OSU
Oregon State
UNIVERSITY

Radiation Center
Oregon State University, 100 Radiation Center, Corvallis, Oregon 97331-5903
T 541-737-2341 | F 541-737-0480 | http://ne.oregonstate.edu/facilities/radiation_center

Designated Original
Redacted version
Public Available

July 27, 2007

Mr. Alexander Adams
U. S. Nuclear Regulatory Commission
Research and Test Reactors Branch A
Office of Nuclear Reactor Regulation
Mail Stop O12-G13
One White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

Reference: Oregon State University TRIGA Reactor (OSTR)
Docket No. 50-243, License No. R-106
Request for Additional Information (RAI) Regarding License Renewal, Oregon State University TRIGA Reactor (TAC NO. MC5155) dated May 21, 2007

Subject: Additional Oregon State University Response to RAI Regarding License Renewal, Oregon State University TRIGA Reactor dated May 21, 2007

Mr. Adams:

In a letter dated May 21, 2007, the U.S. Nuclear Regulatory Commission (NRC) requested that Oregon State University (OSU) provide additional information with regards to the OSU license renewal application of October 5, 2004, as supplemented. Enclosed is a further OSU response to the request. If you have any questions, please call me at the number above. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 7/27/07.

Sincerely,

Steve Reese
Director

Enclosure

cc: Document Control, NRC
Al Adams, NRC
Craig Bassett, NRC
John Cassady, OSU
Rich Holdren, OSU
Todd Palmer, OSU
Mike Hartman, OSU

A020

Oregon State University

Further Responses to RAI Letter of May 21, 2007

1. **Section 3.4. What is the relationship between the UBC 1964 Zone 3 seismic requirements and the maximum ground accelerations given in Table 2-4?**

   Unfortunately, because the accelerations listed in Table 2-4 are peak ground accelerations and accelerations of the 1964 UBC method are building design level accelerations, one cannot do a direct comparison without further calculation. The 1964 UBC does not have a method to calculate peak ground accelerations to compare to Table 2-4, but the building design level forces can be calculated using 2006 IBC criteria to provide a basis of comparison with the 1964 UBC method.

   The criteria calculated using the 1964 UBC results in a building design level acceleration. Design level acceleration is a scaled form of acceleration that includes methods to account for the building response characteristics. This scaling is intended to give forces that are comparable with those observed in actual events and testing. The different Codes have used methods of different sophistication for this scaling. The 1964 method was very rudimentary, while the most recent 2006 IBC methods are more sophisticated and include factors to account for soil response characteristics and the hazard to the public of the building occupancy which were not in the 1964 method. A comparison of the design level forces follows.

   The 1964 UBC method is a single prescriptive formula that results in an acceleration that is intended for use in building element design. The seismic response characteristics of the structure and empirically observed behavior are built into the formulas. The maps included in the 1964 UBC indicate the area in zone 2 (zones ranged from 0 to 3, with 3 being the zone of highest seismic concern), however zone 3 was conservatively used. The resulting building design level was calculated to be 0.2W. Using the 2006 IBC and corresponding site specific data now available, the design level acceleration was calculated to be 0.14W. The 2006 IBC value is lower and also significantly more representative. This means that the facility still conforms to current seismic design level standards.

2. **Section 4.2.1.9. The discussion of fuel swelling at high burnups refers to the agglomeration of fission gases at room temperatures above 1300°F. This swelling is time and temperature dependent. Provide a discussion if there should be a steady state temperature limit to control this type of swelling.**

Reference 4.1 shows that the swelling as a function of time increases with increasing time and increasing temperatures. The tendency for the curve to flatten out as temperatures decrease would suggest that swelling at our current steady-state operating temperature of approximately 350°C would be minimal, if present at all. The same conclusion could also be reached from the correlation between swelling and temperatures at end of life. However, the limited safety system setting for fuel temperature at 510°C is sufficiently below the 705°C temperature referenced that this type of swelling is precluded.

3. **What is the maximum fuel element power for possible core loadings (#8)? What are the peaking factors? Tables 4-11 and 4-12 only contain average power per element.**

The power per fuel element was calculated using MCNP5 with a model for a core similar to core #8 at a power of 1.1 MW. The new (MCNP5) model represents the reference HEU core (i.e., the original core loaded in 1976) and contains 4 less elements than core #8. The MCNP5 model for the reference HEU core was found to have very good predictive capability over a wide range of reactor conditions and was deemed to be well-suited for performing neutronics calculations. The results of this model can be seen in the following table:

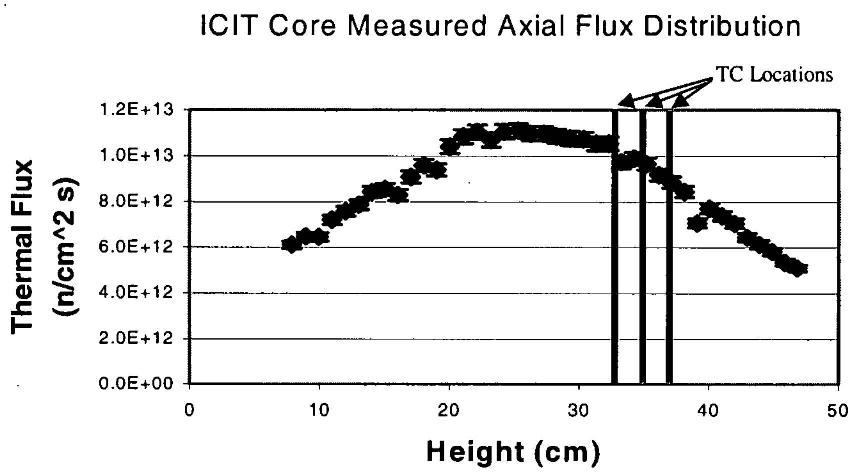
OUO Info Redacted

4. **TS 2.2. Discuss the derivation of the limited safety system setting (LSSS) value of 510°C. Discuss how LSSS protects fuel from exceeding the safety limit considering issues such as instrumented fuel element placement in the core versus the core hot spot, the thermocouple placement in the instrumented fuel element versus the fuel element hot spot, the accuracy of the measuring instrumentation and transient behavior of the reactor safety system.**

The value of the LSSS is designed to protect the [redacted] fuel from exceeding the maximum fuel temperature safety limit (SL) of 1,150°C for [redacted] fuel during non-pulsing reactor operation. It is not applicable to pulsing operations. The value of the LSSS at 510°C was conservatively chosen to be slightly lower than half the SL to account for uncertainties in measurement.

Based upon the analysis described in the answer to question three above, the highest power per fuel element location for the most reactive core was found to be in locations B-3 at a value of [redacted] kW. The Instrumented Fuel Element (IFE) located in grid position B-4 has a calculated power per fuel element of [redacted] kW. This is a difference of approximately 3.5%. The IFE is calibrated annually. Experience has shown that 5% error or less in the true temperature is commonly observed. Thus, the calculated difference in the power per elements is well within the error of the measurement.

Axial flux measurements were made for the In-Core Irradiation Tube (ICIT) core configuration. The results of these measurements are shown in the figure below along with the locations of the three thermocouples within the IFE. The error associated with the flux measurements are slightly larger than the symbols used. The thermocouple at the lowest elevation (which reads highest during steady state operation) is exposed to flux levels no more than 6% less than peak axial flux levels. Per Figure 7.2 in the SAR, the thermocouples are located in the fuel meat halfway between the outside edge of the fuel and the inside edge of the ZrH rod on the interior of the fuel, 0.425 inch from fuel center.



Typical fuel temperatures observed at full power are approximately 350°C. The analysis in section 13.2.2.2.1 shows that an uncontrolled withdrawal of a control rod at an initial power level of 1 MW would result in a trip signal being initiated within 0.28 seconds resulting from a reactivity insertion of

\$0.15. For an uncontrolled withdrawal of a control rod at an initial power level of 100 W, the trip signal would be initiated in 5.06 seconds resulting from a reactivity insertion of \$1.06. Because fuel temperature lags behind power and the power is so low, each of these scenarios would result in high power trips before the fuel temperature trip is reached. This is confirmed by our experience of observed instrument behavior after a pulse. For the loss of coolant accidents described in section 13.2.3, the primary water temperature would trip the reactor or the low level alarm would annunciate and alert the operator long before enough water is lost to initiate a high fuel element temperature trip. Regardless, section 13.2.3.2.2.1 clearly shows that natural convective air cooling of the fuel will keep the maximum fuel temperature well below the SL even after an instantaneous complete loss of primary water at 1.5MW or below.

The TC is located 0.3 inches from the edge of the central zirconium pin, thus giving it an axial position of 0.425 inches. The actual temperature distribution in the fuel pin will be quadratic (convex downward). Assuming a water temp of 30°C, no temperature drop across the clad, temperature at the TC of 350°C, and using a parabolic curve fit, the temperature at the inner radius of the fuel is calculated to be:

$$T(r) = 501.4 - 838r^2$$
$$T(0.425) = 488.3°C$$

The last step needed is to show that given all the uncertainties, measurement errors and biases that the LSSS is sufficient to guarantee that the SL is never exceeded. In other words calculate

(nominal IFE temperature) x (max reactor power / nominal reactor power) x (max power per element in the B-ring / nominal IFE power) x (peak azimuthal flux / nominal azimuthal flux) x (temperature measurement uncertainty factor)

$$(350°C) \times (1.10) \times (1.04) \times (1.40) \times (1.05) = 589°C$$

For a nominal IFE temperature of 510°C (i.e., the LSSS), the calculated value is 858°C. Both of these values are significantly less than the 1150°C. This assures that the IFE can adequately protect the SL during steady-state operations.