlo particle transport

code (MCNP6 version 1.1). The research reactor community uses MCNP extensively to simulate a wide range of particle transport scenarios such as core analysis and performance. The NRC staff verified the fuel characteristics used as inputs to the simulation to ensure the fuel elements were properly characterized. The NRC staff also verified that the core burnup used in the simulation accurately reflected the age of the current core by reviewing UMD's annual reports submitted to the NRC between 2015 through 2020. Even though the MUTR license authorizes operation at a steady-state power level of 250 kWt, UMD analyzed a simulated core containing 109 fuel elements (the total amount that includes the 16 additional fuel elements) at a bounding power level of 300 kWt because the reactor safety power level channel scram is set to 120 percent of full licensed steady-state power level. By letter dated March 10, 2020, UMD stated that the average power per fuel element is 2.75 kWt for the proposed core configuration compared to 3.23 kWt for the current core configuration. Further, UMD determined that the highest powered fuel elements for the current and proposed core configuration are 5.15 kWt and 4.53 kWt, respectively.

	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

Figure 1. Power Distribution for Proposed Core Configuration (kWt/fuel element)

For the proposed core configuration, the highest powered fuel element is in grid plate position D6, as shown as red in Figure 1, "Power Distribution for Proposed Core Configuration." Grid plate positions E7, C6, and E4 represent the locations of the Three-Element Fuel Bundles that each contain a control rod and guide tube. Table 1, "MUTR Core Comparison," summarizes the comparison of the current core to the proposed core configuration, related fuel element average, and highest power as provided in the response to RAI-1.

Table 1. MUTR Core Comparison

Core Configuration	Total Fuel Elements	Design Core Power (kWt)	Average Power per Fuel Element (kWt)	Highest Power Fuel Element (kWt)
Current	93	300	3.23	5.15
Proposed	109	300	2.75	4.53

For the current core configuration, the NRC staff verified that the average power per element is 3.23 kWt and that the highest powered fuel element is 5.15 kWt, as submitted by UMD's letter dated February 2, 2011 (ADAMS Accession No. ML110350175). As stated in the license renewal SER, "Safety Evaluation Report—Renewal of the Facility Operating License for the University of Maryland - Maryland University Training Reactor," issued December 2016 (ADAMS Accession No. ML16075A214), Section 2.6, "Thermal-Hydraulic Design," the results for the neutronic analysis form the basis of the thermal-hydraulic analysis and effectively establish the configuration as the limiting core configuration. In addition, the NRC staff verified by reviewing the UMD letter that the fuel temperature of the highest powered fuel element was determined to be 250 degrees Celsius (C) (482 degrees Fahrenheit (F)) in the current core configuration. The NRC staff reviewed Figure 2-1, "MUTR TRIGA fuel and core grid plate configuration," in the license renewal SER and compared it to the current core configuration in Figure 2, "MUTR Current Core Design," of this safety evaluation (SE). The NRC staff concludes that there have been no changes in the core since the issuance of the renewed license in December 2016.

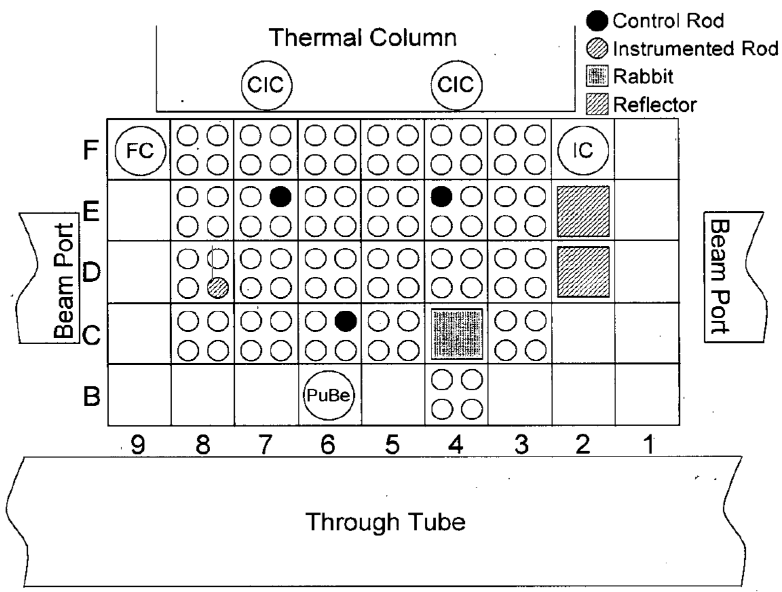

Figure 2. MUTR Current Core Design

The NRC staff evaluated average power, highest power, and power distribution for the proposed core configuration. In its response to RAI-1, UMD provided the fuel element power distribution for the proposed core as shown as Figure 1 of this SE. By letter dated March 10, 2020, UMD stated that the power distribution in Figure 1 for the proposed core configuration is based on a total power distribution of 280 kWt. UMD applied a scaling factor of 1.072 to the fuel element in core grid plate position D6 to determine the highest powered fuel element at 300 kWt. The NRC staff verified that the highest powered fuel element in the proposed core configuration is 4.53 kWt at core grid plate position D6 when operating at a bounding power level of 300 kWt.

1.072 (scaling factor) × 4.23 kWt (highest power in simulated core position D6) = 4.53 kWt

Further, the NRC staff verified the average fuel element power by dividing the reactor bounding power of 300 kWt by the total number of fuel elements.

300 kWt (reactor bounding power) ÷ 109 (fuel elements) = 2.75 kWt

As shown in Table 1 of this SE, the NRC staff finds that both the average and highest powered fuel element are lower in the proposed core configuration than in the current core configuration.

UMD used MCNP to simulate excess reactivity for the proposed core configuration and yielded an excess reactivity value of $2.88 with an uncertainty of ±$0.05. The NRC staff verified that this simulated excess reactivity with the specified uncertainty is less than the proposed excess reactivity limit of $3.50 in TS 3.1, Specification 1. Additionally, UMD used MCNP to simulate control rod worth considering analysis uncertainty for the proposed core configuration, and this is summarized below in Table 2, "MUTR Simulated Control Rod Worth." The NRC staff evaluated the reactivity rate added by each control rod to determine whether the TS 3.2, "Reactor Control and Safety Systems," Specification 2, requirement that the positive reactivity insertion rate by control rod motion not exceed $0.30 per second is met. The NRC staff verified that dividing the simulated control rod worth by the individual control rod withdrawal time results in a maximum control rod reactivity insertion rate of approximately $0.06 per second, which is below and therefore meets, the TS 3.2, Specification 2, requirement that the control rod reactivity insertion rate not exceed $0.30 per second.

Table 2. MUTR Simulated Control Rod Worth

Control Rod	Simulated ($)	Uncertainty ($)
Shim I	3.00	±0.05
Shim II	3.09	±0.05
Regulating	3.26	±0.06
Total (All Rods)	9.35	±0.08
Excess Reactivity (All Rods)	2.88	±0.05

Based on the information above, the NRC staff finds that both the highest and the average fuel element power are lower in the proposed core configuration than in the current core configuration, resulting in lower fuel temperatures. Additionally, the NRC staff finds that the inputs to the MCNP simulation and methodology use reasonable assumptions and methods and that the simulated control rod reactivity insertion rate for the proposed core configuration meets the requirements of TS 3.2. Further, the NRC staff finds that the simulated excess reactivity value of $2.88 is below the proposed excess reactivity limit of $3.50 in TS 3.1, Specification 1, and therefore, is acceptable.

3.1.2 Shutdown Reactivity

The maximum excess reactivity of $3.50 proposed in TS 3.1, Specification 1, helps establish a basis for ensuring that an adequate shutdown margin is available by control rod insertion.

TS 1.3 defines "Shutdown Margin" as follows:

> SHUTDOWN MARGIN—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, and that the reactor will remain subcritical without further operator action.

TS 3.1, Specification 2, states:

> 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.

In response to RAI-4, by letter dated January 13, 2020, UMD provided a shutdown reactivity determination for the proposed excess reactivity limit of \$3.50. TS 3.1, Specification 2, requires that shutdown margin not be less than \$0.50, applying the required core conditions specified in TS 3.1, Specification 2(a), 2(b), and 2(c). Proposed TS 3.1, Specification 1, limits the excess reactivity to \$3.50 with or without experiments in the core. NUREG-1537, Part 2, Section 4.5.3, states that shutdown margin is defined as negative reactivity obtained by control rods to ensure that the reactor is shut down from any condition. Further, NUREG-1537, Part 2, Appendix 14.1, Section 1.3 “Definitions,” states that the analysis should assume that the most reactive scrammable control rods and all nonscrammable control rods are in their most reactive position, and that the reactor will remain subcritical without further operator action. Using the proposed excess reactivity limit of \$3.50, UMD calculated the shutdown reactivity value for the proposed core configuration using simulated data from “Analysis of the Neutronic Behavior of the Maryland University Training Reactor, Revision 4” (ADAMS Accession No. ML20016A314). In response to RAI-4, UMD calculated a shutdown reactivity value of \$2.59, assuming the most reactive control rod (the regulating rod) was fully withdrawn.

The NRC staff performed an analysis, the results of which are shown in Table 3 of this SE, applying conservative uncertainty information provided by UMD to confirm that the licensee demonstrated the ability to satisfy the shutdown margin of \$0.50 specified in TS 3.1.

Table 3. Shutdown Reactivity Value Confirmatory Calculation

	Excess Reactivity (\$)	**Shim Rod I (\$)**	**Shim Rod II (\$)**	**Regulating Rod (Fully Withdrawn) (\$)**	**Calculated SDM (\$)**	**TS 3.1 Shutdown Margin (\$)**
Proposed TS 3.1 Excess Reactivity	3.50	2.95	3.04	3.32	2.49	0.50

$$SDM = \text{Absolute Value (Excess Reactivity} - \text{Shim Rod I} - \text{Shim Rod II)}$$
$$SDM = \text{Absolute Value } (\$3.50 - \$2.95 - \$3.04)$$
$$SDM = \text{Absolute Value } (\$3.50 - \$2.95 - \$3.04)$$
$$SDM = \text{Absolute Value } (-\$2.49)$$
$$SDM = \$2.49$$

As shown in Table 3, the calculated SDM that considers model uncertainty is not less than \$0.50 as required by TS 3.1, Specification 2.

As required by TS 3.1, Specification (b), SDM shall not be less than \$0.50 with all experiments in their most reactive state. The NRC staff verified that UMD determines SDM when experiments are placed in the core. In response to RAI-7, by letter dated January 13, 2020,

UMD stated, regarding reactivity of experiments inserted into the reactor core, that all experiments must be approved by the Reactor Safety Committee (RSC) before insertion into the core as required by TS 6.5, “Experiment Review and Approval.” TS 6.5 states:

> Approved EXPERIMENTS shall be carried out in accordance with established and approved procedures.
>
> 1. All new experiments or class of experiments shall be reviewed by the RSC as required by TS 6.2.3 and implementation approved in writing by the Director or a designated alternate.
>
> 2. Substantive changes to previously approved EXPERIMENTS shall be made only after review by the RSC and implementation approved in writing by the Director or a designated alternate. Minor changes that do not significantly alter the EXPERIMENT may be approved by Level 3 or higher. Changes to experiments shall meet the requirements in accordance with 10 CFR 50.59.

Further, UMD states that reactivity estimates are determined as part of the experimental review process to ensure that TS 3.1, Specifications 1 and 2, are met before the experiment is approved by the RSC.

The NRC staff also evaluated whether experimental measurements to determine operational reactor physics parameters are described in the startup plan. NUREG-1537, Section 12.11, “Startup Plan,” states, in part, that the applicant should submit a startup plan whenever significant core modifications are being made. UMD submitted a startup plan, by letter dated January 13, 2020, for the proposed core configuration. UMD stated that shutdown reactivity measurements will be conducted to ensure that the requirements of TS 3.1 are met after each Four-Element Fuel Bundle is added to the core. The UMD startup plan specifies that UMD will measure control rod reactivity worths, excess reactivity, and SDM, as well as perform a calorimetric power calibration and primary coolant measurements.

The NRC staff finds that the UMD TSs require measurements of reacvitiy control rod worth, excess reactivity, and SDM after each change in core fuel configuration. TS 4.2 “Reactor Control and Safety Systems,” Specification 1, requires that reactivity worth of each control rod be determined after each time the core fuel configuration is changed. Further, TS 4.1, Specifications 1 and 2, requires that excess reactivity and SDM be determined after each time the core fuel configuration is changed. As a result, the NRC staff finds that the TS requirements to measure control rod worth, excess reactivity, and SDM provide reasonable assurance that the reactor is operating as described in UMD’s revised safety analysis included in the LAR, as supplemented. With respect to the proposed increase in the excess reactivity limit from $1.12 to $3.50 in TS 3.1, Specification 1, the NRC staff finds that the resulting shutdown reactivity value of $2.49 is greater than $0.50 and thus, continues to satisfy the shutdown margin requirement of TS 3.1, Specification 2, which ensures that the reactor can be safely shut down from any operating condition.

3.1.3 <u>Location of Instrumented Fuel Element</u>

The IFE allows the temperature of the fuel element in core grid plate position D8 to be measured directly. An automatic protective action (i.e., a scram) is initiated when the IFE

temperature reaches the limiting safety system setting (LSSS) specified in TS 2.2, “Limiting Safety System Settings,” as follows:

> The LIMITING SAFETY SYSTEM SETTING shall be 175ºC as measured by the INSTRUMENTED FUEL ELEMENT (IFE).

As stated in TS 3.2, Table 3.1, “Reactor Safety Channels: Scram Channels,” the fuel element temperature scram is set equal to or less than 175 degrees C (347 degrees F), as measured by the IFE in core grid plate position D8.

The LSSS prevents the fuel element temperature SL from being exceeded. TS 2.1, “Safety Limit,” states:

> The temperature in a standard TRIGA FUEL ELEMENT shall not exceed 1000ºC under any conditions of operation, with the fuel fully immersed in water.

The core grid plate position of the IFE is established by the standard core as defined in TS 1.3, as follows:

> STANDARD CORE — A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate and the IFE in grid position D8.

TS 3.1, Specification 3.(a) states:

> The reactor shall only be operated with a STANDARD CORE.

The NRC staff evaluated the location of the IFE for the proposed core configuration. In the LAR, UMD stated that the power per fuel element was simulated using MCNP to confirm the placement of the IFE for the proposed core configuration. Additionally, in response to RAI-6, by letter dated June 6, 2019, UMD stated that the temperature of the IFE compared to the hottest fuel element should be no more than a factor of 2 higher, in order to allow for a 650-degree C (1,202 degree F) margin to the fuel element temperature SL in TS 2.1, with an IFE trip setting of 175 degrees C (347 degrees F). For the proposed core configuration, UMD determined that the IFE location of D8 remains unchanged since the power generated in the fuel element located at grid position D6 is less than twice the power of fuel element D8.

For the current core configuration, which contains 93 fuel elements, UMD determined that, at a reactor bounding power level of 300 kWt, the power produced in the IFE and in the highest powered fuel element is 2.71 kWt and 5.15 kWt, respectively. The highest powered fuel element was determined to be in grid position E6. The current core averages 3.23 kWt per fuel element when operating at bounding power level of 300 kWt and the licensee determined that the maximum fuel element reaches a temperature of 250 degrees C (482 degrees F). Table A-1, “Key Fuel and Reactor Parameters,” in UMD’s RAI response by letter dated February 2, 2011 (ADAMS Accession No. ML110350175), submitted during the NRC staff review of UMD’s license renewal application, documents these reactor parameters.

For the proposed core configuration, UMD calculated the fuel element power distribution at a reactor design steady-state power level of 300 kWt. By letter dated January 13, 2020, UMD provided Attachment 1, “Table 12—New Core Power Distribution,” which shows fuel element power per element at a steady-state power level of 280 kWt. Table 12 shows that the highest powered fuel element in grid position D6 produces 4.23 kWt, while the IFE located in grid

position D8 produces 2.52 kWt. Because Table 12 shows the power per element at a reactor power of 280 kWt, UMD applied a factor of 1.071 to scale power to 300 kWt that resulted in the highest powered fuel element as 4.53 kWt (position D6) and IFE power as 2.70 kWt (position D8). To verify that the temperature of the highest powered fuel element is maintained below the SL in TS 2.1, the NRC staff calculated the factor of the power produced in fuel element in core grid plate position D8 compared to D6, and this resulted in a factor of 1.68.

Highest Power Fuel Element ÷ Power of IFE =
4.53 kWt ÷ 2.70 kWt = 1.68
4.53 kWt = (1.68) × 2.70 kWt

Applying the 1.68 factor to the IFE temperature scram setting of 175 degrees C (347 degrees F), results in a temperature of 294 degrees C (561 degrees F).

1.68 × IFE Temperature Scram Setting = Temperature in Highest Powered Fuel Element
1.68 × 175 degrees C = 294 degrees C

1,000 degrees C – 294 degrees C = 706 degrees C

Therefore, with a factor of 1.68, which is less than a factor of 2, fuel element temperature will be maintained at less than 350 degrees C (662 degrees F) and the safety temperature margin of 650 degrees C (1,202 degrees F) will be maintained between the highest powered fuel element in core grid plate position D8 and the SL in TS 2.1.

Based on the information above, the NRC staff concludes that the placement of the IFE in grid plate position D8 for the proposed core configuration will continue to ensure that a safety margin of 650 degrees C (1,202 degrees F) between the highest powered fuel element and the SL specified in TS 2.1 is maintained and will ensure the integrity of the fuel element cladding and therefore, is acceptable.

3.1.4 Thermal-Hydraulic Analysis

By letter dated March 10, 2020, UMD stated that the core thermal-hydraulics performed to support the operation of the existing core for license renewal bound the proposed core configuration because the proposed core configuration will have a similar power peaking factor (1.65 versus 1.6 in the existing core), and a lower maximum fuel element power (4.53 kWt versus 5.15 kWt for the existing core at 300 kWt). UMD also stated that the distribution of the same power level across a larger core results in a lower power per fuel element (i.e., 2.75 kWt versus 3.23 kWt in the existing core at 300 kWt). Condition C.1 of the MUTR license authorizes operation of the reactor at a maximum thermal power level of 250 kWt. However, consistent with the power level used in its application for license renewal, UMD performed the thermal-hydraulic analysis of the existing core configuration at the bounding power level of 300 kWt and calculated a departure from nucleate boiling ratio (DNBR), which is the ratio of the critical heat flux to the maximum heat flux at full power, of 5.92. UMD stated that the DNBR would be higher in the proposed core because the peaking factor is similar and the average power per element as well as the peak fuel element power are lower in comparison to the current core configuration. Additionally, UMD stated that the proposed core is more conservative from a thermal-hydraulic aspect and thus is bounded by the current thermal-hydraulic analysis performed for license renewal.

The NRC staff reviewed the thermal-hydraulic parameters and confirmed that the proposed core configuration will have a rod peaking factor of 1.65. This peaking factor is the peak to average fuel element power.

$$\textit{Highest Power Fuel Element} \div \textit{Average Power Element} = \textit{Peaking Factor}$$
$$4.53\ kWt \div 2.75\ kWt = 1.65$$

The NRC staff also evaluated the average power per fuel element and the highest power fuel element for the proposed core configuration in Section 3.1.1 of this SE and verified that the average fuel element power is 2.75 kWt and the maximum fuel element power is 4.53 kWt for the proposed core configuration at a bounding power level of 300 kWt.

A previous NRC staff confirmatory thermal-hydraulic analysis of the existing core, performed at both 250 kWt and 300 kWt, yielded a DNBR of 6.83 and 5.76, respectively. These results are documented in Table 2-4, “DNBR Results,” of the renewal SER. The guidance in NUREG-1537, Part 2, Section 4.6, “Thermal-Hydraulic Design,” recommends that the DNBR should be greater than 2.0, which is an acceptable margin to the onset of nuclear boiling. Because the UMD fuel is cooled by natural convection of the reactor pool water, the flowrate through the core has not changed. The NRC staff verified that the peaking factor is similar to the current core configuration, and that both the average power per fuel element and the highest powered fuel element for the proposed core are less than those in the current core configuration. Because the proposed operating power level remains unchanged and because the proposed core configuration (with the inclusion of the 4 Four-Element Fuel Bundles in the core) results in a decrease of both the average and highest powered fuel element power, the NRC staff finds that (1) the temperature in the average and highest powered fuel element will decrease and (2) these conditions will result in a higher DNBR. Accordingly, the NRC staff finds that these results and conditions provide more margin to the limiting DNBR value of 2.0.

Therefore, the NRC staff concludes that the guidance in NUREG-1537 regarding the DNBR limit is met and that acceptable margins exist for the thermal-hydraulic parameters, to ensure that the fuel is adequately cooled and that the temperature SL of 1,000 degrees C (1,832 degrees F) in TS 2.1 is not exceeded.

3.1.5 Accident Analysis—Insertion of Excess Reactivity

The guidance in NUREG-1537, Part 1, Section 13.1.2, “Insertion of Excess Reactivity,” recommends the evaluation of both a rapid and a slow reactivity insertion event. UMD analyzed multiple scenarios for each reactivity insertion event for an excess reactivity limit of $3.50.

In its response to RAI-3, by letter dated January 13, 2020, UMD analyzed three slow reactivity ramp insertion scenarios resulting from a control rod withdrawal. Using reactor kinetic equations, UMD evaluated the reactivity ramp insertions starting at low power, high power, and with an inoperable reactor safety system. Reactor safety system is defined in TS 1.3 as follows:

> REACTOR SAFETY SYSTEM — Reactor safety systems are those systems, including their associated input CHANNELS, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

UMD assumed in its low power ramp scenario that the reactor is operating at a power of 1 watt with fuel temperature at 20 degrees C (68 degrees F) and that the two independent power level

scrams set at 300 kWt are considered operable. UMD also assumed that control rod withdrawal is terminated by the power level scram at 300 kWt and that there will be a 1-second delay between reaching the power level scram setpoint and the releasing of the control rods used to shut down the reactor. UMD calculated that the maximum reactor power reached would be 366 kWt.

In its high power ramp scenario, UMD assumed that the reactor is operating at just below 300 kWt when a control rod withdrawal starts inserting $0.30 per second into the core and that the two independent power level scrams set at 300 kWt are considered operable. UMD further assumed the power level scram at 300 kWt terminates control rod withdrawal and that there will be a 1-second delay between reaching the power level scram setpoint and the releasing of the control rods used to shut down the reactor. UMD calculated that the maximum reactor power reached is 429 kWt.

The third ramp scenario assumes that all control rods are withdrawn, insertion of $3.50 of reactivity into the core, and an inoperable reactor safety system. UMD applied a power coefficient of reactivity of $0.00053/kilowatt (kW) to the reactivity insertion of $3.50 to determine that the peak reactor power reached is 661 kW during this event. Because TS 3.2, Specification 3, requires the safety channels to be operable, including the reactor power level and rate of power (i.e., period) scram channels, UMD stated that this event is not a plausible scenario. UMD also stated that if any of these safety channels are operable, reactor power would be prevented from reaching 661 kW.

UMD stated that for all three ramp rod withdrawal scenarios above, the resulting peak fuel temperature is less than the fuel temperature SL of 1,000 degrees C (1,832 degrees F) in TS 2.1. To determine peak fuel temperature, UMD used thermal-hydraulic information for the current core configuration submitted by letter, dated February 2, 2011 (ADAMS Accession No. ML110350175), which depicts the peak fuel element temperature as a function of reactor power. For the third scenario, UMD stated that the maximum fuel temperature is 340 degrees C (644 degrees F), which provides a margin of 660 degrees C (1,220 degrees F) to the SL in TS 2.1.

The NRC staff reviewed UMD's response to RAI-3 and the methodology for determining peak power and finds that the analyses of the inadvertent withdrawal of the control rod demonstrate that the period scram and/or the power level scram will terminate the power increase before the reactor power reaches 661 kWt. UMD's current thermal-hydraulic analysis shows that the energy deposited in the hottest fuel element from this limiting reactivity ramp insertion is lower than the fuel temperature SL in TS 2.1, and therefore, fuel integrity would be maintained.

In response to RAI-5, by letter dated January 13, 2020, UMD analyzed rapid reactivity insertions of $4.00 initiated at low and high power and used the Fuchs-Nordhiem model to determine the resulting peak fuel temperature. The NRC staff acknowledges that the Fuchs-Nordheim model employed by the licensee is a longstanding model used to analyze large reactivity insertions for TRIGA reactors. For initial powers of 0.01 kWt and 250 kWt, UMD's analysis shows that a reactivity insertion of $4.00 would result in a peak fuel temperature of 579 degrees C (1,074 degrees F) and 744 degrees C (1,371 degrees F), respectively. UMD's analysis shows that for both scenarios, the peak fuel temperature is below the fuel temperature SL of 1,000 degrees C (1,832 degrees F) in TS 2.1.

The NRC staff used the TRACE/RELAP Advanced Computational Engine (TRACE) computer code to perform calculations to determine if the UMD results are acceptable for a reactivity

insertion of $4.00 at 250 kWt and 300 kWt for the proposed core configuration. The NRC staff's analysis calculated peak fuel temperatures of approximately 870 degrees C (1,598 degrees F) and 877 degrees C (1,610 degrees F), respectively. The NRC staff finds that its calculations confirm that for both rapid reactivity insertion scenarios at $4.00, the resulting peak fuel temperature would be less than the fuel temperature SL of 1,000 degrees C (1,832 degrees F) in TS 2.1, and thus ensure that fuel cladding integrity is maintained. Further, the NRC staff finds that the third rapid scenario is not credible because TS 3.2, "Reactor Control and Safety systems," Specification 3, requires two operable power level safety channels when the reactor is operating. Based on the information provided above, the NRC staff finds the licensee's inadvertent ramp and rapid withdrawal of the control rods scenarios bound the proposed excess reactivity limit of $3.50 in TS 3.1.

The NRC staff also finds that the licensee's analyses of both the slow and the rapid reactivity insertion events use reasonable assumptions and methods and arrive at acceptable results and were conducted in accordance with proposed and existing limits in the TS. Based on the information above, the NRC staff finds the UMD analysis of insertion of excess reactivity, bounded by the proposed excess reactivity limit of $3.50 in TS 3.1, Specification 1, results in a peak fuel temperature below the temperature SL in TS 2.1 and is therefore acceptable.

3.1.6 Conclusion on Excess Reactivity Limit

Based on the NRC staff's review of UMD's neutronic analysis, shutdown reactivity, location of the IFE in the core, thermal-hydraulic parameters, and insertion of excess reactivity accident scenarios, the NRC staff finds that the proposed excess reactivity limit of $3.50 in TS 3.1, Specification 1, meets the 10 CFR 50.36(c)(2)(i) requirement that a limiting condition for operation specifies the functional capability or performance level of equipment required for safe operation of the facility. Therefore, the NRC staff concludes that the proposed change to TS 3.1, Specification 1, is acceptable.

3.2 TS 4.1, "Reactor Core Parameters"

UMD proposed to revise TS 4.1, Specification 4, to visually inspect at least 4 Four-Element Fuel Bundles from rows B and C annually, at intervals not to exceed 15 months. Additionally, UMD proposed to rotate the inspections such that all accessible Four-Element Fuel Bundles are visually inspected in a 2-year period.

The current TS 4.1, Specification 4, states:

> A visual inspection of a representative group of fuel bundles from row C column 8, 7, 6, 5, 3 and row B column 4 shall be performed annually, at intervals not to exceed 15 months. If any are found to be damaged, an inspection of the entire MUTR core shall be performed.

The proposed TS 4.1, Specification 4, states:

> 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.

UMD proposed to remove the visual inspection of fuel in core grid plate position C6 because of a risk of damage to the Three-Element Fuel Bundle as a result of movement for inspection. UMD proposed to add the words "at least 4 FOUR ELEMENT FUEL BUNDLES from rows B and C" describing the locations of the fuel that is inspected annually. Further, UMD proposed to add the sentence "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," to describe the timeframe that all Four-Element Fuel Bundles in Rows B and C would be visually inspected.

The NRC staff evaluated the proposed revision to TS 4.1, Specification 4, using the guidance in NUREG-1537, Parts 1 and 2. TS 4.1, Specification 4, specifies the frequency and methods for fuel element inspection. The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 4.1, recommends that, for nonpulsing TRIGA reactors, the fuel should be inspected and measured for length and bend on at least a 5-year cycle so that approximately 20 percent of the fuel would be inspected and measured annually. The bend and length measurements involve removing from the core and placing fuel elements in special gauges that determine fuel element length and bend.

In the license renewal safety evaluation report (SER), the NRC staff found that visually inspecting the fuel in core grid plate positions listed in current TS 4.1, Specification 4, that includes the fuel elements located in rows B and C, would provide a representative profile of all other fuel elements in the core. This finding was based on the following factors:

- The MUTR facility does not pulse and does not use a forced circulation coolant system
- The MUTR has relatively low fuel burnup given the operating history
- The MUTR uses stainless steel clad fuel elements
- The risk of damage to instrumentation is low
- The facility's current licensed power is 250 kWt
- The MUTR analyses were performed at 300 KWt

Additionally, the NRC staff stated in the license renewal SER that to physically measure fuel element bend and length, the fuel element bundles would need to be disassembled, but this has never been undertaken because it presents a risk of fuel and instrumentation damage. Proposed TS 4.1, Specification 4, retains the existing requirement that the entire core be visually inspected if an annual inspection identifies damaged fuel.

In the LAR, as supplemented by UMD's response to RAI-6 by letter dated January 13, 2020, UMD proposes to visually inspect at least 4 Four-Element Fuel Bundles from rows B and C on an annual basis, not to exceed 15 months and to visually inspect all Four-Element Fuel Bundles in rows B and C every 2 years. The licensee stated that this visual inspection would include the proposed addition of 16 elements configured into 4 Four-Element Fuel Bundles to the core. The licensee further stated that proposed TS 4.1, Specification 4, does not require the visual inspection of grid position C6 since this Three-Element Fuel Bundle is not readily accessible without an excessive risk of damage to the fuel as a result of movement. The Three-Element Fuel Bundle contains a "control rod guide tube" and "control rod," which are defined in TS 1.3 as follows:

> CONTROL ROD — A control rod is a device fabricated from neutron-absorbing material which is used to establish neutron flux changes and to compensate for

routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

CONTROL ROD GUIDE TUBE — Hollow tube in which a CONTROL ROD moves.

The NRC staff finds that visual inspection of grid plate position C6 is not needed because of the potential risk of damage to the control rod, guide tube, and fuel elements. UMD proposes to insert the additional 4 Four-Element Fuel Bundles into row B, positions B8, B7, B5, and B3 of the core grid plate as highlighted in Figure 3, "University of Maryland Proposed Core Configuration." The NRC staff finds that visually inspecting of 4 Four-Element Fuel Bundles annually and all Four-Element Fuel Bundles in a 2-year period from rows B and C accounts for approximately 15 to 18 percent, respectively, of the proposed core. The portion of TS 4.1, Specification 4 that requires inspection of the entire core if a damaged element is identified, remains unchanged.

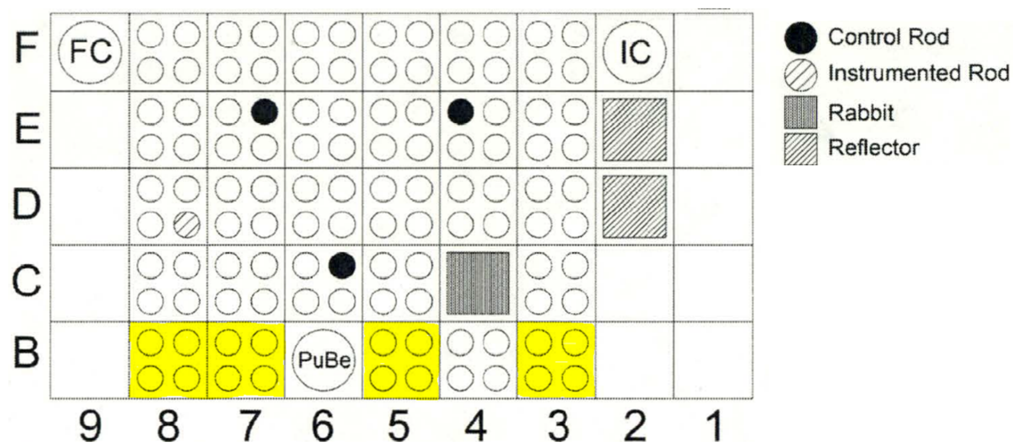

Figure 3. University of Maryland Proposed Core Configuration

In reviewing UMD's 2017 and 2018 annual operating reports (ADAMS Accession Nos. ML18094A116 and ML19094A258, respectively), the NRC staff found that the MUTR continues to maintain relatively low fuel burnup.

Because the MUTR facility 1) retains its license condition precluding pulsing, 2) does not use a forced circulation coolant system, 3) has relatively low fuel burnup given the operating history, 4) uses stainless steel fuel elements, 5) has a low risk of damage to instrumentation, and 6) is currently licensed at a power level of 250 kWt, the NRC staff finds that the visual inspection of Four-Element Fuel Bundles in grid plate rows B and C as proposed in TS 4.1, Specification 4, UMD would continue to provide a competent detection capability of fuel element cladding deterioration. The NRC staff finds that inspecting 15 to 18 percent of the core annually is close to the 20 percent specified in NUREG-1537, Part 1, Appendix 14.1, Section 4.1. The NRC staff finds that the proposed TS 4.1, Specification 4, is more restrictive than the currently approved TS 4.1, Specification 4, because more fuel elements are inspected annually. Proposed TS 4.1 would require a visual inspection of fuel bundles in grid plate positions B3, B5, B7, and B8. The NRC finds that because these fuel bundles have been stored for a significant period time, inspecting these elements would help identify any deterioration as a result of operation of the reactor. Further, the NRC staff recognizes that fuel element failure is typically identified with an increase in pool water radioactivity. TS 4.3, "Primary Coolant System," Specification 3, requires UMD to determine pool water gamma activity monthly, at intervals not to exceed six weeks. In

conducting this surveillance, UMD would identify a fuel element failure. Because UMD is required by TS 4.3, Specification 3, to determine pool water gamma activity monthly, more fuel elements are inspected annually, and the inclusion of fuel bundles in core grid plate positions B8, B7, B5, and B3 are inspected, the NRC staff finds that proposed TS 4.1, Specification 4, to visually inspect at least 15 percent of the fuel elements annually, is acceptable.

Based on the above information, the NRC staff finds that the proposed TS 4.1, Specification 4, is acceptable and provides reasonable assurance that fuel element cladding deterioration and failure will be detected. The NRC staff finds that the proposed TS 4.1, Specification 4, which 1) revises the fuel element inspection surveillance requirement by removing the visual inspection of the fuel element in core grid plate position C6, and 2) requires inspection of all Four-Element Fuel Bundles in Rows B and C in a 2-year period, including the Four-Element Fuel Bundles proposed to be added to the core, meets the 10 CFR 50.36(c)(3) requirement that inspections assure that the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that the limiting conditions for operation will be met. Therefore, the NRC finds that the proposed changes to TS 4.1, Specification 4, are acceptable.

3.3 TS 1.3, "Definitions," Core Configuration

UMD proposed to revise the definition of "Core Configuration," in TS 1.3 to characterize the core configuration with the proposed inclusion of 16 TRIGA fuel elements.

The current definition of "Core Configuration," in TS 1.3 states:

> CORE CONFIGURATION—The core consists of 24 fuel bundles, with a total of 93 fuel elements, arranged in a rectangular array with one bundle displaced for the pneumatic experimental system; three CONTROL RODS; and two graphite reflectors.

The proposed definition of "Core Configuration," in TS 1.3 states:

> CORE CONFIGURATION—The core consists of TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES, arranged in a close-packed array. Bundles may be displaced for the pneumatic experimental system, PuBe source, neutron detectors and graphite reflectors.

The proposed definition of "Core Configuration" would remove the specific number of fuel bundles and total number of fuel elements by deleting the words "24 fuel bundles, with a total of 93 fuel elements." and inserting instead, "TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES." Additionally, the proposed TS would revise the core array description by deleting "rectangular" and inserting "closed packed" and would remove the specific number of bundles displaced by deleting "one" and adding that "PuBe source, neutron detectors" may displace fuel bundles in the core grid plate. UMD also proposed to delete "three CONTROL RODS" from the list of items that are displaced by a fuel bundle and to delete the specific number of graphite reflectors by deleting "two."

The NRC staff evaluated the proposed TS 1.3 definition of "Core Configuration," using the guidance in ANSI/ANS-15.1-2007, Section 1.3, "Definitions," and NUREG-1537, Part 1, Chapter 14. Figure 2, "MUTR Current Core Design," in UMD's RAI response for license renewal, dated February 2, 2011 (ADAMS Accession No. ML110350175), which is included in Section 3.1.1 of this SE, shows the current core configuration for the MUTR. The core grid plate

has a 9-by-5 array of holes that can accommodate fuel bundles, instrumentation, neutron startup source, experimental facility, and graphite reflector assemblies.

The NRC staff reviewed the proposed core configuration provided in UMD's response to RAI-9, dated June 6, 2019, to verify consistency with the proposed definition of "Core Configuration." The UMD RAI response depicts 28 bundles, totaling 109 fuel elements, and states that the fuel elements are displaced in core grid plate position B6 and C4, which are internal core positions, for the plutonium-beryllium (PuBe) source and in-core pneumatic experimental system (often called a "rabbit" system), respectively. Additionally, UMD states that the neutron detectors and graphite reflectors are located on the periphery of the core. In response to RAI-9, UMD provided an illustration of the proposed core configuration as shown below in Figure 4, "MUTR Proposed Core Design," of this SE. The neutron detectors consist of a fission chamber (FC) and ion chamber (IC) located in core grid plate location F9 and F2, respectively.

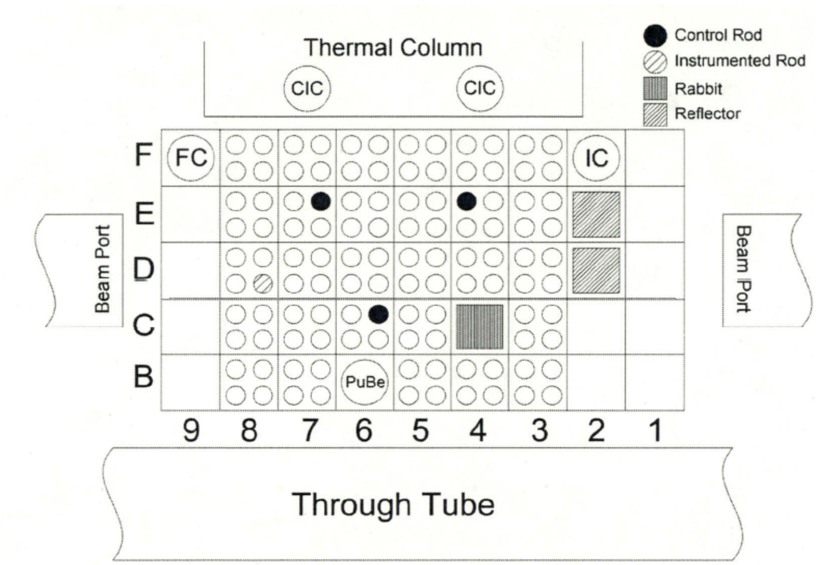

Figure 4. MUTR Proposed Core Design

In the LAR, UMD states that the 16 fuel elements would be assembled into 4 Four-Element Fuel Bundles for installation into the core support structure. MUTR TS 1.3 defines fuel element, Four-Element Fuel Bundle, and the three-element bundle as follows:

> FUEL ELEMENT— A fuel element is a single TRIGA fuel rod.
>
> FOUR ELEMENT FUEL BUNDLE—The four element fuel bundle consists of an aluminum bottom, 4 stainless steel clad FUEL ELEMENTS and aluminum top handle.
>
> THREE ELEMENT FUEL BUNDLE—The three element fuel bundle consists of an aluminum bottom, 3 stainless steel clad FUEL ELEMENTS, 1 CONTROL ROD GUIDE TUBE, and aluminum top handle.

UMD proposed to remove "three CONTROL RODS," which are currently listed among the items that displace a fuel bundle, from the definition of "Core Configuration." The NRC staff finds this change acceptable because the TS 1.3 definition of a "Three Element Fuel Bundle" already specifies that each includes a control rod guide tube, which displaces only a single fuel element,

but not an entire fuel bundle. Further, TS 3.1, “Reactor Core Parameters,” Specification 3(d), requires that the reactor shall be operated with three operable control rods. Therefore, it is not necessary to specify the number of control rods in the “Core Configuration” definition.

UMD proposed to insert the 16 elements grouped as 4 Four-Element Fuel Bundles in core grid plate positions B8, B7, B5, and B3 positions within the fueled core region. Based on its review of the LAR, as supplemented by UMD’s response to RAI-9, the NRC staff finds that the proposed addition of the 4 Four-Element Fuel Bundles, is acceptable because the calculated excess reactivity, the SDM, the thermal-hydraulics evaluation, and the IFE location in the core, result in a core configuration that is: (1) a rectangular arrangement with equal sides (i.e., square) and that (2) contains no vacant internal positions within the fueled core region. In addition, the staff finds that this core configuration can be accurately described as “a close-packed” instead of a “rectangular” array because the described arrangement eliminates vacant grid positions in the core to minimize the possibility of an inadvertent reactivity addition into the core that could result in a significant power excursion, and supports operational flexibility by allowing adjustments in the core as long as UMD maintains a closely packed core configuration. Therefore, the NRC staff finds that revising TS 1.3 to describe the core arrangement as a closely packed configuration is acceptable.

UMD proposed to delete “two,” which specifies the number of graphite reflectors in the core. The NRC staff reviewed the TS definition of core configuration in other TRIGA reactor licenses and found that no other TRIGA facility TSs specified, and thereby required, a specific number of graphite reflectors in the definition of core configuration. However, TS 5.3, Specification 4, “Reactor Core and Fuel,” requires the reflector be a combination of two graphite reflectors that meets the guidance in ANSI/ANS-15.1-2007 that states core configuration includes the number and type of reflector elements. As a result of the review, the NRC staff finds that specifying the number of graphite reflectors is not necessary in the TS 1.3 definition of core configuration since the number of reflectors is specified, and thereby required, by TS 5.3, Specification 4.

UMD proposed to delete “24 fuel bundles, with a total of 93 fuel elements” describing the fuel bundles and elements in the core. The NRC staff reviewed the definition of “core configuration” in the TS in other TRIGA reactor licenses and found that no other TRIGA facility TSs specified, and thereby required, a specific number of fuel bundles or fuel elements in the definition of core configuration. As a result of the review, the NRC staff finds that it is acceptable not to include the number of fuel bundles and elements in the TS 1.3 definition of core configuration.

UMD proposed to add the words “TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES” describing the fuel bundles in the core. ANSI/ANS-15.1-2007, Section 1.3, states that core configuration includes the type of fuel elements occupying the core grid. The MUTR is presently fueled with stainless steel clad TRIGA-type elements assembled into Three-Element or Four-Element Fuel Bundles. UMD states in the LAR that the fuel elements will be assembled into Four-Element Fuel Bundles. The NRC staff finds that, because the proposed additional fuel elements will be assembled into Four-Element Bundles and the existing core only contains TRIGA fuel elements assembled into Three-Element and Four-Element Fuel Bundles, adding “TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES” accurately describes the proposed core configuration shown in Figure 4 of this SE.

The NRC staff finds that the proposed definition of “Core Configuration,” meets the intent of the guidance in ANSI/ANS-15.1-2007, Section 1.3, in that it describes the proposed arrangement of the fuel elements as being in a close-packed array, the arrangement of control rods by

specifying Three-Element Fuel Bundles (each defined in TS 1.3 as including a control rod), and the type of (graphite) reflector elements that would occupy the grid plate with the inclusion of 16 additional fuel elements as 4 Four-Element Fuel Bundles. ANSI/ANS-15.1-2007, Section 5.3, “Reactor core and fuel,” states that the description of core configuration can include any other special information concerning the core. UMD’s proposed inclusion of the PuBe source and neutron detectors describes other components that occupy the core grid positions in the core configuration. The NRC staff finds that adding the PuBe source and neutron detectors as items that displace a fuel bundle in the core grid plate, accurately describes the core configuration.

Therefore, the NRC staff concludes that the proposed changes to TS 1.3 meet the intent of the definition of core configuration as described in ANSI-15.1-2007 and appropriately reflect the proposed addition of 16 TRIGA fuel elements configured into 4 Four-Element Fuel Bundles as analyzed in the LAR and is acceptable.

3.4 TS 5.3, “Reactor Core and Fuel”

UMD proposed to revise TS 5.3, Specification 1, to reflect the proposed normal core configuration with the proposed addition of 16 fuel elements as 4 Four-Element Fuel Bundles.

The current TS 5.3, Specification 1, states:

> 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.

The proposed TS 5.3, Specification 1, states:

> The core shall consist of TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES.

UMD proposed to replace “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” with “TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES” to describe the core configuration.

The NRC staff evaluated the proposed TS 5.3, Specification 1, using the guidance in ANSI/ANS-15.1-2007, Section 5.3, which states, in part, that each type of authorized reactor fuel shall be described.[1] UMD’s proposed TS 5.3, Specification 1, limits the MUTR core configuration to TRIGA fuel elements in fuel bundles of three and four elements. Renewed Facility Operating License No. R-70 authorizes UMD to use TRIGA-type reactor fuel for the MUTR. TS 1.3 defines Three-Element and Four-Element Fuel Bundles as shown in Section 3.1 of this SE. The NRC staff reviewed other similar TRIGA facility TSs, such as Washington State University (ADAMS Accession No. ML14204A625), Texas A&M University (ADAMS Accession No. ML14136A085), and University of Wisconsin (ADAMS Accession No. ML14204A614), regarding core configuration design features and found that the level specificity regarding the number of fuel elements is similar to UMD’s proposed TS 5.3, Specification 1. As a result of the review, the NRC staff determined that the level of specificity is acceptable because UMD’s proposed TS 5.3, Specification 1, contains the same level of specificity as the TSs of similar facilities related to the number to fuel bundles and elements used to describe the design

[1] Extracted from ANSI/ANS-15.1-2007 with the permission of the publisher, the American Nuclear Society.

features of the normal core configuration and supports operational flexibility of potential future core configuration changes. Further, the NRC staff finds that since TS 1.3 definition of Three-Element Fuel Bundle is described as containing a control rod guide tube, “and CONTROL ROD guide tube” is not needed to describe the design features of the normal core configuration. The NRC staff also finds that TS 3.1, Specification 3(d), requires that the reactor be operated with three operable control rods. Therefore, the NRC staff finds that it is not necessary to specify the number of control rod guide tubes in TS 5.3, Specification 1.

The NRC staff verified that the proposed core configuration is composed of a combination of Three-Element and Four-Element fuel bundles by reviewing the UMD’s response to RAI-9 by letter dated June 6, 2019, that depicts an illustration of the proposed core configuration presented in Figure 4 in this SE. The NRC staff finds that the proposed TS 5.3, Specification 1, would continue to ensure that only TRIGA fuel elements are authorized for use in the MUTR core that are configured into either Three-Element or Four-Element Fuel Bundles. The NRC staff finds that proposed TS 5.3, Specification 1, is consistent with the guidance in ANSI/ANS-15.1-2007 to describe each type of authorized reactor fuel.

UMD proposed to revise TS 5.3, Specifications 2, to reflect the proposed normal core configuration with the proposed addition of 16 fuel elements as 4 Four-Element Fuel Bundles.

The current TS 5.3, Specification 2, states:

> The fuel bundles shall be arranged in a rectangular 4 x 6 configuration, with one bundle displaced for the in-core pneumatic experimental system.

The proposed TS 5.3, Specification 2, states:

> The fuel bundles shall be arranged in a close-packed array, with bundles displaced for the pneumatic experimental system, PuBe source, neutron detectors, and graphite reflectors.

UMD proposed to delete “rectangular 4 x 6 configuration” and add “close-packed array” describing the arrangement of the fuel bundles in the core. Additionally, UMD proposed to delete “one bundle,” which specifies the number of fuel bundles displaced by the in-core pneumatic experimental system and to specify other core components that displace fuel bundles by adding “bundles” and “PuBe source, neutron detectors, and graphite reflectors” to describe normal core configuration. UMD proposes to add Four-Element Fuel Bundles to grid plate positions B8, B7, B5, and B3 as shown in Figure 4 of this SE.

The NRC staff evaluated proposed TS 5.3, Specification 2, using the guidance in ANSI/ANS-15.1-2007, Section 5.3, which states that TSs describe the normal core configurations and any other special information concerning the core. The NRC staff finds that proposed TS 5.3, Specification 2, continues to limit the MUTR normal core configuration of the fuel bundles to an array arrangement that is a close-packed configuration, but without specifying the dimensions of the array. The NRC staff determined that the dimensions of the arrangement of the fuel bundles are not a design feature needed to ensure the safe operation of the MUTR because the grid plate limits the core dimensions, the requirement of a closely-packed fuel element array helps reduce the probability of an accidental reactivity insertion. Further, the NRC staff finds by not specifying the array dimensions in TS 5.3, that this change supports future operational flexibility. Figure 4 of this SE shows that the fuel elements in the proposed core configuration are closely packed and are displaced by the pneumatic experimental system,

PuBe neutron source, fission and ion chambers (i.e., neutron detectors), and graphite reflectors. ANSI/ANS-15.1-2007, Section 5.3, states that TSs shall include any other special information concerning the core. The NRC staff finds that, because adding that the PuBe source, neutron detectors, and graphite reflectors displace fuel bundles in the core grid plate describes special information concerning the core configuration in accordance with ANSI/ANS-15.1-2007, Section 5.3, these changes are acceptable. The NRC staff reviewed TS 5.3, Specification 2, and finds that it helps ensure that the core fuel arrangement is closely packed such that there are no empty internal core grid positions. In addition, the staff finds the revised specification accurately describes the proposed normal core configuration, in sufficient detail, consistent with the guidance in ANSI/ANS-15.1-2007 in that TS 5.3, Specification 2, and describes the normal configuration and special information concerning the core.

Additionally, the NRC staff compared UMD's proposed TS 5.3, Specifications 1 and 2, to the TSs of other TRIGA research reactors. The NRC staff finds that UMD's proposed Specifications 1 and 2 in TS 5.3 contain a level of specificity that is similar to the Oregon State TRIGA Reactor TS 5.3.1, "Reactor Core," issued on September 10, 2008, (ADAMS Accession Nos. ML082520047 and ML082530509), and Armed Forces Radiobiology Reactor Institute (AFRRI) TS 5.2.2, "Reactor Core" issued on November 30, 2016 (ADAMS Accession Nos. ML16077A303, and ML16077A302, respectively). Neither the Oregon State TRIGA Reactor TSs nor the AFRRI TSs specify a certain number of fuel elements as a design feature of the reactor core and fuel.

Based on the information provided above, the NRC staff finds that Specifications 1 and 2 of the proposed TS 5.3 are consistent with the guidance in ANSI/ANS-15.1-2007 in that they adequately describe the types of fuel elements and the normal configuration of the core as described in the LAR and limits the core configuration to a closely packed array, which prevents accidental reactivity. The NRC staff finds that the TSs, including the proposed changes, continue to ensure that the design features of the facility. The NRC staff also finds that the description of the core configuration satisfies the requirements of 10 CFR 50.36(c)(4), which require that the TSs include design features, such as geometric arrangements, which if altered or modified, would have a significant effect on safety and are not covered by safety limits, limiting safety system or control setting, limiting conditions for operation, or surveillance requirements. Therefore, the NRC staff concludes that the proposed change to TS 5.3, Specifications 1 and 2, are acceptable.

3.5 Conclusion

The NRC staff evaluated the TS changes proposed in the UMTR TSs in the LAR, as supplemented. The NRC staff finds that the proposed changes to TSs 1.3, 3.1, and 5.3 are consistent with the guidance in both NUREG-1537, Parts 1 and 2, and ANSI/ANS-15.1-2007 and meet 10 CFR 50.36(c)(2)(i) that requires limiting conditions for the safe operation of the facility and 10 CFR 50.36(c)(4) that requires design features to be included that have a significant effect on safety that are not covered in paragraphs (c)(1), (c)(2), and (c)(3) of Section 50.36. The NRC staff further finds that the proposed change to TS 4.1 provides reasonable assurance that fuel element cladding deterioration and failure will be detected and meets the 10 CFR 50.36(c)(3) requirement that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Therefore, based on its review, the NRC staff concludes that the requested changes to the MUTR TSs are acceptable.

4.0 CHANGES TO UNIVERSITY OF MARYLAND RENEWED FACILITY OPERATING LICENSE R-70

In addition to the proposed changes in Section 3.0 of this SE, UMD requested by LAR supplemental letters dated March 26, 2018, February 24, 2020, and May 12, 2020, that Renewed Facility Operating License No. R-70, be revised as follows:

- Remove LC 2.B.2.b that allowed for the receipt and possession of 1,060 grams of contained U-235 enriched to less than 20 percent in the form of “Alternate Reactor Fuel” TRIGA-type reactor fuel.

- Revise LC 2.B.2.f to authorize UMD to receive, possess, and use, but not separate, special nuclear material as may be produced by the operation of other facilities in the form of TRIGA-type reactor fuel.

- Revise LC 2.B.3.b to authorize UMD to receive, possess, and use, but not separate, such byproduct materials as may be produced by the operation of other facilities in the form of TRIGA-type reactor fuel.

- Revise LC 2.B.2.a. to increase the authorized limit for receipt, possession and use, but not separate, in connection with the operation of the facility, of U-235 from 3,441 grams to 4,501 grams of contained U-235 enriched to less than 20 percent in the form of TRIGA-type reactor fuel.

UMD supplemented its license renewal application by letter dated November 1, 2016 (ADAMS Accession No. ML16312A066), to request a change to the MUTR facility operating license to increase the special nuclear material (SNM) possession limit, in the form of TRIGA fuel elements, which UMD identified as “Alternate Reactor Fuel,” and to allow UMD to possess these used TRIGA fuel elements that were irradiated at another research reactor and the associated SNM and byproduct material produced. The NRC staff reviewed and approved these possession limit changes as part of the issuance of Renewed Facility Operating License No. R-70 on December 22, 2016 (ADAMS Accession No. ML16075A211), which added current LCs 2.B.2.b, 2.B.2.f, and 2.B.3.b.

4.1 Revisions to University of Maryland Renewed Facility Operating License R-70

Current LC 2.B.2.b authorizes UMD to receive, possess, but not use or separate, in connection with operation of the facility, up to 1,060 grams of contained U-235 enriched to less than 20 percent in the form of “Alternate Reactor Fuel.” Current LC 2.B.2.a authorizes UMD to receive, possess, and use, but not separate, in connection with the operations of the facility, up to 3,441 grams of contained U-235 enriched to less than 20 percent in the form of TRIGA-type reactor fuel. UMD requested to add the SNM amount authorized in current LC 2.B.2.b to the authorized limit of 3,441 grams in current LC 2.B.2.a by deleting “3,441” and inserting “4,501.” UMD stated in its LAR that the TRIGA-type reactor fuel identified as “Alternate Reactor Fuel” specified in current LC 2.B.2.b is used fuel from another facility and is the same design as the TRIGA reactor fuel specified in current LC 2.B.2.a. UMD also proposed that current LC 2.B.2.b be deleted because this amount of SNM in the form of TRIGA-type fuel would be combined with the quantity of TRIGA-type reactor fuel in current LC 2.B.2.a.

The NRC staff used the guidance in NUREG-1537, Part 1, Section 9.5, that states, in part, that the reactor license (10 CFR Part 50) should contain material for direct operation of the reactor. In its LAR, UMD requested a combined total of 4,501 grams of SNM in proposed LC 2.B.2.a. The NRC staff verified that the requested combined SNM total limit of 4,501 grams is the sum of the existing 3,441 grams of SNM currently authorized by LC 2.B.2.a and the 1,060 grams of SNM authorized by current LC 2.B.2.b, the used fuel that (1) UMD previously identified as "Alternate Reactor Fuel," and (2) is the same design as the TRIGA-type reactor fuel identified in current LC 2.B.2.a (see Section 3.0 of this SE). The NRC staff finds that, because UMD proposes to use this SNM in the form of TRIGA-type reactor fuel for operation of the MUTR and the total possession limit for TRIGA-type reactor fuel does not change, inclusion of the former "Alternate Reactor Fuel" SNM amount in LC 2.B.2.a is acceptable.

Because it proposes to "use" the used TRIGA fuel that it is currently authorized "to possess, but not use," and it proposes to include that SNM amount in the total SNM possession limit set forth in LC 2.B.2.a, UMD requested the deletion of current LC 2.B.2.b. The NRC staff finds that the deletion of current LC 2.B.2.b is acceptable given that UMD plans to use the SNM and the NRC staff found the inclusion of the amount in LC 2.B.2.b in the SNM amount authorized by LC 2.B.2.a is acceptable.

Current LC 2.B.2.f authorizes UMD "to receive, possess, but not use, and not separate, in connection with the operation" of MUTR, such SNM that is produced by the operation of other facilities in the form of "Alternate Reactor Fuel." UMD requested to delete "Alternate Reactor Fuel" and insert "TRIGA-type reactor fuel." In its LAR, UMD stated that this "Alternate Reactor Fuel" is TRIGA-type reactor fuel of the same design of the TRIGA reactor fuel currently being used in the MUTR core. Because the "Alternate Reactor Fuel" is the same design as the TRIGA-type fuel that is currently being used to safely operate the MUTR, the NRC staff finds that deleting "Alternate Reactor Fuel" and inserting "TRIGA-type reactor fuel" is acceptable. UMD also requested to change "to receive, possess, but not use, and not separate" to "receive, possess, and use, but not separate," SNM that may be produced at other facilities that is in used fuel." Since the former "Alternate Reactor Fuel" contains SNM in the form TRIGA-type reactor fuel that was produced at another research reactor, the fuel is the same design as that currently used in the core, and the prohibition on separation is maintained, the NRC staff finds the changes allowing the use of used fuel, which includes SNM generated at other facilities, at the MUTR facility are acceptable. Therefore, based on the information above, the NRC staff finds that the proposed changes to LC 2.B.2.f are acceptable.

Current LC 2.B.3.b authorizes UMD "to receive, possess, but not use, and not separate in connection with the operation" of MUTR, such byproduct material contained in the form of "Alternate Reactor Fuel" that is produced by the operation of other facilities. UMD requested to replace the words "Alternate Reactor Fuel" with "TRIGA-type reactor fuel." UMD stated in the LAR that "Alternate Reactor Fuel" is fuel irradiated at other facilities and is TRIGA-type reactor fuel of the same design as the TRIGA fuel currently used in the MUTR core. Because the "Alternate Reactor Fuel" is the same design of the TRIGA-type fuel that is being used in the MUTR core, the NRC staff finds that deleting "Alternate Reactor Fuel" and inserting "TRIGA-type reactor fuel" is acceptable. UMD also requested the condition be revised by deleting "but not" and inserting "use" to allow the use of byproduct materials that may be produced at other facilities. Since byproduct material is contained in the used reactor fuel that UMD proposes to add to the MUTR and this SE concludes that the such operation is safe, the NRC staff finds that use of byproduct materials at the MUTR facility is acceptable. In addition, the proposed revisions continue to prohibit UMD from separating byproduct materials contained within reactor fuel produced by the operation of other facilities, the NRC staff finds this change acceptable.

Therefore, based on the information above, the NRC staff finds that the proposed changes to LC 2.B.3.b are acceptable.

4.2 Minor Revisions to University of Maryland Renewed Facility Operating License R-70

To reflect the proposed deletion of LC 2.B.2.b, UMD has requested the following changes:

- Relabel LC 2.B.2.c as LC 2.B.2.b.
- Relabel LC 2.B.2.d as LC 2.B.2.c.
- Relabel LC 2.B.2.e to LC 2.B.2.d.
- Relabel LC 2.B.2.f to LC 2.B.3.e.

The NRC staff finds that the proposed changes that would re-designate LC 2.B.2.c, LC 2.B.2.d, LC 2.B.2.e, and 2.B.2.f as LC 2.B.2.b, LC 2.B.2.c, LC 2.B.2.d, and LC 2.B.2.e respectively, in the MUTR license are minor formatting and editorial changes that are appropriate to reflect the deletion of LC 2.B.2.b, which is no longer needed. In Section 4.1 of this SE, the NRC staff finds the deletion of LC 2.B.2.b appropriate because the amount of SNM in the used fuel is combined in the total SNM amount authorized by LC 2.B.2.a to use, in connection with MUTR operation and the total SNM possession limit stated in the license is unchanged. Therefore, the NRC staff finds these minor formatting and editorial changes to the license are acceptable.

4.3 Conclusion

Based on the evaluation set forth above, the NRC staff concludes that the licensee's proposed LC changes are consistent with 10 CFR 50.52, "Combining Licenses," which allows the NRC to include in a 10 CFR Part 50 license, authorization of activities that are licensed under other NRC regulations. In addition, modifying and consolidating conditions to reflect that additional fuel elements may be used in the reactor is consistent with the guidance in NUREG-1537, Part 1, Section 9.5, which indicates that nuclear material used in conjunction with the operation of the reactor, including the possession of 10 CFR Part 70, "Domestic Licensing of Special Nuclear Materials," may be authorized in a 10 CFR Part 50 license. Therefore, the NRC staff finds that the proposed changes to Section 2.B of Renewed Facility Operating License No. R-70 are acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement is required for any action within the category of actions listed in 10 CFR 51.22(c). The Commission has declared these actions to be a "categorical exclusion" as the action does not individually or cumulatively have a significant effect on the human environment.

An amendment authorizing revision of TS 1.3, TS 3.1, TS 4.1, and TS 5.3 involves changes to a requirement in the installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, "Standards for Protection against Radiation." The issuance of the proposed amendment is subject, in part, to the categorical exclusion in 10 CFR 51.22(c)(9), provided that it meets each of the criteria below:

(i) *The amendment involves no significant hazards consideration* [10 CFR 51.22(c)(9)(i)]*;*

The NRC's regulations in 10 CFR 50.92(c) state that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not—

(1) *Involve a significant increase in the probability or consequences of an accident previously evaluated"* [10 CFR 50.92(c)(1)];

The amendment would revise the TS 1.3 core configuration definition and TS 5.3 design features to reflect the close-packed array and additional items in the core grid plate that displace fuel elements, increase the TS 3.1 excess reactivity limit to $3.50, increase the TS 4.1 fuel element inspection requirements to include visual inspection of all Four-Element Fuel Bundles in rows B and C of the revised core configuration within a 2-year period, and revise the license to allow UMD to use SNM and byproduct material in the form of TRIGA fuel elements of the same design as fuel currently in the core. Chapter 13 of the UMD safety analysis report (SAR), as supplemented, and reviewed by the NRC staff in Chapter 4, "Accident Analysis," of the License Renewal SER, previously evaluated a postulated maximum hypothetical accident that bounds all accidents at the facility, including the ramp and rapid reactivity insertion events, and assumes that the release of fission products from a fuel element to the unrestricted environment results in radiological consequences. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the TSs would require that the additional nuclear material authorized for use be in a close-packed core configuration that reduces the potential for inadvertent reactivity events and adequate safety limits LCOs, and inspection requirements are maintained, to reduce accident probability. Further, although the amendment would authorize the use of an increased amount of nuclear material, the proposed core configuration results in a lower peak and average fuel element temperature and a higher DNBR, which reduces the potential for fuel element damage. Also, no changes are being proposed to reactor design or hardware, or to structures, systems, and components (SSCs) that are relied upon for accident detection, mitigation, or response. In addition, the proposed LAR does not change the licensed power level of the reactor or change any potential release paths from the facility. Therefore, the NRC staff concludes that there is no significant increase in the probability or consequences of an accident previously evaluated.

(2) *Create the possibility of a new or different kind of accident from any accident previously evaluated [10 CFR 50.92(c)(2)];*

The amendment would revise the core configuration definition and design features to reflect the close-packed array and additional items in the core grid plate that displace fuel elements, increase the excess reactivity limit, increase the number of fuel bundles required to be visually inspected annually, and allow the use of additional SNM and byproduct material in the form of TRIGA fuel elements of the same design as that currently used in the core. These changes do not create any new or different accident from any accident previously evaluated because no SSCs that are relied upon for accident detection, mitigation, or response to an accident are changed and the changes require inspections of the

newly added fuel bundles at adequate intervals. In addition, the changes would not introduce any new accident scenarios, transient precursors, failure mechanisms, or limiting single failures, and there would be no adverse effect or challenges to any reactor safety related systems as a result of the proposed amendment. Therefore, the amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Involve a significant reduction in a margin of safety" [10 CFR 50.92(c)(3)].

The amendment would revise the license and TSs to allow the use of used TRIGA fuel of the same design as fuel currently used in the core and in a close-packed core configuration, modifying the core configuration definition, core design features, excess reactivity limit, and fuel inspection requirements accordingly. The proposed changes do not authorize any changes in SSCs design, function, operation, or change the authorized steady-state reactor power level. The proposed amendment does not alter how SLs or LSSSs are determined and does not adversely affect existing facility temperature safety margins related to excess reactivity or the reliability of equipment assumed to mitigate accidents in the facility because the margin to the DNBR limit of 2.0 increases due to the lower average and peak temperature in the fuel elements. The proposed LC and TS changes do not adversely affect equipment required to safely shut down the reactor and required to maintain it in a safe shutdown condition. Therefore, the NRC staff finds that this amendment does not involve a significant reduction in a margin of safety.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite [10 CFR 51.22(c)(9)(ii)].

The amendment would revise the license and TSs to allow the use of used TRIGA fuel of the same design as fuel currently used in the MUTR core and in a close-packed core configuration, would modify the core configuration definition and design features, and increase the excess reactivity limit and fuel inspection requirements. Although the amendment would authorize the use of an increased amount of nuclear material that is the same design as that currently used in the reactor, the amendment would not result in a significant change in the types or a significant increase in the amounts of fission products and effluents that are already generated by operation of the reactor and present in the reactor fuel because the addition of reactor fuel results in a lower average power per fuel element and a lower highest powered fuel element. The MUTR Renewed License would continue to require that MUTR operate at a steady-state maximum power level of 250 kWt. The amendment does not change potential release paths from the facility. The required automatic shutdown of the ventilation system, as required by TS 3.4, if a substantial release of airborne radioactivity occurs helps to minimize the amounts of effluents that may be released off site and do not exceed limits in 10 CFR Part 20 and Table 2 of 10 CFR Part 20, Appendix B. Therefore, the NRC staff finds that there is no significant change in the types or increase in the amounts of any effluents that may be released off site.

(iii) “*There is no significant increase in individual or cumulative occupational radiation exposure” [10 CFR 51.22(c)(9)(iii)].*

The proposed amendment would make TS revisions and LC changes to allow use of additional SNM and byproduct material in the form of TRIGA fuel that is of the same design as that currently in the reactor core. Operation of the reactor in a close-packed array with the additional Four-Element Fuel Bundles will be governed by TSs that limit excess reactivity and ensure adequate cooling to maintain reactor fuel temperature below the temperature SL in TS 2.1. Reactor operation with the additional fuel elements would not result in a significant increase in individual occupational radiation exposures because the fission continues to be limited by the licensed power level of 250 kWt. In addition, the TS 4.1 revision requiring visual inspection of the additional fuel elements will not significantly increase individual or cumulative occupational radiation exposure because the inspection will continue to occur with the fuel element in the reactor pool providing radiation shielding to the occupational workers. The MUTR Renewed License, as revised, would not change the maximum power level or routine or postulated accident doses and, occupational doses would remain significantly below 10 CFR Part 20 limits in Section 20.1201, “Occupational dose limits for adults.” Additionally, existing TS 6.4, “Operating Procedures,” requires administrative controls for operating procedures that would help ensure the adequacy of the radiation protection program during the movement of fuel elements, helping to limit individual or cumulative occupational radiation exposure. Furthermore, TS 6.3, “Radiation Safety,” requires the implementation of a radiation safety program as mandated by 10 CFR Part 20. Therefore, the NRC staff finds that there is no significant increase in individual or cumulative occupational radiation exposure.

The issuance of the amendment would also make minor editorial and format changes that relabel the several LCs in Section 2.B.2 to reflect the deletion of LC 2.B.2.b, which authorized possession-only of “Alternate Reactor Fuel” that is now authorized for use in the reactor due to the inclusion of the associates grams of SNM in LC 2.B.2.a. Accordingly, this section of the amendment meets the eligibility criteria for categorical exclusion as stated in 10 CFR 51.22(c)(10)(v) for changes to the format of the license or minor revisions.

5.1 Conclusion

Accordingly, the NRC staff has determined that issuance of this amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area under 10 CFR Part 50. The NRC staff has determined that the amendment involves no significant hazards consideration, as well as no significant increase in the amounts or types of any effluents that may be released off site, and there is no significant increase in individual or cumulative occupational radiation exposure. In addition, the amendment changes the format of the license or otherwise makes editorial, corrective or other minor revisions. Therefore, this amendment meets the eligibility criteria for a categorical exclusion stated in 10 CFR 51.22(c)(9) and (c)(10)(v). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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