November 16, 2007

Mr. William Bonzer, Reactor Manager
University of Missouri–Rolla
226 Fulton Hall
Rolla, MO 65409-0170

SUBJECT: UNIVERSITY OF MISSOURI - ROLLA RESEARCH REACTOR FACILITY – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE RENEWAL (TAC NO. MC5737)

Dear Mr. Bonzer:

We are continuing our review of your application for license renewal of the University of Missouri - Rolla Research Reactor (UMRR). After reviewing your submissions, questions have arisen for which we require additional information and clarification. During a discussion with you on November 8, 2007, you agreed to provide a response to the enclosed Request for Additional Information (RAI) no later than January 31, 2008. Your timely response is needed to support completion of the review. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation.

Should you have any questions, please contact me at 301-415-1128 or John Nguyen at 301-415-4007.

Sincerely,

/RA/

John Nguyen, Project Manager
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-123
License No. R-79

Enclosure: As stated
cc w/enclosure:
Please see next page

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Test, Research, and Training
Reactor Newsletter
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University of Missouri - Request For Additional Information

The following questions apply to the area of site characteristics related to potential accident or radiological release scenarios or conditions. These questions are necessary to verify compliance with 10 CFR 50.36, Technical Specifications, 10 CFR Part 100, Reactor Site Criteria, 10 CFR Part 20 Subpart C, Occupational Dose Limits, 10 CFR Part 20 Subpart D, Radiation Dose Limits for Individual Members of the Public and to ensure that safety limits are not exceeded. As additional guidance, the NRC staff is also relying on the guidance contained in NUREG-1537 in conducting its review.

1. Section 2.1.2. This Section does not specifically address Fort Leonard Wood (or any other military installation). Confirm that the distance from UMRR to Fort Leonard Wood is greater than 8 kilometers. Alternatively, confirm that none of the missions performed at Fort Leonard Wood (i.e., chemical school) or any nearby military facility presents an unanalyzed threat to the safe operation of the UMRR.

2. Section 2.2.1. This Section concludes that none of the industries, transportation routes, or other facilities pose a threat to the UMRR. It appears that the bases for this conclusion are the Preliminary Hazards Evaluation (December 1958) and the Hazards Summary Report (November 1965). Provide references or analyses that take into account current industries, transportation routes, or other facilities.

3. Section 2.3.2. This Section contains detailed wind observation studies from the Vichy Station for the 1948-1954 time period. Explain why this period is selected, and the bases for the acceptability of 50 year-old data. Provide more recent data which can verify the acceptability of the conclusions in the SAR, or provide justification why this data is not needed.

4. Section 2.3.2. This Section has presented wind data for four Missouri cities (Columbia, Kansas, St. Louis, and Springfield City) for the period 1930 to 1996, amongst which Rolla is centrally located. While the mean wind speed for these cities is about the same, there are noticeable variations in the prevailing wind direction and peak wind gusts. Discuss how representative wind conditions were derived from the wind data for the UMRR. Confirm that no additional events occurred in the period 1996 to present that would require modifications of the wind data set.

5. Sections 3.2 and 13.1.8. The evaluation of external events in these Sections states that tornadoes or hurricanes occur infrequently in the Rolla area. Provide historical data, references, or other information on the tornadoes or hurricanes that have affected the region or the reactor facility. Also, discuss the reactor facility design and surveillance to ensure that structures, systems, and components will continue to perform their safety functions as specified in the SAR.

The following questions apply to the area of reactor design, which are necessary to verify compliance with 10 CFR 50.36, Technical Specifications, 10 CFR Part 20 Subpart C, Occupational Dose Limits, 10 CFR Part 20 Subpart D, Radiation Dose Limits for Individual Members of the Public and ensure that safety limits are not exceeded. As additional guidance, the NRC staff is also relying on the guidance contained in NUREG-1537 in conducting its review.

6. Section 4.2.1. This Section has not discussed or provided reference to information on the design and development program for the MTR-type fuel used. Provide a discussion or reference to ensure that this fuel design will continue to perform its safety functions and will not affect public health and safety during the period of extended operation.

7. a. Section 4.5. Provide the standard or most used core configuration and the highest neutron flux density (or highest power level) corresponding to this configuration. Discuss reactor facility design and safety functions to prevent the reactor from exceeding the reactor power limit setting.

b. Section 4.5. Provide the limiting core configuration (the most compact core) and the highest neutron flux density (or highest power level) corresponding to this configuration. Discuss reactor facility design and safety functions to prevent an uncontrolled reactor transient.

c. Section 4.5. Describe rearrangements of the different core configurations. Also, discuss the prevention of uncontrolled reactor transients during these rearrangements. Include a description of how the reactivity worths of various components such as control rods, standard fuel element, half fuel, thermal column, (etc.) change with core configuration.

d. Section 4.5. Provide the reactivity worth of a fuel element in the center of the core.

8. Section 14.2.1. The Bases in this Section state “The melting temperature of ... .fabrication is 588 °C (1076 °F).” Provide the correct conversion from °C to °F value.

9. Section 4.2. Table 4.1 in this Section states that the Void Coefficient of Reactivity is -9.0x10E-7 Δk/k/°C. In Section 4.5.2.2, it states that the Void Coefficient of Reactivity is 9E7Δk/k/cm3. Explain why two values of the Void Coefficient of Reactivity are inconsistent.

10. Section 4.5.2.4. β-eff in this Section is equal to 0.0079, whereas it is equal to 0.0065 in Section 13.1.2. Provide justification why β-eff is shown as two different values in the SAR. Provide analyses or reference to where the value of β-eff is obtained.

11. Section 4.5. Given that proposed TS doesn't explicitly prevent misloading a bundle, excess reactivity may be exceeded. What is their approach in controlling excess reactivity? Also, discuss reactor facility design and surveillance that prevent the reactor from exceeding the limiting reactivity conditions.

12. Sections 4.5.1, 4.6, and 14.2.1. According to NUREG-1313, fission product release from irradiated fuel elements starts at approximately the blister temperature of the cladding, which is ~527°C. The objective of the Safety Limits in TS Section 14.2.1 is "to ensure that the integrity of the fuel cladding is maintained in order to guard against an uncontrolled release of fission products." Provide justification for the proposed Safety Limit in these Sections and provide justification for the adequate margin, or explain why the justification is not needed.

13. Section 4.2. Table 4-1 in this Section states that Shim/Safety Rod drive speed is 6 in/min and Regulating Rod drive speed is 24 in/min. Provide frequency for maintenance and calibration of the Rod drive speed.

The following questions apply to the areas of accident analysis and thermal hydraulics which are necessary to verify compliance with 10 CFR 50.36, Technical Specifications, 10 CFR 50.59, Changes, Tests, and Experiments, 10 CFR Part 20 Subpart C, Occupational Dose Limits, 10 CFR Part 20 Subpart D, Radiation Dose Limits for Individual Members of the Public, and ensure that safety limits are not exceeded.

14. Section 5.1. Please provide a diagram of the reactor water systems for this Section.

15. Section 6.2. The objective of TS Section 14.3.5, Ventilation System, is "to provide for normal building ventilation and the reduction of airborne radioactivity within the reactor bay during reactor operation." The TS Bases for Section 14.3.5 states, "Experience has shown that during normal operation this specification is sufficient to maintain radioactive gaseous effluents below 10 CFR Part 20 Appendix B limits." Provide the historical airborne effluent data, with or without HVAC operation, for when the reactor was at power and the 10 CFR Part 20 limit was exceeded. Also, describe any existing or planned design features and/or procedures that protect reactor operators if the airborne effluents exceeded 10 CFR 20 Appendix B limits under these conditions.

16. Provide justification for the five-minute evacuation time. What specific steps must the operators perform after an alarm sounds and how much time is needed for each of these steps? Under emergency conditions, there is reaction time, diagnosis time, decision time, and perhaps last minute activities prior to evacuating. Considerations could include: do the operators need to verify scram has occurred? Does anyone need to verify the control rods are inserted and the reactor has shut down? Does anyone need to attend to an experiment? Which building systems need to be verified as operating or shut down? An example of the process deriving diagnosis times for a power plant is in Table 12-2 of NUREG/CR-1278. Provide a time analysis for evacuation.

17. The failure of a fuel element cladding outside of the reactor pool could be due to corrosion or manufacturing defect. Provide clarification that the consequence of failed fuel cladding is bounded by the failure of the fueled experiment.

18. Section 11.1.2 refers to containment, but does not provide the definition for containment in TS Section 14.1.2. Provide the definition for containment, or provide justification for why this definition is not needed.

19. In implementing the requirements of 10 CFR 50.36(c)(2), ANS-15.1 provides definitions for reactor operating, reactor shutdown, and reactor secured. The ANS-15.1 also specifies that conditions must exist for the reactor to be secured and the limiting reactivity setting for the movement of a single experiment. However, the definition in the TS Section 14.1.2, differs from the ANS standard. Provide justification for the definition of reactor secured in the Section 14.1.2.

20. Section 14.3.2.2. Table 7.1 specifies two channels: Safety Channel No.1 and Safety Channel No. 2 for scram functions. However, these channels are missing in TS Section 14.3.2.2. Provide justification why these channels are not included in TS Section 14.3.2.2.

21. ANS-15.1 specifies the minimum operating systems or operating limits for the reactor coolant to include the coolant level limits, and leak or lost-of-coolant detections. Provide justification why the detections are not included in the TS.

22. Section 14.3.6. ANS-15.1 specifies the minimum number of radiation monitors for radiation protection while operating the reactor. One of the monitor requirements is the Continuous Air Monitors (CAMs). Provide justification why the CAMs are not included in TS Section 14.3.6.

23. Section 14.2.1. This Section states "The maximum cladding temperature associated with full power (200kWt) operations is only about 90 °C. Furthermore, calculations show that cladding temperature associated with a reactor power of 4.5 MW would only be about 140 °C." Provide analyses or references that were used to obtain the above results, or justify why this calculation is not needed.

24. Section 7.2.2-Table 7.1. Table 7.1 references the term "Run Down" but does not provide a definition for it. Provide a definition or reference. Also, consider adding it to Section 14.1.2 Definitions.

25. Sections 7.4. and 14.3.6.1 The SRP calls for a mechanism to determine and monitor the reactor coolant radioactivity. In addition to the Radiation Area

Monitors (RAMs) listed in Table 14.3, what other methods are available to monitor reactor coolant radioactivity for fuel cladding failure.

26. Section 7.4. When the Continuous Air Monitor (CAM) alarm goes off, it appears that the reactor ventilation dampers do not automatically close. Please describe the reactor facility design, safety function or any other mechanisms that are in place ensuring the reactor staff with manually secure the ventilation system when it goes off. Also, specify how a CAM alarm going off at night is handled when reactor staff are not present at the facility.

27. Section 7.6. This Section states "The Period <30-second rod withdrawal prohibit can be key bypassed at the reactor console by the SRO on Duty as provided for in the SOPs". Describe how a bypass of a period less than 30 seconds is activated. Describe the measuring method that prevents an unintentional activation from staff members.

28. Section 9.2. This Section states that the reactivity of the currently used LEU fuel elements have been shown to be comparable to the measured results obtained from the previously used/stored HEU. Provide a reference to the measurements or calculation which supports this statement.

29. The term "unreviewed safety question" used in Sections 12.3 and 14.3.7.2 (3) is no longer exist in the current regulations 10 CFR 50.59. Provide clarification or revision.

The following questions apply to the area of radiation protection, and are necessary to verify compliance with 10 CFR Part 20 Subpart D, Radiation Dose Limits for Individual Members of the Public and 10 CFR 50.36, Technical Specifications.

30. Section 11.1.1. This Section states that the licensees' environmental monitoring program consists of reading film badges located in strategic areas within the reactor building and one set of measurements taken on exterior facility surfaces in 1984. Section 11.1.1.1 also states that the release of Ar-41 to the air is estimated annually through calculations to be below regulatory limits. Further, the 2007 annual report states "Release of gaseous Ar-41 activity through the building exhausts is determined by relating the operating times of the exhaust fans and reactor power during fan operation to previously measured air activity at maximum reactor power. During this period, an estimated 101,742.35 microcuries of Ar-41 were released into the air." 10 CFR 20.1302(a) requires the licensees conduct surveys of radiation levels in unrestricted and controlled areas to demonstrate compliance with the public dose limits of Part 20.1301. Justify how UMRR ensures that the assumption used in the calculations are still valid for meeting the regulation. Also, describe how the yearly dose to the public outside the reactor building is obtained, since none of the three radiation area monitors referenced in Section 14.3.6.1 are near the ventilation system exhaust. Provide the limit exposure above background (doses/year) to the unrestricted environment due to the discharge of Ar-41. Also, provide a proposed action (such as ceasing the reactor operation for the remainder of the calender year) if the limiting exposure setting is exceeded, or provide justification why the proposed action is not needed.

The following questions pertain to the technical specifications and are necessary to verify compliance with 10 CFR Part 55, Operator's Licenses, 10 CFR Part 50.36(c)(5) –Technical Specifications. As additional guidance, the NRC staff is also relying on the content of NUREG-1537 in conducting its review.

31. Sections 12.1.4 and 14.6.1. In accordance with 10 CFR 55.53 and 10 CFR 55.59, reactor operator requires certain conditions for maintaining the NRC license and meeting the requalification requirements. In implementation of these requirements, the UMRR commits to ANSI/ANS-15.4 (1978) for the selection, training, and requalification of personnel. Provide justification why the latest version of ANSI/ANS-15.4 (1988) is not used.

32. Neither the SAR or TS specifically address 10 CFR 50.54(k) requirements, as listed below.

§50.54 Conditions of licenses.

...

(k) An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility.

UMRR TSs only require an operator in the control room (CR). Are the control room and "at the controls" the same at UMRR? Are there portions of the CR that are not directly accessible to the controls?

33. In implementing the requirement of 10 CFR 50.36(c)(2), ANS -15.1 provides definitions of reactor operating, reactor shutdown, and reactor secured. The SRP and ANS also specify minimum staffing for the reactor in unsecured condition. The TS Section 14.6.1.3 does not include minimum staffing requirements for reactor shutdown and for reactor not secured. Provide justification why the requirements are not included in the TS.

34. Section 12.2. The SRP and ANS 15.1 specify certain rules for the Radiation Safety Committee (RSC), which are not addressed in the SAR or TSs. These may be in the RSC charter which was not submitted to NRC. Please verify that the following items are addressed: dissemination of minutes in a timely manner (no longer than three months after the meetings), appointment of at least one qualified RSC member not on the staff of the Reactor Department, and a written report of the findings and recommendations of the committee submitted to Level 1 in a timely manner after the review is complete.

35. Sections 12.2.3 and 14.6.2.3. In implementing 10 CFR 50.59, ANS-15.1 specifies the responsibilities of the Radiation Safety Committee including the review of new procedures. Provide justification why this item is not included in the TS Section 14.6.2.3.

36. Section 14.6.2.4. ANS 15.1 specifies items that should be audited by the auditors including training, requalification program, emergency plan, security plan, experiments, health physics, and the results of actions taken to correct deficiencies. Justify why these items are not included in the SAR or TSs.

37. Section 14.6.1. In implementing the requirement of 10 CFR 50.36(c)(5), the NRC SRP specifies the review functions for the development of new procedures. It also specifies the responsibility of the reactor staff in reviewing reactor operations, radiation protection, and reactor administration procedures. Justify why these requirements are not included in Section 14.6.1.

38. Section 14.6.5. In implementing the requirements of 10 CFR 50.36(c)(5), ANS 15.1 specifies the responsibilities of the Director of Reactor Facility (DRF) and RSC in reviewing and approving for new experiment and substantive changes to approved experiments. Provide justification why the DRF responsibilities are not included in the requirements. Also, discuss a process for approval of new experiment and substantive changes to approved experiments.

39. Section 14.6.7.1. ANS 15.1 specifies the items to be included in the annual report. One item is missing from the Section 14.6.7.1 namely, “a summarized result of environmental surveys performed outside the facility.” Provide proposed change to Section 14.6.7.1, or provide a reason why this is not included. Also, there are no off-site environmental monitoring surveys required by the TS and listed as required records to be maintained for the lifetime of the facility, as specified in ANS 15.1. Propose such a TS or acceptable alternative, or provide justification that it is not needed.

Editorial comment - at the end of Section 14.6.7.1(4) there is a stray “(6)” tacked on to the end of the line, attached to “10 CFR Part 50” that could potentially be confusing.

40. Sections 12.5.2(1) and 14. 6.7.2(1) state that any required reports will be sent to the NRC Project Manager and the Regional NRC Office. Also, Sections 12.5.1, 12.15.2(2), 14.6.7.1 and 14.6.7.2(2) state the written reports will be sent to the Regional Administrator. Propose revised TS and SAR in accordance with 10 CFR 50.2, 10 CFR 50.4, and 10 CFR 50.36 for correct communications between licensee and NRC, or provide rationale why this change is not needed.

41. The terms rundown is used in the SAR and TSs. Justify why the definition is not included in Section 14.1.2 Definitions.

42. Section 14.1.2. The TS defines “Excess reactivity-that amount of reactivity that would exist if all control rods were fully withdrawn from the core.” The definition differs from the ANS 15.1 guidance. Provide revision in accordance with the guidance of ANS 15.1, or explain why the change is not needed.

43. The TS defines “reference core condition-reactivity condition of the core when.....is negligible (<0.30 dollars).” The unit, dollars, is not consistent with unit used in the TS (% Δ k/k). Provide revision, or explain why this change is not needed.

44. Discuss the process that UMRR will use to incorporate Amendments, responses to NRC RAIs, or changes to the facility or organization structure that are approved by NRC during the license renewal review period (April 2004 to review completion date) into the license renewal SAR and proposed license renewal TSs.

The following questions are editorial in nature in Chapter 14, Technical Specifications.

45. Sections 14.2.1 and 14.3.7.2. Provide justification why the reactor power is shown in two different units, kWt vs. kW, in SAR.

46. Section 14.2.1. It is not consistent with the TS format. Some paragraphs display °C (°F), some just display °C.

47. Section 14.3.2.2. The TS states “The Log N and Period not operative scram shuts the reactor down.” Propose a revision “Period not operative” in accordance with Table 14-2, page 14-8 for consistency.

The following questions apply to the area of Operator Requalification Program, which are necessary to verify compliance with 10 CFR 55.59, Requalification.

48. Section 2- Reactor Requalification Program (RRP). 10 CFR 55.59(a)(1) requires that the requalification program have a 24-month cycle; however, Section 2, Description of the Program, does not define a cycle, but instead requires a “biennial requalification cycle” written examination. How does the facility ensure that no requalification cycle is longer than 24 months?

49. Section 2.1. RRP contains a list of areas, from which the biennial examination is to be prepared. The list is from “A” through “I”, but skips “D”. Is there supposed to be a “D”? Comparing to 10 CFR 55.59(c)(2) it appears that Plant Instrumentation and Control Systems is missing.