October 3, 2006

Dr. Steven R. Reese
Director, Radiation Center
Oregon State University
100 Radiation Center
Corvallis, OR 97331-5903

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE RENEWAL, OREGON STATE UNIVERSITY TRIGA REACTOR (TAC NO. MC5155)

Dear Dr. Reese:

By a letter dated October 5, 2004, as supplemented on August 8, 2005, Oregon State University (licensee) submitted an application pursuant to 10 CFR 50.54 for Oregon State University TRIGA Reactor (OSTR) for review by the U.S. Nuclear Regulatory Commission (NRC). During our review of your license application, questions have arisen for which we require additional information and clarification. Please provide a response the enclosed request for additional information within 30 days of the date of this letter. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation. Please note that your timely response is needed in order to support completion of the review of the renewal application.

Should you have any questions regarding this review, please contact Jessie Quichocho, at (301) 415-1225.

Sincerely,

/RA/

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

Docket No. 50-243

Enclosure: As stated

cc w/encl: See next page

cc:

Mayor of the City of Corvallis
Corvallis, OR 97331

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Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

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REQUEST FOR ADDITIONAL INFORMATION

OREGON STATE UNIVERSITY TRIGA REACTOR

DOCKET NO. 50-243

1. In Section 3.2, Meteorological Damage, the licensee states, “The superstructure of the OSTR has been designed for area wind, rain, snow, and ice loads.” Clarify if the roof structure is also designed to withstand any adverse weather condition (e.g., heavy rain or wind) in the area.

2. How is the leak tightness of the aluminum can (Ref. SAR Sect 4.2.3) assured? If coolant infiltration were to occur, what assurances are there that it will not interfere with safe operation or prevent a safe shutdown?

3. Section 4.5.2.1 is the Core Description. NRC SRP, Chapter 4 requires information on excess reactivity and shutdown margin for the core. In this section several different core configurations were described. What would the excess reactivity and shutdown margins be for the different core configurations? For excess reactivity and shutdown margin determinations, are the Position A inserts mentioned in Section 13.2.2.2.2 considered as part of the core or as part of an experiment?

4. Some of the items requested in NUREG-1537, Part 2, Chapter 4 have not been included in the SAR.
   - How does the excess reactivity and shutdown margin change with U and poison burnup and Pu buildup?
   - What are the neutron flux densities?
   - What is the fuel burnup between reloads or shutdowns?
   - How were the values for the neutron lifetime and effective delayed neutron fraction determined and what are their estimated uncertainties?
   - Have the reactor periods been analyzed?
   - What is the void coefficient (note: according to NUREG/CR-2387 typical void coefficient for the interstitial water in the core is about -0.2% $\Delta k/k$/1% void)?
   - What is the xenon and samarium override?
   - What is the overall power coefficient of reactivity if not accounted for in the items listed above?
   - Are there any credible situations where a flow instability could occur in a fuel channel?

5. Section 4.6 provides the results of the calculations of the maximum heat flux. For which of the cores described in Section 4.5.2.1, and mentioned in Section 13.2.2.2.2, were the calculations performed and will there be significant differences for the other cores?

6. Confusion exists as to the specific power measuring channels that comprise the “power level measuring channel” and/or “power level monitor” referred to in TS 3.2.2. These channels are not defined in the TS or adequately described in the SAR. From the information provided in Sections 7.2.3.1 and 7.4.1, it appears that the Safety and/or the Percent Power Channels could form the “power level measuring channel” referred to in the TS. However, this is not entirely clear, as Section 7.2.3.1 also refers to a “power range monitor” which is apparently composed of the percent power and pulsing

channels. In addition, it is not clear if one or two channels are required to satisfy the power level monitoring capability specified in the TS.

7. Reactor power measuring channels are described in Section 7.2.3.1 of the SAR. TS 3.2.2, "Basis," states: "The power level monitors assure that the reactor power level is adequately monitored for both (sic) steady state, square wave, and pulse modes of operation" TS Table 1, "Minimum Measuring Channels," identifies the "Nvt circuit" as being required while operating in the Pulse Mode. However, the "Nvt circuit" is not described in Chapter 7 of the SAR. Please clarify.

8. SAR Chapter 5 states that cooling of the reactor core is accomplished by natural circulation water flow through the core area combined with a forced-flow circulation of pool water through a tube-and shell type heat exchanger. Chapters 3 and 5 of the SAR indicate that a loss of forced coolant flow will not result in any adverse consequences. For example, Section 3.1 of the SAR states: *"Natural convection cooling is sufficient to dissipate core heat."* This statement should be clarified as it implies that a loss of forced circulation would have no impact on safe reactor operations which is not consistent with the evaluation presented in SAR Chapter 13 (Section 13.2.4.2.1).

9. Section 13.2.4.2.1 of the SAR states: *"The reactor, however, would shut down as the water level dropped passed the top of the fuel."* There does not appear to be an automatic trip (scram) in response to low pool water level. Therefore, this statement should be clarified to describe the specific mechanisms that would cause the reactor to shutdown (e.g. temperature coefficient of the fuel and/or scram in response to high fuel temperature) .

10. As described in Section 7.3 of NUREG 1537 Part 2, the RCS should be designed for reliable operation in the normal range of environmental conditions anticipated within the facility. Please describe the type of electrical isolation/physical protection provided for voltage and signal cables associated with the redundant power monitoring channels, as this is not currently in Section 7 of the OSU SAR. For example, are they routed as separate conduits from the reactor to the control room?

11. Section 7.2.3.1 states that a loss of operating voltage to the percent power channel or pulsing channel (which form the "power range monitor") will initiate a reactor scram in response to a "non-operable condition." Will a loss of operating voltage to the Safety and Percent Power Channels also initiate a "non-operable condition" reactor scram?

12. Section 9.1.2 describes the ventilation system. The Argon manifold exhaust is part of the reactor bay ventilation system, as shown in Figure 9.1. Section 11.1.5.2 states that the reactor bay ventilation system has HEPA filters on all ducts originating from irradiation or sample handling facilities. Please clarify the purpose of the HEPA filters and explain why there are no surveillance requirements in TS 4.5 for filter testing on the two Argon manifold HEPA filters?

13. Section 9.1.2 describes the ventilation system. Section 11.1.1.1.2 notes that the ventilation system keeps occupational building exposures from Ar-41 well below the 10 CFR 20 DAC limits. The Basis of TS 3.5 for the Ventilation System discusses exposure to both the public and occupational dose to workers. It is not clear if the use of the word "ground" in the first sentence of TS 3.5 Basis refers to outside the Radiation Center building or inside the building in unrestricted areas. Please clarify the purpose and need

for the ventilation with respect to both workers and the public for normal operations and for accident scenarios.

14. The SAR describes a specialized chamber inside Beamport #4 to produce Ar-41. Please provide information on what individual components of the argon production facility equipment are physically located inside Beamport #4. Additionally, please provide the operating pressure that the transfer lines or equipment located inside the beamport will be exposed to and state whether periodic inspections are performed on equipment inside the beamport for material degradation.

15. ANS-15.1, Item 6.5 also would require ROC review of substantive changes to approved experiments. Please explain why this requirement is not part of the OSTR Technical specifications?

16. TS 4.8 is basically a restatement of the limiting conditions of operation (TS 3.8.2). TS 4.8 does not mention surveillance or inspection requirements for any experimental apparatus. TS 4.8 should contain requirements for inspecting the appropriate experiments periodically for material degradation to prevent failures, if the experiment is installed in the reactor beyond some pre-determined time frame, as per the guidance in Regulatory Guide 2.2, paragraph C.2.f, Radiation Sensitive Materials.

17. The NRC SRP, App. 14.1, item 6.3, Radiation Safety states that the TSs should state management's commitment to practice an effective ALARA program. This is currently not in TS 6.3. Please address.

18. The NRC SRP and ANS 15.1 specify that the purpose of the review committee is to provide independent oversight. The SRP also states that it is desirable to have members on the committee who are not employed by the reactor owner, and that the operating staff should not constitute the majority of a quorum. The ROC as described in the SAR has all of the reactor line management on the ROC, including Level 1, 2, & 3. This does not appear to provide the desired independence. It is unusual to have the Level 1 manager on the committee, since that is the one that appoints the committee and the one to whom the committee typically reports its independent oversight results. Also, neither the SAR nor the TS address the quorum requirements or the membership of persons not employed by OSU. It would appear from the membership that operations line management would usually be a majority of the committee. Please address.

19. Section 12.2.4 of the SAR discusses the audit function at the OSTR and includes audits of reactor operating areas and records, and reportable occurrences. TS 6.2 requires the performance of an annual audit. However, a number of areas delineated in the SRP and ANS 15.1 for audit are not specified in the TS or the SAR, namely: conformance with the TSs, actions taken to correct deficiencies, emergency plan and implementing procedures, security plan, facility procedures, experiments, surveillances, and the training program. The annual audit of the health physics is listed in Chap. 11 of the SAR but should also be in the TSs. Also, the SAR should specify that no individual responsible for an area may conduct the audit. Please address.

20. Section 12.3 of the SAR discusses procedures. The process in TS 6.4 for minor changes to reactor safety procedures is reasonable, but the separate process for minor ("unsubstantive" is used in the TS) changes to radiation protection procedures is not. Guidance in ANS 15.1 is for such changes to be made by Level 3 management with

approval by Level 2 in 14 days. TS 6.4 allows these changes with no prior approval and final approval by the Senior Health Physicist (Level 2) in 120 days. Please address.

21. The unusual need to deviate from procedures to deal with special circumstances is addressed in ANS 15.1, which states that such actions should receive approval of the SRO and be subsequently documented and reported to Level 2 management. SAR Section 12.3 addresses this need but does mention any controls such as those stated in ANS 15.1. Please address.

22. TS 6.5.1.c gives the required content for the report to the NRC in the event of a Safety Limit violation. This area is not discussed in Chapter 12 of the SAR. The TS agrees with the recommendations of ANS 15.1 with the exception that it does not specify that the report contain the effects (if any) of the violation on the health and safety of personnel and the public. Also in TS 6.6.2.a.1 the word "accidental" should be deleted. And TS 6.6 should note that duplicate telephone and written reports should also be made to the NRC Regional Office. Please address.

23. SAR Section 13.2.3 discusses a loss of coolant accident. What are the expected integrated doses that could be received by the staff and the general public from such an event?

24. The licensee did not specifically discuss prior use of reactor components for the OSTR in the SAR. SRP Chapter 16.1 discusses prior use of components and that fact that degradation mechanisms should be addressed. This is clearly applicable to a license renewal situation. Relevant information on prior component use and how deterioration of existing components is judged not to be a problem during the license renewal period. Please discuss reactor components that are continuing to be used for the OSTR that perform a safety function and address the concerns identified in SRP Section 16.1.