November 7, 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 (CAC/DOCKET/EPID NO. A11010/05000223/L-2015-RNW-0001)

Dear Dr. Chowdhury:

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of 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).

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 safeguards 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 301-415-4067, or by electronic mail at Edward.Helvenston@nrc.gov.

Sincerely,

**/RA/**

Edward Helvenston, Project Manager
Research and Test Reactors Licensing Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

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

Enclosure:
As stated

cc: See next page

cc:

Mayor of Lowell
City Hall
Lowell, MA 01852

Mr. Leo Bobek
Reactor Supervisor
University of Massachusetts - Lowell
One University Avenue
Lowell, MA 01854

Department of Environmental Protection
One Winter Street
Boston, MA 02108

Jack Priest, Director
Radiation Control Program
Department of Public Health
Schrafft Center, Suite 1M2A
529 Main Street
Charlestown, MA 02129

John Giarrusso, Planning and Preparedness Division Chief
Massachusetts Emergency Management Agency
400 Worcester Road
Framingham, MA 01702-5399

Test, Research and Training
Reactor Newsletter
P.O. Box 118300
University of Florida
Gainesville, FL 32611

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 (CAC/DOCKET/EPID NO. A11010/05000223/L-2015-RNW-0001)
DATED: NOVEMBER 7, 2017

**DISTRIBUTION**:

| | | | |
|---|---|---|---|
| Public | RidsNrrDlpPrlb | RidsNrrDlp | RidsNrrDlpProb |
| STraiforos, NRR | NParker, NRR | JEads, NRR | PRLB R/F |
| EHelvenston, NRR | AAdams, NRR | AMendiola, NRR | CMontgomery, NRR |

**ADAMS Accession No.: ML17276B783** ***concurrence via e-mail**

| **OFFICE** | NRR/DLP/PRLB/PM* | NRR/DLP/LA* | NRR/DLP/PRLB/BC | NRR/DLP/PRLB/PM |
|---|---|---|---|---|
| **NAME** | EHelvenston | NParker | AAdams | EHelvenston |
| **DATE** | 10/11/17 | 10/10/17 | 11/7/17 | 11/7/17 |

**OFFICIAL RECORD COPY**

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 of 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 its review of the safety analysis report (SAR) in support of the development of its safety evaluation report. 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" (license renewal ISG), dated October 2009 (ADAMS Accession No. ML092240244)

- American National Standards Institute/American Nuclear Society, (ANSI/ANS)-15.1-2007 (R2013), "The Development of Technical Specifications for Research Reactors."

Enclosure

**RAI-1.1**

The regulation in 10 CFR 50.21(c) states that a class 104c license will be issued to an applicant who qualifies to possess and use “a production or utilization facility, which is useful in the conduct of research and development activities […].”

In the transmittal letter for its license renewal application, UML requests that its class 104c utilization facility license include an allowance to possess, and use in connection with operation of the facility, up to 50 grams of plutonium-beryllium sources, and up to 20 millicuries (mCi) per radionuclide and 50 mCi total of atomic numbers 3 through 83 in any form. SAR Section 9.5.1 states that this material is for checks, calibrations, and characterizations of radiation monitoring instruments.

a) It is not clear to the NRC staff why possession and use of atomic numbers 3 through 83 in “any form” is requested, if this material is contained in check and calibration sources that will be used in connection with facility operation. Clarify why possession and use of “any form” is requested; revise the request to clarify that the material will be limited to sealed (and/or plated) check and calibration sources; or, justify why no additional information is required.

b) It is not clear to the NRC staff whether “50 grams of plutonium-beryllium sources” refers to 50 grams total of plutonium plus beryllium, or 50 grams of plutonium contained in a larger mass of plutonium-beryllium sources. Clarify which of these is being requested, or justify why no additional information is required.

**RAI-9.1**

The regulations in 10 CFR 50.34(b)(2) require that the SAR contain a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements; the bases, with technical justification therefor, upon which such requirements have been established; and the evaluations required to show that safety functions will be accomplished.

The regulations in 10 CFR 50.36(c)(2) require that technical specifications (TSs) include limiting conditions for operation, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

The guidance in NUREG-1537, Part 1, Section 9.2, states that applicants should provide analysis and discuss how subcriticality is ensured under all conditions of fuel storage, except during transportation off site.

The license renewal ISG, Section 9.2, states that as part of the focused licensed renewal process, a review of fuel storage is limited to applicable TSs (and accident analysis, if appropriate).

The guidance in ANSI/ANS-15.1-2007, Section 5.4, states that the TSs should include a requirement that "[f]uel, including fueled experiments and fuel devices not in the reactor, shall be stored in a geometric array where $k_{eff}$ is no greater than 0.90 for all conditions of moderation and reflection using light water […]."

SAR Section 9.2 states that storage of both new and irradiated fuel elements at the UMLRR is provided by fuel storage racks kept within the reactor pool. SAR Section 9.2.1 discusses the in-pool fuel storage racks at the UMLRR, including how the design of the racks helps allow natural convection pool water cooling of the fuel elements being stored. The SAR does not appear to contain an analysis of $k_{eff}$ for the fuel storage racks. UML's response, dated August 11, 2016 (ADAMS Accession No. ML16224A322), to NRC Generic Letter 2016-01, "Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools," dated April 7, 2016 (ADAMS Accession No. ML16097A169), states that a criticality analysis for a fuel storage rack fully loaded with fresh fuel elements has shown that without credit for Boral neutron absorbing material, the linear array has a maximum $k_{eff}$ of less than 0.7. However, the NRC staff requires additional information to fully understand certain details of UMLRR's fuel storage, to determine that the assumptions made for the criticality analysis are conservative and bounding, and to determine that the UMLRR TSs include appropriate TSs for fuel storage. Provide the following, or justify why no additional information is required:

a) While UML's response to NRC Generic Letter 2016-01 states that a criticality analysis has been performed for the fuel storage racks, the methodology and assumptions for this analysis are not fully clear. Discuss the methodology and assumptions used to perform the analysis, including whether the analysis considered that the fuel storage racks are submerged in water, and whether neutronic coupling between other racks in the pool (given that the pool contains eight racks as discussed in SAR Section 9.2.1 and illustrated in SAR Figure 9-3 on SAR page 9-6) and/or the reactor core was considered; and, justify the assumptions used.

b) SAR Section 9.2.1 states that each rack holds nine fuel elements in a planar array. However, SAR Figure 9-4 on SAR page 9-7 appears to illustrate a fuel storage rack which holds 13 fuel elements. Clarify which fuel rack capacity is correct, and clarify whether UML's fuel storage criticality analysis is consistent with the correct fuel rack capacity.

c) The NRC staff notes that the proposed TSs do not appear to include a requirement that fuel and other fissionable material be stored in a criticality-safe configuration, as recommended by ANSI/ANS-15.1-2007. Provide a TS to require that fuel and other fissionable material be stored in a criticality-safe configuration, or justify why no such TS is necessary.

d) The NRC staff notes that the proposed TSs do not appear to include a requirement that irradiated fuel and other irradiated fissionable material be stored such that adequate cooling will be ensured. Provide a TS to require that irradiated fuel and other irradiated fissionable material be stored in a location where it will be adequately cooled, or justify why no such TS is necessary.

**RAI-11.2**

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding 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.

In SAR Section 11.1.1.1.3, UML provides analyses of doses to members of the public from gaseous argon-41 (Ar-41) released from the reactor.

The guidance in NUREG-1537, Part 1, Section 11.1.1.1, states, in part, that assumptions and methods used for calculation of doses from airborne radionuclides released from research reactors should be conservative but physically realistic. NUREG-1537, Part 1, Section 11.1.1.1, also states, in part, that in the unrestricted area, potential doses should be analyzed for the maximally exposed individual, at the location of the nearest permanent residence, and at any locations of special interest.

a) While the analyses in SAR Section 11.1.1.1.3 provide calculations of potential doses at maximum exposure locations, the NRC staff noted that they do not appear to include calculation of doses at the nearest permanent residence.

    i. Provide the location of the nearest permanent residence relative to the facility stack, or justify why no additional information is required.

    ii. Provide a calculation of the dose from Ar-41 at the nearest permanent residence; provide a discussion of how the doses calculated in SAR Section 11.1.1.1.3 are representative of doses at the nearest permanent residence; or, justify why no additional information is required.

b) The HOTSPOT model used for the analyses in SAR Section 11.1.1.1.3 assumes Pasquill stability class F. While stability class F is generally conservative for ground releases, or for elevated releases when the receptor is very far from the release point, the NRC staff notes that stability class F (i.e., moderately stable conditions) may not necessarily be conservative (or representative of average conditions) for releases from the UMLRR stack. Justify why stability class F is conservative and/or representative of average weather conditions at the UMLRR; provide a calculation of the dose from Ar-41 for another stability class, along with a justification that that stability class would be conservative; provide a calculation of the dose from Ar-41 for neutral or average stability conditions; or, justify why no additional information is required.

c) The analyses in SAR Section 11.1.1.1.3 consider the Ar-41 releases from the UMLRR to be an elevated release from the 100 foot facility stack. NRC Regulatory Guide 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” Revision 1, states that for routine effluents released from points at the level of or above adjacent solid structures, but less than twice the height of adjacent solid structures, determination of whether the gaseous effluent plume should be considered as an elevated release is based on the vertical exit velocity of the plume, and the horizontal wind speed.

i. Based on SAR Figure 3-2 on SAR page 3-13, it appears that the top of the reactor building is approximately 80 feet above the beam level, which is close to grade level. Based on SAR Figure 3-3 on SAR page 3-14, it appears that there is approximately 70 feet between the top of the reactor building and the beam/grade level. Depending on which figure is correct, it appears that the top of the 100 foot UMLRR stack is approximately 20 or 30 feet above the top of the reactor building. Clarify which figure in the SAR, 3-2 or 3-3, shows the correct dimensions.

ii. Given that the 100 foot UMLRR stack appears to be less than twice the height of the adjacent reactor building, justify the consideration of releases from the stack as elevated; provide a revised calculation that considers that during certain conditions, it may not be conservative to consider stack releases to be elevated at the full height of the stack, due to downwash of the effluent plume; or, justify why no additional information is required.

d) SAR Figures 11-1 and 13-1 on pages 11-9 and 13-28, respectively, provide wind roses for Hanscom Air Force Base for 2013. However, the two wind roses appear to be based on different data sets. Clarify the difference between the two wind roses, and which wind rose was used for the Ar-41 dose calculation in SAR Section 11.1.1.1.3; or, justify why no additional information is required.

**RAI-11.3**

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding 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.

The regulations in 10 CFR 20.2003(a) state that licensees may discharge licensed radioactive material to sanitary sewerage if the material is readily soluble in water, and if the monthly concentrations of the material do not exceed the limits in 10 CFR Part 20, Appendix B, Table 3.

SAR Section 2.1.1.2 states that the reactor operations boundary is the reactor containment [(confinement)] building.

SAR Sections 3.5.9, 11.1.1.2, and 11.2.3.2 state that any radioactive liquid effluents produced at the reactor are transferred to the liquid waste hold-up tanks outside the reactor building, and that liquid in these tanks is analyzed for its radioactive content before being discharged.

It is not clear to the NRC staff how storage and disposal of these radioactive liquid effluents would be controlled once they are transferred out of the reactor building, since they may no longer be within the licensed area of the reactor.

Provide the following information, or justify why no additional information is required:

a) Clarify whether the reactor operations boundary (the reactor building) is the area controlled under the reactor Facility Operating License No. R-125, or whether the area controlled under Facility Operating License No. R-125 extends beyond the reactor building.

b) Clarify whether radioactive liquid effluents transferred from the reactor building to the liquid waste hold-up tanks are still considered to be controlled (for storage and disposal) under the reactor Facility Operating License No. R-125, or whether they become under the control of another license (i.e., a State of Massachusetts radioactive materials license) when they are transferred out of the reactor building.

c) If radioactive liquid effluents are transferred to the control of another license, describe how UML ensures that the other license is able to receive and possess the radioactive material in the effluents before the material is transferred from the reactor license.

d) Describe how the analyses performed on liquid radioactive waste in the hold-up tanks prior to its discharge ensure compliance with the release limits and solubility requirement of 10 CFR 20.2003.

**RAI-11.4**

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding 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.

The guidance in NUREG-1537, Part 1, Section 11.1.4, states, in part, that licensees should provide summary descriptions of all radiation monitoring equipment employed throughout the facility, including locations and functions of each device and system. NUREG-1537, Part 1, Section 11.1.4, also states, in part, that licensees should describe how routine monitoring provided at the facility will ensure that radiation exposures to workers and the public can be detected.

Radiation monitoring systems at the UMLRR are described in SAR Sections 7.7 and 11.1.4.2. The SAR discusses the functionality of the radiation monitors, including alarms and scrams which are triggered by combinations of radiation monitors reaching their setpoints. However, the setpoints that are used for radiation monitors at the UMLRR, and how these setpoints are determined, do not appear to be discussed in the SAR.

Provide representative setpoints for radiation monitors (including area monitors, the stack effluent monitor, the stack gas monitor, continuous air monitors, and other monitors, as applicable) at the UMLRR, and discuss how these setpoints are determined; or, justify why no additional information is required.

**RAI-11.5**

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding 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.

The guidance in NUREG-1537, Part 1, Section 11.1.7 states, in part, that licensees should describe their environmental monitoring and surveillance programs, and the bases for these programs should be described.

a) SAR Table 7-8 includes an "Environmental Gamma" radiation monitor which monitors environmental radiation levels within the UMLRR. SAR Table 11-6 includes an "EcoGamma" radiation monitor which monitors environmental radiation levels around the UMLRR. SAR Section 11.1.7 states that environmental monitors are used throughout the Pinanski Building attached to the UMLRR, and area dosimeters are located in the first and third floor airlocks and in Pinanski rooms 101 and 301. However, it is not clear whether the "Environmental Gamma" monitor in SAR Table 7-8 is the same as the "EcoGamma" monitor in SAR Table 11-6, and it is also not clear whether the other area dosimeters in the airlock and Pinanski building are part of the "EcoGamma" monitor, or are separate, standalone monitors (i.e., thermo-luminescent dosimeters or other passive dosimeters). Provide clarification of these items, or justify why no additional information is required.

b) The SAR does not appear to describe any environmental monitoring that is performed beyond the reactor building and attached Pinanski building. Describe and provide a basis for the level of monitoring that is performed outside these buildings.

**RAI-13.5**

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding 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.

The regulations in 10 CFR 50.9(a) require that information provided to the NRC by a licensee shall be complete and accurate in all material respects.

The guidance in NUREG-1537, Part 1, Chapter 13, describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include a maximum hypothetical accident (MHA). NUREG-1537, Part 1, Section 13.1.1, states that the MHA bounds all credible accidents and can be used to illustrate the analysis of events and consequences during an accidental release of radioactive material.

SAR Section 13.2.1 provides the MHA analysis for the UMLRR. The SAR MHA analysis considers five scenarios, Scenario A through Scenario E. The NRC staff requires additional information to fully understand certain details of the UMLRR MHA analysis, and to determine that the assumptions made in the analysis are conservative and bounding. Provide the following, or justify why no additional information is required:

a) The equation in SAR Section 13.2.1.1, page 13-8, appears to contain a possible typographical error in that the term "$3.1 \times 10^{11}$" should read "$3.1 \times 10^{10}$." Clarify what this term should read, and if it is a typographical error, whether any MHA calculations in SAR Section 13.2.1 are affected.

b) SAR Section 13.2.1.3, page 13-16, states that Federal Guidance Report (FGR)-11 and FGR-12 dose conversion factors were used for "Scenarios A through D." However, the dose conversion factors used for Scenario E are not discussed. Clarify whether "Scenarios A through D" is a typographical error that should read "Scenarios A through E," and clarify whether any MHA calculations in SAR Section 13.2.1 are affected.

c) SAR Section 13.2.1.5, page 13-25, Table 13-3, appears to contain a possible typographical error in that the whole-body inhalation dose conversion factor for iodine-135, "3.32E10" should read "3.32E-10." Clarify what this term should read, and if it is a typographical error, whether any MHA calculations in SAR Section 13.2.1 are affected.

d) SAR Section 13.2.1.2, page 13-16, appears to contain a possible error in that the statement "radioactive decay considered for the gamma shine component and the *ground* release" should read "[…] and the *elevated* release" because Scenario E considers gamma shine and an elevated release, not a ground release. Clarify what this statement should read, and if it is an error, whether the results of any MHA calculations in SAR Section 13.2.1 are affected.

e) SAR Section 13.2.1.1 discusses the source term and release fractions for the UMLRR MHA. SAR Section 13.2.1.1 states that all iodine isotopes released from the failed fuel plate to the reactor pool are released in elemental form, and are readily dissolved in the primary coolant. SAR Section 13.2.1.1 also describes a methodology, based on mole fractions and partial pressure of iodine in the reactor building air, which is used to calculate that the release fraction of iodine from the reactor pool to the reactor building air is approximately 4.6E-5. However, the NRC staff notes that this methodology may not be valid, because it is based on the ideal gas law which does not apply to heavy gases such as iodine vapor. Additionally, the methodology does not consider that additional iodine could be released from the pool after some airborne iodine was released from the reactor building, since the partial pressure of iodine would drop once iodine was released from the building. Justify the use of the 4.6E-5 release fraction of iodine from the reactor pool to the reactor building air; or, provide an MHA calculation based on a justifiable release fraction from iodine from the reactor pool to the reactor building air.

f) SAR Section 13.2.1.5, Table 13-2, provides calculated values of fission product activities released from the failed fuel plate into the reactor pool, and from the reactor pool into the reactor bay. SAR Section 13.2.1.1 states that the recoil distance (thickness of fuel plate from which fission products would be released) is 1.37E-3 centimeters, which, based on the fuel thickness, corresponds to a release of approximately 2.7 percent of the fuel plate fission product inventory. However, the NRC staff notes that for xenon-133m and xenon-135m, SAR Table 13-2 appears to show releases to the reactor pool that do not correspond to the 2.7 percent release. Additionally, SAR Section 13.2.1.1 states that all (i.e., 100 percent) of noble gases released to the reactor pool are also released to the reactor bay, but the NRC staff noted that for xenon-131m, xenon-133m, and xenon-135m, the activities released to the reactor bay do not match the activities released to the reactor pool. Justify the reason for these discrepancies, or provide revised calculations that are consistent with the assumptions discussed in SAR Section 13.2.1.1.

g) The MHA analyses in SAR Section 13.2.1 do not appear to consider the short-lived noble gas fission products krypton-90, xenon-137, and xenon-139. The NRC staff notes that although the half-lives of these three fission products are short, given the speed with which they could potentially be released to the reactor building during an accident, they could potentially contribute significantly to the MHA occupational doses (Scenario A). Provide an occupational dose calculation for the MHA that includes the contribution from krypton-90, xenon-137, and xenon-139; or, justify why the contribution of these fission products to occupational doses would be small and they do not need to be included.

h) SAR Section 13.2.1.5, Table 13-4, provides calculated values of occupational doses for Scenario A. The NRC reviewed this table, and noted that the columns labelled “Thyroid Inhalation CEDE” and “Thyroid Submersion CEDE” appear to be committed dose equivalents (CDEs) rather than committed effective dose equivalents (CEDEs). Additionally, the NRC staff noted that for iodine-131 through iodine-135, there are large differences in the thyroid submersion doses and whole-body submersion doses, but given the similarity in dose conversion factors (assuming the thyroid submersion doses are CDEs), thyroid submersion CDEs and whole-body submersion total effective dose equivalents (TEDEs) should be very similar. Clarify these discrepancies, and, if necessary, provide revised occupational dose calculations for Scenario A.

i) While the MHA analyses in SAR Section 13.2.1 provide calculations of potential doses at maximum exposure locations for members of the public, the NRC staff notes that they do not appear to include calculation of doses at the nearest permanent residence. Provide an MHA dose calculation for the nearest permanent residence; or, provide a discussion of how the doses calculated in SAR Section 13.2.1 are representative of doses at the nearest permanent residence.

j) The public dose calculations in SAR Section 13.2.1 include calculations assuming no ventilation or leakage from reactor building and external gamma shine from radioactive material in building only (Scenario B); leakage from the truck door, plus external gamma shine (Scenario C); ventilation to the stack with normal ventilation system lineup, plus external gamma shine (Scenario D); and, ventilation to the stack through the emergency exhaust system, plus external gamma shine (Scenario E). SAR Section 13.2.1.2 states that Scenario E represents the most likely evolution of a potential MHA. As discussed in SAR Section 6.2.3 and in UML's response to RAI-6.1 dated March 31, 2017 (ADAMS Accession No. ML17090A350), during an accident that released radioactive material to the reactor building air, the ventilation system lineup would normally automatically change such that the reactor building would be vented to the stack through the emergency exhaust system. The NRC staff notes, however, that based on the calculated doses listed in SAR Section 13.2.1.5, page 13-27, Tables 13-5 through 13-8, Scenario D, rather than Scenario E, could potentially result in the lowest public dose. Compared to Scenario D, Scenario E shows higher external gamma shine at publicly-accessible locations near the reactor building truck door. Additionally, the stack release dose contributions for both Scenario D and Scenario E are small, and the NRC staff notes that, if similar meteorological conditions were used for Scenario D and Scenario E (Scenario E used average meteorological conditions, which may not be conservative, as noted in SAR Section 13.2.1.3), the stack release dose contributions for Scenario D and Scenario E may be more similar. Furthermore, the NRC staff notes that Scenario D does not consider radioactive decay, which results in a conservative estimate of external gamma shine and stack release doses relative to Scenario E. Justify how the use of the Scenario E ventilation system lineup (which delays the release of radioactive material from the reactor building) following a radioactive material release into the reactor building air would be as low as is reasonably achievable (ALARA), given that Scenarios B, C, and E show the elevated external gamma shine doses that could occur if the material were held up in the building longer, and the public doses from the stack release may be comparable regardless of whether the normal ventilation or emergency exhaust system is operating; or, provide an alternate scenario which represents actions that UML would take during an actual release of radioactive material to the reactor room air to maintain doses ALARA, but which still uses conservative, bounding assumptions (for release fractions, meteorology, etc.) consistent with the MHA guidance in NUREG-1537.

k) SAR Section 13.2.1.5, Tables 13-7 and 13-8, provide public dose calculation results for Scenarios D and E, respectively. For Scenarios D and E, the doses at the maximum elevated dose location 1.7 kilometers from the stack are 0.15 millirem and 0.047 millirem, respectively. The NRC staff notes that the Scenario D elevated dose is approximately a factor of three greater than the Scenario E dose. However, the NRC staff also notes that, as discussed in SAR Section 13.2.1.2, Scenario D assumes no radioactive decay and that the wind blows toward the receptor at one meter per second for the entire release duration, while Scenario E assumes radioactive decay and that the winds are based on average weather conditions (which, based on the wind rose in SAR Figure 13-1, would result in the wind blowing toward the receptor for approximately 17 percent of the release duration, because that is the greatest fraction of the year in which the wind blows in any one direction). Given the degree of conservatism used for the Scenario D elevated release dose calculation compared to the Scenario E calculation, it is not clear to the NRC staff why there is only a factor of approximately 3 difference (instead of a greater difference) between the Scenario D and E results. Explain the reason for this discrepancy.

l) In SAR Section 13.2.1, public dose Scenarios B, C, and E take credit for radioactive decay. Clarify whether the dose contributions from decay products are considered for these scenarios. If they are not considered, provide Scenario B, C, and E calculations which include these dose contributions; justify why these dose contributions are small and do not need to be included; or, clarify if Scenarios B, C, and E are not necessary to determine the doses from a scenario representing what would occur during the MHA, and are provided only for comparison and sensitivity analysis purposes.

m) SAR Section 13.2.1.5, page 13-27, Table 13-6, lists the calculated public doses for Scenario C. Given that Scenario C represents a ground release from the truck door, it is not clear why the dose is higher 70 meters from the truck door than it is 20 meters from the truck door. Clarify why the dose is higher at 70 meters from the truck door.

n) SAR Section 13.2.1.2 states that public dose Scenario C assumes a leakage rate from the truck door of one percent of the reactor building volume per day. Given that, as stated in the response to RAI-6.1 dated March 31, 2017 (ADAMS Accession No. ML17090A350), the reactor building will be changed from a containment to a confinement, leak rate surveillances will no longer be performed on the building. Justify the one percent leakage rate given that no surveillances will be performed for building leakage; or, clarify if Scenario C is not necessary to determine the doses from a scenario representing what would occur during the MHA, and is provided only for comparison and sensitivity analysis purposes.

o) SAR Section 13.2.1.2 states that public dose Scenario D assumes Pasquill stability class F and a constant wind speed and direction. SAR Section 13.2.1.2 also states that public dose Scenarios C and E assume that wind directions are based on a historical wind rose for Hanscom Air Force Base for 2013 provided in SAR Figure 13-1. However, the Pasquill stability class assumed for Scenarios C and E is not clear. Additionally, while stability class F is generally conservative for ground releases, or for elevated releases when the receptor is very far from the release point, the NRC staff notes that stability class F (i.e., moderately stable conditions) may not necessarily be conservative (or representative of average conditions) for releases from the UMLRR stack when potential receptors during accident conditions could be located close to the release point.

   i. Given that SAR Figures 11-1 and 13-1 both represent a historical wind rose for Hanscom Air Force Base for 2013, clarify that the wind rose provided in SAR Figure 13-1 is the wind rose used for the Scenario C and E public dose calculations.

ii. Clarify what Pasquill stability class was assumed for Scenarios C and E. Additionally, justify why this stability class is conservative and/or representative of average weather conditions at the UMLRR; provide revised Scenario C and E calculations for another stability class, along with a justification that that stability class would be conservative; or, clarify if Scenarios C and E are not necessary to determine the doses from a scenario representing what would occur during the MHA, and are provided only for comparison and sensitivity analysis purposes.

iii. Justify why stability class F is conservative for Scenario D; or, provide a Scenario D calculation for another stability class, along with a justification that that stability class would be conservative.

p) The analyses for Scenarios D and E in SAR Section 13.2.1 consider the radioactive material releases from the UMLRR during the MHA to be elevated releases from the 100 foot facility stack. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, states that for conservative estimation of dose consequences from accidents, releases from points that are effectively lower than 2.5 times the height of adjacent solid structures should be treated as ground releases. Given that the 100 foot UMLRR stack appears to be less than twice the height of the adjacent reactor building, justify the consideration of releases from the stack as elevated releases; or, provide a revised calculation that considers that it may not be conservative to consider stack releases to be elevated at the full height of the stack.

**RAI-13.6**

The regulations in 10 CFR 50.34(b)(2) require that the SAR contain a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements; the bases, with technical justification therefor, upon which such requirements have been established; and the evaluations required to show that safety functions will be accomplished.

The regulations in 10 CFR 50.36(c)(2) require that TSs include limiting conditions for operation, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

The guidance in NUREG-1537, Part 1, Chapter 13, describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include a loss of coolant accident (LOCA), which is discussed in NUREG-1537, Part 1, Section 13.1.3. NUREG-1537, Part 1, Section 13.2, describes how accidents should be analyzed in the SAR, and states that the applicant should discuss each event giving information consistently and systematically for gaining a clear understanding of the specific reactor and making comparisons with similar reactors. NUREG-1537, Part 1, Section 13.2 also states that the most limiting conditions should be used in the analysis.

SAR Section 13.2.4, as supplemented by the response to RAI-13.2 dated March 31, 2017 (ADAMS Accession No. ML17090A350), provides the LOCA analysis for the UMLRR. The response to RAI-13.2 describes how the design of the facility helps minimize the likelihood and consequences of a severe LOCA. The response to RAI-13.2 also discusses a LOCA analysis performed for the Rhode Island Nuclear Science Center (RINSC) research reactor, and provides discussion and comparisons to show that the consequences of a LOCA at the UMLRR would be bounded by the RINSC LOCA analysis. However, the NRC staff requires additional information to fully understand certain details of the UMLRR LOCA analysis, and to determine that the assumptions made in the analysis are conservative and bounding. Provide the following, or justify why no additional information is required:

a) The response to RAI-13.2(a), (b), and (d) references a LOCA analysis performed by Dr. Earl Feldman for the RINSC reactor, but the NRC staff noted that this analysis, which is used as part of the basis for demonstrating the acceptability of the LOCA analysis for the UMLRR, does not appear to be included with the RAI response. Provide a copy of the version(s) of the Feldman analysis which are referenced and used as part of the basis for the UMLRR LOCA analysis, or provide a reference as to where the version(s) of the Feldman analysis can be found in NRC ADAMS.

b) The response to RAI-13.2(a), (b), and (d) states that the RINSC LOCA analysis is relevant to the UMLRR because the RINSC and UMLRR facilities have similar physical characteristics. However, the NRC staff noted that there are some differences in the RINSC and UMLRR cores for which it is not clear how the LOCA analysis could be affected. The RINSC reactor uses exclusively silicide fuel, and has a beryllium reflected core, while the UMLRR could include a mix of silicide and aluminide fuel, and is graphite reflected. Explain whether, and how, these or other differences between the cores could affect the applicability of the RINSC LOCA analysis to the UMLRR.

c) SAR Section 4.2 describes the reactor core, including the design of the core box. However, it is not clear to the NRC staff whether the sides and bottom of the core box are watertight, or whether some water can flow between the inside of the core box and the bulk pool through penetrations or other small openings in the sides and bottom of the core box (i.e., whether the core box and the bulk pool are hydraulically connected). The NRC staff needs to understand this aspect of the core box design in order to perform confirmatory calculations of the UMLRR LOCA analysis. Clarify whether the core box is designed to be watertight, or whether the core box is not designed to be watertight and may be assumed to be hydraulically connected to the bulk pool.

d) The response to RAI-13.2(a), (b), and (d) states that for each of the two 6-inch and one 8-inch beam ports penetrating the reactor pool, a heavy lead shutter located within the pool shield wall and a bolted shield plug at the outer shield would help prevent accidental drainage of the pool through one of these beam ports (for example, in a scenario where the pool end of one of the beam ports was sheared off). The RAI response also states that due to radiological considerations, the lead shutter is not raised and the shield plug removed at the same time when the reactor is in the stall end of the pool, even after an extended shutdown period. However, the NRC staff notes that there do not appear to be proposed TS controls preventing the lead shutter being raised and the shield plug removed at the same time. Provide a TS that would help mitigate the consequences of a LOCA by requiring that during and for an appropriate period following reactor operation (at any pool location), for each beam port, the lead shutter shall not be raised and the shield plug shall not be removed at the same time; or, justify why no such TS is required.

e) The response to RAI-13.2(a), (b), and (d) states that a conservative, average pool surface area of 372 square feet is used for the UMLRR LOCA analysis. This surface area is based on the surface area of the entire pool (i.e., it includes both the stall and bulk pool regions); therefore it is assumed that the pool divider gate would be open during a LOCA. However, the NRC staff notes that there do not appear to be proposed TS controls requiring the pool divider gate to be open. Provide a TS that would help mitigate the consequences of a LOCA by requiring that during and for an appropriate period following reactor operation, the pool divider gate shall be open; or, justify why no such TS is required.

**<u>RAI-13.7</u>**

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding 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.

The regulations in 10 CFR 50.9(a) require that information provided to the NRC by a licensee shall be complete and accurate in all material respects.

The guidance in NUREG-1537, Part 1, Chapter 13, describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include a LOCA, which is discussed in NUREG-1537, Part 1, Section 13.1.3. NUREG-1537, Part 1, Section 13.2, describes how accidents should be analyzed in the SAR, and states that potential radiological consequences should be evaluated using realistic methods.

SAR Sections 13.2.4.1 through 13.2.4.11 provide the LOCA shine analysis (i.e., the analysis of the doses that could result if the reactor pool drained, and the fuel and cobalt-60 sources in the pool became unshielded) for the UMLRR. The NRC staff requires additional information to fully understand certain details of the UMLRR LOCA shine analysis, and to determine that the assumptions made in the analysis are realistic and conservative.

As noted in RAI 11-2(c)(i), it is not clear whether the top of the reactor building is approximately 70 feet or approximately 80 feet above the grade/beam level. SAR Section 13.2.4.5 describes the geometry used to model indirect dose rates for the LOCA shine analysis, and states that the analysis assumes the height of the vertical walls of the reactor building is 69.3 feet, and the distance from the bottom of the pool to the curved dome ceiling is 80 feet. Clarify whether these geometry assumptions used in the LOCA shine analysis are consistent with the correct reactor building dimensions, or justify why no additional information is required.

**RAI-13.8**

The regulations in 10 CFR 50.34(b)(2) require that the SAR contain a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements; the bases, with technical justification therefor, upon which such requirements have been established; and the evaluations required to show that safety functions will be accomplished.

The guidance in NUREG-1537, Part 1, Chapter 13, describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include a loss of coolant flow, which is discussed in NUREG-1537, Part 1, Section 13.1.4. As discussed in NUREG-1537, Part 1, Section 13.1.4, initiators of a loss of coolant flow can include blocking or significant decrease in flow in one or more coolant channels.

SAR Section 13.2.3 provides a loss of coolant flow analysis which assumes that the loss of flow event results from a pump failure, loss of electrical power, or operator error. The analysis assumes that all forced flow to the core stops, and the reactor scrams. However, the analysis does not appear to consider an event such as a blockage (i.e., from a foreign object dropped into the reactor pool), that could cause coolant flow to drop or decrease in only some of the coolant channels. The NRC staff notes that in this scenario, the blocked coolant channels could, potentially, have a significant localized temperature increase before any reactor scram setpoint would be reached.

Provide an analysis for a loss of coolant flow resulting from a coolant channel blockage; demonstrate that the consequences of any coolant channel blockage would be bounded by those of the scenarios analyzed in SAR Section 13.2.3; describe how UMLRR procedures and/or design features would prevent channel blockage by a foreign object; or, justify why no additional information is required.