UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 16, 2019

Ms. Amber Johnson, Director
Nuclear Reactor and Radiation Facilities
University of Maryland
Department of Materials Science
and Engineering
4418 Stadium Drive
College Park, MD 20742-2115

SUBJECT: UNIVERSITY OF MARYLAND – REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST FOR THE USE OF 16 ADDITIONAL FUEL ELEMENTS IN THE MARYLAND UNIVERSITY TRAINING REACTOR (EPID NO. L-2018-LLA-0037)

Dear Ms. Johnson:

By letter dated January 29, 2018 (Agencywide Documents Access and Management System Accession No. ML18032A096), as supplemented by letter dated March 26, 2018 (ADAMS Accession No. ML18092A086), the University of Maryland (UMD) submitted a request for an amendment to Renewed Facility Operating License No. R-70 for the Maryland University Training Reactor. The requested amendment would authorize the use of 16 additional fuel elements in the reactor core.

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the license amendment request identified the need for additional information, as described in the enclosed request for additional information (RAI). Within 60 days from the date of this letter, provide either a response to the RAI, or a written request for additional time to respond which includes the proposed response date and a brief explanation of the reason. Following receipt of the response to the RAI, the NRC staff will continue its review of the amendment request.

The response to the RAI must be submitted in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.4, "Written communications," and, per 10 CFR 50.30(b), "Oath or affirmation," be executed in a signed original under oath or affirmation. Information included in the response that is considered sensitive or proprietary and sought to be withheld from public disclosure, must be marked in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Any information related to security should be submitted in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements."

If you have any questions regarding the NRC staff's review of the license amendment request, or if you intend to request additional time to respond, please contact me at 301-415-3398 or by electronic mail at Cindy.Montgomery@nrc.gov.

Sincerely,

/RA/

Cindy K. Montgomery, Project Manager
Research and Test Reactors Licensing Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No. 50-166
License No. R-70

Enclosure:
As stated

cc: See next page

cc:

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Test, Research and Training
Reactor Newsletter
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SUBJECT: UNIVERSITY OF MARYLAND – REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST FOR THE USE OF 16 ADDITIONAL FUEL ELEMENTS IN THE MARYLAND UNIVERSITY TRAINING REACTOR (EPID NO. L-2018-LLA-0037), DATE: APRIL 26, 2019

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ADAMS Accession No. ML19024A291 ***Concurrence via e-mail**

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DATE	4/16/2019				

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING AMENDMENT TO

RENEWED FACILITY OPERATING LICENSE NO. R-70

THE UNIVERSITY OF MARYLAND

MARYLAND UNIVERSITY TRAINING REACTOR

DOCKET NO. 50-166

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the LAR using the appropriate regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), and the following guidance:

- NUREG-1537 Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," issued February 1996 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML042430055)

- NUREG-1537 Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria," issued February 1996 (ADAMS Accession No. ML042430048)

- American National Standards Institute/American Nuclear Society, (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors"

Based on its review, the NRC staff requires the following additional information to continue its review of the LAR.

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 listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Section 4.5.2, "Reactor Core Physics Parameters," states that licensees should discuss the core physics parameters and show the methods and analyses used to determine them.

In your letter dated March 26, 2018, you indicated that the technical analysis for the addition of 16 fuel elements to the existing core configuration was provided by Oregon State University (OSU) Radiation Center. This report is listed in the references section of the LAR.

Additional information is needed for the NRC staff to understand the core physics parameters and the methods and analyses used to determine them for the proposed core configuration.

Provide a copy of the OSU Radiation Center report titled, “Analysis of the Neutronic Behavior of the Maryland University Training Reactor,” dated July 2017.

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 structures, systems, and components 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 and a detailed description of the methods used in the thermal-hydraulic analysis should be provided.

Additional information is needed for the NRC staff to understand that sufficient cooling exists for the proposed core geometry to prevent fuel 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 cooling.

Provide a thermal-hydraulic analysis for the proposed core configuration or explain why the thermal-hydraulic analysis for the current core configuration, as referenced in the LAR, bounds the proposed core configuration.

Alternatively, justify why additional information is not necessary.

RAI 3. The regulations in 10 CFR 50.9, “Completeness and accuracy of information,” require that information submitted, or information required to be maintained by the applicant be complete and accurate in all material respects.

The regulations in 10 CFR 50.36, “Technical specifications,” paragraph (c)(3) require technical specifications (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 limiting conditions for operation will be met.

The LAR requests a change to MUTR TS 4.1, “Reactor Core Parameters,” to modify the fuel bundles that are inspected annually associated with the proposed core. The LAR states TS 4.1, Specification 4 as follows:

4.1 Reactor Core Parameters

4. *A visual inspection of a representative group of fuel bundles from row C column 8, 7, 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.*

An apparent discrepancy may exist in that TS 4.1 as documented in the LAR does not match TS 4.1 in the current MUTR TSs as referenced in the LAR. The MUTR TS titled, “Technical Specifications License No. R-70, Docket No. 50-166, dated 2 December 2016,” states TS 4.1, Specification 4 as follows:

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

4. *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.*

Confirm that TS 4.1, Specification 4, in the LAR should include the inspection of bundle C6. If not, explain the apparent discrepancy between the information provided in the LAR and current MUTR TSs.

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 structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 2, Section 4.5.2, “Reactor Core Physics Parameters,” states that the calculation 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.

Table 3, “Rod worth measurements and calculations,” of the LAR compares the current core configuration simulated and measured reactivity worths for the Regulating Rod, Shim 1, and Shim 2. The difference between the simulated and measured reactivity worths for Shim 1 and Shim 2 differ by $1.72 and $1.12, respectively.

Additional information is needed for the NRC staff to understand the differences between the simulated and measured reactivity worths for Shim 1 and Shim 2 and whether the model supplying those results is suitably predictive.

Provide an explanation of the substantial difference between the calculated and measure reactivity worths for Shim 1 and Shim 2.

Additionally, state any ranges of acceptability between the simulated and measured reactivity worths and what action will be taken if the range is exceeded.

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 structures, systems, and components 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 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 MUTR TS 3.1, “Reactor Core Parameters,” Specification 1, states that 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.

By letter dated December 18, 2006 (ADAMS Accession No. ML101480913), UMD provided information in response to a request for additional information (RAI) during license renewal review. In response to RAI 84, UMD provided information on ramp reactivity insertions caused by inadvertent rod withdrawal at both low and high power conditions.

Additional information is needed for the NRC staff to understand if the information in the rod withdrawal analysis provided by letter December 18, 2006, is bounding given the proposed increase in the excess reactivity limit to $3.50 and that TS limitations give reasonable assurance that a rapid insertion of reactivity is not credible.

Provide a justification that the rod withdrawal analysis provided by the December 18, 2006, letter bounds the proposed excess reactivity LCO of $3.50. If not, provide a revised rod withdrawal analysis that considers an excess reactivity of $3.50.

Alternatively, justify why additional information is not necessary.

RAI 6. The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Appendix 14.1, “Format and Content of Technical Specifications for Non-Power Reactors,” states that the safety analysis should show the relationship between the measured fuel temperature and the maximum fuel temperature.

The current MUTR TS 2.2, “Limiting Safety Systems Settings,” states the following:

Specification

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

UMD states in the “Background Information,” of the LAR that the IFE would still trip before another element exceeds the limiting safety system setting (LSSS).

Additional information is needed for the NRC staff to understand positioning of the IFE and the margin to the safety limit for the proposed core configuration.

Provide the rationale for the statement in the LAR that the IFE would trip prior to another element exceeding the LSSS. The LSSS is 175 ºC.

Provide an analysis that shows the relationship between the measured fuel temperature in the IFE D8 position and the maximum fuel temperature within the proposed core configuration.

Alternatively, justify why additional information is not necessary.

RAI 7. The regulations in 10 CFR 50.9 require that information submitted, or information required to be maintained by the applicant be complete and accurate in all material respects.

The guidance in NUREG-1537, Part 1, “General Requirements – Physical Specifications of the Application,” states that the applicant should specify measurements in the units used in the design of the facility.

LAR Figure 2, “Current core power distribution,” does not specify any units of measure.

Clarify the units of measure used in Figure 2 of the LAR or explain why units of measure are not needed.

RAI 8. The regulations in 10 CFR 50.9 require that information submitted, or information required to be maintained by the applicant be complete and accurate in all material respects.

The guidance in NUREG-1537, Part 1, “General Requirements – Physical Specifications of the Application,” states that the applicant should specify measurements in the units used in the design of the facility.

LAR Figure 3, “Proposed core configuration power peaking factors,” provides information on power peaking factors. However, the ratio used to develop these peaking factors are not clear.

Provide an explanation of the ratio used to develop the power peaking factors presented in LAR Figure 3.

Alternatively, justify why additional information is not necessary.

RAI 9. The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Section 4.5.1, “Normal Operating Conditions,” states that licensees should present information on core geometry and configurations, including the limiting core configuration (LCC) (the core yielding the highest power density and fuel temperature using the fuel specified for the reactor), and other proposed operating core configurations that are demonstrated to be encompassed by the safety analysis of the LCC.

UMD states in “Proposed Changes to the Technical Specifications,” of the LAR that the new fuel will be assembled into three- or four-element fuel bundles, arranged in a closed-packed configuration. The configuration will match the grid plate to ensure that 1) the core fuel arrangement is closely-packed, and 2) there are no open internal positions except as identified for the in-core pneumatic experimental systems, plutonium-beryllium source, neutron detectors, and graphite reflector elements.

In the LAR, the proposed TS definition for Core Configuration is as follows:

> ***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.*

Additional information is needed for the NRC staff to understand which fuel bundles in the proposed core configuration may be displaced such that there are no open internal positions.

Provide an illustration of the proposed core configuration showing the locations of fuel bundles that may be displaced for the in-core pneumatic experimental system, plutonium-beryllium source, neutron detectors, and graphite reflector elements.

Alternatively, justify why additional information is not necessary.

RAI 10. The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components 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 the most reactive control rod is fully withdrawn (one stuck rod), non-scrammable control rods are at their most reactive position, and 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.

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. A lower 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.*

LAR Table 3, “Rod worth measurements and calculations,” provides measured and simulated control rod reactivity worths for the current core configuration. Additionally, Table 3 provides simulated control rod reactivity worths for the proposed core configuration.

With the substantial differences between the simulated and measured control rod reactivity worths for the current core configuration, the NRC staff needs additional information to understand how the shutdown margin will always be maintained under actual conditions (i.e., measured) for the proposed core configuration. Because of this substantial control rod reactivity difference in the current core configuration, the UMD simulation of the proposed core configuration may not provide adequate predictions of control rod worths for determining that the shutdown margin will be maintained for the proposed core configuration.

Provide a shutdown margin analysis that includes relevant uncertainties, error limits, and worst-case conditions and takes into account the difference between the simulated and measured control rod worths for the current core configuration.

Provide an explanation of why the proposed core configuration simulation provides acceptable predictions of control rod worths for determining that the shutdown margin will be maintained for the proposed core configuration.

Alternatively, justify why additional information is not necessary.

RAI 11. The regulations in 10 CFR 50.9 require that information submitted, or information required to be maintained by the applicant be complete and accurate in all material respects.

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.

The LAR states the following:

> *As an upward bound, a **$4.00** insertion of excess reactivity will be analyzed as the credible option for a prompt insertion of reactivity. This number is taken from technical specification 3.6.2, the total reactivity worth of an experiment.*

MUTR TS 3.6, "Limits on Experiments," Specification 2 states the following:

> *The total absolute reactivity worth of EXPERIMENTS shall not exceed **$3.00,** including the potential reactivity which might result from experimental malfunction and EXPERIMENT flooding or voiding.*

Provide an explanation for the apparent discrepancy between the statements in the LAR above referring to a $4.00 insertion of excess reactivity and MUTR TS 3.6, Specification 2. In addition, for the proposed core configuration, provide the basis for selecting $4.00 of reactivity as a bounding analysis for a credible prompt insertion of reactivity, especially given that your proposed excess reactivity is $3.50.

RAI 12. The regulations in 10 CFR 50.9 require that information submitted, or information required to be maintained by the applicant be complete and accurate in all material respects.

UMD states in "Proposed R-70 License Changes," of the LAR that License Condition (LC) 2.B.2.d is updated for grammar. However, it is not clear to the NRC staff that the current LC 2.B.2.d should be updated for grammar. The current Renewed Operating License No. R-70, LC 2.B.2.d states the following:

> *to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 80 grams of plutonium contained in encapsulated plutonium-beryllium neutron sources;*

Clarify the request for the grammar change of the current LC 2.B.2.d.

RAI 13. 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 limiting conditions for operation will be met.

NUREG-1537, Part 1, Appendix 14.1, Section 4.1.6, “Fuel Parameters,” states that for non-pulsing TRIGA reactors, approximately 20 percent of the fuel could be inspected and measured annually.

UMD proposes a change to MUTR TS 4.1, “Reactor Core Parameters,” to modify the fuel bundles that are inspected annually as follows:

4.1 Reactor Core Parameters

4. *A visual inspection of a representative group of fuel bundles from rows B and C 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, as written, is unclear on which and how many fuel bundles will be visually inspected annually. Additional information is needed for the NRC staff to understand if UMD intends to perform a visual inspection of at least 20 percent of the fuel bundles in the core on an annual frequency as referenced in NUREG-1537, Appendix 14.1, Section 4.1.6.

Provide an explanation if UMD can perform an adequate visual inspection of the remaining fuel bundles (i.e., other than fuel bundles in rows B and C) in the core. If so, state which fuel bundles can be adequately inspected.

Include additional information in TS 4.1, Specification 4, to explicitly state which and how many fuel bundles will be inspected on an annual frequency.

Alternatively, justify why additional information is not necessary.