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April 14, 2008

Mr. Alexander Adams
U. S. Nuclear Regulatory Commission
Research and Test Reactors Branch A
Office of Nuclear Reactor Regulation
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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. MD7360) dated March 19, 2008

Subject: Oregon State University Response to RAI Regarding License Renewal, Oregon State University TRIGA Reactor dated March 19, 2008

Mr. Adams:

In a letter dated March 19, 2008, you requested that Oregon State University (OSU) provide additional information related to our license renewal submitted on October 5, 2004, as supplemented. Answers to the RAI can be found in the attached enclosure.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 4/14/08.

Sincerely,

Steve Reese
Director

Enclosure

cc: Document Control, NRC
Craig Bassett, NRC
John Cassady, OSU
Rich Holdren, OSU
Todd Palmer, OSU
Todd Keller, OSU (w/o Enclosure)

A020

Oregon State University

Responses to RAI Letter of March 19, 2008

1. **Were any data (like excess reactivity or control rod worth) available at other than BOL to use to benchmark the neutronic model for the HEU core? If so, why weren't they used?**

   The HEU FLIP core that was loaded in 1976 and served as the HEU modeling basis was only operated for a short duration before the core was significantly reconfigured. As such, the available plant data beyond the BOL benchmark is inconsistent with the HEU modeling basis and unsuitable for direct comparison with the model. The overall features of the HEU depletion curve are in accord with those experienced during actual operation of the facility. The ability of the HEU neutronic model to accurately predict measured BOL data, combined with its ability to properly account for the gross features of the burnup profile for the HEU core, give us confidence in the validity of the modeling technique.

2. **Table 1 – Why does the calculated transient rod worth deviate from the measurement for the HEU core at BOL while the other three control rods showed much closer agreement between calculated and measured worth?**

   A review of the rod calibrations performed on the HEU BOL core shows that the reactor period (and hence reactivity) measurements were based upon observing an increase in reactor power by a factor of 1.5. The time for the increase was measured with a stopwatch and the power trace was recorded on a chart recorder. Given the crude nature of this procedure, it is likely that either the recorded time or the actual power levels had associated errors. As an example, consider the calibration of a rod using a power increase by a factor of 1.5 from 40 percent to 60 percent of a measurement scale. Here the time of the power increase is taken to be 8.0 seconds with a Beta-effective of 0.0070 and a prompt-neutron lifetime of 30 μsec (Note: these are representative values from the rod calibration for the HEU BOL core). If the initial power is assumed to be exactly the same for two hypothetical measurements, but the final power in one case is 59% while that in the second is 61%, a calculated reactivity difference of $0.012 results. Likewise, an error in timing of 0.1 second, from 8.0 seconds to 7.9 seconds, results in a calculated reactivity difference of $0.002. If one assumes an error of $0.015 for each rod pull, this corresponds to an error of ~5% for each measurement. The transient rod calibration consisted of a total of eight individual measurements with an error of ~14%. The difference between the MCNP5 calculated and the measured HEU BOL value for the transient rod worth is 26.6%.

   The MCNP5 calculation of the transient rod worth used the rod positions from the HEU BOL testing data, hence the discrepancy between the predicted and measured values

cannot be attributed to systematic error (e.g. rod shadowing effects) between the two methods.

The exact cause of the discrepancy cannot be determined, however, one likely reason is the uncertainty associated with the initial HEU BOL measurements.

3. **Table 1 – Total measured rod worth data is inconsistent with data shown in the SAR Table 13-12. Do the data reflect different sets of measurements?**

   Yes, the data in Table 1 represents data taken and calculated at BOL while SAR Table 13-12 represents data typically seen in 2004 for the HEU normal core configuration.

4. **Table 2 – Please elaborate on the aspects of the evaluation techniques that contribute to the discrepancy between the measured and calculated shutdown margin for the HEU BOL core when the regulating rod is fully withdrawn. How does the calculated shutdown margin compare with the measured value for the other control rods?**

   The discrepancy is due to the difference by which core excess was calculated and how it is actually measured. For the calculated MCNP5 model, one rod is completely withdrawn while the remaining three are completely inserted into the core. This calculation method allows determination of the shutdown margin exactly as defined. This method, however, cannot be used to actually measure shutdown margin because reactivity cannot be directly measured by the period method when the core is significantly subcritical. To measure core excess, the reactor is brought to 15 W with three rods at a specified withdrawal height, and the fourth rod withdrawn to criticality. The reactivity value of the three rods at their reference height is known from their calibration curves. The value of the fourth rod at its critical height is also determined from its calibration curve. The three rods are typically 50% withdrawn which creates shadowing and rod worth effects that can't be otherwise accounted for. The inherent difference between the two methods was not calculated. The shutdown margin for rods other than the most reactive rod was not calculated because, by definition, the shutdown margin requires that the highest worth rod is fully withdrawn from the core.

5. **In Table 3 the hot channel peak factor for BOL is listed as 1.442 with corresponding thermal power of 18.02 kW. For MOL the thermal power is increased to 18.37 kW while the peak factor decreased to 1.420 indicating a potential inconsistency, if the number of fuel elements remained the same. Please clarify.**

   This was a typographical error. Table 3 should be replaced with the following:

Table 3 Hot-Rod Power Summary.

Hot Channel Power Summary						
	Hot Channel Location	Hot Channel Thermal Power [kW]	Relative Error Associated to Thermal Power [%]	Hot Channel Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Axial Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Radial Peak Factor [P_{max}/P_{avg}]
FLIP-BOL NORMAL Core	B3	18.02	0.0017	1.376	1.236	1.907
FLIP-MOL NORMAL Core	B6	18.37	0.0009	1.403	1.209	1.518
FLIP-EOL NORMAL Core	B6	16.48	0.0011	1.258	1.234	1.708

6. **What is the effect on power peaking of replacing a burnt fuel rod (e.g., an IFE) with a fresh fuel rod? What controls are in place with respect to loading new fuel? Provide a calculation to indicate the change in the peaking factor when fresh fuel is placed in an EOL core at the worst location.**

   Conversion of the core from HEU to LEU fuel should happen in the next year. Given the timing of the conversion, this question is essentially moot as there is no anticipate need to replace any burnt fuel rod with a fresh fuel rod for the HEU core in the coming year. However, we agree to perform this calculation if such a change is needed. That being said, it is possible to change the core configuration in many different ways. These changes would be implemented through the processes allowed by existing procedures and that describe in 10 CFR 50.59.

7. **Please provide analytical justification for ignoring cross flow between neighboring flow channels.**

   One, two, and eight channel RELAP5-3D models were individually analyzed against the BOL core. Cross flow was incorporated in the two and eight channel models through junctions connected at each individual axial nodal location between adjacent subchannels. The axial and radial fuel temperature distributions were assumed to be identical in each model. This is because the hot channel fuel elements in each model produce the same internal heat generation rate, are made of the same material, and have the same geometry, therefore their temperature distributions are the same.

   Figure 1 provides a quantitative comparison of the bulk coolant temperature distribution found in the one, two and eight channel models. This comparison

provides evidence that with an increase in the number of subchannels a corresponding decrease in exit bulk coolant temperature will result as expected. The coolant equilibrium quality and subchannel mass flux are presented in Figure 2 as a function of axial position. These properties, along with system pressure, are the primary parameters for CHF. With an increase in the number of subchannels the mass flux is perturbed greater due to cross flow in the lower axial portion of the core. The equilibrium quality for the single, two, and eight channel models remain similar though the majority of the axial length of the core. As a result of the increased mass flux in the eight channel model near the exit of the subchannel, less energy is deposited into the fluid producing lower equilibrium quality values for the eight channel model relative to the two and one channel models.

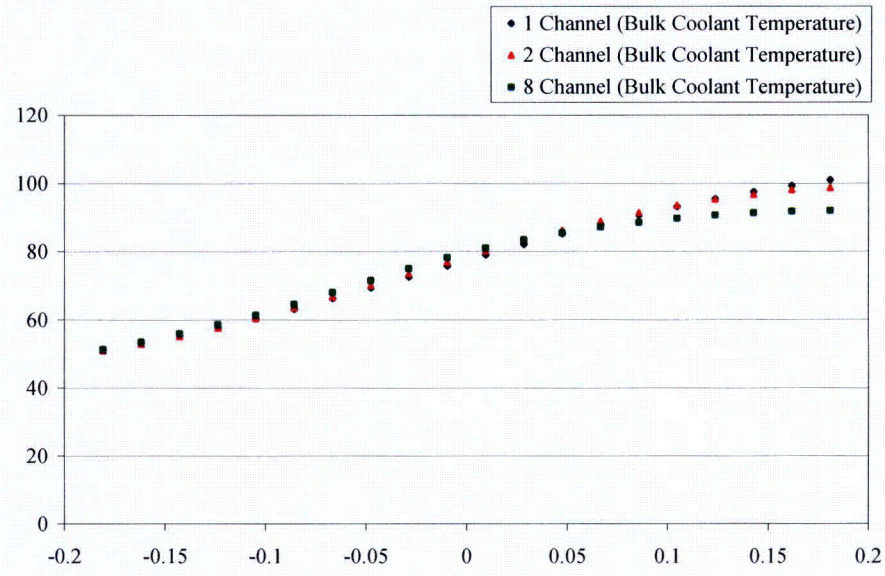

Figure 1: Axial Bulk Coolant Temperature Distribution for 1, 2, & 8 Channel Model

(HEU Beginning of Life Normal Core)

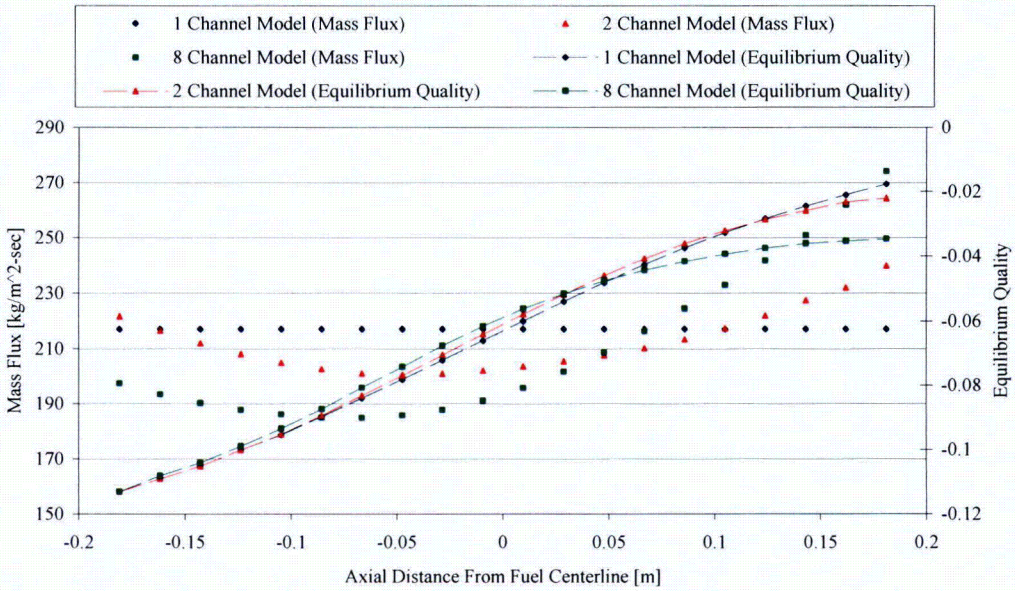

Figure 2: Mass Flux and Equilibrium Quality Distribution for 1, 2, & 8 Channel Model

(HEU Beginning of Life Normal Core)

Figure 3 presents the CHFR determined using the 2006 Groeneveld AECL look-up tables and Bernath correlation for the one, two, and eight channel models. Small differences in the CHFR values for the different models compared can be accounted for by the observations made above. The exit CHFR value for the eight channel model is larger than for the two channel model, which is larger than the one channel model. This is due to a decrease in equilibrium quality near the exit of the subchannel. Although there are small deviations in the CHFR axial distributions it is important to note that the minimum critical heat flux ratio (MCHFR) differs by only 0.3% on average for the look-up tables and 0.6 % for the Bernath correlation. This is well within the error margin associated with each CHF method of calculation.

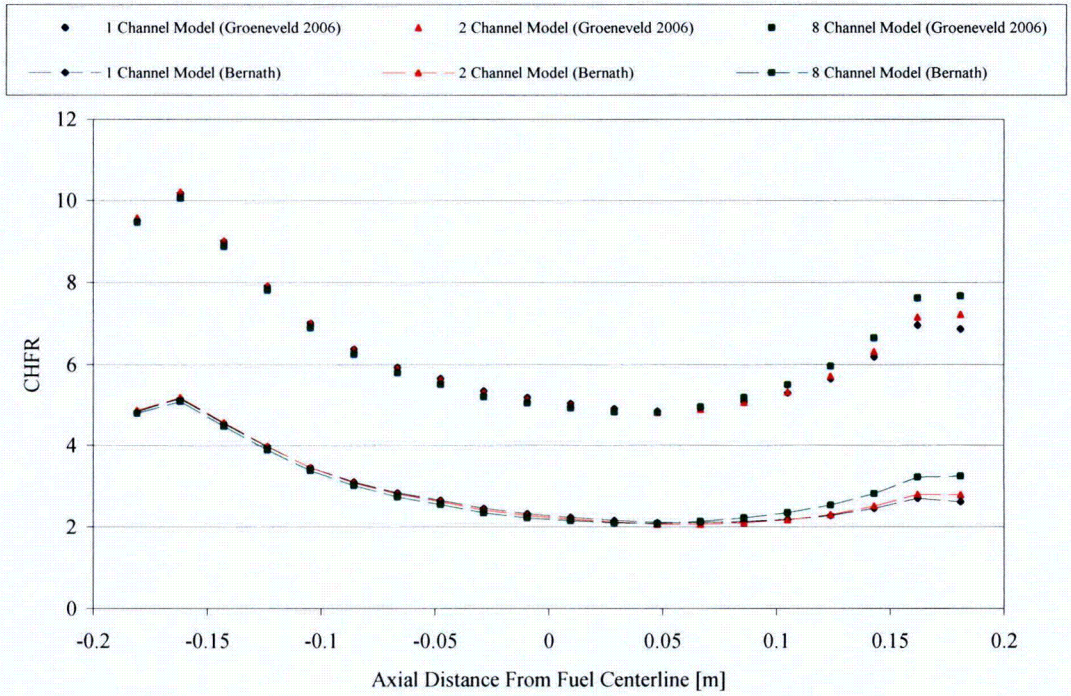

Figure 3: Axial CHFR Distribution for 1, 2, & 8 Channel Model

(HEU Beginning of Life Normal Core)

Based on a conservative method for safety analysis, the single channel model provides critical heat flux results using the 2006 Groeneveld look-up tables within ~1.0 % of those produced from the eight channel model. The single channel model produced the most conservative results relative to the two and eight channel models.

8. **Was the transient rod also removed for the calculation of the rod power distribution in the core?**

Yes, the transient rod was also removed for the calculation of the rod power distribution in the core.

9. **Does the peak power density (maximum local heat flux) at steady-state occur in the fuel element (rod) with the maximum rod power?**

The peak power density was assumed to occur in the fuel element with the maximum rod power in the analysis. The RELAP5-3D model utilized the smallest pitch in the core, the most restrictive flow channel, no cross flow between channels, and it assumed that all fuel elements bordering the most restrictive flow channel were also operating at the maximum rod power. Although additional calculations were not performed on lower power fuel elements, the conservative assumptions in the RELAP5-3D model were

implemented in order to compensate for potential variations in intrafuel fuel element power distributions.

10. Why did the thermal analysis not use the Groeneveld Look-up table build into RELAP5-3D?

Six different methods for calculating critical heat flux with reference to the core conversion study are considered. In order to quantitatively compare the relationship of these correlations, the HEU Beginning of Life NORMAL core was chosen to be analysed.

RELAP5-3D internally calculates the critical heat flux with reference to the 1986 AECL Groeneveld look-up tables. All other correlations were calculated externally using the thermal hydraulic properties resulting from the RELAP5-3D model. In order to verify that these external calculations were being completed in the correct manner, the critical heat flux was calculated externally with reference to the 1986 AECL Groeneveld look-up tables. The result of this comparison is shown in Figure 4. The external calculations produced values that deviated from those conducted in RELAP5-3D by ~1.0% on the conservative side. The deviation was potentially caused by round off error. Qualitatively, this deviation was estimated to be within an acceptable margin. The axial CHFR distribution for the RELAP5-3D internally calculated values as well as all external calculated values are presented in Figure 4.

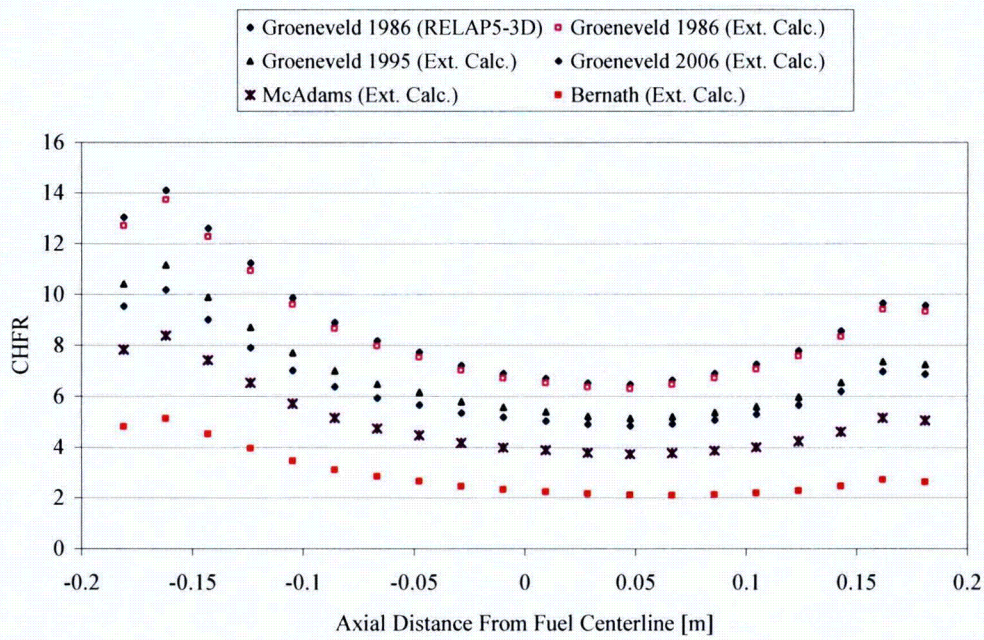

Figure 4: Axial CHFR Distribution, Correlation Comparison

(HEU Beginning of Life Normal Core using the single channel model)

As presented in Figure 4, the 1986 AECL Groeneveld look-up tables produce the least conservative CHFR distribution values, followed by the 1995 and 2006 Tables. The McAdams correlation and Bernath correlation, although having the largest error associated with them, produce the most conservative CHFR distribution values. Based on the results presented in Figure 4, the 2006 Groeneveld look-up tables and the Bernath correlation were used in the final version of the core conversion project. They were chosen for the following reasons:

Bernath -

- It is traditionally used as a supplement in research reactor SARs with respect to the RERTR program including the recent submission of the Washington State University Research Reactor [1], the University of Massachusetts Lowel Research Reactor [2], and the South African MNSR [3].
- The correlation produces the most limiting CHFR values over all other correlations considered during this study.

2006 AECL Groeneveld look-up Tables -

- The correlation is the most accurate method for calculating CHF values over all others considered during this study.

11. What is the number of HEU fuel elements assumed in calculating the hot rod peak factors for the HEU core in Table 3?

The number of HEU fuel elements assumed in calculating the hot rod peak factors for the HEU core in Table 3 was 85.

12. Are changes in the gap properties (e.g., gap size, oxidation at the fuel boundary, gas composition) over time taken into account when calculating fuel temperatures?

The gap properties were held constant over time for this analysis. For gap thickness, a sensitivity study looking at the affect of increasing gap thickness from 0.05 mil to 0.40 mil in 0.05 mil increments on radial fuel temperature was conducted. Calculated values of fuel temperature for a gap value of 0.10 mil were found to be just above that measured in the Instrumented Fuel Element at BOL, accounting for differences in element location and thermal couple position within the element.

The only gap thermal model used was that listed as the "gap" thermal model listed in the material properties table of RELAP5-3D. This model was chosen because it incorporates fission product build up in fuel elements. This will produce a larger thermal resistance compared with air and is conservative over the lifetime of the fuel.

Oxidization was not considered in the thermal analysis for OSTR fuel elements.

13. Have the consequences of pulsing the reactor from full power operation been considered? If yes, what are the results; if no, why not?

The HEU Beginning of Life Normal Core Configuration was analysed for consideration of the potential contingency that the OSTR were pulsed to its licensed limit of reactivity insertion ($2.55) from its maximum licensed steady state power level (1.1 MW_{th}). Figure 5 and Table 1 below summarize the results produced from RELAP5-3D using the HEU Beginning of Life fissile fuel characteristics.

Table 1 – Results of Pulsing at Full Power

LEU-EOL ICIT Core Configuration Pulse at 1 MW Study		
Initial Steady State Power [kW]	1.0	1100.0
Reactivity Insertion [$]	2.55	2.55
Peak Total Core Power [MW_{th}]	8473	4950
Time of Peak Total Core Power [sec]	0.0260	0.0151
Full Width at Half Maximum [Δsec]	0.0045	0.0052
Energy at 0.1 sec [MJ_{th}]	42.751	29.662
Peak Fuel Temperature [°C]	862.5	782.2
Time of Peak Fuel Temperature [sec]	0.030	0.024

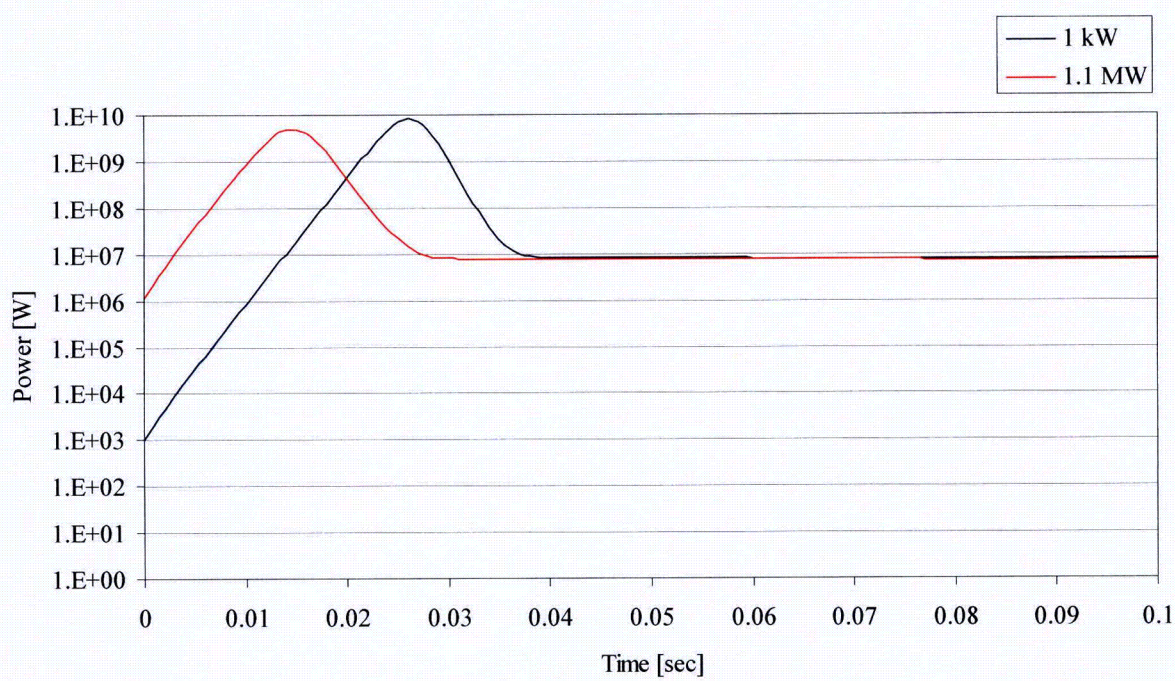

Figure 5 – Comparison of Pulsing from 1 kW and 1 MW

As a result of this comparison it is demonstrated that although the initial steady state power is much greater for the case where the reactor is pulsed from 1.1 MW_{th} it results in a lower peak power than that of the 1.0 kW_{th} initial steady state power. This is due to the temperature dependant prompt fuel temperature coefficient that is inherent to the OSTR fuel. However, pulsing from 1MW requires both a failure of the 1 kW interlock and a failure of the reactor operator to follow written procedures.

1. INC., G.A.E.S., *Safety Analysis for the HEU to LEU Core Conversion of the Washington State University Reactor*, N.R. Commission, Editor. 2007, General Atomics: TRIGA Reactors Division of General Atomics-ESI. p. 1-112.
2. Bousbia-Salah, A., et al., *Assessment of RELAP5 Model for the University of Massachusetts Lowell Research Reactor.* Nuclear Technology & Radiation Protection, 2006. **21**: p. 3-12.
3. Dunn, F.E., et al. *MNSR Transient Analyses and Thermal Hydraulic Safety Margins for HEU and LEU Cores Using RELAP5-3D Code*. in *2007 International Meeting on Reduced Enrichment for Research and Test Reactors*. 2007. Prague, Czech Republic.