April 17, 2015

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

SUBJECT: UNIVERSITY OF MISSOURI AT COLUMBIA - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE RENEWAL OF FACILITY OPERATING LICENSE NO. R-103 FOR THE UNIVERSITY OF MISSOURI AT COLUMBIA RESEARCH REACTOR (TAC NO. ME1580)

Dear Mr. Butler:

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of your application for the renewal of Facility Operating License No. R-103, dated August 31, 2006 (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 Nos. ML062540114 - cover letter; ML092110573 - Safety Analysis Report, Chapters 1-9; ML092110597 - Safety Analysis Report, Chapters 10-18), as supplemented, for the University of Missouri - Columbia Research Reactor. During our review, questions have arisen for which additional information is needed. The enclosed request for additional information (RAI) identifies the additional information needed to complete our review. We request that you provide responses to the enclosed RAI within 45 days from the date of this letter. If you need additional time to respond, please contact me.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.30(b), you must execute your response in a signed original document under oath or affirmation. Your response must be submitted in accordance with 10 CFR 50.4, "Written communications." Information included in your response that is considered sensitive or proprietary, that you seek to have withheld from the public, must be marked in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Any information related to security should be submitted in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." Following receipt of the additional information, we will continue our evaluation of your renewal request.

If you have any questions regarding this review, please contact me at (301) 415-0893.

Sincerely,

/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

Enclosure:
Request for Additional Information

cc: See next page

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Test, Research, and Training
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202 Nuclear Sciences Center
Gainesville, FL 32611

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REQUEST FOR ADDITIONAL INFORMATION

FOR THE RENEWED LICENSE FOR

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 your application for the renewal of Facility Operating License No. R-103, dated August 31, 2006 (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 Nos.: ML062540114 - cover letter; ML092110573 – Safety Analysis Report (SAR), Chapters 1-9; ML092110597 - SAR, Chapters 10-18), as supplemented, for the University of Missouri - Columbia Research Reactor (MURR). 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 complete our review. We request that you provide responses to the enclosed RAI within 45 days from the date of this letter.

1. In the MURR SAR, Sections 1.4.2, 4.2.2.4, and 4.5.3, the control blade drop time is expressed as "insertion to 20% of the withdrawn position in less than 0.7 seconds." SAR Section 3.5.2 describes the control blade drop process including the effect of the dashpot, but does not describe the method for determining the drop time nor does it explain the basis for the 80 percent insertion times. The scram times and reactivity worths used or assumed for the various analyses in the SAR are not clearly described or provided. NUREG-1537, Section 4.5.3, "Operating Limits," provides guidance that the analysis for the shutdown reactivity for all operational conditions should be described.

   a. Explain the MURR process for determining the control blade insertion times and the associated control blade insertion reactivity per blade. Provide typical control blade full insertion scram times and reactivities, or justify why no additional information is needed.

   b. Explain which analyses documented in the SAR utilize the assumptions described in Item a. above regarding control blade insertions, withdrawals, and scrams (e.g., blade withdrawal from subcritical, control blade run in, insertion of excess reactivity, etc.). For each such event, provide the control blade motion speeds and reactivities utilized to provide the SAR analyses, or justify why no additional information is needed.

2. NUREG-1537, Section 9.2, "Handling and Storage of Reactor Fuel", provides guidance that the licensee provide analyses and methods to demonstrate the secure storage of new and irradiated fuel with a criticality limit of keff $\leq$ 0.90. The NRC staff's review of the MURR SAR and Hazards Summary Report could not find a criticality analysis supporting the use of any fuel storage locations outside of the core. Identify the locations that may be used for the storage of new or irradiate fuel, and provide supporting criticality analyses, or justify why no additional information is needed.

Enclosure

3. NUREG-1537, Section 4.5.1, “Normal Operating Conditions,” and Section 4.5.2, “Reactor Core Physics Parameters,” provide guidance that the licensee should identify their analytical methods, including calculations of individual control blade worths, core excess reactivity, and coefficients of reactivity, and compare the results with experimental measurements. The MURR SAR, Section 4.5 states that analyses have been performed using PDQ, EXTREMINATOR, and BOLD-VENTURE codes using Rθ, RZ, and RθZ models. The NRC staff noted other analyses (e.g., the RAI responses supporting the NRC staff review of License Amendment No. 36, ADAMS Accession Nos. ML11237A088 and ML12150A052) used Estimated Critical Position (ECP) comparisons with the Monte Carlo Neutron Production code. The design code used to support the T&H analysis appears to be DIF3D. The NRC staff is not clear as to which analytical method is the final supporting analysis to be reviewed for the MURR license renewal application. The final supporting analysis should be the source for information used in accident and event analysis (e.g., peaking factors, control blade worths). Furthermore, in response to RAI 4-14.c., (ADAMS Accession No. ML103060021), it is not clear how the stuck control blade was determined, what the relative reactivity worth is for the other control blades in the shutdown margin (SDM) analysis, and whether they are calculated, measured, or compared. The following information is needed:

   a. Identify the neutronics code used as the basis for the MURR License Renewal Application, or justify why this information is not needed.

   b. Using results from that code provide the results of calculations and comparisons of the corresponding measurements for the ECP (or excess reactivity) for a known critical control blade configuration at zero power, no xenon condition, or justify why this information is not needed.

   c. Provide calculated and measured control blade worths (Shim-1, Shim-2, Shim-3, Shim-4, and Regulating blades) for a given core configuration at a low power, no xenon condition, or justify why this information is not needed.

   d. Provide a calculated and measured temperature coefficient for a given core configuration at a low power, no xenon condition, or justify why this information is not needed.

4. NUREG-1537, Section 4.5.3, “Operating Limits,” provides guidance that licensees demonstrate that their facility has sufficient control blade worth to achieve the required shutdown reactivity assuming that all scrammable control blades are released upon scram, but the most reactive blade remains in its most reactive position. The NRC staff could not find this information in the MURR SAR, but noted a reference in the 1971 Low Power Testing Program that indicated that the shutdown margin control blade reactivity was determined using 66 percent of the 4 shim blade insertion worth. Explain how MURR ensures adequate SDM, whether and if so, how the 66 percent factor from the 1971 Low Power Testing Program is used, or justify why this information is not needed.

5. NUREG-1537, Section 11.1.1.1, “Airborne Radiation Sources,” provides guidance for the licensee to characterize the dose for the maximally exposed individual, at the location of

the nearest permanent residence, and at any locations of special interest in the unrestricted area.

a. The MURR SAR, Appendix B, contains summary information regarding the radiological impacts of the MURR generated release of Argon 41 (Ar-41) during normal operations. The MURR methodology includes an equation on SAR page B-10 that is used to alter the effective stack height used in the dose calculations to compensate for elevation changes of the receptor due to the local topography. Although unreferenced in the SAR, the NRC staff reviewed "Plume Rise" by Briggs (TID-25075) and it seems that this equation is based on the Davidson empirical model which has limited supporting data. Describe how the effective stack height calculations are performed for the unique topography surrounding MURR, and how the results are sufficiently conservative for the estimation of dose, or justify why no additional information is needed.

b. SAR page B-11 has an equation for X/Q that includes the σ_y and σ_z dispersion factors. The NRC staff was unable to validate some of the dispersion values used in Tables B-2 and B-3. Explain how these values were determined or justify why no additional information is needed.

6. NUREG-1537, Section 13, provides guidance that the applicant should demonstrate that the facility design features, safety limits, limiting safety system settings, and limiting conditions for operation have been selected to ensure that no credible accident could lead to unacceptable radiological consequences to people or the environment. The NRC staff review examined the analyses provided in the MURR SAR, Chapter 13, including the assumptions regarding the initial conditions (e.g., reactor power, reactivity insertion, etc.), analytical input (e.g., peaking factors and decay times), and results. The following information is needed:

a. Regarding Insertion of Excess Reactivity - The initial power is 10 MW rather than the Limiting Safety System Setting setpoint in TS 2.2 (12.5 MW). The temperature feedback coefficient used is -7.0×10^{-5} $\Delta k/k$ rather than the TS 5.3.a value of -6×10^{-5} $\Delta k/k$. It is unclear what peaking factors are employed. SAR Figure 13.2 seems to indicate that the scram time used is faster than the value in TS 3.2.c. The acceptability of the results is based upon whether the power for burnout is achieved rather than the safety limit identified in TS 2.1. Provide additional information justifying and supporting the analysis and the safety conclusions or provide a justification for why such information is not required.

b. Regarding Loss of Primary Coolant and Loss of Primary Coolant Flow - The initial power is 11 MW rather than the LSSS setpoint in TS 2.2 (12.5 MW). It is unclear what peaking factors are employed. The acceptability of the results is based upon the peak fuel temperature attained rather than the safety limit identified in TS 2.1. Provide additional information justifying and supporting the analysis and the safety conclusions or provide a justification for why such information is not required.

c. Regarding the maximum hypothetical accident (MHA) and Failed Fueled Experiment - these events use a 10 minute and 2 minute evacuation time

respectively. Provide additional information identifying the limiting evacuation time and then use that time to justify and support the analysis and the safety conclusions or provide a justification for why such information is not required.

7. NUREG-1537, Section 13.1.1, “Maximum Hypothetical Accident,” provides guidance for the licensee to postulate a failed fuel element scenario and analyze the consequences. The MURR SAR, Section 13.2.1.2, provides the analysis and related consequences for a fuel failure involving the melting of four number 1 fuel plates in a core region where the power is at a maximum. The fuel fails submerged and it is assumed that all iodine, krypton, and xenon isotopes are released into the primary coolant system (PCS) while in Modes I or II (PCS closed).

    a. The iodine and noble gases core inventories are based on a 1200 MWD burnup consisting of twelve 10-day cycles over a 300-day period. These values were then adjusted using a peaking factor of 1.6. However, in the response to RAI A.27 (ADAMS Accession No. ML120050315), a peaking factor of 3.0 has been used. In the MURR SAR, Section 4.5, the peaking factor is listed as 3.676. Clarify the discrepancies in the peaking factors used, and provide a revised calculation of the source using the peaking factors determined from the final analysis, or justify why no additional information is needed.

    b. The release is assumed to occur into the PCS with a volume of 2,000 gallons. Identify what components comprise this volume and provide information to confirm the 2,000 gallon volume assumption, or justify why no additional information is needed.

    c. The release is assumed to remain in the PCS except for the amount that will enter the pool cooling system as part of the PCS to pool cooling system leakage. Therefore, the concentration of iodine that is released first enters the pool cooling system and is diluted once again. This seems to reduce the consequences of this accident to a fraction of the consequences of the failed fueled experiment as provided in your response to RAI 13.9 (ADAMS Accession No. ML103060018). As such, this event (four failed fuel plates) may not be the MHA. Provide a confirmation of the dilution assumptions stated above and clarification as to the MHA for MURR.

    d. The released concentrations in the containment are based on the 10-minute leakage between the PCS and the pool cooling system. However, the NRC staff questions whether the release into the PCS will collect in the vent tanks and other places in the PCS and eventually be released to the environment after decay. Provide an explanation for this leakage path, including assumptions and calculations of the possibility of the isotopic concentrations being released to the environment, or justify why no additional information is needed.

    e. In determining the offsite doses in the unrestricted areas from the releases, the concentrations of the released isotopes are calculated using a method described in the MURR SAR, Appendix B, which used a simplified joint frequency distribution of weather data that was prepared in the 1960s. Given the changes in weather conditions over the last 50 years, it is not clear to the NRC staff

whether the listed probabilities and wind speeds for the stability classes are still applicable. Provide available current weather data, and state whether changes warrant reconsideration of the cited data, or justify why no additional information is needed.

f. It is not clear to the NRC staff which dispersion factors were used to arrive at the listed concentrations in the cited unrestricted location, which is also not specified. The calculation of the ratio of the average concentration in the unrestricted location to the corresponding concentration in containment results in the reduction factor for iodine twice as large as the value for the noble gases. For example, for Krypton-85 the ratio is $7.5\times10^{-14}/3.0\times10^{-8}$ or a reduction of about 4.0×10^{5}. For I-131, the ratio is $1.36\times10^{-14}/1.1\times10^{-8}$ or a reduction of about 8.1×10^{4}. Provide an explanation of all assumptions relating to the calculation of average isotope concentrations, specify all locations where these concentrations are determined, and explain how dispersion factors are determined and used, or justify why no additional information is needed.

g. In determining occupational doses, it appears that the MURR SAR calculations use a combination of dose conversion factors (DCFs). It appears that for radioiodine, the calculation uses DCFs from Federal Guidance Report (FGR) No. 11 for inhalation pathway (thyroid) and FGR No. 12 for submersion dose (external-deep-dose), whereas for submersion doses from noble gases, it uses the derived air concentrations from 10 CFR Part 20, Appendix B, Table 1. FGR 12 revises the dose coefficients for air submersion used in FGR 11. Those DAC values are based on International Commission on Radiation Protection (ICRP)-2 DCFs, whereas the FGR 11 values are based on ICRP-38. In addition, neither FGR 11 nor FGR 12 lists DCFs for isotopes with very short-half lives. In 10 CFR Part 20, Appendix B Table 1, the regulation provides a DAC value of 1×10^{-7} micro-Ci/ml for those isotopes with a half-life of less than 2 hours. Overall, the differences in the calculated DCFs result in high values of calculated doses from noble gas isotopes with a very short half-life. Provide dose calculations using uniform data and methodology.

8. NUREG-1537, Section 13.1.3, “Loss of Coolant,” provides guidance to the licensee to consider the consequences of a loss of coolant accident (LOCA). MURR SAR Section 13.2.3.2 describes the LOCA event for the loss of the PCS integrity, and states that the accident of greatest consequence is a rupture in the short section of the PCS piping (either the cold leg or the hot leg) between the reactor pool and either isolation valves (507B or 507A). The SAR describes the consequences of a cold leg break between the isolation valve 507B and the reactor pool in significant detail. The hot leg break discussion is more succinct. The SAR also states that how “the anti-siphon system ensures that the core remains covered differs depending on the location of the rupture.”

   The NRC staff reviewed the event as described in the SAR and is considering the hot leg break sequence. It is our understanding that after isolation occurs the coolant surrounding the core heats up, and because of natural buoyancy it flows upward and out of the reactor pressure vessel into the in-pool heat exchanger. After passing through the heat exchanger, the cooled water may then flow downward through what is normally the upward flow path of the inverted loop and then into the bottom of the pressure vessel.

As this process continues, the water will fill up the downward inverted loop to the bottom of the core reaching to the inverted loop creating an open condition for releasing the PCS coolant through the broken hot leg pipe. Explain the credibility of this event, and, if credible, provide a supporting analysis demonstrating acceptable core cooling and peak fuel temperatures, or justify why no additional information is needed.

9. NUREG-1537, Section 13.1.5, "Mishandling or Malfunction of Fuel" provides guidance that the licensee analyze the consequences of a mishandled fuel event. MURR SAR Section 13.2.5.2.1 describes damage to a fuel element due to mishandling. It states that the mishandling could occur during movement and packaging of the irradiated fuel, damage could only occur to the inner or the outer fuel plate, and could only occur during fuel element relocation activities. Because this accident occurs while the PCS is open there is minimal containment of fission products by the PCS. The response to RAI A.27 (ADAMS Accession No. ML120050315), provides an analysis of such an occurrence assuming that the fuel element has decayed for 60 days as part of the spent fuel movement from storage to a shipping container. However, the NRC staff questions whether this event could also occur during the initial stages of refueling which would invalidate the assumption of 60 days of decay. The NRC staff also performed a confirmatory calculation based on this inventory using the cited values for the MHA analysis, and it results in an inventory that is seven percent larger than reported by MURR.

    a. Explain the possibility of this event occurring during the initial stages of refueling, and the applicability of using the stated decay time in the dose calculation. Also, describe any radioactivity release alarms that are expected to actuate, and whether containment isolation is expected, including the time required to verify containment isolation, or justify why no additional information is needed.

    b. Provide the details of how the source term is determined, or justify why no additional information is needed.

10. NUREG-1537, Section 13.1.6, "Experiment Malfunction" provides guidance that the licensees analyze the consequences of a failed fueled experiment. SAR Section 13.2.6.2 describes that limiting fueled experiments to 150 curies of radioiodine will result in a projected dose well within the limits of 10 CFR Part 20. The response to RAI 13.9.a (ADAMS Accession No. ML103060018) provides radioiodine and noble gas activities for a 5-gram low-enriched fuel target. The response uses a method similar to that used in the MHA analysis and lists the gaseous fission products to be released into the pool cooling system. The occupational dose calculation assumes a 2-minute evacuation time. The NRC staff notes that the submersion dose calculations were performed using the DAC values, but the DAC data for isotopes with half-lives of less than 2 hours that are not listed in Table 1 of Appendix B are not consistent with the recommended value of 1×10^{-7} μCi/ml. The NRC staff notes that the 2-minute evacuation time is not consistent with the 10-minute evacuation time assumed in the MHA analysis, or the SAR Section 13.2.1.2 statement that it takes the operations staff approximately 5 minutes to secure the PCS and verify containment isolation following a containment isolation signal.

    a. Please clarify the sequence of events, state which alarms are expected to provide indication that evacuation is required, justify the evacuation time, and use

that time to revise the dose assessment employing consistent DAC values, or justify why no additional information is needed.

b. SAR Section 13.2.6.2 states that “Fueled experiments containing inventories of Iodine-131 through Iodine-135 greater than 1.5 curies or Strontium-90 greater than 5 millicuries shall be vented to the facility ventilation exhaust stack through high efficiency particulate air and charcoal filters which are continuously monitored for an increase in radiation levels.” This is inconsistent with TS 3.8.o which states that a fueled experiment can be encapsulated or vented. Clarify whether fueled experiments are vented or not and revise the TS if required, or justify why no additional information is needed.

c. If such venting is permitted then explain why those contributions are not included in the inventory of normally released material (such as Ar-41), or justify why no additional information is needed.