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

February 1, 2017

Dr. Partha Chowdhury, Director
Nuclear Radiation Laboratory
University of Massachusetts-Lowell
One University Avenue
Lowell, MA 01854

SUBJECT: UNIVERSITY OF MASSACHUSETTS AT LOWELL – REQUEST FOR ADDITIONAL INFORMATION REGARDING THE RENEWAL OF FACILITY OPERATING LICENSE NO. R-125 FOR THE UNIVERSITY OF MASSACHUSETTS AT LOWELL RESEARCH REACTOR (MF7199)

Dear Dr. Chowdhury:

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review for your application for the renewal of Facility Operating License No. R-125, for the University of Massachusetts at Lowell Research Reactor dated October 20, 2015 (a redacted version of the application is available on the NRC's public Web site at www.nrc.gov under Agencywide Documents Access and Management System Accession No. ML16042A015), for the University of Massachusetts-Lowell Research Reactor.

During our review, questions have arisen for which additional information is needed. The enclosed request for additional information (RAI) identifies the additional information needed to continue our review. We request that you provide responses to the enclosed RAIs within 60 days from the date of this letter.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.30(b), "Oath or affirmation," you must execute your response in a signed original document under oath or affirmation. Your response must be submitted in accordance with 10 CFR 50.4, "Written communications." Information included in your response that is considered sensitive or proprietary, that you seek to have withheld from the public, 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." Following receipt of the additional information, we will continue our evaluation of your renewal request.

If you have any questions, or need additional time to respond to this request, please contact me at 301-415-4246, or by electronic mail at Eben.Allen@nrc.gov.

Sincerely,

/RA/

Eben Allen, Project Manager
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-223
License No. R-125

Enclosure:
As stated

cc: See next page

UNIVERSITY OF MASSACHUSETTS AT LOWELL – REQUEST FOR ADDITIONAL INFORMATION REGARDING THE RENEWAL OF FACILITY OPERATING LICENSE NO. R-125 FOR THE UNIVERSITY OF MASSACHUSETTS AT LOWELL RESEARCH REACTOR DATE: February 1, 2017

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ADAMS Accession No.: ML16112A006 ***concurrence via e-mail**

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE RENEWAL OF

THE UNIVERSITY OF MASSACHUSETTS AT LOWELL RESEARCH REACTOR

LICENSE NO. R-125; DOCKET NO. 50-223

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review for your application for the renewal of Facility Operating License No. R-125, for the University of Massachusetts at Lowell (UML) Research Reactor (UMLRR) dated October 20, 2015 (a redacted version of the application is available on the NRC public Web site at www.nrc.gov under Agencywide Documents Access and Management System (ADAMS) Accession No. ML16042A015).

In the course of reviewing the UMLRR renewal application, the NRC staff has determined that additional information or clarification is required to continue the review of the safety analysis report (SAR) in support of the development of its safety evaluation report. The UMLRR facility as described in the SAR, is primarily evaluated 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 (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)

- “Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research and Test Reactors,” dated October 2009 (ADAMS Accession No. ML092240244)

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

Enclosure

SAR-4.1

The regulations in 10 CFR 50.34, “Contents of applications; technical information,” paragraph (b)(2) require that a description and analysis of the structures, systems, and components of the facility required in 10 CFR 50.34(b)(2)(i) be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

SAR Section 4.5, “Nuclear Design,” states, in part, “The data in Table 4-1 were used in all the physics and thermal hydraulics models developed for the current UMLRR analyses. As detailed later in this chapter, the UMLRR physics and safety analysis studies were performed for a variety of core arrangements containing only UMLRR fuel, only WPI fuel, and for a variety of mixed core configurations. In all cases, the original UMLRR uranium silicide fuel is shown to be more limiting than the WPI fuel primarily because it is more reactive due to its higher U-235 loading and it has a higher average plate power due to the number of smaller fuel plates per element.”

The guidance in NUREG-1537 Part 1, Section 4.5.3, states, in part, “The applicant should present the following information on reactor operating limits: The limiting core configuration [LCC] that is possible with the planned fuel in this reactor. The limit should be imposed by the maximum neutron flux density and thermal power density compatible with coolant availability. The safety limit [SL] and limiting safety system settings [LSSSs] for the reactor should be derived from this core configuration.”

The NRC staff requires additional information to perform an evaluation to determine if your application allows the findings in NUREG-1537, Part 2, Section 4.5.1 that the assumptions and methods for core configuration, all reactivity conditions, and proposed technical specifications (TSs) (e.g., LSSSs) have been justified and validated.

Provide the following items, or justify why additional information is not necessary:

a. Results from your VENTURE and/or Monte Carlo N-Particle Transport Code models that demonstrate the ability to acceptably predict core performance by using these models to calculate measured core parameters from one or more operating core configurations (e.g., the eigenvalue for known critical conditions [k_{ecp}], control rod worths, isothermal temperature coefficients, power distributions, etc.);

b. Characterize and justify any analytical bases that result from using these models;

c. A core power map by assembly position (i.e., relative amount of power generated by the fuel assembly in each position of the core grid) for the LCC;

d. Power distribution and peaking factor information from the LCC that is then used in the thermal-hydraulic (T&H) analysis of the LCC; and

e. Information that supports the assumption in the SAR that the UMLRR fuel is more limiting than the Worcester Polytechnic Institute (WPI) fuel.

SAR-4.2

The regulations in 10 CFR 50.34(b)(2) require that a description and analysis of the structures, systems, and components of the facility required in 10 CFR 50.34(b)(2)(i) be sufficient to permit understanding of the system designs and their relationship to safety evaluations.
SAR Section 4.2.1.1, "Evaluation of the Fuel," states, in part, "The Onset of Nucleate Boiling (ONB) was used to determine operational limits of the Fuel," and "The thermal analysis later in this chapter provides the detailed results of the thermal performance of the reactor fuel and core configurations."

SAR Section 4.6.1, "Steady State Operating Limits," states, in part, "For forced flow steady state operation, a relationship between the reactor power and the pump flow rate at which the onset of nucleate boiling (ONB) point is reached needs to be established. Similarly, for steady state natural convection operation, the power level where ONB occurs also needs to be determined."

The guidance in NUREG-1537 Part 2, Section 4.6, states, in part, "The reviewer should confirm that the thermal-hydraulic analyses for the reactor are complete and address all issues that affect key parameters (e.g., flow, temperature, pressure, power density, and peaking). The basic approach is an audit of the SAR analyses, but the reviewer may perform independent calculations to confirm SAR results or methods."

Provide the following information to allow the NRC staff to complete confirmatory analysis, or justify why additional information is not necessary:

a. The correlation(s) that are used to demonstrate how flow instabilities are precluded by UMLRR analysis and the basis for using it (them);

b. The calculation process or approach used for the determination of the onset of flow instability;

c. Document whether the onset of nucleate boiling (ONB) and/or departure from nucleate boiling ratio determination include consideration of non-uniform heating in the calculation of bulk boiling;

d. The PLTEMP model input used to characterize T&H performance (e.g., is this the highest power density fuel assembly consisting of a full complement of plates or is it a group of assemblies, etc.);

e. Comprehensive T&H results (e.g., peak fuel temperature, peak cladding temperature, ONB, etc.) for the LCC demonstrating that operation at LSSS conditions results in temperatures that are bounded by the SL for the LCC for both forced flow and natural circulation conditions; and

f. Model geometry information including:

    i. The elevation of the bottom of active fuel ("meat") relative to the bottom of the whole fuel plate, or another elevation reference from which this could be derived;

ii. Drawings or diagrams with dimensions (i.e. Figure 4-3 of the SAR) of an end box and its connection to the fuel element, including the items below, or provide the numerical values separately:

1. Total height of an end box;

2. Height by which the end box overlaps the fuel element side plates;

3. Thickness of the end box walls;

4. End box interface details to the fuel plates;

5. End box flow areas at the junction to the core lower or upper plenum; the junction to the interior of the fuel element; and the minimum or choke point in the interior of the box;

iii. Geometry and configuration of the core upper and lower plena, a schematic or a scaled drawing illustrating the upper and lower plena including such features as the size; orientation and flow areas of connections to the pump-driven cooling loop and to the bulk pool. Also indicate the orientation of flow to and from these junctions and the fuel element inlets and outlets;

iv. Clarify the physical boundaries of the flow region applicable to this cooling channel pressure drop. Table 1-2 of the UMLRR SAR states that the pressure drop through the core is 0.30 pounds per square inch (e.g., does it include only the fuel plate length, or also the end boxes and junctions of those end boxes to the inlet and outlet plena, etc.).

SAR-6.1

The regulations in 10 CFR 50.34(b)(2) require that a description and analysis of the structures, systems, and components of the facility required in 10 CFR 34(b)(2)(i) be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

SAR Section 6.1, "Summary Description," states, in part, "The principal engineered safety feature for the University of Massachusetts Lowell Research Reactor (UMLRR) is the reactor containment building and its associated ventilation system."

The guidance in NUREG-1537 Part 1, Chapter 6, states, in part, "The need for ESFs is determined by the SAR analyses of accidents that could occur, even though prudent and conservative design of the facility has made the incidence of an accident very unlikely. It is also possible that for a particular design the SAR analyses will show that ESFs are not needed."

The NRC staff requires additional information to perform an evaluation to determine if your application allows the finding in NUREG-1537, Part 2, Section 6.2.1 to be made that the radiological consequences from accidents to the public, the environment, and the facility staff will be reduced by an engineered safety feature (ESF) to values that do not exceed the applicable limits of 10 CFR Part 20 for research reactors, and are as far below the regulatory limits as can be reasonable achieved.

Provide the following, or justify why additional information is not necessary:

a. Clarify if the ESF is to be evaluated as a confinement for this licensing action.

b. Discuss how the structure provides the necessary confinement analyzed in SAR Chapter 13, with cross reference to other chapters for discussion of normal operations (e.g. Chapter 4, and Chapter 11), as necessary.

c. Identify operational limits, design parameters, surveillances, and surveillance intervals that will be part of the TSs.

SAR-11.1

The regulations in 10 CFR 50.34(b)(3) requires the SAR to contain the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

SAR Section 1.3.3, "Reactor Core," Table 1-2, "Reactor Design Characteristics," indicates primary water pressure at the heat exchanger of approximately 50 pounds per square inch gauge, and secondary water pressure at the heat exchanger of approximately 35 psig.

The guidance in NUREG-1537 Part 1, Section 5.3, states, in part, "The applicant should discuss how the pressure in the secondary coolant system is maintained above that in the primary coolant system for all operating conditions, or analyze the radiological effect of leakage of contaminated primary coolant into the secondary coolant system."

The NRC staff requires additional information to perform an evaluation to determine if your application allows the finding in NUREG-1537, Part 2, Section 11.1.1 that liquid effluent volumes and radionuclide concentrations should be shown to be within the requirements of 10 CFR Part 20.

Provide an analysis to support coolant leakage limiting conditions for operation (LCO) and surveillance requirements in the TSs, include radiological consequences to members of the public, or justify why additional information is not necessary.

SAR-13.1

The regulations in 10 CFR 50.34(b)(2) require that a description and analysis of the structures, systems, and components of the facility required in 10 CFR 50.34(b)(2)(i) be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

SAR Section 13.2.2, “Insertion of Excess Reactivity,” describes three types of reactivity insertion events pertaining to UMLRR: (1) step insertions, (2) ramp insertions, and (3) cold water insertions. Table 13-9: “Max. step reactivity insertion allowed w/o hitting the ONB limit,” shows the final cladding temperatures that are higher than the SL in the TS supplied (118 °C).

The guidance in NUREG-1537 Part 1, Section 13.1.2, states, in part, “Insertion-of-excess-reactivity accidents can also be used to show that limiting conditions for operation on reactivity are justified.”

The NRC staff requires additional information to perform an evaluation to determine if your application allows the finding in NUREG-1537 Part 2, Chapter 13 that the resulting temperature from a malfunction of the control rod drive mechanism or operator error during reactor startup is lower than the SL temperature for fuel cladding.

Describe how the analyses you performed are limiting. Justify the initiating event (during reactor startup), sequence of events, automatic protective actions, and malfunctions considered; or justify why additional information is not necessary.

SAR-13.2

The regulations in 10 CFR 50.34(b)(3) require the SAR to contain the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

SAR Section 13.2.4, “Loss of Coolant,” refers to Oak Ridge National Laboratory (ORNL)-2892 (SAR Section 13.3, Ref. xiii) and ORNL-TM-632 (SAR Section 13.3, Ref. xiv) to indicate that the UMLRR core can withstand a total loss of pool water with no expectation of fuel failure.

The guidance in NUREG-1537 Part 1, Section 13.2, states, in part, “State the initial conditions of the reactor and equipment. Discuss relevant conditions dependent on fuel burnup, experiments installed, core configurations, or other variables. Use the most limiting conditions in the analyses.”

It is not clear if the fuel, fuel burnup, installed experiments, core configurations, or other decay heat producing variables used in Oak Ridge research reactor tests are comparable to the UMLRR and WPI fuel at the UMLRR facility. The NRC staff requires additional information to perform an evaluation to determine if your application allows the finding in NUREG-1537 Part 2, Chapter 13 to confirm what maximum decay heat is available at the start of the event, when the coolant level reaches the TS level, and when the coolant level reaches the top, and bottom of the core.

Provide the following, or justify why additional information is not necessary:

a. Clarify how the characteristics of the UMLRR or WPI fuel element are bounded by the results in ORNL 2892 and ORNL-TM-632;

b. The heatup characteristics of UMLRR and WPI fuel in the environment occurring during the accident;

c. The maximum amount of primary coolant that could be lost from a leak before discovery and mitigation could occur;

d. Identify the limiting break size and location, explain the sequence of events, show that the minimum possible drain time for a credible failure is acceptable, and that the assumptions in the sequence are controlled by TS, if appropriate;

e. Provide information pertaining to mitigation measures available such as pool level alarms and notification as well as mitigation measures available such as manual or automatic fill systems.

SAR-13.3

The regulations in 10 CFR 50.34(b)(2) require that a description and analysis of the structures, systems, and components of the facility required in 10 CFR 50.34(b)(2)(i) be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

SAR Section 13.2.6, "Experiment Malfunction," states, in part, "The amount of explosive materials which can be irradiated, or which is allowed to generate in any experiment, has been limited to 25 milligrams of TNT-equivalent explosives in order to reduce the likelihood of damage to the reactor or pool should the explosive material detonate. The irradiation container for this material is required to be tested to ensure the no damage to the container would result from detonation of the explosive."

Regulatory Guide 2.2(C)1c, "Mechanical Stress Effects," states, "(1) Every experiment should be evaluated with respect to the storage and possible uncontrolled release of any mechanical energy (2) Experiments involving a potential for creating objects with substantial momentum (missiles) should be oriented in such a way as to minimize the probability of damage to the reactor system (3) Materials of construction and fabrication and assembly techniques utilized in experiments should be so specified and used that assurance is provided that no stress failure can occur at stresses twice those anticipated in the manipulation and conduct of the experiment or twice those which could occur as a result of unintended but credible changes of, or within, the experiment (4) Prototype testing under experiment conditions should be employed to demonstrate the ability to withstand failure."

While there is a process for testing the encapsulations, the NRC staff requires additional information to determine if that testing is sufficient. Analysis showing mechanical stresses of the encapsulation twice those anticipated consistent with (3) above have been previously accepted by the NRC to demonstrate that detonation will not result in radioactive material escaping into the reactor room or the air exhaust stream to the unrestricted environment.

Document the methods and assumptions used to support calculations that demonstrate compliance with the factor of two margin with suitable conservatism, or justify why additional information is not necessary.

SAR-13.4

The regulations in 10 CFR 50.9, “Completeness and accuracy of information,” paragraph (a) require that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.

SAR Section 3.5.3, “Reactor Protection System (RPS) Function,” states, in part, “The reactor protective system complies with the single failure criterion of IEEE-279. A malfunction in one of the reactor safety system trip actuator amplifiers (TAAs) could result in, at most, the failure to interrupt the current to two control blade electromagnets, in which case the other two control blades would drop and successfully shut down the reactor.”

SAR Section 13.2.7, “Loss of Normal Electrical Power,” states, in part, “Passive safety features exist so that on loss of electricity the reactor will shut down and the reactor building ventilation will be isolated. Specifically, the four reactor control blades are attached to their drives by electromagnets. On loss of power, the magnets de-energize and the blades drop into the core by gravity.”

The NRC staff requires additional information to perform an evaluation to determine if your application allows the finding in NUREG-1537 Part 2, Chapter 13 that the design bases of the electromechanical systems and components give reasonable assurance that the facility systems and components will function as designed to ensure safe operation and safe shutdown of the reactor.

Clarify if a malfunction in one of the reactor safety system TAAs could prevent the reactor protective and safety systems from performing its intended safety function.

SAR-14

The regulations in 10 CFR 50.36, “Technical specifications,” paragraph (a)(1) require each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. 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 technical specifications.

SAR Chapter 14, Section 1.2.1, “Purpose,” states, “The technical specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission (NRC) on administrative controls, operational parameters, and equipment requirements, for safe reactor operation and for dealing with abnormal situations. They are typically derived from the safety analysis report (SAR). These specifications represent a comprehensive envelope for safe operation. The operational parameters and equipment requirements directly related to preserving this safe envelope are included.”

The guidance in NUREG-1537 Part 1, Appendix 14.1, Section 1.2.2, states, “Technical specifications that use the SAR as a basis should explicitly reference the SAR section number. In addition, any other sources used to support the technical specification should be explicitly referenced.”

While the TSs generally follow the guidance in NUREG-1537 Part 1, and ANSI/ANS-15.1-2007, those documents do not provide sufficient guidance to meet the regulatory requirements in 10 CFR 50.36(a)(1). Prepare a basis for all the design features in Section 5 of your TSs.