September 7, 2017

Mr. Ralph Butler, Executive Director
University of Missouri-Columbia
Research Reactor Center
1513 Research Park Drive
Columbia, MO 65211

SUBJECT: UNIVERSITY OF MISSOURI AT COLUMBIA - REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST TO IMPLEMENT SELECTIVE GAS EXTRACTION TARGET EXPERIMENTAL FACILITY AT THE UNIVERSITY OF MISSOURI RESEARCH REACTOR (CAC NO. MF9524)

Dear Mr. Butler:

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of your license amendment request (LAR) to produce molybdenum-99 using the General Atomics, Selective Gas Extraction process, provided by letter dated May 3, 2017 (redacted versions of the application are available on the NRC's public web site at www.nrc.gov under Agencywide Documents Access and Management System Accession Package No. ML17132A252).

The NRC staff has reviewed your proposed LAR and identified the items in the attached enclosure which need additional information or clarification. We request that you provide responses within 30 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 LAR.

If you need additional time to complete this request, or have any questions regarding this review, please contact me at (301) 415-0893, or by electronic mail at Geoffrey.Wertz@nrc.gov.

Sincerely,

/Michael Balazik Acting For RA/

Geoffrey A. Wertz, Project Manager
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-186
License No. R-103

Enclosure:
As stated

cc: See next page

cc:

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SUBJECT: UNIVERSITY OF MISSOURI AT COLUMBIA - REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST TO IMPLEMENT SELECTIVE GAS EXTRACTION TARGET EXPERIMENTAL FACILITY AT THE UNIVERSITY OF MISSOURI RESEARCH REACTOR (CAC NO. MF9524) DATED: SEPTEMBER 7, 2017

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REQUEST FOR ADDITIONAL INFORMATION

FOR THE LICENSE AMENDMENT REQUEST TO IMPLEMENT THE SELECTIVE GAS

EXTRACTION TARGET EXPERIMENTAL FACILITY AT

THE UNIVERSITY OF MISSOURI-COLUMBIA RESEARCH REACTOR

LICENSE NO. R-103; DOCKET NO. 50-186

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of the University of Missouri-Columbia Research Reactor (MURR) license amendment request (LAR) to conduct an experiment that would produce molybdenum-99 using the General Atomics, Selective Gas Extraction (SGE) process, provided by letter dated May 3, 2017 (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 (ADAMS) Accession Package No. ML17132A252). The LAR is Part 1 of 2, and consists of the changes needed to perform irradiation of the target material. The NRC staff has reviewed your proposed LAR, Part 1, and identified the items in the attached enclosure which need additional information or clarification. We request that you provide responses within 30 days from the date of this letter.

These request for additional information (RAIs) have been developed based on the following requirements and guidance applicable to your LAR:

- The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR).

- The regulations in 10 CFR Part 20, "Standards for Protection against Radiation," require that radiation doses to workers and members of the public be limited. To support meeting the public dose limits, 10 CFR Part 20, also limits the release of radioactive materials from the licensed facility to the environment (e.g., 10 CFR Part 20, Appendix B, Table 3).

- The regulations in 10 CFR 50.9, "Completeness and accuracy of information," require that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

- The regulations in 10 CFR 50.34, "Contents of applications; technical information," require information related to design bases and the principal design criteria of the instrumentation and control (I&C) systems. Furthermore, 10 CFR 50.34(b)(6)(v) requires research reactor licensees to develop and implement an emergency plan. Regulatory Guide (RG) 2.6, Revision 1, "Emergency Planning for Research and Test Reactors," issued March 1983 (ADAMS Accession No. ML003740234), American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.16-2015, "Emergency Planning for Research Reactors," issued February 11, 2015, and NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," issued October 1983 (ADAMS Accession No. ML062190191) provides guidance for the development of emergency plans.

- The regulations in 10 CFR 50.36, "Technical Specifications [TSs]," require each applicant to propose TSs. Additionally, 10 CFR 50.36(c) provides requirements to include safety limits, limiting safety system settings, limiting conditions for operation (LCO), surveillance requirements (SRs), design features, and administrative requirements. These TSs are derived from the analyses and evaluation included in the safety analysis report (SAR) and submitted pursuant to 10 CFR 50.34. Furthermore, ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors," as discussed in NUREG-1537 Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," Chapter 14, "Technical Specifications," provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate.

- The regulations in 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," require that the applicant submit an application fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

- The regulations in 10 CFR Part 55, "Operators' Licenses," provided requirements for the issuance of operator licenses for utilization facilities.

- The regulations in 10 CFR Part 73, "Physical Protection of Plants and Materials," require each licensee who possesses, uses or transports special nuclear material (SNM) of moderate or low strategic significance to establish and maintain a physical protection system. RG 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," provides guidance for the development of a physical security plan (PSP).

- 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).

- "Final Interim Staff Guidance [ISG] Augmenting NUREG-1537, Part 1, 'Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,' for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," dated October 17, 2012 (ADAMS Accession No. ML12156A069).

- "Final Interim Staff Guidance Augmenting NUREG-1537, Part 2, 'Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,' for Licensing Radioisotope Production

Facilities and Aqueous Homogeneous Reactors," dated October 17, 2012 (ADAMS Accession No. ML12156A075).

- Draft ISG augmenting Chapter 7 of NUREG-1537 Part 1 & Part 2, dated November 9, 2015 (ADAMS Accession Nos. ML15134A484 and ML15134A486, respectively). This draft chapter of NUREG-1537 provides revised guidance for preparing and reviewing applications for I&C for non-power production or utilization facilities. This guidance also expands the applicability of Chapter 7 for non-power reactors to all non-power production or utilization facilities, including medical isotope facilities, for licensing under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

LAR Attachment 1

1. LAR, Attachment 1, Section 1.1, "Proposed Experiment Description," states, in part, that "The SGE Target Experimental Facility (TEF) will be operated by MURR staff in concert with MURR's routine reactor operations." The LAR provides changes to the facility that include the addition of a target cooling system (TCS), and associated reactor trips and TSs. Additionally, LAR, Section 5.2.3, states, in part, that "Loading two TAs [target assemblies] in the MURR graphite reflector region causes perturbations in neutronic and thermal performance of the MURR core as follows: The MURR core excess reactivity will increase slightly, which will be compensated by the reactivity control devices." During its review, the NRC staff could not find any information in the LAR discussing training requirements associated with the TEF, or the operators' requalification training in response to a malfunction of an automatic control system affecting reactivity; and thus, it is not clear to the NRC staff how the operator training will satisfy the requirements associated with 10 CFR 55.45, "Operating tests," and 10 CFR 55.59, "Requalification," as described below.

   The regulations in 10 CFR 55.59(a)(2)(ii) require an operating test that ensures the operator or senior operator demonstrate an understanding of and the ability to perform the actions necessary to accomplish a comprehensive sample of items specified in 10 CFR 55.45(a)(2) through (13) inclusive to the extent applicable to the facility. The regulations in 10 CFR 55.45(a)(8) require the operating test to demonstrate that the operators can "[S]afely operate the facility's auxiliary and emergency systems, including operation of those controls associated with plant equipment that could affect reactivity or the release of radioactive materials to the environment."

   The regulations in 10 CFR 55.59(c)(3)(i)(W) require licensed operators to demonstrate that they can manipulate controls in response to a "[M]alfunction of an automatic control system that affects reactivity."

   Provide a description of the operator training applicable to the proposed changes implemented with this LAR, or justify why no additional information is needed.

2. LAR, Attachment 1, Section 1.2, "Normal Operation," states that, following irradiation, the reactor is shutdown and the TA is cooled by forced flow from the TCS for a specified time period to remove fission product decay heat, after which direct conduction with the pool

water is sufficient to maintain the temperature of the target rods in a safe condition. However, the NRC staff is not clear as to the resulting condition of the TA should the TCS fail, or fail to remain in service for the specified time period.

The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. The regulations in 10 CFR 50.90 require that the applicant submit an application fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. Furthermore, the guidance in NUREG-1537, Part 2, Chapter 13, “Accident Analysis,” indicates that the LAR should also describe how equipment will work when needed in potential accident situations.

Describe or reference the result of an analysis, which provides the results of the TA temperature during a loss of forced flow prior to the specified time period where direct conduction is sufficient to maintain the TA in a safe condition, or justify why no additional information is needed.

3. LAR, Attachment 1, Section 3.3.1, “Target Flow Assembly Design,” provides a description of the Target Flow Assembly Design, including the inlet piping, and states that there is a single inlet suction line pipe that is fitted with a replaceable screen to prevent any flow blockage caused by debris. The NRC staff review identified that additional information was needed to accurately model the target flow assembly.

   The regulations in 10 CFR 50.90 require that the applicant submit an application fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. Furthermore, the guidance in NUREG-1537, Part 2, Chapter 13, indicates that the LAR should also describe how equipment will work when needed in accident situations (e.g., loss of coolant flow). Provide the following information, or justify why no additional information is needed.

   3.1. A detailed description and drawings of the pipe inlet and screen.

   3.2. The dimensions of the smallest piece of debris that could cause a rapid reduction in flow worse than what is assumed in the RELAP accident analysis.

   3.3. The dimensions of the largest piece of debris that could flow through the inlet screen.

   3.4. A description if there is any protection from debris entering the target decay heat removal valves when they are open.

   3.5. Any design consideration given to minimize or prevent the introduction of air into TCS lines.

4. LAR, Attachment 1, Section 4.1, “Summary Description,” provides a description of the the TEF I&C systems which include the following seven subsystems: (a) TCS Control System; (b) TCS Protection System; (c) TCS Parameter Indication, Recording and Alarm system; (d) TCS Secondary Coolant Control System; (e) TCS Secondary Coolant Parameter Indication, Recording, and Alarm System; (f) N-16 Reactor Power Monitoring System; and (g) Pool Coolant Monitoring System. For each of these systems, the NRC staff review was

not able to completely identify sufficient design information to complete its review. Therefore, as described in RAIs 4.1 through 4.9 below, additional information is needed in order for the NRC staff to determine the adequacy of the design basis of the TEF I&C systems and subsystems.

The regulations in 10 CFR 50.90 require that the applicant submit an application fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. The regulations in 10 CFR 50.34 require information related to design bases and the principal design criteria of the I&C systems. NUREG-1537, Part 1, Chapter 7, "Instrumentation and Control Systems," and Section 7.1, "Summary Description," provides guidance that the applicant should describe the I&C systems of the reactor, including block, logic, and flow diagrams showing major components and subsystems, and connections among them. The draft ISG augmenting Chapter 7 to NUREG-1537, Part 1, Section 7.2.1, "Design Criteria," provides additional design criteria for I&C systems (specific draft ISG sections are provided in [brackets] for reference).

4.1. For each of the systems listed above, (a) through (g), provide an overall block logic, and process flow diagrams showing major components and subsystems, and connections for the I&C, or justify why no additional information is needed.

4.2. For each of the systems listed above, (a) through (g), provide more detail for the implementation of "Hardwired isolated interface with MURR SCRAM loop," (LAR, Attachment 1, Figure 30, page 54, "Target Cooling System Control System Architecture") used to describe the independent hard-wired safety function with regard to specific means of providing isolation or justify why no additional information is needed [draft ISG 7.1, 7.3, 7.4, 7.6, and 7.7].

4.3. The LRA indicated that numerous indication and alarm systems will be added for the TEF I&C systems, but does not state how the operator will manage the additional task load successfully. For each of the systems listed above (a) through (g), provide a summary of the human-machine interface principles used in the location of the associated control console and other status display instruments for the TEF I&C systems, as listed below [draft ISG 7.6].

4.3.1. Explain how control, safety, and associated TEF position indication systems and alarm and warning lights are displayed and readily accessible and understandable to the reactor operator or justify why no additional information is needed.

4.3.2. Explain the other controls and displays of important parameters that the operator should monitor to keep TEF parameters within a limiting value, and those that can affect the reactivity of the core, are readily accessible and understandable to the reactor operator or justify why no additional information is needed.

4.3.3. Explain how displays and controls provided for manual system-level actuation and control of safety equipment will be functional under conditions that may require manual actions or justify why no additional information is needed.

4.3.4. Provide diagrams that illustrate the various control panel layouts for controls and indicators, including labeling and color schemes (similar to Figure 31 for the TCS Control Panel). Additionally, provide a general depiction of where all of the system control and indicators are located in relation to the operator or justify why no additional information is needed.

4.4. The NRC staff review did not find a complete a description of the TEF I&C system instruments by type [e.g., hardwired analog, computerized digital that uses stored programs (software, which includes firmware), or combinations of these], and there was no discussion or identification of digital systems. The NRC staff review found that some systems are hardwired analog, but not all systems. Provide a description of the TEF I&C system instruments by type, or justify why no additional information is needed.

4.5. The NRC staff review did not find a discussion of any additional area and effluent radiation detection systems to be installed to facilitate the TFE system that monitor and alarm to provide indication of potentially hazardous radiation levels, including alarms or signals to other subsystems, as applicable. Provide information on any additional area and effluent radiation monitors, or justify why no additional information is needed.

4.6. The NRC staff review did not find a discussion of access control features (e.g., variable frequency drives with accessible control pads for local maintenance control shown in LAR, Attachment 1, page 54, Figure 30), to includes both preventing unauthorized access and allowing authorized access for both analog and digital systems (as applicable). Provide information on access controls such as alarms and locks on panel doors, or administrative control of access to rooms, and the protection of any networked systems (if applicable) or justify why no additional information is needed [draft ISG 7.2.5].

4.7. The NRC staff review finds that the keyed bypass switches to the TCS Protection System (LAR, Attachment 1, Section 4.3.3, “Target Cooling System Protection Bypass Capability”), allow for reactor operation with the TCS not operating, one TCS branch operating, or both TCS branches operating. The keyed bypass removes the protection signal inputs for any part of the TCS that is not operating. However, the following information was not provided. Provide the following information or justify why no additional information is needed:

4.7.1. The procedural administrative control of authorized use for the keys for the bypass switches (i.e., MURR standard operating procedure or TS).

4.7.2. The location of the bypass keys on the reactor console, and specifically, if any visual indication is provided for the bypass status of the individual green and yellow safety legs.

4.7.3. Electrical isolation of the relay inputs to the MURR Reactor Safety System (RSS) between the yellow and green safety legs and each leg and the MURR RSS.

4.7.4. LAR, Attachment 1, Section 4.3.3, discusses a failure or misuse of a single switch with two TAs operating, but does not fully discuss the single component failure criteria associated with the operating loop with the implications of the other loop being intentionally bypassed due to being non-operating. Discuss the single component failure criteria when only one TA loop is operational and the failure exists in that operating loop.

4.8. The NRC staff review did not find a complete description of the hardware detectors for the TCS Flow Elements, TCS Flow Transmitters, Pool Coolant Monitoring System detector, and the Target Cooling Automatic Temperature Control Valves (S-3A and S-38). Provide a description of the hardware detectors, or justify why no additional information is needed.

4.9. LAR, Attachment 1, Section 3.4.9, "Electrical Power System Supporting the TCS," indicates that the new electrical system and power provided from MCC-4 is safety-related. The NRC staff review did not find information related to whether an electrical breaker coordination study was performed to ensure that downstream breakers do not trip breakers located upstream. Provide an analysis to ensure that the selected breaker and cable for electrical equipment is properly sized (i.e., to ensure adequate voltage is available at the terminals of the equipment—e.g., cooling pumps), or justify why no additional information is needed.

5. LAR, Attachment 1, Section 8.3, "Solid Sources," provided general information on the SNM content for two target assemblies. However, the NRC staff was unable to find information relative to the total amount of SNM that would be used during the production process. As such, the NRC staff is unable to establish if the LAR would require any changes to the licensee's current license conditions, emergency plan, or PSP.

   The regulations in 10 CFR 50.34(b)(6)(v) require research reactor licensees to develop and implement an emergency plan. RG 2.6, ANSI/ANS-15.16, and NUREG-0849, provides guidance for the development of emergency plans.

   The regulations in 10 CFR Part 73 require each licensee who possesses, uses or transports SNM of moderate or low strategic significance to establish and maintain a physical protection system. RG 5.59 provides guidance for the development of the PSP.

   Provide information on the quantity of SNM which is planned to be processed in the TEF, and indicate if this quantity will result in any changes to the current licensee conditions for SNM possession, or the emergency and/or PSPs, as a result of this LAR, or justify why no additional information is needed.

6. LAR, Attachment 1, Section 9.3, "Material Control & Accounting," states, in part, that "Irradiated target rod material control will be discussed in the Part 2 License Amendment application." The LAR for Part 1 of the SGE experiment involves the necessary reactor and facility modifications, target fabrication, and all associated steps up to and including the irradiation of target rods. The LAR indicates that the remaining SGE process systems (i.e. the hot cells necessary to perform the target dissociation and the actual SGE process) and the associated safety analyses will be addressed in the future, when Part 2 of the SGE experiment is submitted for review and approval. The NRC staff notes that the possibility

exists that the LAR Part 1 could be approved and the LAR Part 2 could be denied. Approval of Part 1 without the approval of Part 2 would leave the applicant with the ability to irradiate the target rods with no clear pathway as to how those target rods would be handled once irradiated. As such, the NRC staff finds that there is no discussion of what would happen to the irradiated target rods if the future LAR for Part 2 of the SGE experiment is not approved. This information is needed in order for the NRC staff to determine that the irradiated target rods are handled safely.

The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 2, Chapter 11, "Radiation Protection Program and Radioactive and Waste Management," indicates that the LAR should provide information on radiation protection and waste management.

Provide a description of how all irradiated target rods will be handled in the instance that the Part 2 license amendment is denied. Also, provide all associated impacts (e.g., radiological, environmental, etc.) involved with the irradiated target rods in the instance that the Part 2 license amendment is denied, or justify why no additional information is needed.

7. LAR, Attachment 1, Section 10.1, "Target Experimental Facility Maximum Hypothetical Accident," provides a description of the maximum hypothetical accident (MHA) associated with the TEF (known as the TEF MHA versus the SAR MHA which results from the Failed Fueled Experiment accident). The NRC staff review of the TEF MHA has identified that the following items need additional information, for the NRC staff to ensure that radiation doses to the workers and public remain below the limit in 10 CFR Part 20.

   The regulations in 10 CFR Part 20 require that doses to workers and members of the public be limited. Furthermore, the guidance in NUREG-1537, Part 2, Chapter 13, states the applicant should discuss and analyze a postulated accident scenario whose potential consequences are shown to exceed and bound all credible accidents. For non-power reactors, this accident is called the MHA.

   Note: The methodology used to calculate the occupational and public doses in the LAR appears to include several assumptions and/or calculations that were previously used in the MHA analysis provided by the licensee in its responses to RAIs for the license renewal. In the NRC staff issued SER for the license renewal (ADAMS Accession No. ML16124A887), Section 13.1, "Maximum Hypothetical Accident—Failed Fueled Experiment," the NRC staff documented those assumptions and/or calculations that were determined to be inconsistent or non-conservative.

   7.1. LAR, Attachment 1, Section 10.1.1, "Description of Maximum Hypothetical Accident Scenario," identifies the MHA as an accident involving the release of radioactive material from one target rod due to assumed target rod cladding failure or end cap weld failure. The NRC staff review of the MHA observes that the postulated release is limited to the failure of only one (1) target rod as the credible accident. Based on the information provided in the application, it is not clear why the postulated accident does not involve more than a single target rod failure for the TEF MHA. Provide a justification for limiting the MHA postulated target rod failure to a single target rod, including supporting information as to what controls would be effective to prevent damage to multiple target rods in a postulated target rod assembly handling malfunction or common mode failures, or justify why no additional information is needed.

7.2. LAR, Attachment 1, Section 10.1.4, "Radionuclide Concentration in Reactor Pool Water," states that the failed target rod releases its fission product gases over a 5 minute time period, from a postulated pressure of approximately 80 pounds per square inch. Based on the information provided in the application, it is not clear to the NRC staff why it would take the fission gas 5 minutes to release to the pool water? Provide a justification for the 5 minute fission gas release time under the assumed accident conditions, or justify why no additional information is needed.

7.3. LAR, Attachment 1, Section 10.1.4, states that the target rod failure results in fission gas inventory mixed instantly and uniformly with the entire reactor pool water volume (20,000 gallons). The NRC staff is not clear if this assumption is conservative (i.e., over-predicts the dose calculations) since it dilutes the fission gas concentration to a minimum concentration. Provide a justification for instant mixing of the gas inventory with the entire pool water inventory that demonstrates its conservative nature, or justify why no additional information is needed.

7.4. LAR, Attachment 1, Section 10.1.5, "Radionuclide Concentration in Containment," provides calculations of radionuclides released from the reactor pool to the containment air, which assume that 0.1 percent (based on 20 gallons out of the 20,000 gallon reactor pool evaporating) of the iodines released to the pool are subsequently released to containment. However, the LAR, Attachment 1, Section 10.1.1, states that it is only the fission gas in the void volume of the target rod that is released to the target cooling water which is then discharged to the reactor pool water. Given that the iodines released to the pool are entirely in gaseous form (rather than solid form), the NRC staff note that a scenario in which the iodines become completely dissolved in the pool, and then are slowly released due to pool water evaporation, may not be realistic. Because the iodines released to the pool are entirely gaseous, greater than 0.1 percent of the iodines released from the target rod to the pool could potentially be released to the containment air via bubbles that rise to the pool surface. Justify the assumption that no iodine is released to the containment air by this mechanism; provide a revised calculation that considers additional iodine that could be released to containment by this mechanism; or, justify why no additional information is needed.

7.5. LAR, Attachment 1, Section 10.1.7, "Dose Consequences to Members of the Public," provides example calculations of Iodine-131 and Krypton-85 in containment which appear to assume that radionuclides released to the confinement air are mixed within a containment volume of 229,800 cubic feet. Current MURR TS 5.5.a, states, "The reactor and fuel storage facilities shall be enclosed in a containment building with a free volume of at least 225,000 cubic feet." The NRC staff notes that the assumption of a 229,800 cubic foot containment volume does not appear to be consistent with the TS 5.5.a value. Provide an explanation for the use of a containment volume that is greater (less conservative) than the TS value for the calculations in the LAR, Section 10.1.7; provide revised public dose calculations that utilize the TS value; or, justify why no additional information is needed.

7.6. LAR, Attachment 1, Section 10.1.7, discusses the method used to calculate the mixing and release of the radionuclides from the reactor laboratory building. The NRC staff review finds that the free air volume of the reactor laboratory building is not provided, and no discussion whether or not the reactor building ventilation system would provide mixing with the entire free air volume of the reactor laboratory building or only a portion of the free air volume. The NRC staff also needs clarification if the reactor laboratory building ventilation system would intake an air flow rate from outside the building, at the exhaust air flow rate of 30,500 cubic feet per minute (which would result in a "double" constant dilution of the laboratory building radionuclide concentration due to inflow of uncontaminated air and outflow of contaminated air). Provide the free volume of the reactor laboratory building used in the analysis, and a description of the mixing and release concentrations, or justify why this information is not needed.

7.7. LAR, Attachment 1, Section 10.1.7, states that most leakage pathways from containment discharge into the reactor laboratory building, which surrounds the containment structure. The NRC staff review questions if ignoring any direct leakage of radionuclides from the reactor containment building to the environment is a conservative assumption. Furthermore, the NRC staff finds that these direct leakage paths would result in essentially a ground level release to the environment without benefit of diluted dispersion through the laboratory building exhaust stack. Provide justification for ignoring possible direct leakage paths from the containment building to the environment, or justify why no additional information is needed.

7.8. LAR, Attachment 1, Section 10.1.7, states the assumed leakage rate out of containment results in a radiological release to the environment for 16.5 hours (elapsed time for the containment building pressure to reach equilibrium with the laboratory building pressure, effectively ending the radiological release to the environment). The NRC staff review notes that the licensee's calculation of post accident reactor building radionuclide concentrations is based on the evaporation from the reactor pool water that occurs in the first 5 minutes and ignores any additional pool evaporation that could occur during the post accident time period from 5 minutes to 16.5 hours. Provide justification for ignoring any additional radionuclide addition to the reactor building atmosphere due to pool water evaporation during the time period from 5 minutes to 16.5 hours.

7.9. LAR, Attachment 1, Section 10.1.8, "Dose Assessment in Unrestricted Area," provides public dose calculations that appear to be based on the assumption that the airborne effluent concentrations (AECs) for noble gases in Appendix B, Table 2, of 10 CFR Part 20, are based on a total effective dose equivalent (TEDE) of 50 millirem per year (mrem/yr). The NRC staff notes that the AECs for noble gases in Appendix B, Table 2, of 10 CFR Part 20, are based on a TEDE of 100 mrem/yr, and therefore dose calculations based on the assumption that the AECs for noble gases in Appendix B, Table 2, of 10 CFR Part 20, are based on a TEDE of 50 mrem/yr may not be conservative. Provide an explanation for the assumption, used in the calculations in the LAR, Section 10.1.8, that the AECs for noble gases in Appendix B, Table 2, of 10 CFR Part 20, are based on a TEDE of 50 mrem/yr; provide revised public dose calculations that use the 100 mrem/yr; or, justify why no additional information is needed.

7.10. LAR, Attachment 1, Section 10.1.8, provides public dose calculations that are based on an effluent dilution factor of 292, which assumes that the maximum public dose would occur for an individual 760 meters north of MURR (the location of the nearest residence to MURR) during Class F stability conditions. However, the basis for the use of this dilution factor is not clear to the NRC staff. The NRC staff notes that higher accident doses could potentially occur at locations other than the nearest residence, even if those locations are not continually occupied. Additionally, it is not clear to the NRC staff whether this dilution factor is based on worst-case meteorological conditions. Justify the use of an effluent dilution factor of 292 for determining the dose to the maximally-exposed member of the public during worst-case meteorological conditions, and provide a revised calculation using an alternate dilution factor that considers the worst-case level of dilution that could occur for any location and reasonable meteorological condition (include consideration of fumigation conditions and/or conditions that could result in plume downwash due to building wake, if appropriate).

8. LAR, Attachment 1, Section 10.5, “Pipe Break Locations Out of the Reactor Pool,” provides an analysis of a pipe break in the air region of the TA flow loop with opening of the decay heat removal valves. The NRC staff review of the analysis find that there appears to be no heatup of the target rods because there is a forward and reverse “chugging” flow between the pool and the TA. The NRC staff is not clear on details associated with the “chugging analysis.”

   The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.6, “Thermal-Hydraulic Design,” provides guidance that the LAR should include all necessary information on the primary coolant hydraulics and thermal conditions of the fuel, including design parameters and correlations, test results, for all modes of normal operation and accident scenarios.

   8.1. Provide an explanation of why the positive flow in the broken loop persists for more than 10 seconds and then reverses for tens of seconds (as counter current flow limitation could prevent coolant water from entering the exit flow restriction of the TA), or justify why no additional information is needed.

   8.2. Provide a model of the counter current flow limitation at the exit of the test assembly in the RELAP5 model of the test loop, if one was done, or justify why no additional information is needed.

9. LAR, Attachment 1, Section 10.5, states that, the operator is assumed to trip the TCS pump on the high flow alarm. The NRC staff is not clear the effect on the accident scenario that would result if the operator would fail to trip the TCS pump.

   The regulations in 10 CFR 50.90 require that the applicant submit an application fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. Furthermore, the guidance in NUREG-1537, Part 2, Chapter 13, indicates that the LAR should also describe how equipment will work when needed in accident situations.

Describe or reference the result of an analysis, which provides the results on the scenario should the operator action not occur as assumed, or justify why no additional information is needed.

10. LAR, Attachment 1, Section 10.8, “Mishandling of Target Cartridge or Target Rods,” provides a description of the mishandling of a target cartridge or target rods and states that based on additional decay prior to handling, the consequences of mishandling the target cartridge or target rods are bounded by the TEF MHA. However, the NRC staff review finds that a quantitative analysis of the potential doses from target mishandling accidents was not provided. Furthermore, LAR Section 10.8, states, in part, that “Multiple barriers separate the fission products in the target material from potential release locations within the TA.” The NRC staff can not identify the multiple barriers.

    The regulations in 10 CFR Part 20 require that radiation doses to workers and members of the public be limited. Furthermore, the guidance in NUREG-1537, Part 2, Chapter 13, indicates that the LAR should also describe the dose consequences of accidents.

    10.1. Provide a quantitative analysis to show that a target mishandling accident is bounded by the TEF MHA, or justify why no additional information is needed.

    10.2. Provide the multiple barriers to potential release locations with the TA, or justify why no additional information is needed.

11. LAR, Attachment 1, Section 12.1.2, “Critical Heat Flux Tests,” states, in part, that “[T]he NRC-accepted correlations show that the CHFR for the system will continue to provide sufficient margin”. The NRC staff notes that thermal limit calculations were performed with a variety of critical heat flux (CHF) correlations. However, the NRC staff is not aware of an NRC-accepted CHF correlation for the unique application provided by the LAR design.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.6, provides guidance that the LAR should include all necessary information on the primary coolant hydraulics and thermal conditions of the fuel, including design parameters and correlations, and test results, for all modes of normal operation and accident scenarios.

    Provide a description of the CHF correlation that was used, and provide the supporting information used to determine its acceptability for this application, or justify why no additional information is needed.

12. LAR, Attachment 1, Section 12.1.3, “System Integration and Cooling Flow Tests,” Table 57, indicates that the system integration and cooling flow test report would be completed by June 23, 2017. The NRC staff notes that, as of the issuance of this RAI, the test report has not been provided. Additionally, the NRC staff did not find any RELAP5 modeling results against data from the System Integration and Cooling Flow Tests section of the LAR.

    The regulations in 10 CFR 50.90 require that the applicant submit an application fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 5, "Reactor Coolant Systems," provides guidance that the LAR should include a functional analysis of the cooling system, including test results, for all modes of normal operation and accident scenarios.

12.1. Provide the system integration and cooling flow test report described in LAR Section 12.1.3, Table 57, or justify why the report is not needed.

12.2. Provide assessment of the RELAP5 modeling against data from the System Integration and Cooling Flow Tests described in Section 12.1.3, or justify why no additional information is needed.

<u>LAR Attachment 2</u>

13. LAR, Attachment 2, requested changes to the current MURR TSs. The NRC staff review finds that the proposed changes generally were not accompanied by a justification, basis or reference to the SAR.

    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, "Technical Specifications," provides guidance that the TSs should include a justification, basis and/or reference to the SAR.

    Provide a justification, basis, and/or SAR reference to each proposed TS, or justify why no additional information is needed.

14. Proposed TSs, Section 1.0, "Definitions," provided additional definitions to support the operation of the TEF. However, during its review, the NRC staff noted that the following items were not clearly understood:

    14.1. The numbering for the proposed TS Definitions, was not completely sequential, as definitions for 1.36 through 1.41 are repeated for different items.

    14.2. The TS definition for item 1.36, "Secured Experiment," was provided twice (i.e., repeated) within the definitions.

    14.3. The TS page numbers were not provided, as such, the NRC staff is not clear as to the completeness of the proposed TSs.

    14.4. The TS definition 1.39, "Selective Gas Extraction (SGE) Target Experimental Facility (TEF)" does not provide a description of the major components or systems which constitute the TEF.

    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance for definitions, and that the TSs should follow the format in ANSI/ANS-15.1-2007.

Provide TSs with complete and accurate format and definitions, or justify why no changes are needed.

15. Proposed TS 3.2, Specification g, added items 22 through 26 to the Reactor Safety System Instrument Channel table, which provided low flow, high flow, and high temperature setpoints of 95 gallons per minute (gpm), 129 gpm, and 105 degrees Fahrenheit (°F), respectively.

    15.1. The NRC staff noted that the setpoints were provided in the LAR, Attachment 1, Section 10.4. However, the NRC staff could not find a description or justification, relative to the safety analysis, for the determination of the setpoints (+ 15 percent) for the proposed TS 3.2, Specification g, items 22 through 26.

    Provide a justification for the setpoints in proposed TS 3.2, Specification g, items 22 through 26, or justify why no changes are needed.

    15.2. The NRC staff noted that footnote (9) states the SGE Heat Exchanger Outlet Water Temperature limit is not needed if the SGE TEF is "secured." Secured is not defined in the TSs, and the NRC staff does not understand the criteria needed to establish the secured condition of the TEF.

    Provide a definition or description to indicate the criterial associated with the term "secured" in TS 3.2, footnote (9), or justify why no change is needed.

The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36, and each of the TSs should be supported by the SAR.

Provide additional description or justification needed to assess the acceptability of the proposed TS 3.2, Specification g, items 22 through 26, and the footnote (9) for TEF "secured," or justify why no changes are needed.

16. Proposed TS 3.8, Specification n, describes "Amendment No. X." It is not clear to the NRC staff what "Amendment No. X" refers to in the application.

The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should describe facility-specific terms needed to clarify terms used in the TSs.

Remove, revise or describe "Amendment No. X."

17. Proposed TS 3.11, includes the Safety Limits and Limiting Safety System Settings, and other specifications. However, it is not clear to the NRC staff which specifications are specifically considered LCOs.

    The regulations in 10 CFR 50.36(c) provide requirements to include safety limits, limiting safety system settings, LCOs, SRs, design features, and administrative requirements. NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should include LCOs.

    Provide some indication to differentiate the LCOs and identify the corresponding SR in the TS for these LCOs, or justify why no change is needed.

18. Proposed TS 3.11, Specification b, Mode 1 Operation, SGE Heat Exchanger Outlet Water Temperature, provides a limit of 105 °F (Maximum). The NRC staff is unable to find the corresponding analysis which provides the basis for the temperature limit of 105 °F, which is needed to determine the adequacy of the basis for this safety limit.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should be supported by their respective safety analysis.

    Provide an analysis for the proposed TS 3.11, Specification b, temperature limit of 105 °F, or justify why no additional information is needed.

19. Proposed TS 3.11, Specification d, provides a limit for the target irradiation of 480 hours at 10 megawatts (MWs). The NRC staff is not clear if this limit could be interpreted as an integrated neutron fluence, i.e., 4,800 MW-hours, or simply 480 hours at full power operation. This information is needed to determine the adequacy of the basis for this safety limit.

    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff, and the public in accordance with 10 CFR 50.36, and each of the TSs should be supported by the SAR.

    Provide a clarification for the proposed TS 3.11, Specification b., limit of 480 hours at 10 MWs.

20. Proposed TS 3.11, Specification g, provides the required instrument channels for operation of the TEF. However, the SGE Heat Exchanger Outlet Water Temperature, provided as an LCO in proposed TS 3.11, Specification b, is not listed as a required instrument channel in proposed TS 3.11, Specification g. Clarification is needed for the NRC staff to determine the adequacy of proposed LCOs for required instrument channels.
    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless

acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36, and each of the TSs should be supported by the SAR.

Provide a TS requirement and associated SR for operability of the SGE Heat Exchanger Outlet Water Temperature instrument channel, or justify why no change is needed.

21. Proposed TS 4.11, Applicability, states “Select” versus “Selective,” which is used throughout the LAR. The inconsistent use of terminology, creates uncertainty of the use of facility-specific terms used in the TSs and LAR.

    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36, and each of the TSs should be supported by the SAR.

    Revise TS 4.11, Applicability, to use “Selective” consistent with the LAR, or justify why no change is needed.

22. Proposed TS 4.11, Specification c, states “tested for operability” which does not match the surveillance definitions in ANSI/ANS-15.1-2007 which include “channel check,” “channel test,” and “channel calibration.” The inconsistent use of terminology, creates uncertainty of the use of facility-specific terms used in the TSs and referenced guidance.

    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36, and each of the TSs should be supported by the SAR.

    Provide a description or definition to ensure consistent understanding of the term “tested for operability,” or revise to match the definitions in ANSI/ANS-15.1-2007, or justify why no change is needed.

23. Proposed TS 4.11, Specification e, states “checked for operability” which does not match the surveillance definitions in ANSI/ANS-15.1-2007 which include “channel check,” “channel test,” and “channel calibration.”
    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless

acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36, and each of the TSs should be supported by the SAR.

Provide a description or definition to ensure consistent understanding of the term "checked for operability," or revise to match the definitions in ANSI/ANS-15.1-2007, or justify why no change is needed.

24. Proposed TS 4.11, Specification f, provides a channel calibration for the instruments listed in TS 3.11, Specification g. However, it is not clear if an operability, or channel test, is required prior to startup. This information is needed to ensure the safe operation of the facility.

    The regulations in 10 CFR 50.36 require each applicant to propose TSs. Furthermore, ANSI/ANS-15.1-2007 provides guidance acceptable to the NRC staff, and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate. Furthermore, NUREG-1537, Part 2, Chapter 14, provides guidance that the TSs should state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36, and each of the TSs should be supported by the SAR.

    Provide a TS which requires a verification of channel operability or a channel test, prior to startup of the TEF, or justify why no change is needed.

<u>LAR Attachment 3</u>

25. LAR, Attachment 3, "Codes and Standards," states that the TEF was designed and fabricated in accordance with applicable codes and standards. However, the NRC staff review was unable to find the specific codes and standards to which the I&C system was designed and fabricated (only American Society of Mechanical Engineers standards are listed). This information is needed to determine the adequacy of the design basis for the I&C system.

    The regulations in 10 CFR 50.34 require information related to design bases and the principal design criteria of the I&C systems. NUREG-1537, Part 1, Chapter 7, Section 7.2.1, provides guidance that the applicant should discuss the criteria for developing the design bases for the I&C systems. The basis for evaluating the reliability and performance of the I&C systems should be included. All systems and components of the I&C systems should be designed, constructed, and tested to quality standards commensurate with the safety importance of the functions to be performed. Where generally recognized applicable codes and standards are used, they should be named and evaluated for applicability, adequacy, and sufficiency. The draft ISG augmenting Chapter 7 of NUREG-1537, Part 1 & Part 2, provides revised guidance for preparing and reviewing applications for I&C systems for non-power production or utilization facilities.

Furthermore, the draft ISG guidance states that design criteria should be specified for each structure, system, or component that is assumed in the safety analysis to perform an operational or safety function, and design criteria should include references to applicable up-to-date standards, guides, and codes. Specific references to the draft ISG are provided in [brackets].

Provide the following information on design criteria for the TEF I&C systems [draft ISG 7.2.1], or justify why no additional information is needed:

25.1. Design for the complete range of operating conditions for both system and facility (environmental) parameters to cope with anticipated transients and potential accidents,

25.2. Design <u>redundancy</u> to protect against unsafe conditions in case of single failures of the TEF or reactor protective and safety systems,

25.3. Design <u>independence</u> to protect against TEF system failures or malfunctions preventing the reactor protective system from performing its safety function or preventing safe reactor shutdown,

25.4. Design <u>fail-safe</u> when required by the safety analysis, to assume a safe state on loss of electrical power or has a reliable source of emergency power sufficient to sustain the operation of those specific devices credited in safety analysis,

25.5. Design to facilitate inspection, testing, and maintenance (e.g., readily testable and capable of being accurately calibrated), including justification for test intervals for the surveillance testing proposed as part of the facility TSs,

25.6. Design surveillance test and self-test features for a digital computer-based protection systems, where applicable, to address failure detection, self-test features (e.g., monitoring memory and memory reference integrity, using watchdog timers or processors, monitoring communication channels, monitoring central processing unit status, and checking data integrity), and TEF system actions taken upon failure detection,

25.7. Design to limit the likelihood and consequences of fires, explosions, and other potential manmade conditions. (This can include mitigating features of other facility or system design features),

25.8. Design quality standards commensurate with the safety function and potential risks, and

25.9. Analysis of function, reliability, and maintainability of systems and components.

<u>LAR Attachment 6</u>

26. LAR, Attachment 6, Section 2.3.1.2, “Driver fuel element loading pattern,” the NRC staff notes that a reference is made to the “extreme burnup core” having the “worst power peaking in both the driver and target assemblies” (page 10). The elemental and core

average burnups for all four Xenon/burnup configurations is presented Table 2-1. The peaking factors for the extreme burnup and maximum burnup cores are presented in Tables A-1 through A-16 for all eight fuel elements. The NRC staff is not clear if peaking factors were calculated for the other two burnup/Xenon combinations demonstrated in Table 2-1 (e.g., the Minimum and Average burnup cores), and if so, what were the calculated peaking factors for these other two combinations. Also, nuclide compositions for all eight-fuel elements for the xenon/burnup configurations referenced in Table 2-1 were not provided.

The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, "Nuclear Design," provides guidance that applications should include all necessary information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup, as discussed in the safety analysis.

26.1. Provide the peaking factors for the Minimum and Average burnup cores, if calculated, provide which peaking factors were used, or justify why no additional information is needed.

26.2. Provide the nuclide compositions of all eight fuel elements for the xenon/burnup configurations referenced in Table 2-1, or justify why no additional information is needed.

27. LAR, Attachment 6, Section 5.1, "Validation for Computational Methods and Calculations at MURR," provides reference to a library of beryllium reflectors at various stages of depletion that were created with MONTEBURNS (page 73). The NRC staff is not clear as to the burnup methodology for the beryllium structure (i.e., are these libraries distinguished from each other solely by changes in composition or density?). Additionally, reference is made to the central flux trap composition, but no other details are provided (i.e., what are the compositions of these targets at the most limiting core configuration?).

The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, provides guidance that the LAR should include all necessary information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup, as discussed in the safety analysis.

27.1. Provide the burnup methodology and the burnup libraries, or justify why no additional information is needed.

27.2. Provide the compositions of the targets at the most limiting core configuration, or justify why no additional information is needed.

28. LAR, Attachment 6, Section 5.2, "Validation of MURR Control Blade Depletion Model," Figures 5-6 and 5-7, list the Boron-10 (neutron, alpha) reaction rate profiles for the control

blades, and Table 5-4 provides a demonstration model that the control blade burnup has a significant effect on the deviation of the k-effective (page 84). However, the NRC staff notes that the control blade composition at the most limiting core configuration is not provided.

The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, provides guidance that the LAR should include all necessary information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup, as discussed in the safety analysis.

Provide the control blade composition at the most limiting core configuration, or justify why no additional information is needed.

<u>LAR Attachment 7</u>

29. LAR, Attachment 7, Section 4.2, “FRAPCON Generic Code Assumptions,” provides assumptions for use of the FRAPCON computer code. However, the NRC staff review noted that the use of FRAPCON’s thermal model with the stated Linear Heat Generation Rate (LHGR) of 54.2 kilowatts per meter (kW/m) (Attachment 6, page 59), specifically with respect to fuel centerline measurements at these temperatures, is higher than any of the documented thermal integral assessment cases in the FRAPCON-4.0: Integral Assessment (PNNL-19418, Vol.2 Rev.2). Additionally, the NRC staff noted that FRAPCON’s radial power distribution model (tubrnp) used an enrichment of 19.75 percent uranium (U)-235; however, FRAPCON is validated for light water reactor fuel with a maximum enrichment of 5 percent U-235. The NRC staff needs additional clarification on the use of FRAPCON given these values used.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, provides guidance that the LAR should include all necessary information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup, as discussed in the safety analysis.

    29.1. Provide data to support the use of FRAPCON’s thermal model with the stated LHGR of 54.2 kW/m, specifically with respect to fuel centerline measurements at these temperatures, or justify why no additional information is needed.

    29.2. Provide an explanation for using FRAPCON’s radial power distribution model (tubrnp) with an enrichment of 19.75 percent U-235, or justify why no additional information is needed.

30. LAR, Attachment 7, Section 4.3, “FRAPTRAN Generic Code Assumptions,” provides assumptions for use of the FRAPTRAN computer code. However, the NRC staff noted that the use of a temperature limit of 2,860 degrees Celsius (°C) (Attachment 1, page 14) for

transients for uranium dioxide (UO_2) appears to exceed the melting temperature for fresh fuel in FRAPTRAN.

The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, provides guidance that the LAR should include all necessary information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup, as discussed in the safety analysis.

Provide data to support the temperature limit of 2,860 °C for transients for UO_2, or justify why no additional information is needed.

31. LAR, Attachment 7, Section 5.1, Table 2, provides a value of 0.055 centimeters (cm) for the dish shoulder. The NRC staff noted that Attachment 6, Section 3.1.1, Table 3-1 provides a value of 0.031 cm.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, provides guidance that the LAR should include all necessary information on the nuclear parameters and characteristics of the reactor core.

    Provide a clarification as to the dish shoulder value, or justify why no additional information is needed.

32. LAR, Attachment 7, Section 6.3, "Loss of Target Coolant," provides analyses, including the break location and size, for two loss-of-coolant accident (LOCA) scenarios: 1) in-pool; and 2) out-of-pool. The NRC staff needs a more comprehensive understanding of the LOCA analyses, including if other break/size locations were considered, rupture times, or break sizes less than a full double ended break at the limiting in-pool location.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.6, provides guidance that the LAR should include all necessary information on the primary coolant hydraulics and thermal conditions of the fuel, including design parameters and correlations, and test results, for all modes of normal operation and accident scenarios.

    32.1. Indicate if other LOCA analyses were performed (i.e., other break/size locations were considered, rupture times, or break sizes less than a full double ended break at the limiting in-pool location) and if so, provide the results of for these analyses, or justify why no additional information is needed.

    32.2. The limiting LOCA scenario provided is a full double ended break at the inlet to the target fuel assembly. The calculation assumes a rupture time of 0.5 seconds. The NRC staff needs additional information to evaluate the system response. Provide sensitivity calculations showing the system response with assumed rupture times of 0.25, 0.1, and 0.01 seconds, or justify why no additional information is needed.

32.3. Provide a description or analysis of any review of potential break sizes less than a full double ended break to establish the bounding break condition, or justify why no additional information is needed.

33. LAR, Attachment 7, Section 6.4, "Loss of Target Flow," provides a description of the loss of target flow analysis, which includes consideration for a loss of pump flow from a loss of electrical power to a pump. The NRC staff notes the possibility that a more severe reduction in coolant flow could occur from debris blocking the suction inlet screen of a locked pump rotor, and did not find any description or discussion of this possibility.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.6, provides guidance that the LAR should include all necessary information on the primary coolant hydraulics and thermal conditions of the fuel, including design parameters and correlations, and test results, for all modes of normal operation and accident scenarios.

    Provide sensitivity analyses of rapid reduction in flows that occur on a timescale significantly shorter that what was analyzed (e.g., debris blocking the suction inlet screen of a locked pump rotor), or justify why no additional information is needed.

34. LAR, Attachment 7, provides assumptions for the use of the FRAPCON and FRAPTRAP computer codes. The NRC staff reviewed the assumptions and identified the following items needing additional information: gas composition (see Attachment 1, Section 2.2.3); uncertainties used in the analyses; and power curve data. This information is needed for the NRC staff to complete its independent modeling and verification.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, provides guidance that the LAR should include all necessary information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup, as discussed in the safety analysis.

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

    34.1. Gas composition (Attachment 1, Section 2.2.3, page 30, states, in part, that "[T]he target rod is filled with > 95% Helium (He), at ≈1 atmosphere"). If it is not 100 percent, please state the remainder of the gas composition assumed in the analysis.

    34.2. Please provide a description of the model uncertainties used in the analysis.

    34.3. Please provide the following FRAPTRAN power curves in tabular form (LHGR vs. time):

34.3.1. The 600 percent mille step reactivity insertion (Attachment 1, Section 10.2.1, page 156).

34.3.2. The control blade withdrawal (Attachment 1, Section 10.3, Figure 79).

35. LAR, Attachment 7, provides the results of various accident scenarios. The NRC staff is not clear as to why some of the safety analyses assumed 115 percent power on the first and last day, with 100 percent power on the remaining days (such as the step reactivity insertion) while others assumed 100 percent power on all days analyzed.

    The regulations in 10 CFR 50.9 require that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Furthermore, the guidance in NUREG-1537, Part 1, Chapter 4, Section 4.5, provides guidance that the LAR should include all necessary information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup, as discussed in the safety analysis.

    Provide an explanation why the safety analyses assumed 115 percent power on the first and last day, with 100 percent power on the remaining days, versus other analyses which assumed 100 percent power on all days analyzed, or justify why no additional information is needed.

Physical Security Plan Considerations

36. The NRC staff review finds that the current PSP does not discuss the type of SNM, nor the additional amount of SNM, that will be stored and used as a result of this amendment. The amendment seems to indicate that the category of SNM will be low strategic significance (10 CFR 73.2 Category III). Therefore additional information is needed for the NRC staff to determine whether MURR's PSP properly accounts for the amount of SNM stored and used in connection with this amendment request.

    The regulations in 10 CFR 73.2 define SNM of low strategic significance as less than 10,000 grams but more than 1,000 grams of U-235 (contained in uranium enriched to 10 percent or more but less than 20 percent in the U-235 isotope).

    Revise this PSP to describe the added SNM, its type, and the amount, or justify why no change is needed.

37. The NRC staff review finds that the current PSP does not discuss the location of use or storage of the SNM used for the target assemblies described in the LAR.

    The regulations in 10 CFR 73.2 define SNM of low strategic significance as less than 10,000 grams but more than 1,000 grams of uranium-235 (contained in uranium enriched to 10 percent or more but less than 20 percent in the U–235 isotope). The regulations in 10 CFR 73.67, "Licensee fixed site and in-transit requirements for the physical protection of special nuclear material of moderate and low strategic significance," requires licensees to store or use SNM within a controlled access area (CAA).

Verify the added SNM is Category III, and clarify its use and storage location. If there will be a new CAA revise the plan to include a description of the location of the SNM while in storage and in use (add locations to the maps as well). Describe the added physical barriers, if any, the means and criteria for controlling access, and the access control points of the CAA, or justify why no additional information is needed.

38. The NRC staff review is not clear if any licensed material will be stored within a new CAA not already described in the PSP. If so, discuss the means for the monitoring the CAA and responding to indications of intrusion into the CAA.

    The regulations in 10 CFR 73.67 require the CAA to be monitored by an intrusion alarm, device, or procedures to detect unauthorized access.

    Revise the PSP to include a description of the means to monitor for unauthorized access, and procedures for responding to an alarm, or justify why no additional information is needed.