July 20, 2007

Bill:

In accordance with our telephone conversation on July 19, 2007, please find attached a preliminary Request for Additional Information (RAI). This e-mail is to provide you the opportunity to identify areas in the preliminary RAI, which may require additional clarification before issuance of the final RAI. No formal response is required to the preliminary RAI at this time. We will discuss the preliminary RAI with you shortly to see if you need additional clarification on it, and to agree upon a schedule for responding to the final RAI. Please confirm that you got this e-mail and propose a date and time in the next two weeks, which would be acceptable to you to discuss the preliminary RAI and schedule to answer the final RAI request.

If you have any questions regarding this review, please contact me at (301) 415-1128 or John Nguyen at (301) 415-4007.

Sincerely,
Marvin Mendonca

Research and Test Reactors Branch B
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

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Junk Mail Handling Evaluation Results
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JULY 19, 2007

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 20 Subpart C, Occupational Dose Limits, 10 CFR Part 20 Subpart D, Radiation Dose Limits for Individual Members of the Public and ensure1 that safety limits are not exceeded. As additional guidance, the NRC staff is also replying on the guidance contained in NUREG-1537 in conducting its review.

1. Section 2.1.2. The SAR does not specifically address Fort Leonard Wood (or any other military installation) in this section. Confirm that the distance to Fort Leonard Wood is greater than 8 kilometers or that none of missions performed at Fort Leonard Wood (i.e., chemical school), or any other nearby military facility, present a unanalyzed threat to the safe operation of the UMRR.

2. Section 2.2.1. The SAR 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 is the Preliminary Hazards Evaluation (December 1958) and the Hazards Summary Report (November 1965). Provide references or analyses that encompass current industries, transportation routes, or other facilities.

3. Section 2.3.2. The SAR has reported on detailed wind observation studies from the Vichy Station for the 1948-1954 time period. Why was this period selected, and what is the bases for the acceptability of this data, which is over 50 years old, for the period of extended operations? Is there more recent data which can verify the acceptability of the conclusions obtained from the analysis of this data?

4. Section 2.3.2. The SAR has presented wind data for four Missouri cities, amongst which Rolla is centrally located for the period 1930 to 1996. 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. Section 3.2. and Section 13.1.8 The SAR states that tornados occur infrequently in the Rolla area. Provide data, references or other information on the tornados

which did affect the region, including maximum wind speed, to verify the conclusions discussed in 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 replying on the guidance contained in NUREG-1537 in conducting its review.

6. Section 4.2.1. The SAR 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 on this to provide reasonable assurance that this fuel design will function safely and not affect public health and safety during the period of extended operation.

7. a. Section 4.5. What is considered to be the standard or most used core configuration and what is the associated relative power distribution?

b. Section 4.5. What is considered the limiting core configuration and what is the associated relative power distribution?

c. Section 4.5. Describe or reference how different core configurations are considered or bounded in the accident analyses. Include description of how do all of the coefficients quoted in Section 4.5 of the SAR change with core configuration.

8. Section 4.5. What is the reactivity worth of a fuel element in the center of the core?

9. Section 4.5.2.2. Describe how the value of the void coefficient was determined? Verify that the quoted value, 9E+7 $\Delta k/k/cm^3$ is correct.

10. Section 4.5.2.4. What are the uncertainties associated with the delayed neutron fraction numbers?

11. Section 4.5. What is the overall power coefficient of reactivity?

12. Section 4.5.1. According to NUREG-1313, fission product release from irradiated fuel elements starts around the blister temperature of the cladding which is ~527°C. The stated purpose of the Safety Limit, TS14.2.1 is "To ensure that the integrity of the fuel cladding is maintained in order to guard against an uncontrolled release of fission products." Should the Safety Limit be changed to a value lower than the present 580°C and does this still ensure adequate margin?

13. Section 4.2.1.4. SAR refers to fuel plate positions; provide a fuel core diagram.

14. Section 4.2. Table 4-1 listed that Shim/Safety Rod drive speed is 6 in/min and Regulating Rod drive speed is 24 in/min. How often are those speeds checked?

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 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.

15. Section 5.1. Please provide diagram of the reactor water systems.

16 Section 6. The objective of TS 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 Bases of this TS states "Experience has shown that during normal operation this specification is sufficient to maintain radioactive gaseous effluents below 10 CFR 20 (Appendix B) limits". Confirm that during normal operations radiation exposure to the operators due to airborne effluents would exceed 10 CFR 20 (Appendix C) limits without HVAC operation

17. Section 7.2.2-Table 7.1. Provide or reference a definition of "Run Down." Also consider adding that definition to the definition section of the TSs.

18. Section 7.4. The SRP calls for a mechanism to determine and monitor reactor coolant radioactivity. Do the Gamma RAMs in TS Table 14.3 serve this purpose? The note for TS Table 14.3 states that the rationale for RAM detector locations, setpoints and functions is presented in SAR Section 7.4, however this is not the case. Other than the gamma guard on the demineralizer, if a leak in the fuel were to occur, what other methods are available for operating personnel to determine that a leak had occurred?

19. Section 7.4. When a CAM alarm goes off, it appears the reactor ventilation dampers will not automatically close. How do you ensure that the reactor staff will manually secure the ventilation system in the event of a CAM alarm? Include a discussion of CAM alarms at night.

20. Section 7.6. Table 7.2. Describe how a bypass of a period of less than 30 sec is activated. Describe the measuring to ensure that it is not inadvertently activate.

21. Section 9.2 SAR 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 support this statement.

The following question apply to the area of radiation protection, which 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.

22. Section 11.1. SAR 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. SAR 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 latest annual reports say:

> "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) states that licensees shall make surveys of radiation levels in unrestricted and controlled areas to demonstrate compliance with the public dose limits of Part 20.1301. Please justify how UMRR ensures that the assumption used in the calculations are still valid for meeting the regulation.

Describe how is the yearly dose to the public outside the reactor building obtained, since none of the three radiation area monitors referenced in TS 14.3.6.1 are near the ventilation system exhaust. How would an increase in Ar-41 leaving the building be identified in order to mitigate the situation?

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

23. Section 12.1.4. and Section 14.6.1 10 CFR 50.36(c)(5) requires administrative controls, including reviews, to assure operation of the facility in a safe manner. In implementation of that requirement, the UMRR commits to ANSI/ANS-15.4 (1978) for the selection, training, and requalification of personnel in SAR Chapter 12 and TS 14.6.1.4. This should be updated to the 1988 version of the standard per the SRP or justification provided.

24. Neither the SAR or TS specifically address 10CFR50.54 (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. Are the control room and "at the controls" the same at UMRR, or are there portions of the CR that are not directly accessible to the controls?

25. 10 CFR 50.36(c)(2) requires limiting conditions of operation to ensure safe operation. In implementation of that requirement, ANS -15.1 provides definitions of reactor operating, reactor shutdown, and reactor secured. The TS provide similar definitions. The SRP and ANS specify minimum staffing when the reactor is not secured. TS 14.6.1.3 provides minimal staffing when the reactor is operating. Thus, the TS do not provide for staffing of a shutdown, but not secured reactor. TS 14.6.1.3 should be changed from operating to not secured or an alternative proposed, or provide justification why the change is not required.

26. 10 CFR 50.36(c)(2) requires limiting conditions of operation to ensure safe operation. In implementation of that requirement, ANS -15.1 provides definitions of reactor operating, reactor shutdown, and reactor secured. The TS provide similar definitions. However, the definition for reactor secured differs from that recommended by the ANS standard. In the standard, 4 conditions must exist for the reactor to be secured, one of which is "No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, which ever is smaller." The maximum value allowed for a single experiment at UMRR per Ch. 10 of the SAR is 0.4 % Δk/k. The UMRR TS definition for reactor secured includes the following item: "no experiments worth more than 0.4% Δk/k are near the core." This does not meet the definition in the standard to limit the amount of reactivity addition by the movement of experiments whose sum adds up to more than the maximum value allowed for a single experiment. Please revise definition to conform to the ANS standard's definition or acceptable alternative, or provide justification why the change is not needed.

27. 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 items are addressed: dissemination of minutes in a timely manner (no longer than 3 months from meetings), appointment of at least one qualified RSC member(s) not on the staff of the University, 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.

28. Section 12.2.3 and Section 14.6.2.3. ANS Standard 15.1 lists the items that should be reviewed by the RSC. The following item from the ANS Standard, Section 6.2.3 was not included in the SAR or TSs as part of the review responsibility of the RSC: "All new procedures." 10 CFR 50.59 requires such reviews. Please modify TS 14.6.2.3 to add the responsibility of the RSC to review all new procedures and major revisions to procedures having safety significance or provide acceptable alternative, or reason the review should not be required.

Additionally, the responsibilities of the RSC and Director of the Reactor Facility for procedures does not list either review or the approval of new procedures or procedure changes. However, the section on procedures notes that the RSC must review and approve substantive changes to procedures. Please clarify who has the review and the approval responsibility for procedures and procedure changes.

29. ANS Standard 15.1 lists the items that should be audited by the audit group. The following items from the ANS Standard, Section 6.2.4 and the NRC SRP were not included in the SAR or TSs: retraining and requalification program for the operating staff, emergency plan, security plan, experiments, health physics, and results of action taken to correct deficiencies in reactor safety. Please provide TS or give a rationale why they are not needed.

30. 10 CFR 50.36(c)(5) requires administrative controls, including reviews, to assure operation of the facility in a safe manner. In implementation of that requirement, the NRC SRP notes that the procedure development process should include reviews by staff from reactor operations, radiation protection, and reactor administration. Please provide TS or a SAR commitment to include these reviews or give a rationale why they are not needed.

31. 10 CFR 50.36(c)(5) requires administrative controls, including reviews, to assure operation of the facility in a safe manner. In implementation of that requirement, ANS 15.1 states that new experiments be approved by Level 2 management (the DRF for UMRR). Further, substantive changes to approved experiments should be reviewed by the RSC and approved in writing by the DRF. Propose TS to require these approvals , reviews and experimental controls or give a rationale why they are not needed. Also, do the UMRR procedures for the control of experiments address NRC Reg. Guide 2.2, items c.3.a(2) and c.3.b, and c.3.c (as noted in NRC SRP 14.1, # 6.5)?

32. Section 14.6.6.2 (1) states that, "Reactor conditions shall be returned to normal or the reactor shall be shutdown." 10 CFR 50.36(c)(1)(ii)(A) and (2)(I) require bases to prevent recurrence and remedial actions specified by TS, respectively. In implementation of these requirements the NRC SRP (App. 14.1, Section 6.6.2) notes that, when a licensee chooses to add to the TSs the option to return the reactor to normal, the TS should be written to establish, in advance, specific criteria for returning the reactor to normal. An example may be: a reactor scram from a known condition such as an electronic transient. Propose such a TS or acceptable alternative, or provide justification that it is not needed.

33. Section 14.6.7.1. ANS 15.1 specifies the items to be included in the annual report. One item is missing from the TS list for UMRR, namely, "a summarized result of environmental surveys performed outside the facility." As required by 10 CFR 50.36(c)(8), please revise TS 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 TS14.6.7.1(4) there is a stray "(6)" tacked on to the end of the line, attached to "10 CFR 50" that could potentially be confusing.

34. SAR sections 12.5.2(1) and 14. 6.7.2(1) indicate reports are to be sent to "NRC Project Manager and the regional NRC office." Similarly, SAR sections 12.5.1, 12.15.2(2), 14.6.7.1 and 14.6.7.2(2) indicate written reports to the Regional Administrator. Proposed revised TS and SAR in accordance with 10 CFR 50.2, 10 CFR 50.4, 10 CFR 50.36 or rationale why this change is not needed.

The following questions apply to the area of accident analysis, which are necessary to verify compliance with 10 CFR Part 20 Subpart C and D, Occupational and Public Dose Limits. As additional guidance, the NRC staff is also relying on the content of NUREG-1537 in conducting its review:

35. Provide justification for the five minutes evacuation time. Even though the building has been evacuated in three minutes during a drill does not mean that in the event of an emergency, the building will be so rapidly evacuated. 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 to accomplish prior to evacuating. Consideration 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; what 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.

36. The failure of a fuel element cladding outside of the reactor pool could be due to dropping it, corrosion, or manufacturing defect. Was failure of a fuel element out of the pool considered and is it bounded by the failure of the Fueled Experiment?

The following question apply to the area of technical specifications, which are necessary to verify compliance with 10 CFR 50.36, Technical Specifications:

37. The terms "rundown" is used in the Technical Specifications. Provide definition in Sec 14.1.2, Definitions, does not contain the following definitions

38. Sec 14.1.2, page 14-2. 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 clarification or define excess reactivity in accordance with the guidance of ANS 15.1.

39. 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 clarification or revision.

40. Sections 11.1.2 and 14.6.7.2 refer to containment vice confinement. Provide clarification or correction as Technical Specifications only define confinement.

41. Sec 14.2.1, page 14-4. The TS 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". Describe these calculations or provide them.

42. Sec 14.2.1, page 14-4. The TS state "The melting temperature ....fabrication is 588 °C (1076 °F)." Provide verification of the correct value and units for this temperature.

43. Sec 14.3.2.2, page 14-8,Table 14.2. Provide an explanation why Table 14.2, page 14-8, has only one channel, Reactor Power Channel, for scram function. Where as Table 7.1, page 7-2, has scram channels for Safety #1 and Safety # 2. Provide clarification and revision that both Safety #1 and Safety #2 channels are required by the Technical Specifications.

44. Provide clarification on the need for scram function associated with water level due to loss of the reactor pool water.

45. Sec 14.3.2, Page 14-9. Provide clarification why the TS do not have a section that corresponds to safety system interlocks or RWP.

46. The term "unreviewed safety question" is used in SAR Sections 12.3 and 14.3.7.2 (3), page 14-5. This term is no longer used in the regulations 10CFR50.59. Provide clarification or revision.

47. The CAMs referred to in the section 7.4 of the SAR are not required by TSs in accordance with NUREG-1537 guidance. Provide rationale why this should not be required or propose change to the TSs to include such in accordance with the NUREG.

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

48. Sec 14.2.1, page 14-4. Provide an explanation why the Bases in Sec 14.2.1 use different units to describe the reactor power. One uses kWt and the other use MW. Same comment for Sec 14.3.7.2, page 14-14

49. Sec 14.2.1, page 14-4 and page 14-5. It is not consistent with the TS format. Some paragraphs display °C (°F), some just display °C.

50. Sec 14.3.2.2, page 14-9. The TS states " The Log N and Period not operative scram shuts the reactor down". Revise "Period not operative" in accordance with

Table 14-2, page 14-8 for consistence.

51. Sec 14.3.3, page 14-9. Add a space between paragraphs (1) and (2).

52. Sec 14.3.3, Page 14-10. Delete a space between (2) and (3).

53. Sec 14.4.2.2, page 14-17. Add a space between 2) and 3) in Specifications section.

54. Sec 14.6.1.3, page 14-24. Edit "igure 14.1" to "Figure 14.1" and place it correct in position

55. Sec 14.6.6.1 page14-27. Add a space between paragraphs c and 5

56. Sec 14.6.7.2 page 14-29. Correct indents for paragraphs i, ii, and vi

57. Sec 14.6.7.2 page14-30. Add a space between paragraphs 2) and a)

58. Sec 14.6.7.1, page 14-28. Indent paragraph 2) that corresponds to the entire

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

59. 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. Lately, some facilities have interpreted biennial as the same as Technical Specifications twenty-four, but not exceed 30 months. How does the facility ensure that no requalification cycle is longer than 24 months?

60. 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.