UNIVERSITY OF WISCONSIN
NUCLEAR REACTOR
LICENSE NO. R-74
DOCKET NO. 50-156

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING HEU/LEU CONVERSION

REDACTED VERSION

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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April 10, 2009 RSC 1004

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Docket 50-156, License R-74
Response to Request for Additional Information
for Amendment No. 17 to Facility License No. R-74
University Of Wisconsin Nuclear Reactor
TAC No. MD9592

Dear Sirs:

By letter, dated February 26, 2009, the Commission has requested additional information in order to complete the review for the University of Wisconsin Nuclear Reactor's (UWNR) request to amend facility license number R-74 and technical specifications to facilitate the conversion of the reactor from high enriched uranium (HEU) to low enriched uranium (LEU) in accordance with 10 CFR 50.64(b)(2)(ii).

Enclosed are the responses to the request for additional information. The responses are provided in the same order as the Commission's requests. The format of the enclosure is to restate the request followed by the response. The original request is counter shaded to aid in the separation between request and response.

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

Sincerely,

Robert J. Agasie Executed on: 4-10-09
Reactor Director

Enclosure

A020
NRR

Responses to LEU Conversion Request for Additional Information

1. Sections 1.1, 1.3, 4.2.2, and 4.2.3. In Section 1.1 and 1.3, your application states that only the fuel and new fuel storage will be changed as part of the conversion, however in Section 4.2.2 you state that the transient rod guide tube will be replaced. Is the transient road guide tube being replaced as a part of the conversion? If so, please provide justification.

Licensee's Response:

Yes. The transient rod guide tube is being replaced as part of the conversion. This will minimize handling of the previously irradiated guide tube in order to keep staff doses ALARA. The new guide tube is constructed in accordance with GA drawing T4S210C152, as is the existing guide tube, and therefore is a direct like-for-like replacement. See attachment 1.

2. Sections 1.1, 1.3, 4.2.2, and 4.2.3. In Section 1.1 and 1.3, your application states that only the fuel and new fuel storage will be changed as part of the conversion, however in Section 4.2.3 you state four additional reflectors will be installed.

   a. Are additional reflectors being added as part of the conversion? If so, please provide justification.

Licensee's Response:

Yes. Four additional reflectors are necessary as part of the conversion. As described in section 4.5.2 (pages 37-38), a reduction in the number of fuel bundles, from 23 to 21, is necessary to ensure shutdown margin. One of the current approved HEU operational cores uses 12 reflectors (I23-R12), so a proposed LEU core was analyzed with 12 reflectors (J21-R12). However, as shown in figure RAI-2-1 below, a much reduced core lifetime would result when compared to the HEU core. Calculations show that the core lifetime of 150 MW-days (with subsequent recovery out to 1100 MW-days) using 12 reflectors can be increased to 1800 MW-days using 2 additional reflectors for a total of 14 reflectors.

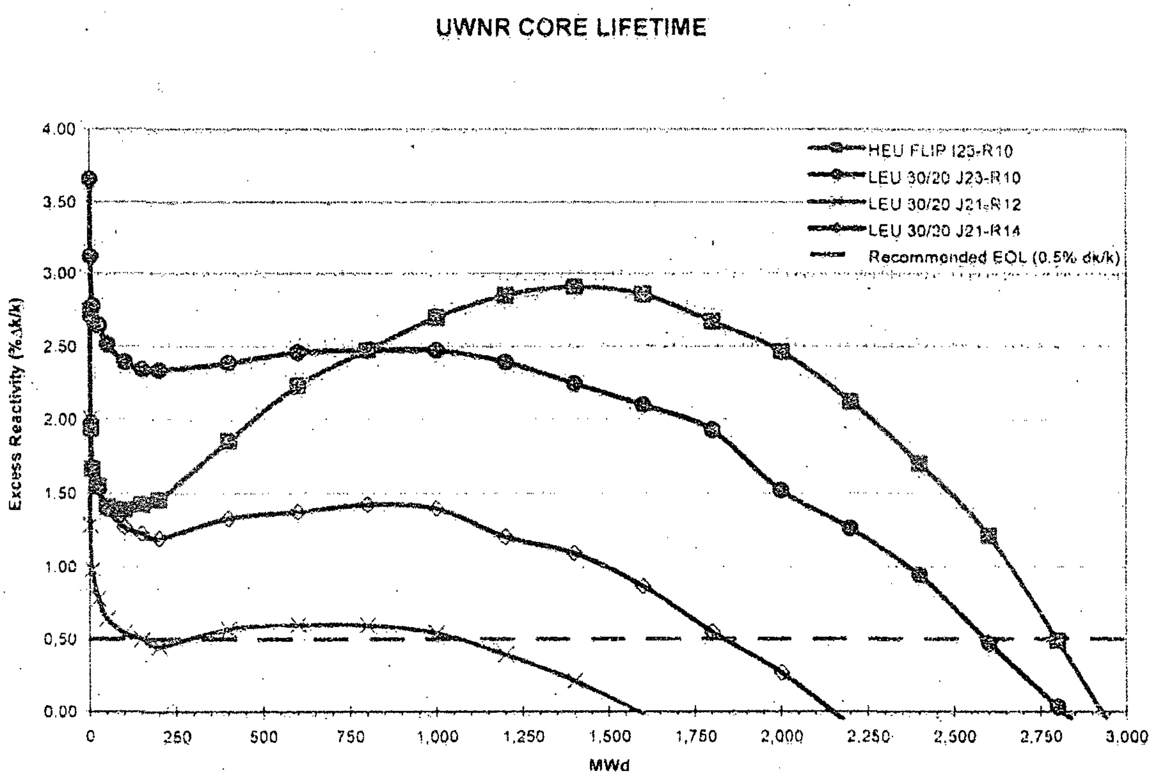

Figure RAI-2-1, UWNR Core Lifetime

b. Is the design of new graphite reflectors consistent with the design of the current reflectors? If not, please provide details.

Licensee's Response:

Yes. The new graphite reflectors are constructed in accordance with Idaho National Laboratory drawing 600855. The new reflector design is consistent with the existing reflector design GE drawing 612D489. The GE drawing was confirmed via measurement to match existing reflectors. See attachments 2 and 3.

3. Section 1.3: Are any other core components being added or changed as part of conversion? If so, please provide justification.

Licensee's Response:

Yes. As a MTR conversion type TRIGA reactor, the bottom adapter and top handle cluster hardware (to include locking plates and bolts) used to create the 4-element cluster are being replaced. This will minimize handling of previously irradiated hardware in order to keep staff doses ALARA. The existing and new bottom adapters are constructed in accordance with GA drawing T4S210D104. The existing and new top handles are constructed in accordance with GA drawings T4S210C101 (4 element) and T4S210D111 (3 element). See attachments 4, 5, and 6.

4. Section 4.2.1, Table 4.2.2: In Section 1.1 and 1.3, the units for uranium enrichment appears as percent U-235, however in Table 4.2.2 units of enrichment appear as atomic percent. Should enrichment be stated in units of weight percent?

Licensee's Response:

Yes. Enrichment should be stated in units of weight percent.

5. Section 4.2.3 and 4.2.5. In Section 4.2.5, your application states that no changes will be required to in-core experimental facilities however in Section 4.2.3 it states that graphite reflectors will be placed in positions D3 and D7 in lieu of currently installed irradiation baskets. Please clarify.

Licensee's Response:

There are currently two approved operational HEU cores, I23-R10 and I23-R12. The difference between these cores is the replacement of the irradiation baskets in D3 and D7 with graphite reflectors. Because the I23-R12 operational HEU core already allows removing the irradiation baskets, this is not a change to an experimental facility related to the LEU conversion. However, this application does not seek to approve the proposed J21-R14 LEU core with irradiation baskets in D3 and D7. Any future core modifications would be performed under 10 CFR 50.59 and existing procedures.

6. Section 4.5.1: In Table 4.5.3, your application states the calculated integral worth of the transient rod is 1.467 %Δk/k, however on page 27 you state the value is 1.334 %Δk/k (BOL, HEU). Please discuss the difference between these two values.

Licensee's Response:

The value of 1.334 ± 0.0453 %Δk/k (page 27) was calculated in MCNP by modeling the rod drop methodology of measuring reactivity. The value of 1.467 ± 0.105 %Δk/k (page 32) was calculated using MCNP and curve fitting by modeling the rising period rod bump methodology of measuring reactivity. The difference between these two methodologies accounts for the importance of the flux shape on the measurement of the worth of a control element. These values are identical within their respective uncertainties and agree well with the measured value of 1.374 %Δk/k (page 32).

7. Section 4.5.1. Are there any other measurements that have been performed on the reactor which could be used to help benchmark the MCNP model with the present HEU fuel?

Licensee's Response:

The only data available from the first all-FLIP HEU core is in the core loading report from January 1980 consisting of differential and integral control element worth curves and axial plots of detector response in a number of fuel bundle locations. While comparisons to this data are possible, they do not result in a quantitative indication of computational bias. Such computational bias is routinely established by comparing simulated results of known critical configurations. The discrepancy between the simulated eigenvalue and the experimental eigenvalue (k_{exp} = 1 by definition) is used to establish the bias. Comparison of simulated axial distributions, whether control element worth or detector response, does not provide the same quantitative basis. Discrepancies between such results can vary axially and do not indicate a specific bias in the eigenvalue. Furthermore, the control element worth curves are based on a manual fit to a small number of measurements and while a record exists of the curve generated by that manual fit, the measurements themselves are not recorded. Finally, the axial detector response data is in a form that does not give a clear indication of how it was measured and what simulation technique would be most appropriate for comparison.

8. Section 4.5.1. In Tables 4.5.4, 4.5.5, 4.5.6, 4.5.7, and 4.5.8, the units for differential worth curve are given as [%Δk/k in]. Should the units be [%Δk/k / in]?

Licensee's Response:

Yes. The units should be [%Δk/k / in]. Note that within section 4.5.1, these units appear in Figures 4.5.4, 4.5.5, 4.5.6, 4.5.7, 4.5.8, and Table 4.5.3 (pages 30-32).

9. Section 4.5.1/4.5.2. In Tables 4.5.6 and 4.5.13, the units for the void coefficients are given as [Δk/k / %void]. Is that correct or should the units be [% Δk/k / %void]?

Licensee's Response:

The reported units of [Δk/k / %void] are correct.

10. Section 4.5.1/4.5.2. In Tables 4.5.6 and 4.5.13, the void coefficient are stated as negative values, however the coolant temperature coefficient are stated as positive values. Should the coolant temperature coefficient be stated as negative or positive values?

Licensee's Response:

The coolant temperature coefficients are positive as reported on pages 35 and 45. Although they are calculated as positive with MCNP, the values are small and comparable to values reported in the HEU SAR. Previous attempts to experimentally measure the value have been difficult because raising the coolant temperature will also raise the fuel temperature, and the negative fuel temperature coefficient is much larger than the calculated coolant temperature coefficient.

11. Section 4.5.1/4.5.2, in Tables 4-5.5 and 4-5.[illegible], the coolant temperature coefficients are stated in units of [Δk/k / K]. Is that correct or should the units be [%Δk/k / K]?

Licensee's Response:

The reported units of [Δk/k / K] are correct.

12. Section 4.5.1/4.5.2, in Tables 4-5.[illegible] and 4-5.[illegible], the units for the prompt temperature coefficients are given as [%Δk/k-K]. Should the units be [%Δk/k / K]?

Licensee's Response:

Yes. The units should be [%Δk/k / K].

13. Section 4.5.2. Please provide more specific information on the calculation of the shutdown margin for the LEU fuel. Demonstrate how the calculation satisfies the Technical Specification 3.1 requirement for shutdown margin of 0.2 % Δk/k?

Licensee's Response:

The shutdown margin for the LEU core was calculated to be 0.294 %Δk/k with the maximum allowable experiment installed, control blade 3 and the regulating blade fully withdrawn. This satisfies the requirements of Technical Specification 3.1. This calculation is based on the shutdown margin of 0.994 %Δk/k as reported on page 38 of the analysis report with no experiments and blade 3 and the regulating blade fully withdrawn. The Technical Specification shutdown margin was calculated by subtracting 0.7 %Δk/k, which is the maximum allowable reactivity of a non-secured experiment, to arrive at 0.294 %Δk/k.

14. Section 4.7.1. RELAP5/MOD3.3 had a fundamental error in the point kinetics model that has recently been fixed. Does the version of the code used in the conversion analysis have the fixes implemented? If not, confirm that the UWNR analysis model is giving results consistent with the transient, e.g. by checking results as a function of time step or with another stand-alone point kinetics model.

Licensee's Response:

No, the RELAP5/MOD3.3 code used does not have the fixes implemented for the fundamental error in the point kinetics model. In 2007 Idaho National Laboratory reported that errors had been found and corrected in the point kinetics routine of RELAP5-3D, which uses the same point kinetics routine as RELAP5/MOD3.3. Two changes in the point kinetics routine were made. The first change was related to an index used in the calculation of the delayed neutron precursors. The second change related to logic that determined when to apply a quasi-steady form of the point reactor kinetics equation. In 2008 INL decided that the first correction was erroneous: the index was correctly calculated in the original code. Now INL recommends applying only the second change.

Argonne National Laboratory has the source code for the UNIX version of RELAP5-3D/ version2.3.7t and has compiled the original version of the code with no point kinetics correction, as well as versions with the 2007 corrections and the 2008 corrections. To investigate the impact of these corrections to the code, calculations were made for a \$2.0 step reactivity insertion transient in the University of Wisconsin reactor using all three code versions. Table RAI-14-1 shows powers calculated by these three versions at times near the power peak. The uncorrected version and the latest version give results that are identical to 6 significant figures. The 2007 version gives results that differ from the others by about 1%. Thus, for pulse calculations of interest for the University of Wisconsin reactor the differences between the uncorrected results and the results from the latest version are non-existent or negligible. Even the differences between the 2007 version and the latest version are minor.

Table RAI-14-1, Calculated Powers for a \$2.0 Step Reactivity Insertion in the University of Wisconsin Reactor

Transient time, s	0.035	0.040
Reactor power, GW, for the 2008 code corrections	1.00128	2.57338
Reactor power, GW, for the 2007 code corrections	0.990375	2.59909
Reactor power, GW, for the uncorrected code	1.00128	2.57338

15. Section 4.7.1. Is there a TS or administrative control on the maximum allowable bulk coolant temperature? If not, please discuss why a limit on bulk coolant temperature is not needed.

Licensee's Response:

Yes, the maximum allowable bulk coolant temperature is administratively controlled by an automatic scram from the reactor protection system if the temperature reaches 130°F. However, adding a Technical Specification for pool water temperature is consistent with Table 1 of Technical Specification 3.3.3. The proposed change is included in response to question 56.

16. Section 4.7.2 In Table 4.7-1. What is the sensitivity of the steady-state results, such as flow rate and critical heat flux, associated with the uncertainty in the derived inlet and outlet pressure loss coefficients?

Licensee's Response:

A sensitivity study on the impact of the maximum fuel temperature, exit bulk temperature, flow rate, and power to CHF and MDNBR as a function of the inlet and outlet pressure loss coefficients at 1.5 MW for HEU-BOL has been provided in the tables below. The inlet and outlet pressure loss coefficients were independently altered by ±20%.

Table RAI-16-1, Altering the Lower Pressure Loss Coefficient Only

Adjustment	-20%	Nominal	20%
Lower Pressure Loss Coefficient	1.616	2.020	2.424
Max Temperature (°C)	642.03	642.03	642.03
Max Clad Temp (°C)	140.90	140.90	140.90
Exit Bulk Temp (°C)	99.66	100.44	101.12
Mass Flow Rate (kg/s)	0.13904	0.13665	0.13462
% difference	1.75%	0.00%	-1.49%
Power to CHF using Groenveld 2006 (kW)	52.770	52.376	52.036
Power to CHF Bernath (kW)	34.299	33.993	33.730
MDNBR Groenveld 2006	1.998	1.983	1.971
% difference	0.75%	0.00%	-0.65%
MDNBR Bernath	1.299	1.287	1.277
% difference	0.90%	0.00%	-0.77%

Table RAI-16-2, Altering the Upper Pressure Loss Coefficient Only

Adjustment	-20%	Nominal	20%
Upper Pressure Loss Coefficient	1.104	1.380	1.656
Max Temperature (°C)	642.02	642.03	642.03
Max Clad Temp (°C)	140.90	140.90	140.91
Exit Bulk Temp (°C)	99.90	100.44	100.94
Mass Flow Rate (kg/s)	0.13829	0.13665	0.13517
% difference	-1.20%	0.00%	1.08%
Power to CHF using Groenveld 2006 (kW)	52.647	52.376	52.129
Power to CHF Bernath (kW)	34.204	33.993	33.802
MDNBR Groenveld 2006	1.994	1.983	1.974
% difference	0.52%	0.00%	-0.47%
MDNBR Bernath	1.295	1.287	1.280
% difference	0.62%	0.00%	-0.56%

Table RAI-16-3, Altering the Both the Upper and Lower Pressure Loss Coefficients

Adjustment	-20%	Nominal	20%
Upper Pressure Loss Coefficient	1.104	1.380	1.656
Lower Pressure Loss Coefficient	1.616	2.020	2.424
Max Temperature (°C)	642.02	642.03	642.03
Max Clad Temp (°C)	140.89	140.90	140.91
Exit Bulk Temp (°C)	99.04	100.44	101.56
Mass Flow Rate (kg/s)	0.14099	0.13665	0.13336
% difference	-3.18%	0.00%	2.41%
Power to CHF using Groenveld 2006 (kW)	53.086	52.376	51.822
Power to CHF Bernath (kW)	34.545	33.993	33.565
MDNBR Groenveld 2006	2.010	1.983	1.962
% difference	1.35%	0.00%	-1.06%
MDNBR Bernath	1.308	1.287	1.271
% difference	1.63%	0.00%	-1.26%

As seen in these tables, altering the lower or upper pressure loss coefficient does not have a significant impact on either the temperature, mass flow rate, or the MDNBR. This is consistent with previous analyses by General Atomics, "TRIGA Reactor Thermal-Hydraulics Study STAT-RELAP5 Comparison," TRD 070.01006.04 (April 2008).

17. Section 4.7.4: In Figure 4.2-1, the location of one instrumented fuel element is shown as E3 NE. The title of Figure 4.7.10 indicates that the instrumented fuel element may have moved from E3 NE to E4 SE for these temperature measurements. Please explain.

Licensee's Response:

During startup testing with the all TRIGA-FLIP HEU core, the fuel temperatures in one quarter of the core were measured as reported in the HEU 2000 license renewal SAR, page 4-45. These measurements were done in support of the reload and startup testing. Once startup testing was complete, the operational core was chosen to have IFEs located in D4 SW and E3 NE. In the analysis report it was chosen to compare the model with a pin having a similar peaking factor which was not the case for E3 NE. Therefore, E4 SE was chosen because both the model and the measured peaking factors were 1.10 as described on page 73.

However, data for E3 NE is available and the measured vs. predicted curve for the instrumented fuel element at E3 NE is shown in the figure below. The bottom thermocouple measurement was not reported in the reload report due to the thermocouple burning out during startup testing.

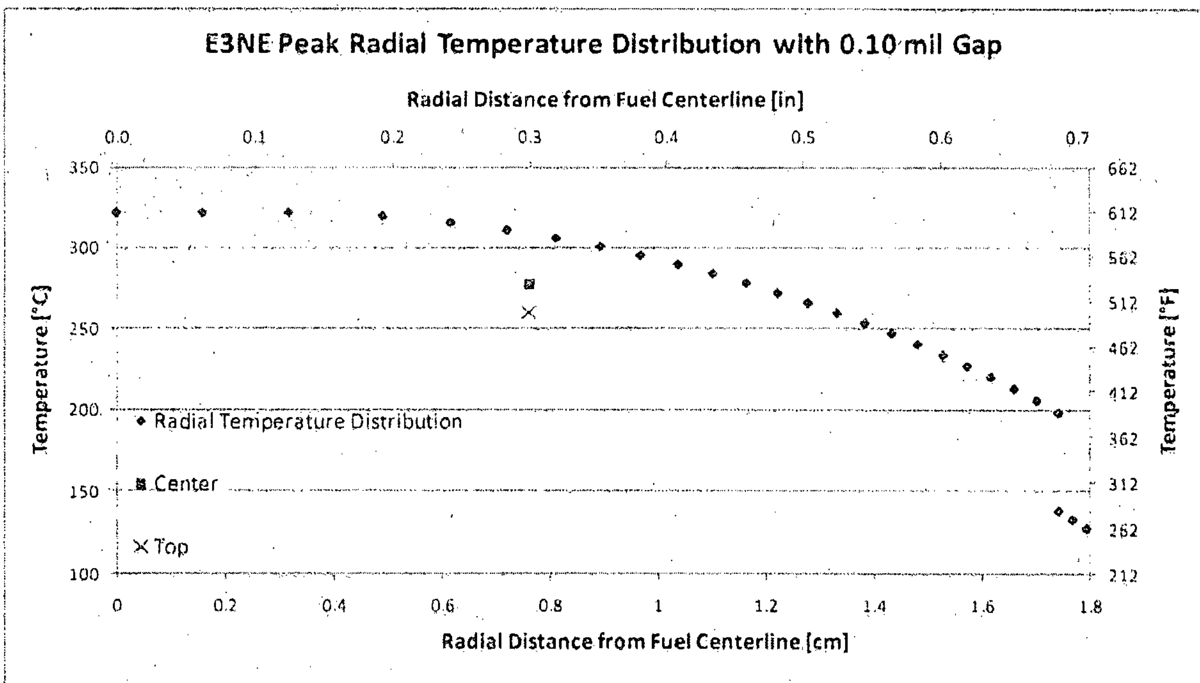

Figure RAI-17-1, E3 NE Peak Radial Temperature Distribution

18. Section 4.7.4: Are there comparisons of measured versus calculated instrumented fuel element temperatures at D4 SW?

Licensee's Response:

There are measured instrumented fuel element temperatures at D4 SW as seen in the figure below.

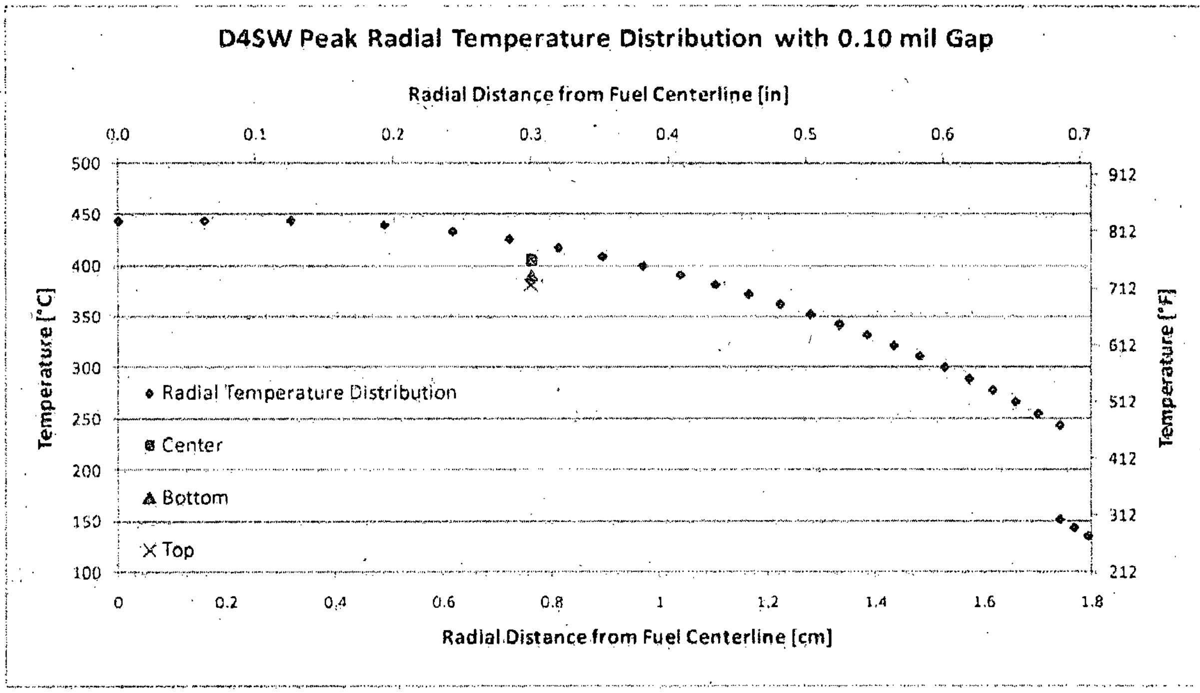

Figure RAI-18-1, D4 SW Peak Radial Temperature Distribution

19. Section 4.7.4: What is the axial location of the radial temperature results shown in Figure 4.7.11?

Licensee's Response:

Figure 4.7.11 on page 77 shows the radial temperature profile where the highest axial power peaking factor is, 5.5 inches (13.97 cm) from the bottom of the active fuel.

20. Sections 4.7.7 and 4.7.8: Please explain why the maximum fuel temperature for the LEU core is the highest at MOL and yet the hot rod power at MOL is lower than that at BOL. Also, please discuss the opposite trends in the MDNBRs, shown in Tables 4.7.13 (LEU-BOL) and 4.7.16 (LEU-MOL), as calculated by the Groneveld 2006 and the Bernath correlation.

Licensee's Response:

An error was discovered with the LEU-BOL axial power shape from MCNP5 that was input into the RELAP5 LEU-BOL models. The following figure shows the comparison between the original axial power shape and the revised MCNP5 axial power shape.

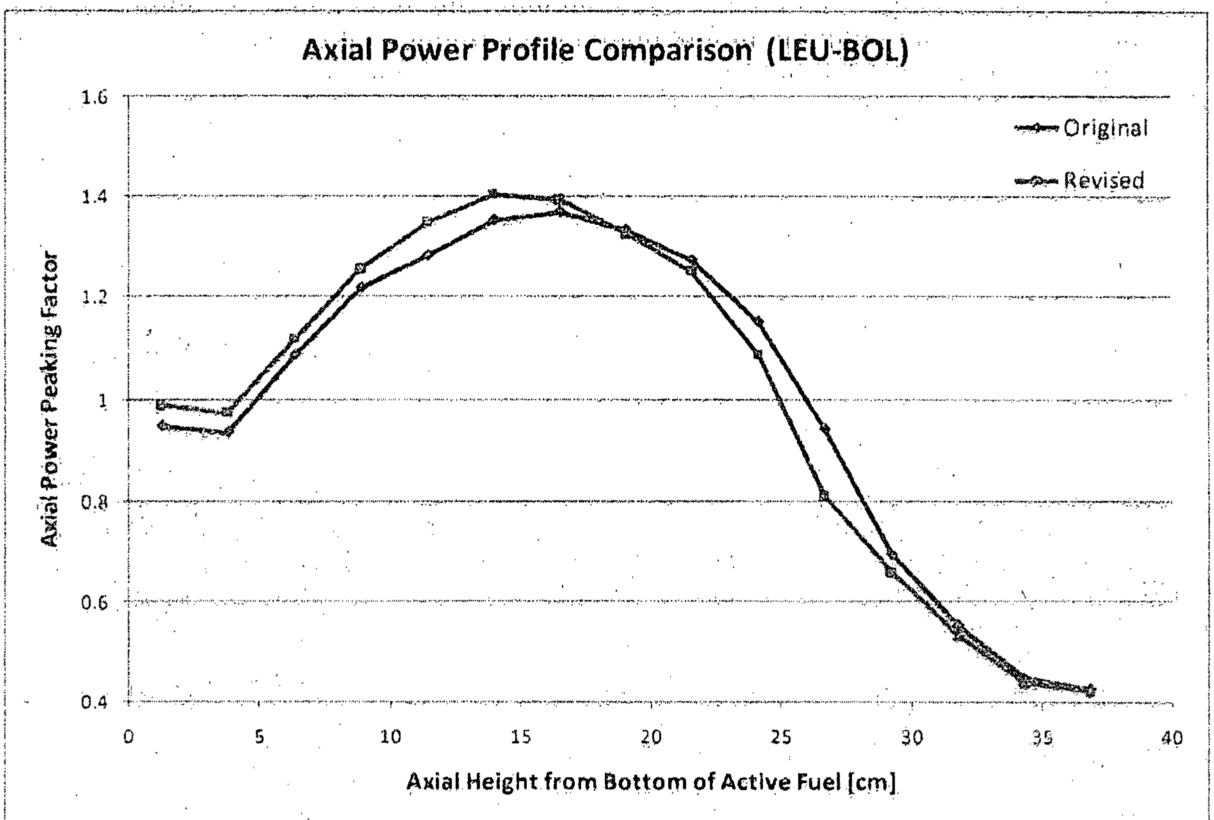

Figure RAI-20-1, Axial Power Shapes Comparison Between Original and Revised (LEU-BOL)

As is evident by Figure RAI-20-1, the peak axial power changed from 1.368 in the original analysis to 1.4032 in the new MCNP5 calculations for LEU-BOL. The pin power peaking factor did not change. This changed the maximum fuel temperature from 662.83°C to 673.86°C at 1.5 MW which is higher than the LEU-MOL maximum fuel temperature at 1.5 MW of 665.06°C. Therefore LEU-BOL has the highest maximum fuel temperature with the highest rod power.

The new LEU-BOL steady state analysis is presented in the table below:

Table RAI-20-1, T/H Comparison between Original and Revised LEU-BOL results

Parameter	Core Power [MW]	LEU Conversion SAR	With Revised Axial / Radial Power Shape	% Difference
Rod Power in D5SW [kW]	1.5	29.041	29.041	0.00%
	1.3	25.169	25.169	0.00%
	1.0	19.361	19.361	0.00%
Mass Flow Rate [kg/s]	1.5	0.14878	0.14861	1.64%
	1.3	0.13143	0.13105	1.61%
	1.0	0.10535	0.10503	1.54%
Maximum Fuel Centerline Temperature [°C]	1.5	662.83	673.86	0.30%
	1.3	594.40	604.10	0.28%
	1.0	490.15	497.81	0.26%
Maximum Outside Clad Temperature [°C]	1.5	141.60	142.02	0.30%
	1.3	139.60	139.99	0.28%
	1.0	136.30	136.66	0.26%
Exit Outer Clad Temperature [°C]	1.5	127.47	127.78	0.24%
	1.3	127.14	127.06	-0.06%
	1.0	125.09	125.02	-0.06%
Exit Bulk Coolant Temperature [°C]	1.5	101.32	100.95	-0.37%
	1.3	100.04	100.17	0.13%
	1.0	98.23	98.37	0.14%
Critical Rod Power Groeneveld 2006	1.5	53.465	53.112	-0.66%
	1.3	52.733	51.453	-2.49%
	1.0	51.884	49.891	-3.99%
Critical Rod Power Bernath	1.5	35.716	35.631	-0.24%
	1.3	33.488	33.403	0.25%
	1.0	29.437	29.599	0.55%
Power to Reach CHF at last non-oscillatory flow rate	Groeneveld 2006	52.786	53.112	0.61%
	Bernath	35.164	35.631	1.31%
MDNBR – Groeneveld 2006	1.5	1.818	1.829	0.59%
	1.3	2.095	2.044	-2.48%
	1.0	2.680	2.577	-4.00%
MDNBR - Bernath	1.5	1.211	1.227	1.30%
	1.3	1.331	1.327	-0.29%
	1.0	1.520	1.529	0.58%

Where % Difference is defined as:

$$\%Difference = \frac{revised - original}{revised} * 100\%$$

As can be seen in the table above, the changes in the axial flux profile lead to insignificant differences.

In regards to the MDNBRs shown in Tables 4.7.13 (page 114) and 4.7.16 (page 121), the overall MDNBR trends are identical. As the power increases, the MDNBR decreases. For the old LEU-BOL analysis, the Groeneveld 2006 correlation has a calculated MDNBR of 2.680 at 1.0 MW and a MDNBR of 1.818 at 1.5 MW. For LEU-MOL, the Groeneveld 2006 correlation has a calculated MDNBR of 2.678 at 1.0 MW and a MDNBR of 1.809 at 1.5 MW. A similar trend for the Bernath MDNBR also results as can be seen from the table.

However, it can also be seen that the MDNBR for LEU-MOL is more limiting in 2 out of 6 cases in the table below.

Table RAI-20-2, MDNBR Comparison between LEU-BOL and LEU-MOL

MDNBR Correlation	Power [MW]	LEU-BOL Conversion SAR	LEU-BOL Revised	LEU-MOL	% LEU-BOL Revised higher than LEU-MOL
Groeneveld 2006	1.5	1.818	1.829	1.829*	0.00%
	1.3	2.095	2.044	1.982	3.03%
	1.0	2.680	2.577	2.678	-3.92%
Bernath	1.5	1.211	1.227	1.240*	-1.06%
	1.3	1.331	1.327	1.339	-0.90%
	1.0	1.520	1.529	1.527	0.13%

* Pseudo-transient calculated stable flow rate at 1.5 MW, thus the MDNBR no longer is being calculated using the critical rod power from a lower power level that calculated a stable flow.

The percentage of LEU-BOL higher than LEU-MOL column shown is calculated as:

(LEU-BOL – LEU-MOL) / LEU-BOL * 100%

In all cases, LEU-MOL has the MDNBR located at 19.05 cm above the bottom of the active fuel. Whereas, LEU-BOL the Groeneveld 2006 MDNBR is located at 16.51 cm above the active fuel and the Bernath MDNBR is located at 21.59 cm above the active fuel. Since the Bernath case at 1.0 MW gives essentially the same result, the only evident discrepancy appears for Groenveld 2006 at 1.3 MW.

The important thermal hydraulic parameters used at 16.51 cm, 19.05 cm and 21.59 cm can be seen in the table below. Note that the critical heat flux ratio (CHFR) is defined as:

CHFR = q"(CHF) / q"(local)

Table RAI-20-3, Important Thermal Hydraulic Parameters for LEU-BOL vs. LEU-MOL

Parameter (at 1.3 MW)	LEU-BOL Revised	LEU-MOL	Percent LEU-BOL higher than LEU-MOL
Mass Flow Rate [kg/s]	0.13105	0.13049	0.43%
Local heat flux at 16.51 cm [W]	816.957	793.381	2.89%
Local heat flux at 19.05 cm [W]	777.822	782.666	-0.62%
Local heat flux at 21.59 cm [W]	733.290	726.828	0.88%
Local quality at 16.51 cm	-0.06136	-0.06230	-1.53%
Local quality at 19.05 cm	-0.05372	-0.05458	-1.60%
Local quality at 21.59 cm	-0.04652	-0.04742	-1.93%
Local fluid velocity at 16.51 cm [m/s]	0.283613	0.282283	0.47%
Local fluid velocity at 19.05 cm [m/s]	0.284573	0.283249	0.47%
Local fluid velocity at 21.59 cm [m/s]	0.285479	0.284147	0.47%
Predicted Groeneveld 2006 CHF at 16.51 cm [kW/m^2]	2079.638	2086.678	-0.34%
Predicted Groeneveld 2006 CHF at 19.05 cm [kW/m^2]	1989.248	1995.506	-0.31%
Predicted Groeneveld 2006 CHF at 21.59 cm [kW/m^2]	1911.717	1917.814	-0.32%
Predicted Bernath CHF at 16.51 cm [kW/m^2]	1370.584	1380.707	-0.74%
Predicted Bernath CHF at 19.05 cm [kW/m^2]	1285.578	1294.833	-0.72%
Predicted Bernath CHF at 21.59 cm [kW/m^2]	1205.403	1215.050	-0.80%
CHFR with Groeneveld 2006 at 16.51 cm	2.546	2.630	-3.30%
CHFR with Groeneveld 2006 at 19.05 cm	2.557	2.550	0.27%
CHFR with Groeneveld 2006 at 21.59 cm	2.607	2.639	-1.23%
CHFR with Bernath at 16.51 cm	1.678	1.740	-3.69%
CHFR with Bernath at 19.05 cm	1.653	1.654	-0.06%
CHFR with Bernath at 21.59 cm	1.644	1.672	-1.70%
Critical Rod Power with Groeneveld 2006	51.453	49.625	3.55%
Critical Rod Power with Bernath	33.403	33.517	-0.34%

From this table, it is evident that LEU-MOL has a more limiting local heat flux at 19.05 cm than LEU-BOL, thus giving a lower MDNBR at this axial location. However, the minimum CHFR shows a less noticeable difference (0.27%) between LEU-MOL and LEU-BOL than the MDNBR does. The reason the MDNBR results are not satisfactory is due to the method of calculating the necessary rod power for the Groeneveld 2006 to reach a MDNBR of 1.0 while keeping the flow rate constant. This is shown in the table above where the critical rod power at 1.3 MW with the Groeneveld 2006 correlation is 3.55% higher for LEU-BOL than LEU-MOL. This is a very large discrepancy and is due to code convergence problems with the K_2 and K_4 terms with switching from negative to positive quality.

Therefore, the differences between LEU-BOL and LEU-MOL are small and generally within the errors of the correlations themselves. Because the Groeneveld 2006 and Bernath correlations were not developed for use in TRIGA analysis, the more limiting Bernath correlation was used. However, Anderson, et al from the University of Wisconsin has proposed to ANL to precisely determine CHF for the three TRIGA fuel assembly types (hexagonal, circular and rectangular). The results displayed in Tables 4.7.13 and 4.7.16 show that the reactor will not reach CHF even at 1.5 MW for LEU-BOL and LEU-MOL.

21. Figures 4.7.16 (p. 84), 4.7.45 (p. 115), 4.7.51 (p. 122), and 4.7.57 (p. 128): These graphs depict hot rod power to reach CHF as a function of flow rate at HEU-BOL, LEU-BOL, LEU-MOL, and LEU-EOL. Please provide the numerical values of the critical rod power as determined by the Groneveld 2006 and the Bernath correlation for core powers of 1 MW, 1.3 MW and 1.5 MW at HEU-BOL, LEU-BOL, LEU-MOL, and LEU-EOL.

Licensee's Response:

The values are provided in the table below, where both the original reported and revised LEU-BOL numbers are provided. The revised numbers account for the revised axial and radial power distributions for LEU-BOL.

Table RAI-21-1, Critical Rod Powers

CHF Correlation	Power [MW]	HEU-BOL [kW/rod]	LEU-BOL Conversion SAR [kW/rod]	LEU-BOL Revised [kW/rod]	LEU-MOL [kW/rod]	LEU-EOL [kW/rod]
Groneveld 2006	1.5	52.376	53.465	53.112	52.832	54.553
	1.3	49.573	52.733	51.453	49.625	54.651
	1.0	47.579	51.884	49.891	51.579	51.314
Bernath	1.5	33.993	35.716	35.631	35.820	35.492
	1.3	31.849	33.488	33.403	33.517	33.153
	1.0	28.206	29.437	29.599	29.406	29.124

22. Section 4.7.4, 4.7.7, 4.7.8, and 4.7.9: When you calculated coolant flow rate versus hot rod power for various core configurations, RELAP5 calculated flow oscillations at some power. Above this hot rod power, you provide graphs showing projections of coolant flow rate. Is the extrapolation of flow calculated by RELAP5 above the last predicted stable flow realistic? Please discuss.

Licensee's Response:

The normal steady state method of solving the RELAP5/MOD3.3 transient failed to get a stable solution for core powers around 1.5 MW. However, another method of solving the steady state cases using a 'pseudo transient' was used that was able to extend the RELAP5/MOD3.3 region of applicability before having to use the extrapolated region. By setting the initial mass flow rate to nearly zero and setting the power to nearly zero and then ramping up the power until RELAP can calculate a steady state solution produced the following graphs for LEU-BOL, MOL and EOL. The revised axial and radial power distributions for LEU-BOL were used.

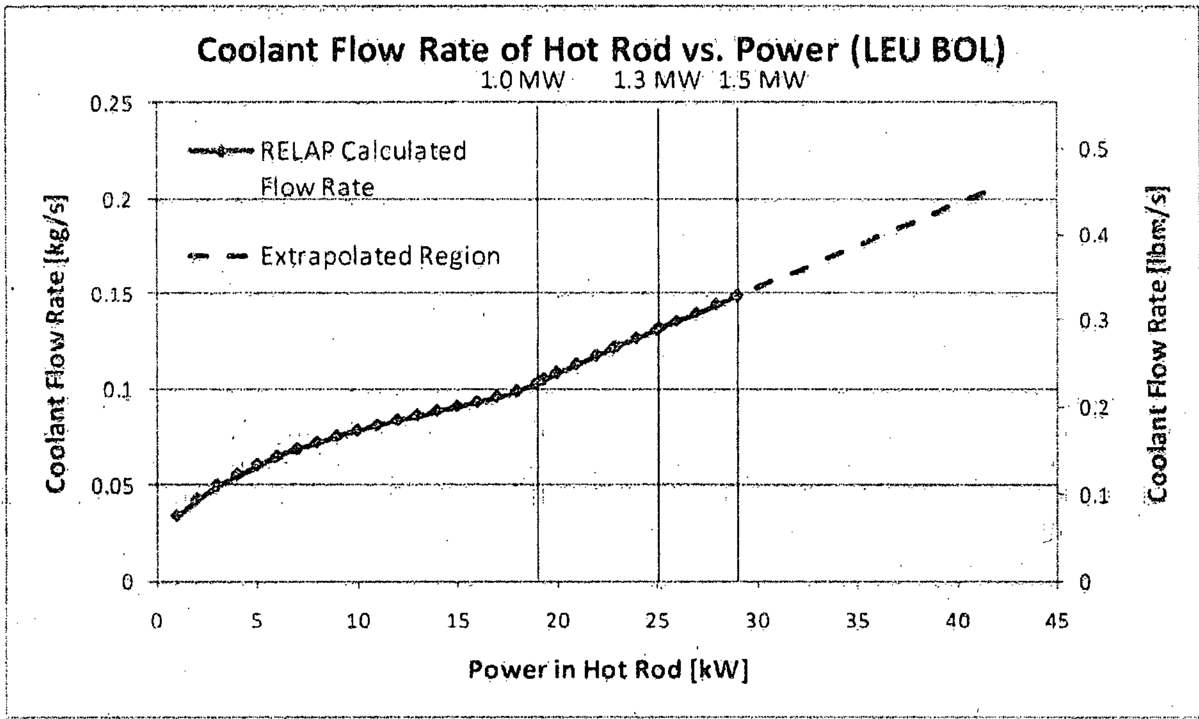

Figure RAI-22-1, Coolant Flow Rate vs. Power LEU-BOL

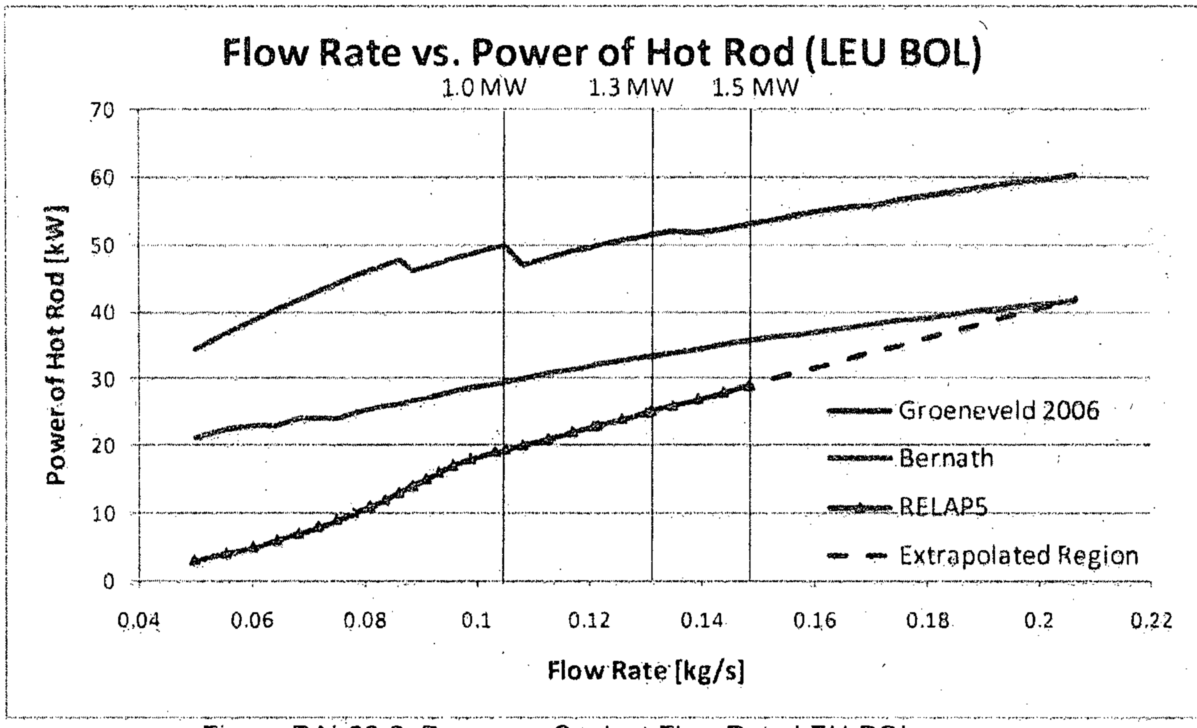

Figure RAI-22-2, Power vs. Coolant Flow Rate LEU-BOL

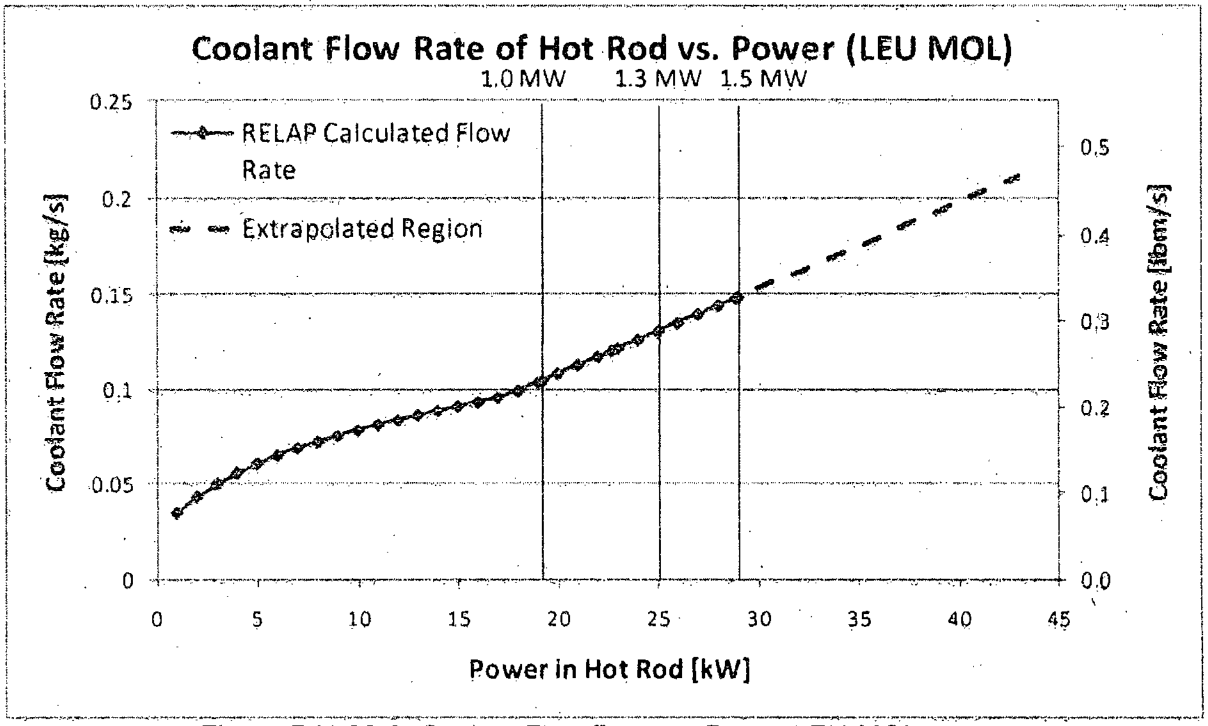

Figure RAI-22-3, Coolant Flow Rate vs. Power LEU-MOL

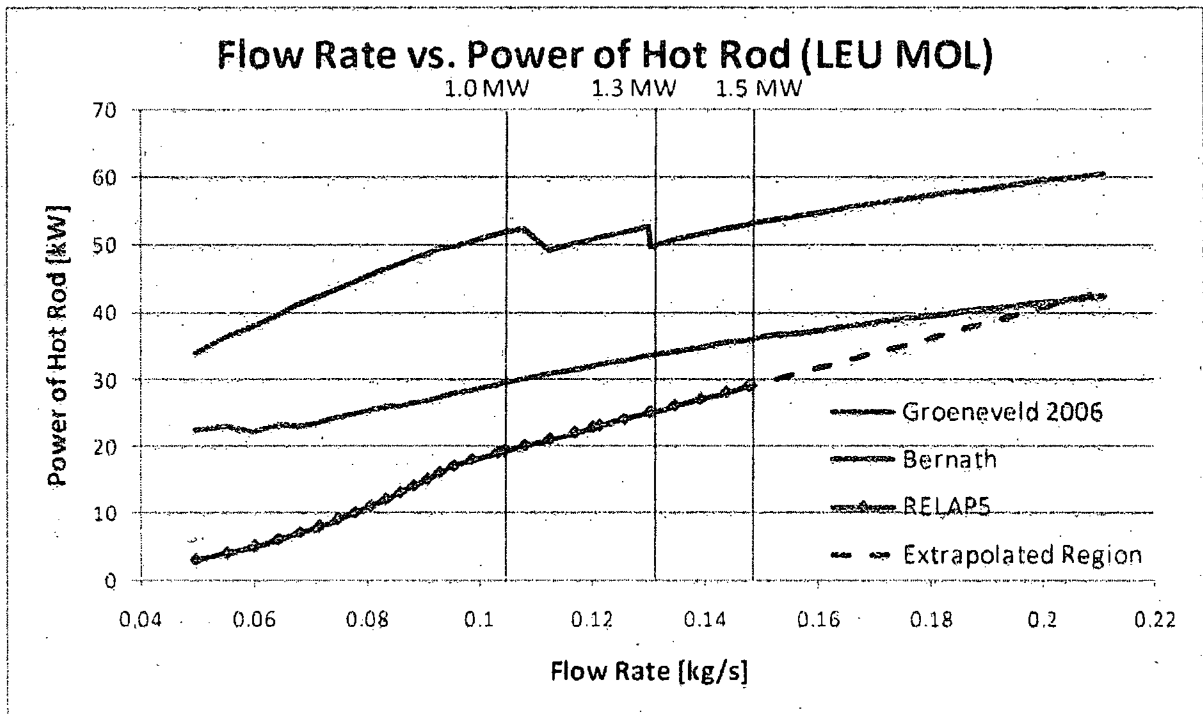

Figure RAI-22-4, Power vs. Coolant Flow Rate LEU-MOL

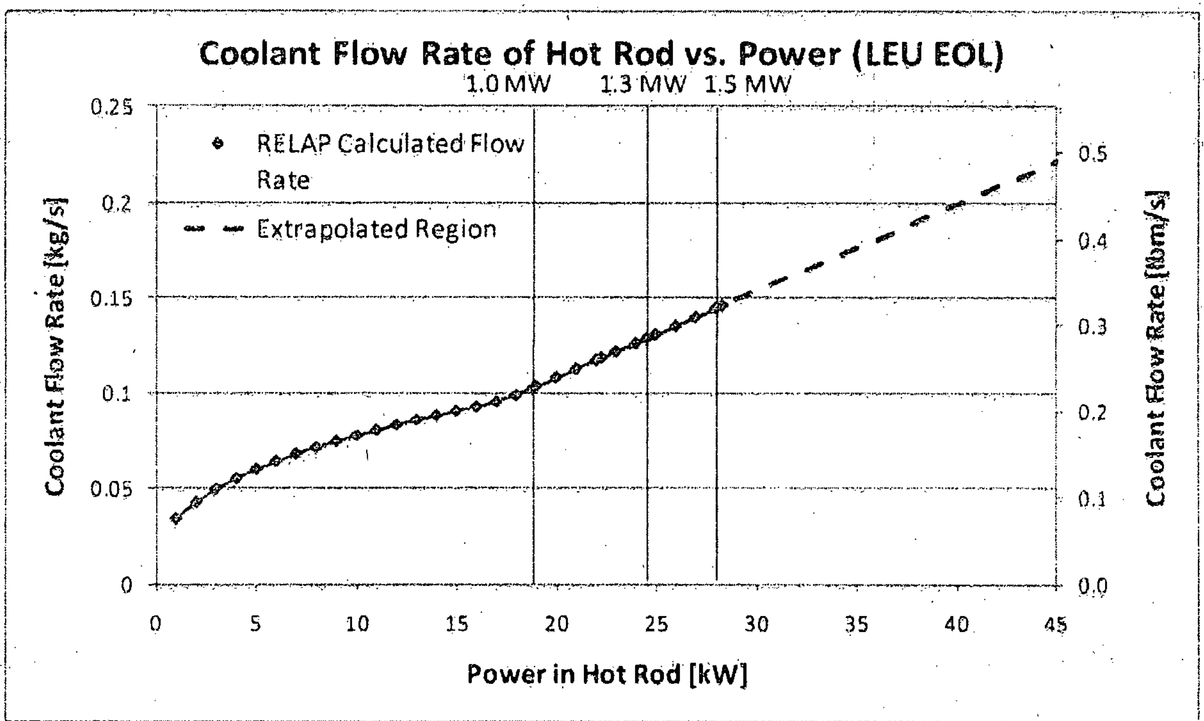

Figure RAI-22-5, Coolant Flow Rate vs. Power LEU-EOL

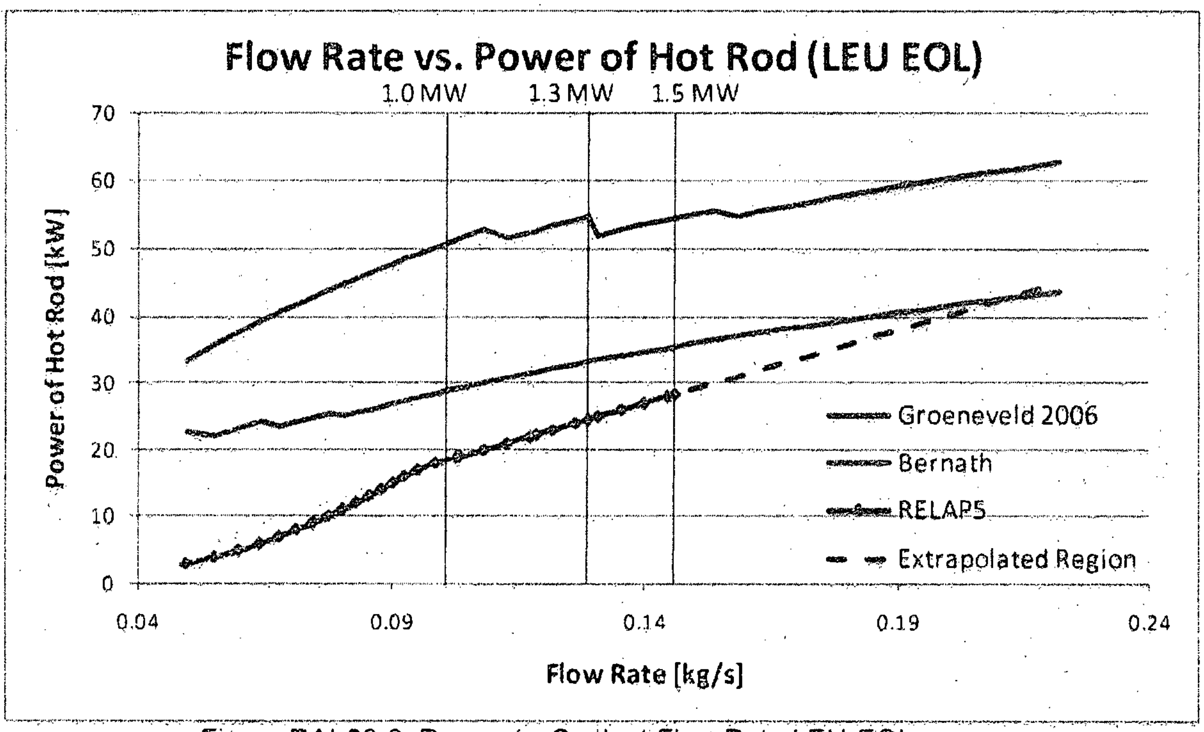

Figure RAI-22-6, Power vs. Coolant Flow Rate LEU-EOL

As can be seen from these graphs, RELAP5/MOD3.3 predicts a stable flow regime at 1.5 MW for all LEU statepoints. In addition if the inlet coolant temperature lowers from 54.44°C to 43°C or 30°C than the stable flow regime would be predicted until 37 kW/rod and 43 kW/rod respectively. Therefore, no flow oscillations are predicted at 1.5 MW or below during steady state operation.

The direct answer to the question "Is the extrapolation of flow calculated by RELAP5/MOD3.3 above the last predicted stable flow realistic?" is that it is simply not known. It may be possible that stable subcooled nucleate boiling flow occurs considerably above the powers that RELAP5/MOD3.3 predicted, but without applicable experimental data it would be difficult to say with certainty. Anderson, et al from the University of Wisconsin has proposed to ANL to precisely determine CHF for the three TRIGA fuel assembly types (hexagonal, circular and rectangular). However, it is important to emphasize that a reactor power of 1.5 MW with an inlet temperature of 54.44°C is far beyond not only normal operating conditions at 1.0 MW with about a 30°C inlet coolant temperature, but also the anticipated faulted conditions where the power trip is no higher than 1.3 MW.

23. Section 4.7.4, 4.7.7, 4.7.8, and 4.7.9. RELAP5 calculated flow oscillations at a power of around 28 kW/rod. Demonstrate that DNB is a more conservative criterion than flow instability in determining the thermal limit of the UWNR.

Licensee's Response:

It may be possible that DNB is not a more conservative limit than flow instability, but without applicable experimental data it would be difficult to say with certainty. Also, the inability of RELAP5/MOD3.3 to predict a stable flow above a specific power may not be indicative of anything more than a limitation of the code. It is important only that the power levels at which either of the two undesirable phenomena occurs be far above where the reactor is operated and they are. For LEU BOL, for example, at the normal reactor power level of 1.0 MW, RELAP5/MOD3.3 predicts stable values of flow up to 1.5 MW. These calculations were performed for a coolant inlet temperature of 54.44°C, which is the maximum administrative limit and far above the normal value of about 30°C. For an inlet coolant temperature of 30°C, the corresponding maximum power level for which RELAP5/MOD3.3 predicts a stable flow is 2.2 MW. It is worth noting that decreasing the inlet temperature from 54.44°C to 30°C will also increase the power levels at which the Bernath and Groeneveld correlations predict DNB to occur. Thus, decreasing the coolant inlet temperature improves both the maximum predicted power for which stable flow is obtained and the powers at which DNB are predicted to occur.

As seen in the new figures in Question 22 for the LEU flow rate using the pseudo transient, the flow is stable at 1.5 MW and thus flow oscillations are not predicted in steady state operation of the LEU core. Since the reactor is limited to operating below 1.5 MW of steady state operation at all times, core power is the determining power of the thermal limit of the UWNR.

Furthermore, Anderson, et al from the University of Wisconsin has proposed to ANL to precisely determine CHF for the three TRIGA fuel assembly types (hexagonal, circular and rectangular) which could provide the necessary experimental data.

24. Section 4.7.5 and 4.7.10. Was a weighted or averaged fuel temperature used in the calculation of the reactivity feedback?

Licensee's Response:

When implementing the 2 channel model for pulsing analysis, the radial nodilization was constructed so each radial zone in the fuel meat had equal radial volume. RELAP5/MOD3.3 gives the choice of defining the reactivity feedback as a function of volume density or volumetric average fuel temperatures. Having equal radial volumes, the simple average of nodal temperatures is automatically volume-weighted.

In addition, RELAP5/MOD3.3 performs point reactor kinetics by computing the core average fuel temperature to use in the reactivity calculation. Since the height of each node is the same across the core, then a simple nodal averaging is used. Thus with nodal temperatures volume-weighted and the reactivity calculated with simple nodal averaging, the reactivity feedback used in RELAP5/MOD3.3 is volume-weighted averaging across the whole core.

25. Section 4.7.5 and 4.7.10. Was the effect of direct gamma heating of the coolant incorporated in the RELAP5 model?

Licensee's Response:

Yes, the effect of gamma heating was incorporated into the pulsing models in sections 4.7.5, 4.7.10, and 13.2. In RELAP5/MOD3.3 card 30000001, the default gamma was chosen to provide the gamma heating from the standard fission product decay calculations. RELAP5/MOD3.3 automatically calculated the amount of gamma heating in addition to the fission power to give the total power produced.

26. Section 4.7.5 and 4.7.10. Was the power distribution in the core maintained constant during the pulse and was the assumption conservative?

Licensee's Response:

Initially, the pulsing analysis was performed using the same radial, axial, and pin peaking factors as those used in the steady state analysis. To verify that this was a conservative assumption, an MCNP case was run for the most limiting case, LEU-MOL, with the transient rod full out and the blades at the cold critical bank height to determine what the radial, axial, and pin power peaking factors would be. The LEU-MOL pin power peaking factor for D5SW was 1.797, axial peaking factor of 1.284, and radial peaking factor of 1.566 as seen in the table below. These new power profiles were put into the LEU-MOL RELAP5/MOD3.3 model to give a new prompt peak fuel temperature of 790.45°C in the hot rod. This maximum fuel temperature is 8.74% higher than the previous LEU-MOL results of 726.95°C. These results are more limiting, but they do not exceed the maximum fuel operating temperature of 830°C.

Table RAI-26-1, Key Parameters Comparing Power Distributions

Parameter	LEU-MOL Original	LEU-MOL T-Rod out	Percent Difference
Pin Power Peaking Factor	1.598	1.797	12.453%
Axial Peaking Factor	1.359	1.284	-5.519%
Radial Peaking Factor	1.438	1.566	8.901%
Peak Pulse Temperature [°C]	726.95	790.45	8.735%
Peak Pulse Power [GW]	2.52	2.52	0.000%

Additionally, the RELAP5/MOD3.3 pulsing model includes the following limiting assumptions:

- Instantaneous firing of the transient rod
- D5 SW channel includes heated perimeter of the transient rod making power
- No reactivity feedback in the hot rod
- No moderator temperature/void reactivity feedback
- Power profile remains constant through transient

Therefore this model is conservative and still demonstrates that the fuel temperature will not exceed 830°C during a 1.4% Δk/k pulse for the most limiting state-point.

27. [illegible] Figures 4.7.22, 4.7.59, 4.7.66, 4.7.73. These figures show power at about 10 MW at 0.25 second following pulse initiation. What is the power profile until the reactor scrams?

Licensee's Response:

The power profile and the maximum temperature from pulse initiation until reactor scram (15 seconds after pulse initiation) for the four figures in question are given below. The revised LEU-BOL axial and radial power distributions were used.

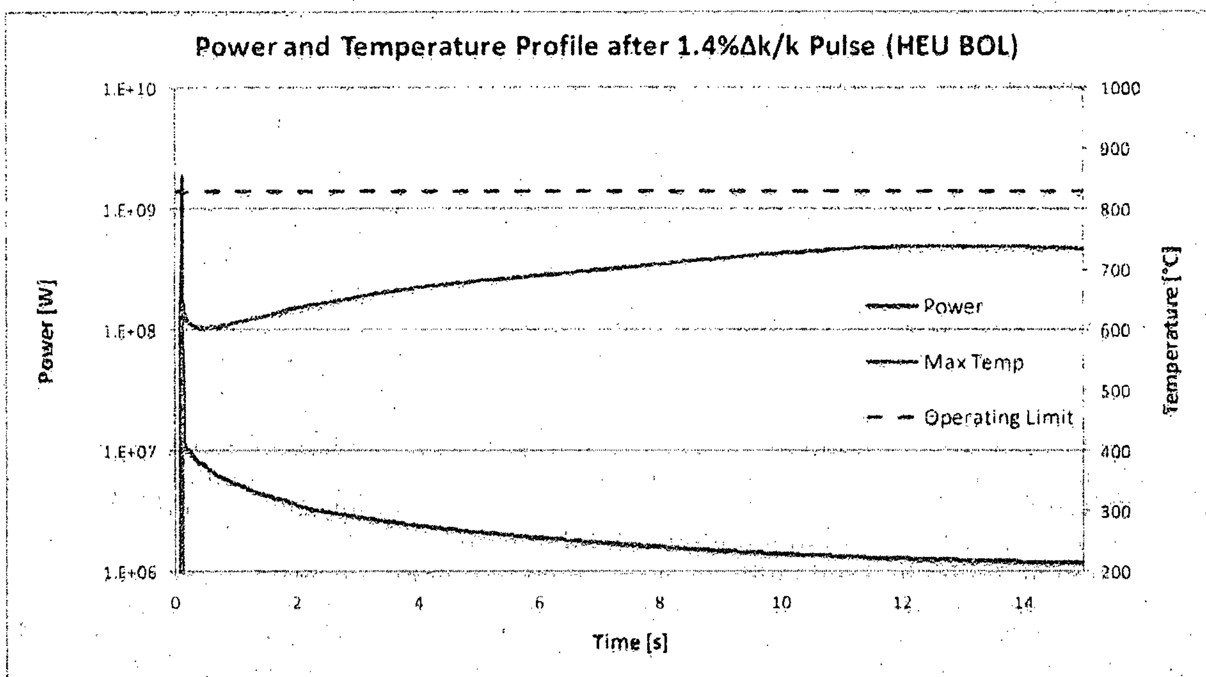

Figure RAI-27-1, Power and Temperature Profiles after Pulse HEU-BOL

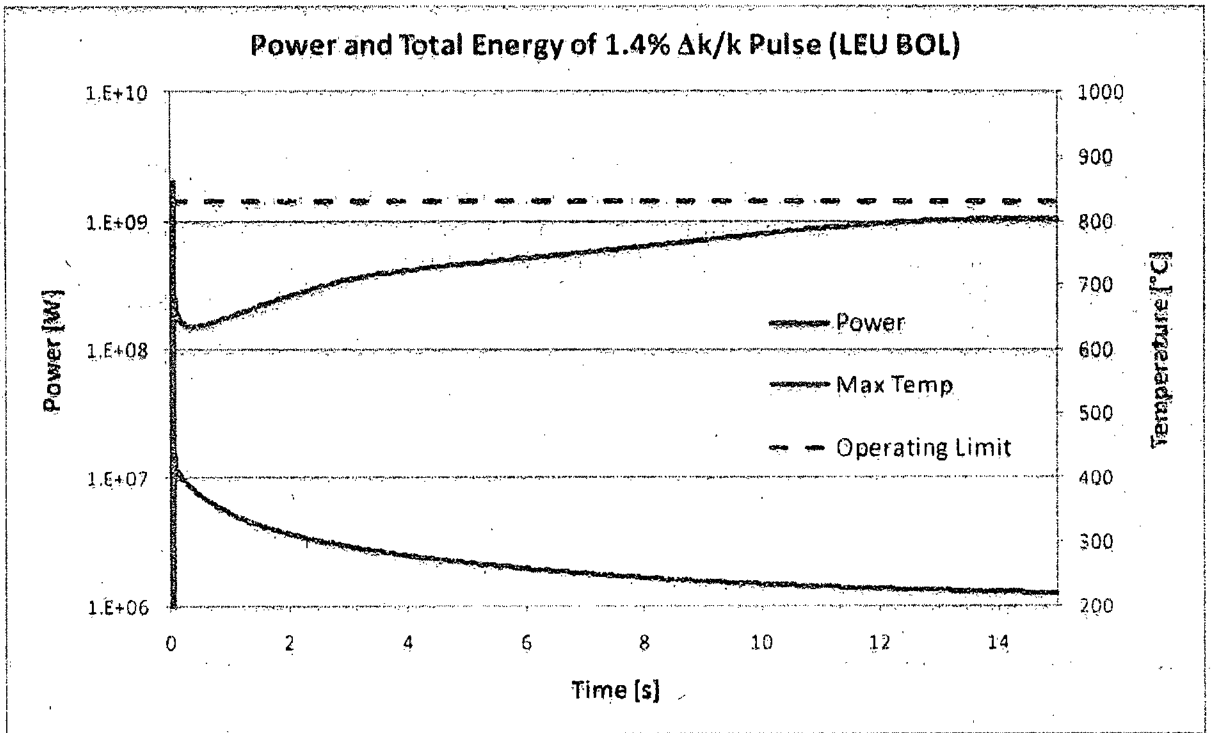

Figure RAI-27-2, Power and Temperature Profiles after Pulse LEU-BOL

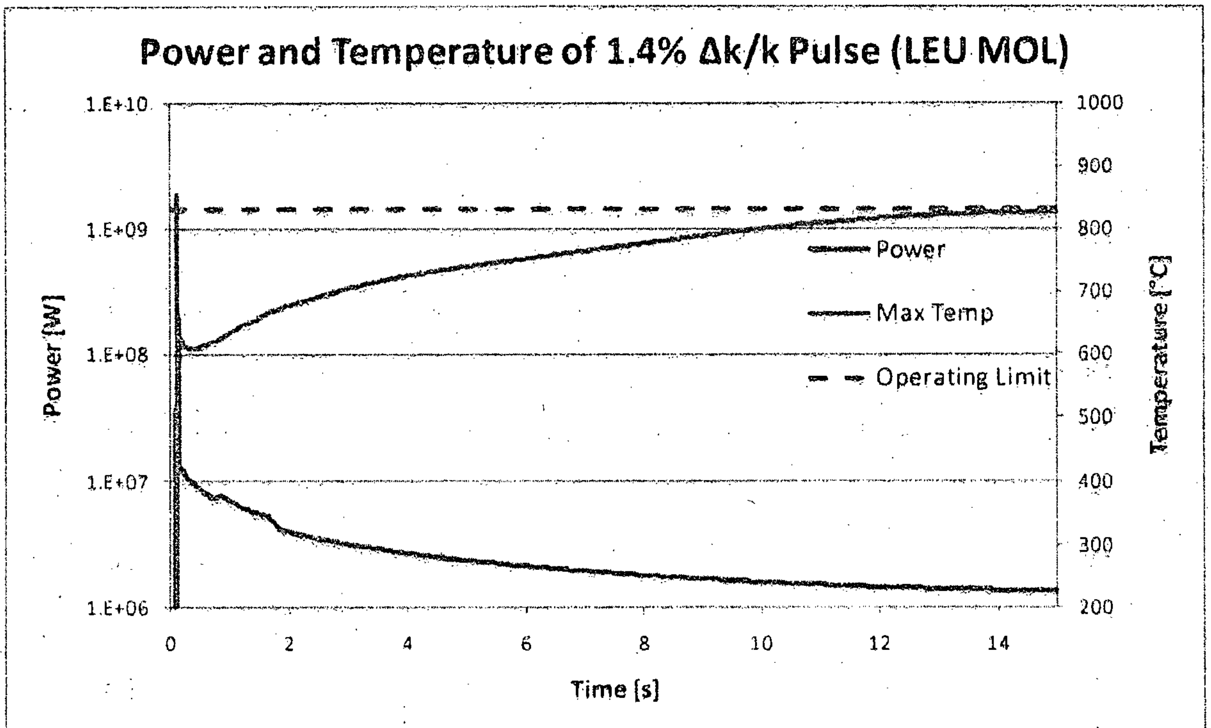

Figure RAI-27-3, Power and Temperature Profiles after Pulse LEU-MOL

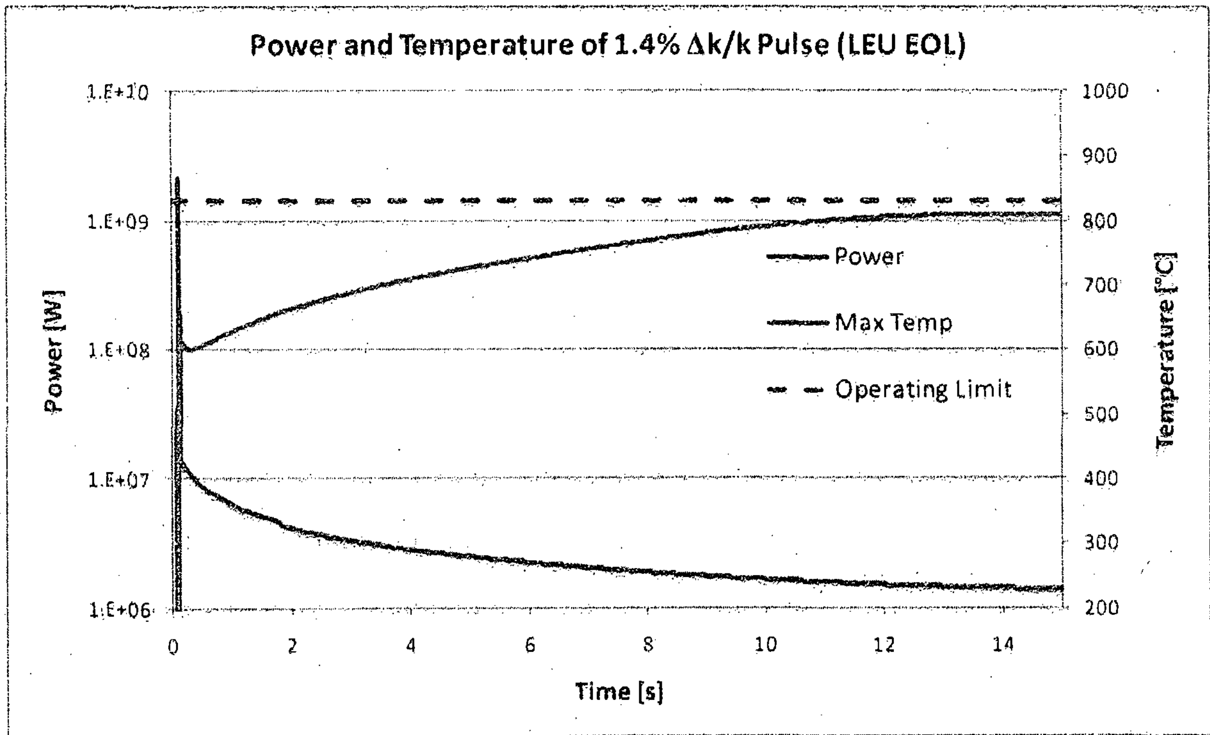

Figure RAI-27-4, Power and Temperature Profiles after Pulse LEU-EOL

These graphs were produced with the following assumptions of the 2-channel model.

- D5 SW channel flow area adjusted to be a typical cell so that the heated perimeter does not include the transient rod making power. This was the same change made in the LOCA analysis.
- Instantaneous firing of the transient rod changed to 0.1 seconds to fire the transient rod
- Incorporating Doppler reactivity feedback in the hot rod
- Incorporating Moderator Temperature feedback from Tables 4.5.6 and 4.5.13
- Incorporating Moderator Density feedback from Tables 4.5.6 and 4.5.13

The major change to this model was the incorporation of the additional feedback mechanisms, specifically the moderator density feedback. Due to void production during the pulse, this adds an additional negative feedback in addition to the Doppler feedback. As can be seen from these graphs at no time does the maximum temperature exceed 830°C. The maximum predicted temperatures over the 15 seconds are 738.35°C, 756.45°C, 826.15°C, and 810.25°C for HEU-BOL, LEU-BOL, LEU-MOL, and LEU-EOL respectively.

Characteristically for pulses in TRIGA reactors, the fuel temperature peaks near the outer surface of the fuel meat near the peak power and shortly thereafter. As time passes, heat is redistributed toward the center of the fuel meat and the temperature rises, becoming larger at the fuel center than the fuel outer surface temperature, as shown in the figure below.

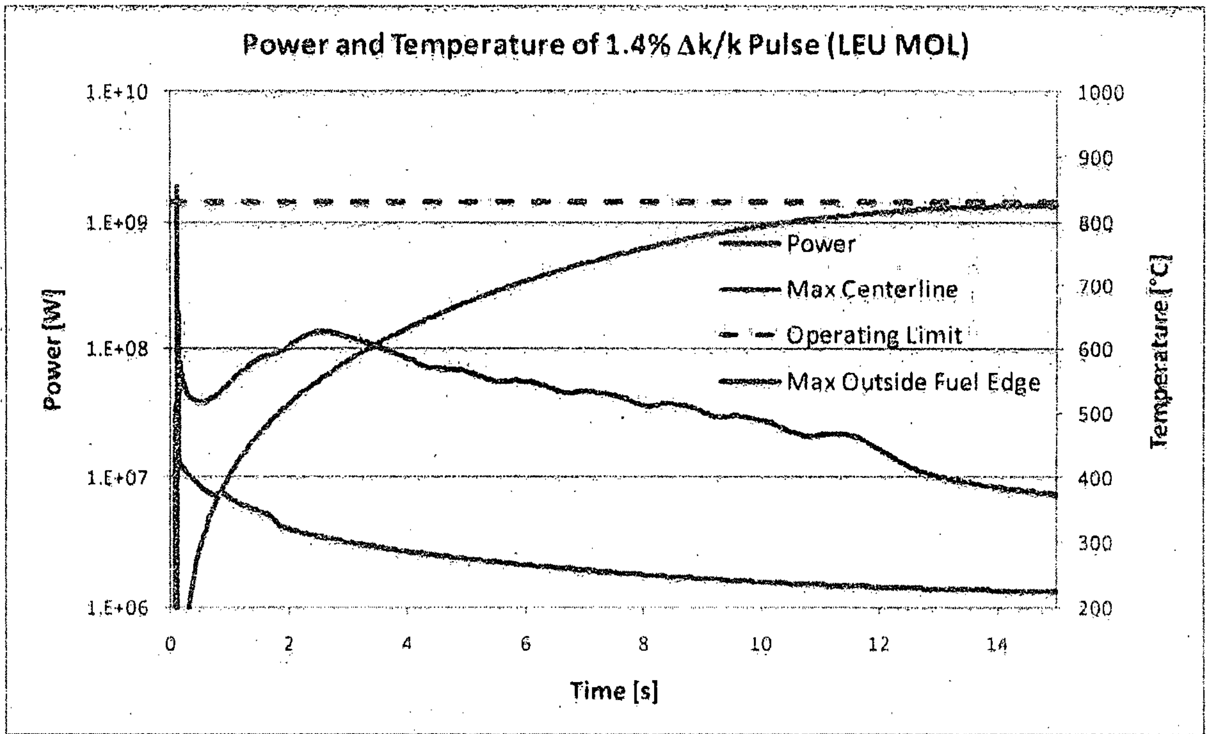

Figure RAI-27-5, Power and Temperature after Pulse LEU-MOL

From the figure above, the prompt peak temperature on the outside edge of the fuel is 613°C, and is substantially less than 830°C. However, as the heat is redistributed, the maximum temperature of the rod occurs at the fuel centerline at 15sec and in the limiting case of LEU-MOL, the fuel temperature approaches the operating limit of 830°C with a maximum of 826°C.

The consequence of the fuel centerline temperature approaching the operational limit of 830°C is not significant when considering the basis for this limit. It is known that after extensive steady-state operations at 1MW, the hydrogen in the ZrH_x matrix will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions, thereby increasing the ZrH_x ratio from the nominal value of 1.6. When the fuel is pulsed, the instantaneous prompt temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher prompt temperatures in the outer regions occur in fuel with hydrogen to zirconium ratios that have now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grow with each pulse (General Atomics, "Pulsing Temperature Limit for TRIGA LEU Fuel," GA-C26017, December 2007). However, at the center of the rod, the ZrH_x ratio has decreased below the nominal value. Thus, as the center of the fuel rod approaches 826°C, in the limiting case of LEU-MOL, the fuel rod will not produce hydrogen gas pressures in excess of the expected $ZrH_{1.6}$.

Finally, the analysis is still conservative in that it neglects cross-flow between channels, which is anticipated to be significant during a pulse transient. Also, additional margin exists because operationally the reactor scrams within 5 seconds of the pulse rather than 15 seconds.

28. Section 4.7.6. In Figure 4.7.34, does the axial power distribution reflect the movement of the control blades out of the core as a function of burnup?

Licensee's Response:

Yes, the axial power profiles were derived from MCNP using critical blade heights. The critical bank height moves out to compensate for core burnup. The revised LEU-BOL curve is shown in the following figure, to replace Figure 4.7.34 on page 102.

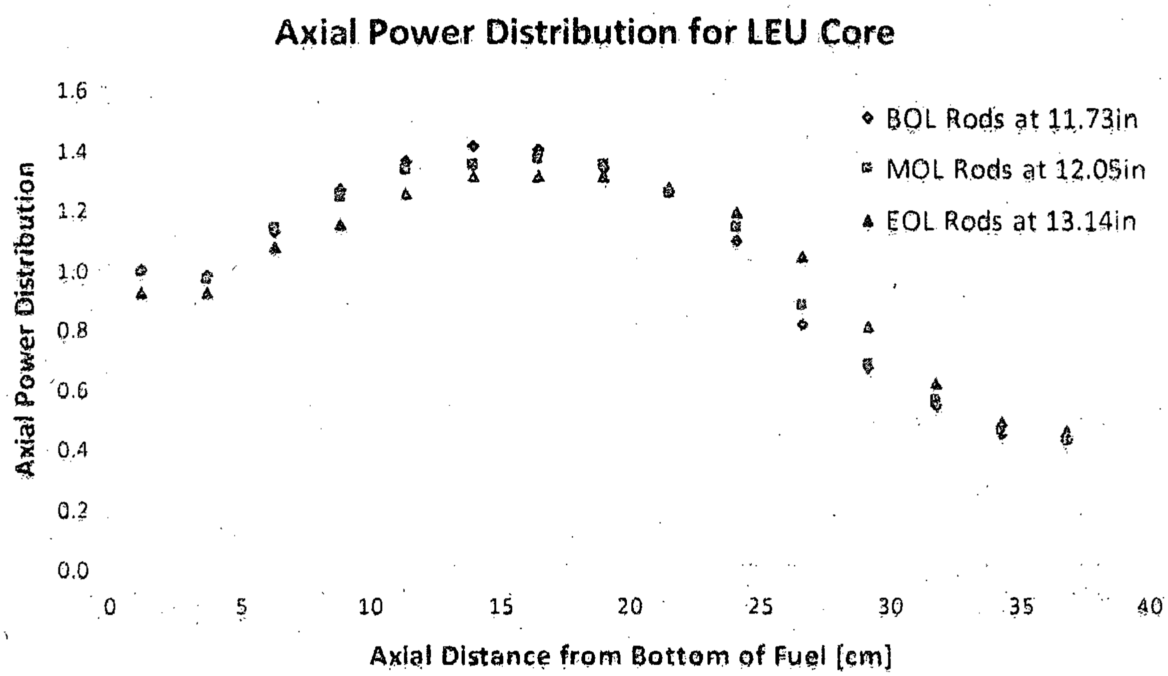

Figure RAI-28-1, Revised LEU Axial Power Distribution

29. Section 4.7.6. Are there core locations other than D5 SW that will have pin power peaking factors greater than 1.61 if a fresh LEU fuel pin is inserted in those locations at MOL or EOL?

Licensee's Response:

It is possible that there will be core locations other than D5 SW that will have pin power peaking factors greater than 1.61 if a fresh LEU fuel pin is inserted in those locations at MOL or EOL. However, D5 SW will always have the greatest pin power peaking factors throughout core life if replaced with fresh fuel. This is because D5 SW will have the greatest burnup of any pin, and therefore the reactivity insertion caused by replacing any other pin location with a fresh fuel pin will be less than replacing the fuel pin at D5 SW. As stated on page 103 of the analysis report, inserting a fresh fuel pin in D5 SW at EOL will produce the largest pin power peaking factor of 1.74. However, there will be no other core location that will produce a pin power peaking factor greater than 1.74 at EOL.

30. Section 4.7.6. Your application states that if the hot rod at core location D5 SW needed to be replaced that the CHF limit would not be exceeded. What acceptance criteria is used following replacement of the fuel? Would a 10 CFR 50.59 review be performed as part of the fuel rod replacement?

Licensee's Response:

The acceptance criteria for replacement of fuel at core location D5 SW would be to ensure that the pin power peaking factor would not exceed 1.61. This would ensure that the design basis analysis of sections 4.7.6 – 4.7.9 would remain valid. It should be noted that replacement of fuel at core location D5 SW could not be with fresh fuel and still meet the acceptance criteria.

In addition, while the analysis of sections 4.7.6 – 4.7.9 have demonstrated that the hot rod at core location D5 SW is the limiting rod with respect to CHF with a pin power peaking factor of 1.61, inserting a fresh fuel pin next to a control blade shroud could be more limiting due to the increased wetted perimeter which decreases the margin to CHF due to reduced flow. In order to prevent the fresh fuel from decreasing the margin to CHF when placed in a location next to a control blade shroud, the pin peaking factor must be less than 1.47. This ensures that the margin to CHF is no less than what was analyzed for the hot rod (D5 SW) in sections 4.7.6 – 4.7.9.

A 10 CFR 50.59 review would be performed as part of a fuel rod replacement and core rearrangement. The analysis would need to show that the acceptance criteria for loading new fuel would be met; specifically, the fresh pin power peaking factor be ≤ 1.47 when placed next to a control blade shroud and ≤ 1.61 in all other locations.

31. Section 4.7.7 (p. 110). At what coolant temperatures and pool levels are the LEU core calculations performed? If temperatures and levels used are not licensed limits, please explain.

Licensee's Response:

The LEU calculations assume a coolant temperature of 130°F and a pool level of 19 feet above the core. The pool level of 19 feet is at the limit given in Technical Specification 3.3.3(d). The coolant temperature is at the administrative limit according to UWNR 100. In response to RAI question 15, the coolant temperature limit is being added to the Technical Specifications. See question 56.

32. Table 4.7.12 (p. 111), 4.7.15 (p. 119), and 4.7.17 (p. 124). Why is the maximum fuel temperature at LEU-EOL lower than that at LEU-MOL?

Licensee's Response:

The key parameters for the differences in the maximum fuel temperature between LEU-MOL and LEU-EOL are summarized in the table below. The percentage of LEU-MOL higher than LEU-EOL column shown is calculated as:

$$(\text{LEU-MOL} - \text{LEU-EOL}) / \text{LEU-MOL} * 100\%$$

Table RAI-32-1, Key Parameters LEU-MOL vs. LEU-EOL for Steady State Analysis

Parameter (at 1.5 MW)	**LEU-MOL**	**LEU-EOL**	**Percent LEU-MOL higher than LEU-EOL**
Pin Power Peaking Factor	1.598	1.567	1.940%
Axial Power Peaking Factor	1.359	1.304	4.047%
Outside Radial Power Peaking Factor	1.438	1.358	5.563%
Interior Radial Power Peaking Factor	0.784	0.817	-4.209%
Centerline Temperature [°C]	665.06	641.91	3.481%
Outside Cladding Temperature [°C]	141.43	140.48	0.672%
Temperature Difference [°C]	523.63	501.43	4.240%

The pin power peaking factor , the axial peaking factor, and the outside radial power peaking factor for LEU-MOL are 1.940%, 4.047%, and 5.563% higher respectively than LEU-EOL. However, the interior radial power peaking factor for LEU-MOL is 4.209% lower than LEU-MOL. In the hottest LEU-MOL fuel rod, more of the power is deposited closer to the surface for the fuel rod and less is deposited closer to the center of the fuel rod than in the hottest LEU-EOL rod. Since on average the power has further to travel in the LEU-EOL rod than in the LEU-MOL, the radial temperature rise in the LEU-EOL is a little bit closer to the LEU-MOL radial temperature rise than it would otherwise be.

This can be seen best by looking at the temperature difference between the clad and the centerline temperature. For LEU-MOL the temperature difference is 523.63°C and for EOL the temperature difference is 501.43°C. Thus, the LEU-MOL temperature difference is 4.240% larger than LEU-EOL.

It is also interesting to note that the pulse analysis shows the maximum power of the pulse is higher for LEU-EOL than LEU-MOL, but the temperature of LEU-MOL is higher by 3.46°C than LEU-EOL. The key parameters for the pulse analysis can be seen in Table RAI-32-2.

Table RAI-32-2, Key Parameters LEU-MOL vs. LEU-EOL for Pulse Analysis

Parameter (at 1.5 MW)	LEU-MOL	LEU-EOL	Percent LEU-MOL higher than LEU-EOL
Pin Power Peaking Factor	1.598	1.567	1.940%
Axial Power Peaking Factor	1.359	1.304	4.047%
Outside Radial Power Peaking Factor	1.438	1.358	5.563%
Interior Radial Power Peaking Factor	0.784	0.817	-4.209%
Maximum Pulse Power [GW]	2.52	3.06	-21.429%
Maximum Temperature [°C]	726.95	723.49	0.476%
Total Negative Temperature Coefficient entered into RELAP [$/K]	-9.30979	-8.60498	7.571%

The maximum pulse power for LEU-EOL is higher than LEU-MOL since the total negative temperature coefficient entered into RELAP is 7.571% lower for LEU-EOL. While the maximum pulse power is 21.429% higher for LEU-EOL, the pin power, axial power, and outside radial power peaking factor are all higher for LEU-MOL. In a pulse, more power is being produced at the outer edge of the fuel and thus the maximum temperature occurs in this region. Therefore, with a higher hot rod power, axial peaking and outside radial power peaking factors, LEU-MOL has a higher fuel temperature despite having a lower maximum pulse power. . The difference between the maximum fuel temperatures is only 0.476% or 3.46°C and is not a significant difference.

33. Tables 4.7.12 (p. 111) and 4.7.15 (p. 119). The hot rod power shown at LEU-MOL is lower than that at LEU-BOL, however the maximum fuel temperature is higher at LEUMOL. Please discuss.

Licensee's Response:

This response is taken in its entirety from Question 20.

An error was discovered with the LEU-BOL axial power shape from MCNP5 that was input into the RELAP5 LEU-BOL models. The following figure shows the comparison between the original axial power shape and the revised MCNP5 axial power shape.

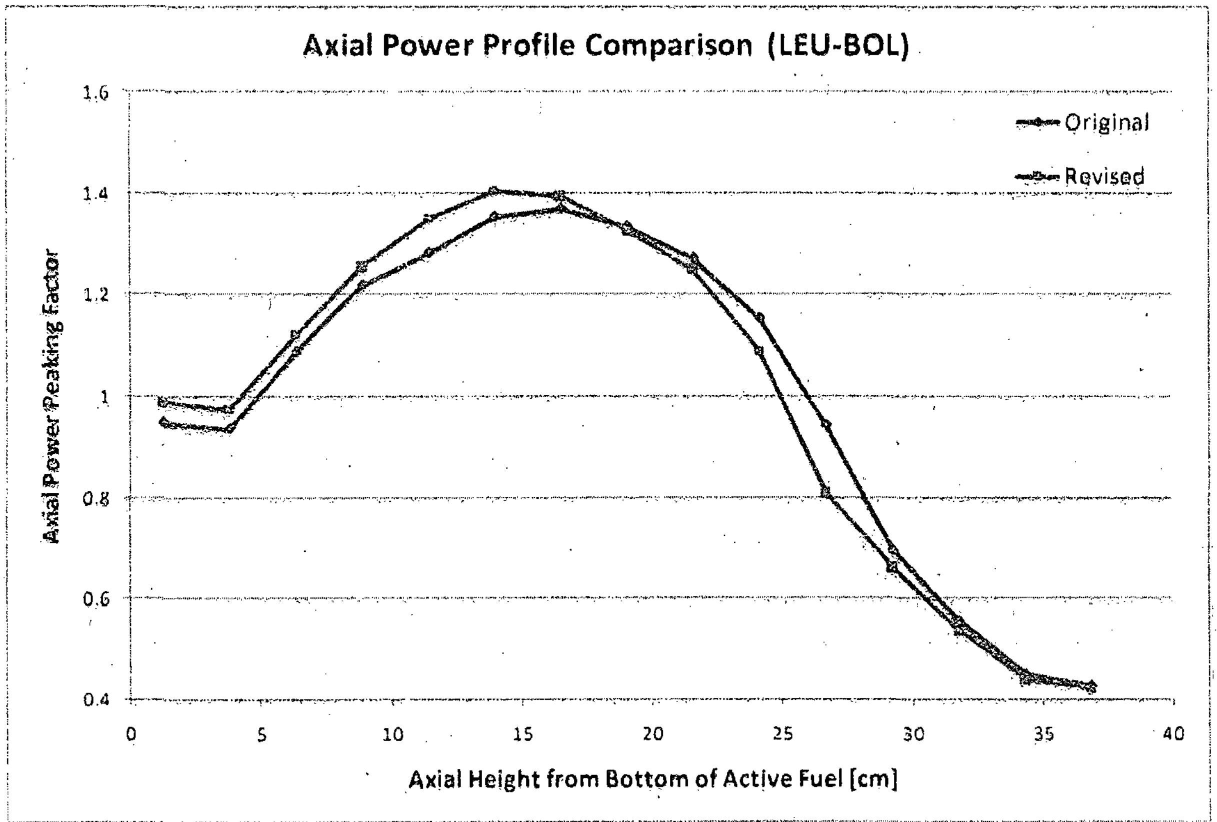

Figure RAI-33-1, Axial Power Shapes Comparison Between Original and Revised (LEU-BOL)

As is evident by Figure RAI-33-1, the peak axial power changed from 1.368 in the original analysis to 1.4032 in the new MCNP5 calculations for LEU-BOL. The pin power peaking factor did not change. This changed the maximum fuel temperature from 662.83°C to 673.86°C at 1.5 MW which is higher than the LEU-MOL maximum fuel temperature at 1.5 MW of 665.06°C. Therefore LEU-BOL has the highest maximum fuel temperature with the highest rod power.

The new LEU-BOL steady state analysis is presented in the table below:

Table RAI-33-1, T/H Comparison between Original and Revised LEU-BOL results

Parameter	Core Power [MW]	LEU Conversion SAR	With Revised Axial / Radial Power Shape	% Difference
Rod Power in D5SW [kW]	1.5	29.041	29.041	0.00%
	1.3	25.169	25.169	0.00%
	1.0	19.361	19.361	0.00%
Mass Flow Rate [kg/s]	1.5	0.14878	0.14861	1.64%
	1.3	0.13143	0.13105	1.61%
	1.0	0.10535	0.10503	1.54%
Maximum Fuel Centerline Temperature [°C]	1.5	662.83	673.86	0.30%
	1.3	594.40	604.10	0.28%
	1.0	490.15	497.81	0.26%
Maximum Outside Clad Temperature [°C]	1.5	141.60	142.02	0.30%
	1.3	139.60	139.99	0.28%
	1.0	136.30	136.66	0.26%
Exit Outer Clad Temperature [°C]	1.5	127.47	127.78	0.24%
	1.3	127.14	127.06	-0.06%
	1.0	125.09	125.02	-0.06%
Exit Bulk Coolant Temperature [°C]	1.5	101.32	100.95	-0.37%
	1.3	100.04	100.17	0.13%
	1.0	98.23	98.37	0.14%
Critical Rod Power Groeneveld 2006	1.5	53.465	53.112	-0.66%
	1.3	52.733	51.453	-2.49%
	1.0	51.884	49.891	-3.99%
Critical Rod Power Bernath	1.5	35.716	35.631	-0.24%
	1.3	33.488	33.403	0.25%
	1.0	29.437	29.599	0.55%
Power to Reach CHF at last non-oscillatory flow rate	Groeneveld 2006	52.786	53.112	0.61%
	Bernath	35.164	35.631	1.31%
MDNBR – Groeneveld 2006	1.5	1.818	1.829	0.59%
	1.3	2.095	2.044	-2.48%
	1.0	2.680	2.577	-4.00%
MDNBR - Bernath	1.5	1.211	1.227	1.30%
	1.3	1.331	1.327	-0.29%
	1.0	1.520	1.529	0.58%

Where % Difference is defined as:

$$\%Difference = \frac{revised - original}{revised} * 100\%$$

As can be seen in the table above, the changes in the axial flux profile lead to insignificant differences.

34. Please refer to questions 56 and 58 when responding to the following question:

Table 4.7.14. The calculated thermocouple temperatures at 1 MW for location D4 SW appear to be above the LSSS limit of 400°C. Please explain. What are the thermocouple temperatures at 1.3 MW and 1.5 MW?

Licensee's Response:

The original basis for the LSSS of 400°C was based on the 1973 SAR estimate of peak fuel temperatures at the UWNR from the Torrey Pines TRIGA Mark III reactor analysis, despite the fact that these two reactors are geometrically dissimilar. During the refueling of the UWNR to the TRIGA core, measured temperatures for D4 SW were reported to exceed 400°C at 1MW, as reported in the startup program and included in the HEU 2000 license renewal SAR (page 4-45). Therefore, the IFE connected to the fuel temperature safety channel was placed in a location that would not exceed 400°C at 1MW, specifically E3 NE. It is fully expected that fuel temperatures in the interior of the core will be greater than 400°C. This is why this application proposes a change to technical specification 2.2 to provide greater flexibility in placing an IFE in the central region of the core, if desired, that could be connected to the fuel temperature safety channel. See proposed change to technical specification 2.2 in the response to RAI question 56.

The calculated thermocouple temperatures at 1.0 MW, 1.3 MW and 1.5 MW using the same methodology that created Table 4.7.14 are shown in the three tables below. The IFE temperatures for D4 SW were updated in order to incorporate the revised axial LEU-BOL shape for the hot rod.

Table RAI-34-1, IFE Temperatures at 1.0MW

IFE Summary Table at 1.0 MW							
IFE Location		0.1 mil gap		0.05 mil gap		0.15 mil gap	
		°C	°F	°C	°F	°C	°F
D4 SW	Bottom	444.37	831.86	397.20	746.95	488.17	910.71
	Center	429.50	805.10	384.22	723.60	471.64	880.95
	Top	412.55	774.58	369.47	697.04	452.75	846.95
E3 NE	Bottom	299.27	570.69	270.90	519.62	326.27	619.29
	Center	291.37	556.47	264.14	507.44	317.34	603.20
	Top	279.87	535.77	254.30	489.74	304.30	579.74

Table RAI-34-2, IFE Temperatures at 1.3MW

IFE Summary Table at 1.3 MW							
IFE Location		0.1 mil gap		0.05 mil gap		0.15 mil gap	
		°C	°F	°C	°F	°C	°F
D4 SW	Bottom	535.23	995.41	476.65	889.96	589.16	1092.48
	Center	516.39	961.49	460.09	860.15	568.30	1054.93
	Top	494.89	922.80	441.24	826.22	544.45	1012.01
E3 NE	Bottom	348.95	660.11	313.48	596.26	382.38	720.28
	Center	338.97	642.15	304.89	580.79	371.16	700.08
	Top	324.47	616.05	292.42	558.36	354.81	670.66

Table RAI-34-3, IFE Temperatures at 1.5MW

IFE Summary Table at 1.5 MW							
IFE Location		0.1 mil gap		0.05 mil gap		0.15 mil gap	
		°C	°F	°C	°F	°C	°F
D4 SW	Bottom	594.79	1102.62	528.99	984.18	654.94	1210.89
	Center	573.32	1063.98	510.07	950.13	631.29	1168.32
	Top	548.82	1019.87	488.53	911.35	604.26	1119.67
E3 NE	Bottom	382.41	720.34	342.28	648.10	420.00	788.00
	Center	371.06	699.90	332.46	630.43	407.27	765.09
	Top	354.55	670.18	318.23	604.81	388.73	731.71

35. Section 10.2. How much cadmium was assumed in the calculation for Table 10.2.1?

Licensee's Response:

The assumed masses of cadmium were 1.04kg for the pneumatic tube and 1.57kg for the whale tube.

36. Section 12.6. Are there any quality assurance tests that University of Wisconsin will apply upon receipt of the fuel? If yes, please briefly describe.

Licensee's Response:

Yes. The fuel rods are inspected by the CERCA Quality Inspectors at the fabrication facility. Following the CERCA inspection, the Idaho National Laboratory performs an on-site Source Inspection of all of the CERCA QA inspection/verification records. The INL then performs a visual inspection of all fuel elements, records all imperfections, and verifies the imperfections are within the established design criteria. The INL also performs a verification of all accessible dimensions on a statistical sampling of fuel elements. If TRIGA fuel fabrication is underway at the facility during the inspection visit, the INL QA inspector will observe the fuel element assembly process to ensure that the process is being carried out as expected.

The UWNR will perform a Receipt Inspection of each fuel element while the INL Quality Assurance Inspector is at the UWNR facility. The Receipt Inspection ensures that fuel elements were not damaged during packaging or transport. The Receipt Inspection is a visual inspection, and observed imperfections can be compared to the Source Inspection records with the INL QA Inspector. The UWNR will also review all of the CERCA and INL QA inspection/verification records to ensure each fuel element complies with design specifications.

Following the receipt inspection, fuel elements will be measured in accordance with UWNR 142, Procedure for Measuring Fuel Element Bow and Growth, to establish the baseline for future annual measurements.

37. Section 12.7: At what stage of the startup will the graphite reflectors be inserted?

Licensee's Response:

The graphite reflectors will be inserted after loading to the proposed 21 bundle core. They will be inserted only after measuring their individual reactivity worth and verifying that adequate shutdown margin will be maintained. After loading to the full J21-R14 core the shutdown margin will be measured and verified to meet Technical Specification limits.

38. Section 12.7: Your application states that "IFE [instrumented fuel element] bundles will be loaded and the IFE then installed before updating the 1/M plot for that bundle." Will the IFE be tested and calibrated? If so, when?

Licensee's Response:

Each IFE will be tested prior to loading into the core. Upon receipt, the resistance values will be verified to be consistent with manufacturer reported values to rule out a possible short or open circuit in the thermocouple. Than the signal from the IFE will be read with a calibrated process meter and the IFE reading will be verified to be consistent with a known reference temperature.

39. Section 12.7: Your application states you will determine shutdown margin using the rod drop method. Will you determine excess reactivity? If so, please discuss?

Licensee's Response:

Yes. Excess reactivity will be determined upon reaching criticality and the operational core using the rising period rod bump method. Excess reactivity will not be determined while loading from initial criticality to the operational core; however shutdown margin will be determined using the rod drop method after each fuel bundle addition to ensure compliance with Technical Specification 3.1.

40. Section 12.7. How many power increment steps will be utilized and how large are the increments?

Licensee's Response:

10 increments in power level will be used, from low power (less than 1kW) up to 1MW full power in 100kW steps.

41. Section 12.7: Will the power and fuel temperature coefficient of reactivity calculations be based on measured data or based on computer models?

Licensee's Response:

Power and fuel temperature coefficients of reactivity will be calculated based on measured data during startup testing.

42. Section 13. From a review of your accident analyses, it appears that some of the scenarios (e.g. maximum hypothetical accident (MHA), loss-of-coolant accident (LOCA), etc.) may have a potential radiological impact outside the reactor facility. From a review of your emergency plan, dated 5/14/04, it is not clear how response is handled in any potentially impacted areas outside the operations boundary in the engineering building. Please discuss.

Licensee's Response:

In accordance with UWNR 150 emergency procedure, "Reactor Accident, Fission Product Release, or Major Spill of Radioactive Materials," the site boundary (as defined in the emergency plan revision 4) is evacuated. It is recognized that an inconsistency exists between the emergency plan definition of the emergency planning zone (EPZ) and the evacuation zone defined in UWNR 150. Therefore, it is proposed that the emergency plan be revised such that the emergency planning zone is defined by the site boundary, rather than the operations boundary. The revision 6 of the emergency plan is attached for approval*. See attachment 7.

The emergency plan was also revised to account for updated dose calculations for four accident events in Table 2, as well as to update the emergency action levels due to revised released inventories as a result of the conversion to LEU. After correcting for the LEU-BOL power distribution, the BOL case was found to be more limiting than the MOL case which was reported in the LEU conversion SAR, therefore the changes to revision 6 of the emergency plan use the revised LEU-BOL power distribution. The dose calculations reflect the revised LEU 30/20 core design as well as current methodologies of calculation in the analysis report as reported in sections 13.1.5.2, 13.1.6.3, 13.1.7.3, and 13.1.8.1. However, the results as reported in the analysis report were updated to use more appropriate fission product release fractions. The analysis report calculated the release fraction based on the maximum centerline temperature, but a more accurate approach is to use an effective release fraction calculated by volume integrating the release fraction equation across the fuel temperature distribution, both axial and radial. This is appropriate since the release fraction measurements were made on small isothermal fuel samples (General Atomics, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuels." GA E-117-833, February 1980, page 5-5). The revised release fractions are , approximately 10% of the values listed in chapter 13 of the analysis report. This results in changes to the emergency action levels and potential exposure as detailed below.

The emergency action levels were changed in revision 5* of the emergency plan as a result of modifications to the ventilation system. The original calculations for ventilation system operable assumed that the release was instantaneous, and that it was vented at a constant rate for the amount of time it would take to exchange one confinement volume, where the confinement volume was assumed to be 2000m^3.

The revision 4 emergency action level was derived by assuming the insoluble beta emitter activity of was released. This activity was released to the confinement volume and vented at a constant rate thereby producing a concentration of:

$$\frac{\quad}{2000m^3} =$$

This was rounded down to for revision 4 of the Emergency Plan.

Revision 5 of the emergency plan was performed in accordance with a 10 CFR 50.54(q) analysis which included changes resulting from the new ventilation system. The new ventilation system is designed to sweep air into the reactor laboratory from public access space through the auxiliary support spaces surrounding the reactor confinement. This limits the potential spread of airborne contamination. To accomplish this, exhaust is taken from the reactor confinement and the auxiliary spaces and combined in a common plenum, prior to release from the stack. The new ventilation system has a nominal exhaust flow rate from confinement of 2700 scfm and an exhaust flow rate of 9600 scfm ($4.531 m^3/s$) in the mixing plenum where the stack sample is taken. The additional dilution would decrease the release concentration and therefore the emergency action level was revised.

As calculated in the 2000 license renewal SAR Rev 2, Appendix A, page A-4, the time to vent confinement is 1569s. The total volume of air exhausted in 1569s is $4.531 m^3/s * 1569s = 7109m^3$. Therefore the revision 5 action level is:

$$\frac{\quad}{7109m^3} =$$

This was rounded down to 1E-4μCi/ml for revision 5 of the emergency plan.

For revision 6, only the activity of was revised. By using the revised BOL power distribution and the revised release fractions, the activities of the insoluble beta emitters are BOL, MOL, and EOL. The insoluble beta emitters are Kr-85m, Kr-85, Kr-87, Kr-88, Kr-89, Xe-133, Xe-135, Xe-137, and Xe-138. If the more limiting value of at EOL is used, then the revision 6 action level is:

$$\frac{\quad}{7109m^3} =$$

This is rounded down to for revision 6 of the emergency plan, when the ventilation system is operable. When the ventilation system is inoperable, the revision 6 action level is:

$$\frac{\quad}{2000m^3} =$$

This is rounded down to for revision 6 of the emergency plan.

The following 4 accident events are therefore revised in revision 6:

1. Event: Severe fuel clad leak-approaching MHA size with pool level normal and ventilation system operative.
   Potential Release: whole body, thyroid revised to whole body thyroid
   Action Level: revised to

2. Event: Severe fuel clad leak-approaching MHA size with pool near empty and ventilation system normal.
   Potential Release: whole body, thyroid revised to whole body thyroid
   Action Level: revised to

3. Event: Severe fuel clad leak-approaching MHA size with pool level normal and ventilation system inoperative.
   Potential Release: whole body, thyroid revised to whole body. thyroid
   Action Level: revised to

4. Event: Severe fuel clad leak-approaching MHA size with pool near empty and ventilation system inoperative.
   Potential Release: whole body, thyroid revised to whole body, thyroid
   Action Level: revised to

* Note: Action levels for ventilation system normal are reported as in revision 4 of the emergency plan. These values were recently changed, as a result of a new ventilation system, to in revision 5, which was submitted under separate cover following a 10 CFR 50.54(q) analysis. Changes to the emergency plan as a result of the LEU conversion are submitted here as revision 6. For convenience, revisions 4, 5 and 6 are included in attachment 7, both with and without strikeouts indicating changes from the previous version.

43. Section 13.1. It is stated that certain isotopes (e.g., I-130m, I-136, Kr-89, Xe-137) were not included in the estimates for whole-body and thyroid dose because of their "short half-lives (less than 10 minutes)." Given the short exposure time of five minutes, these isotopes will make a contribution to the doses. Please justify their exclusion, or submit revised doses.

Licensee's Response:

The primary reason as stated in the analysis report for neglecting these isotopes is the lack of any published dose coefficients. However, using the methodology in reference 24 of the analysis report, the whole-body effective dose coefficients were manually computed for the short-lived isotopes and revised results are shown below (Table numbering represents the original numbering in the LEU Conversion Analysis SAR). The revised LEU-BOL power distribution was used. Also, the revised results use the more realistic temperature distribution integrated release fraction as described in the response to question 42. The thyroid dose contributions were not revised. The source for the thyroid dose coefficients, "Federal Guidance Report No. 11: Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," states on page 25 that the biological half-time to transport iodine from the blood to the thyroid is 6 hours. Any short-lived isotopes would therefore have a negligible contribution to the thyroid dose.

Revised Table 13.1.4 MHA Occupational External Dose by Isotope

Isotope	Effective Dose Coef. (rem-m^3/Ci-s)	Revised HEU External Worker Dose (rem)	LEU BOL External Worker Dose (rem)	LEU MOL External Worker Dose (rem)	LEU EOL External Worker Dose (rem)
Br-82	4.810E-01	8.029E-07	6.870E-07	5.344E-06	1.029E-05
Br-83	1.413E-03	1.052E-06	3.753E-06	3.555E-06	2.769E-06
Br-84	3.482E-01	4.872E-04	1.735E-03	1.635E-03	1.265E-03
Br-85	7.898E-03	1.374E-05	4.884E-05	4.601E-05	3.554E-05
Br-87	2.680E-01	8.022E-04	2.853E-03	2.678E-03	2.059E-03
I-130m	1.637E-02	5.174E-08	3.119E-08	3.591E-07	7.408E-07
I-131	6.734E-02	2.702E-04	9.491E-04	9.421E-04	7.643E-04
I-132	4.144E-01	2.480E-03	8.866E-03	8.609E-03	6.954E-03
I-133	1.088E-01	1.018E-03	3.621E-03	3.498E-03	2.796E-03
I-134	4.810E-01	5.084E-03	1.809E-02	1.742E-02	1.387E-02
I-135	2.953E-01	2.573E-03	9.156E-03	8.840E-03	7.064E-03
I-136	3.931E-01	1.668E-03	5.967E-03	5.717E-03	4.531E-03
Kr-83m	5.550E-06	4.132E-09	1.472E-08	1.396E-08	1.087E-08
Kr-85m	2.768E-02	4.868E-05	1.730E-04	1.631E-04	1.260E-04
Kr-85	4.403E-04	1.142E-08	5.143E-09	5.828E-08	9.676E-08
Kr-87	1.524E-01	5.421E-04	1.926E-03	1.809E-03	1.392E-03
Kr-88	3.774E-01	1.896E-03	6.736E-03	6.327E-03	4.865E-03
Kr-89	1.411E-01	8.994E-04	3.196E-03	2.994E-03	2.296E-03
Xe-131m	1.439E-03	5.716E-08	1.963E-07	2.017E-07	1.592E-07
Xe-133m	5.069E-03	1.387E-06	4.455E-06	4.800E-06	3.868E-06
Xe-133	5.772E-03	5.402E-05	1.821E-04	1.856E-04	1.485E-04
Xe-135m	7.548E-02	1.195E-04	4.242E-04	4.165E-04	3.394E-04
Xe-135	4.403E-02	2.550E-04	9.873E-04	9.367E-04	7.222E-04
Xe-137	2.604E-02	2.166E-04	7.703E-04	7.421E-04	5.919E-04
Xe-138	2.135E-01	1.848E-03	6.571E-03	6.284E-03	4.959E-03

Revised Table 13.1.6 MHA Total Occupational Dose during 5 minute evacuation

	External Dose (mrem)	Thyroid Dose (mrem)	TEDE (mrem)
Previous HEU SAR	10	N/A	N/A
Revised HEU Analysis	20.3	2,110	83.7
LEU BOL Analysis	36.1	3,730	148
LEU MOL Analysis	34.6	3,670	145
LEU EOL Analysis	27.4	2,960	116

Revised Table 13.1.8 MHA Building Occupant Doses for Ground Release

	External Dose (mrem)	Thyroid Dose (mrem)	TEDE (mrem)
Revised HEU Analysis	1.26	131	5.18
LEU BOL Analysis	2.24	231	9.16
LEU MOL Analysis	2.14	227	8.95
LEU EOL Analysis	1.70	183	7.20

Revised Table 13.1.11 Near MHA with Pool Intact Occupational Dose during 5 minute evacuation

	External Dose (mrem)	Thyroid Dose (mrem)	TEDE (mrem)
Previous HEU SAR	N/A	18,900	N/A
Revised HEU Analysis	7.32	211	13.7
LEU BOL Analysis	13.1	373	24.2
LEU MOL Analysis	12.4	367	23.4
LEU EOL Analysis	9.69	296	18.6

Revised Table 13.1.12 Near MHA with Pool Intact Building Occupant Doses for Ground Release

	External Dose (mrem)	Thyroid Dose (mrem)	TEDE (mrem)
Revised HEU Analysis	0.453	13.1	0.845
LEU BOL Analysis	0.808	23.1	1.50
LEU MOL Analysis	0.768	22.7	1.45
LEU EOL Analysis	0.600	18.3	1.15

By including the short-lived isotopes, the TEDE numbers are higher than previously reported by at least 4%, but no more than 11%, and are all still within limits. However, a further reduction by a factor of approximately 10 is achieved using the more realistic release fraction. This is still a conservative calculation since no credit is taken for radioactive decay of the isotopes.

44. Section 13.1.2. In equation 13.1.1, the exponent is given as $\exp(-1.34\times10^{-4}/T)$. Should the exponent be $\exp(-1.34\times10^{4}/T)$?

Licensee's Response:

Yes, the exponent should be $\exp(-1.34\times10^{4}/T)$.

45. Section 13.1.5.1. Why was the fifth floor of the Mechanical Engineering Building specifically excluded from the dose calculation? Are individuals on the fifth floor evacuated? If not, what would their non-occupational dose be?

Licensee's Response:

Individuals on the fifth floor are evacuated with the rest of the building. The volume of the fifth floor was neglected in dose calculations for conservatism and because the open air atrium in the central wing of the building would allow for readily mixing of building air between the first through fourth floors, while the fifth floor is isolated from the atrium. The reported doses to building occupants apply to all individuals in the building, including those on the fifth floor.

46. Section 13.1.7.1. During winter months, you mention the buoyancy rise of the stack exhaust would be significant. What is the buoyancy and momentum rise during the winter and summer months, respectively? What assumption is most conservative?

Licensee's Response:

The momentum rise, as calculated using Equation 13.1.4 on page 187, is 11.3m regardless of the time of year.

The methodology for calculating the buoyancy rise is taken from "Workbook of Atmospheric Dispersion Estimates" by Turner, 1994 (page 3-2), which is reference 27 of the LEU conversion analysis report. First, the intermediate variable of buoyancy flux is calculated using the equation below:

$$F = gvd^2\Delta T/(4T_s)$$

Where: F = buoyancy flux (m^4/s^3)
- g = acceleration of gravity (9.8 m/s^2)
- v = stack gas exit velocity (m/s)
- d = top inside stack diameter (m)
- ΔT = stack gas temperature minus ambient air temperature (K)
- T_s = stack gas temperature (K)

The monthly average temperatures as reported in the HEU 2000 license renewal SAR are assumed. The coldest month is January with a temperature of 16.8°F (264.7K) and the warmest month is July with a temperature of 71.4°F (295.0K). The stack outlet temperature is assumed to be 72°F (295.4K) year-round. The stack gas exit velocity is 17.272m/s and the top inside stack diameter is 0.7747m as reported in the LEU conversion analysis report. The buoyancy flux is therefore calculated to be 2.6m^4/s^3 in the winter, and 0.2m^4/s^3 in the summer.

The buoyancy rise is given by the following equation:

$$\Delta H = 21.425F^{3/4}/u \qquad \text{if } F \text{ is} < 55$$

Where: u = wind speed at top of stack (m/s)

The buoyancy rise is calculated to be 12.4m in the winter and 1.8m in the summer.

A lower buoyancy rise is more conservative because it will result in a lower effective stack height which in turn will cause higher ground-level concentrations. Because the buoyancy rise is not steady year-round, it was assumed to be 0m in the analysis report, which is more conservative.

47. Section 13.1.7, 13.1.8 and 13.1.9. For those scenarios where the ventilation system is in operation, what is the dose to persons in the Mechanical Engineering Building from shine from the volumetric source term in the confinement building until it is ventilated to the environment?

Licensee's Response:

The dose was calculated using an MCNP model of the confinement structure. The revised LEU-BOL power distribution was used, as was the revised temperature distribution integrated release fraction as discussed in the response to question 42. If the total released inventory is assumed to be uniformly dispersed in the confinement volume, then the dose rate in the nearest unrestricted area of the building is calculated to be If this dose rate existed during the time required to exhaust the entire confinement, 26 minutes, it would result in a dose of approximately

48. Section 13.1.7. Can a person in the unrestricted environment receive a dose from shine from the plume passing overhead greater than the immersion dose when the plume reaches the ground?

Licensee's Response:

No. The shine dose from an overhead plume was calculated using an MCNP5 model of the plume as a solid cone source. The cone was sub-divided along its length into 10m segments (frustums) and the plume was modeled as a puff release. The source term was defined as the entire released inventory, which was inserted into a single 10m segment of the plume. After correcting for the LEU-BOL power distribution, the BOL case was more limiting than the MOL case as reported in the LEU Conversion SAR, therefore the revised BOL case was analyzed. Also, the revised temperature distribution integrated release fraction was assumed as discussed in the response to question 42. The dose rates from this source term were calculated at various fixed receptor locations, and then the source term was moved into the next 10m segment of the plume to simulate the puff cloud moving down-wind. For each calculation, the calculated dose rate, in mrem/hr, was multiplied by the time required for the puff cloud to travel the 10m distance assuming the minimum monthly average wind speed of 3.54m/s. This time of travel for the puff cloud is 2.8s, or 7.8E-4hr for each 10m length of down-wind travel. In this manner, the dose contributions from the passing puff cloud were calculated at each receptor location individually for each 10m distance and then summed together. The farthest receptor location was at 150m, because this corresponds to the distance of highest ground-level concentration reported on page 190, therefore any exposure beyond this point is a result of immersion in the plume rather than shine from overhead. The puff was modeled out to a distance of 250m, because beyond this distance the dose contribution to the receptor located at 150m was negligible. The puff distance of 250m corresponds to a total exposure time of 71s (assuming 3.54m/s). The radius of the cone at each down-wind distance was determined by the class A vertical dispersion coefficient used in the Gaussian plume model (Equation 13.1.7 on page 189). Class A was chosen because it is the most limiting, since it results in the most rapid expansion of the cone radius bringing the edge of the cloud closer to the ground, therefore decreasing the distance from the plume to the receptor and increasing dose. Within each segment of the cone source, the concentration of the plume was uniform. The geometry of the problem is shown in the figure below.

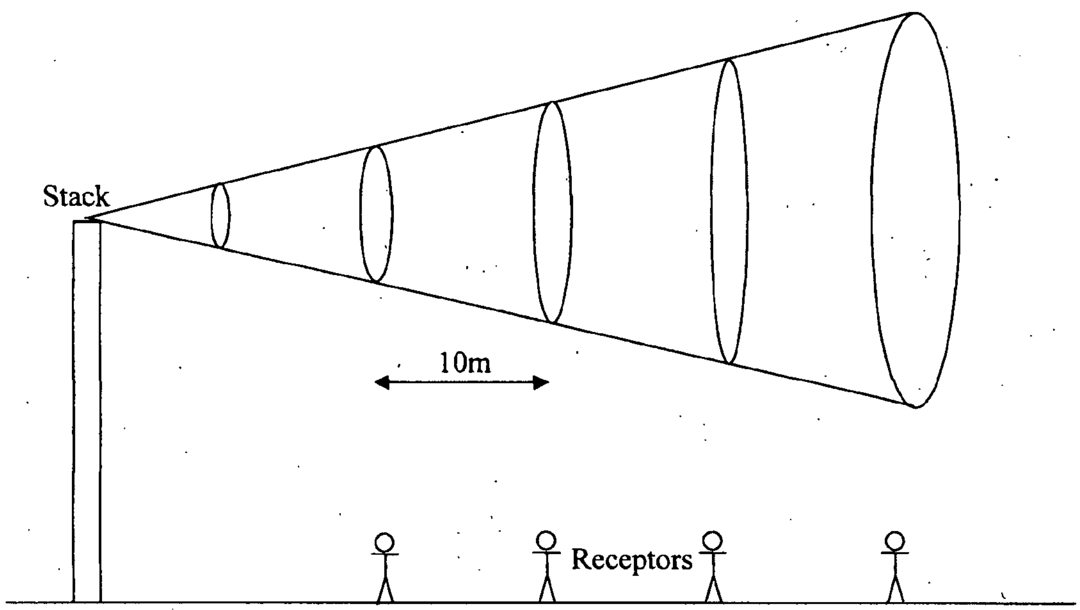

Figure RAI-48-1, Plume Shine Model

Calculations of the external dose from the overhead plume do not exceed 0.01 mrem. This is well below the maximum dose due to immersion of 0.324 mrem at 148m as reported in section 13.1.7. Even when the shine dose is added to the immersion dose, the maximum combined dose is 0.329 mrem compared to 0.324 mrem reported in section 13.1.7. The combined dose does not exceed the previously reported 0.324 mrem until the down-wind distance approaches 150m, which is approximately the point of maximum ground-level concentration. Therefore, the maximum dose due to immersion, as calculated in the conversion analysis, is more limiting than the shine dose from the plume passing overhead. The total shine dose is given in the table below as a function of receptor distance.

Table RAI-48-1, Plume Shine Doses

Receptor Distance (m)	Dose (µrem)
26	2.9
30	3.1
40	3.4
50	3.6
60	3.8
70	4.0
80	4.1
90	4.3
100	4.4
110	4.5
120	4.6
130	4.8

140	5.0
150	5.2

49. Section 13.2 In the Rapid Insertion of Reactivity Accident it appears the total scrammable worth of the blades, 8.86 %Δk/k was instantaneously inserted into the core 2 seconds after the transient. If the reactor was operating at 1.3 MW at BOL, is 8.86 %Δk/k blade worth available to scram the reactor? What is the calculated position of the blade at BOL?

Licensee's Response:

No, 8.86 %Δk/k of reactivity is not available to scram the reactor. The shutdown margin with no experiments and all control elements fully inserted was determined to be 5.677 %Δk/k, where the critical bank height was 10.13in. The power defect from low power to 1.3MW, with a bank height of 11.73in, was determined to be 1.411 %Δk/k at LEU BOL. Therefore, the shutdown margin at 1.3MW would be 7.088 %Δk/k. Using the revised value for shutdown margin inserted 2 seconds after the transient, the revised results of section 13.2 are given below. As can be seen in the LEU BOL plots the change is negligible and this trend is similar for HEU, LEU-MOL, and LEU-EOL plots.

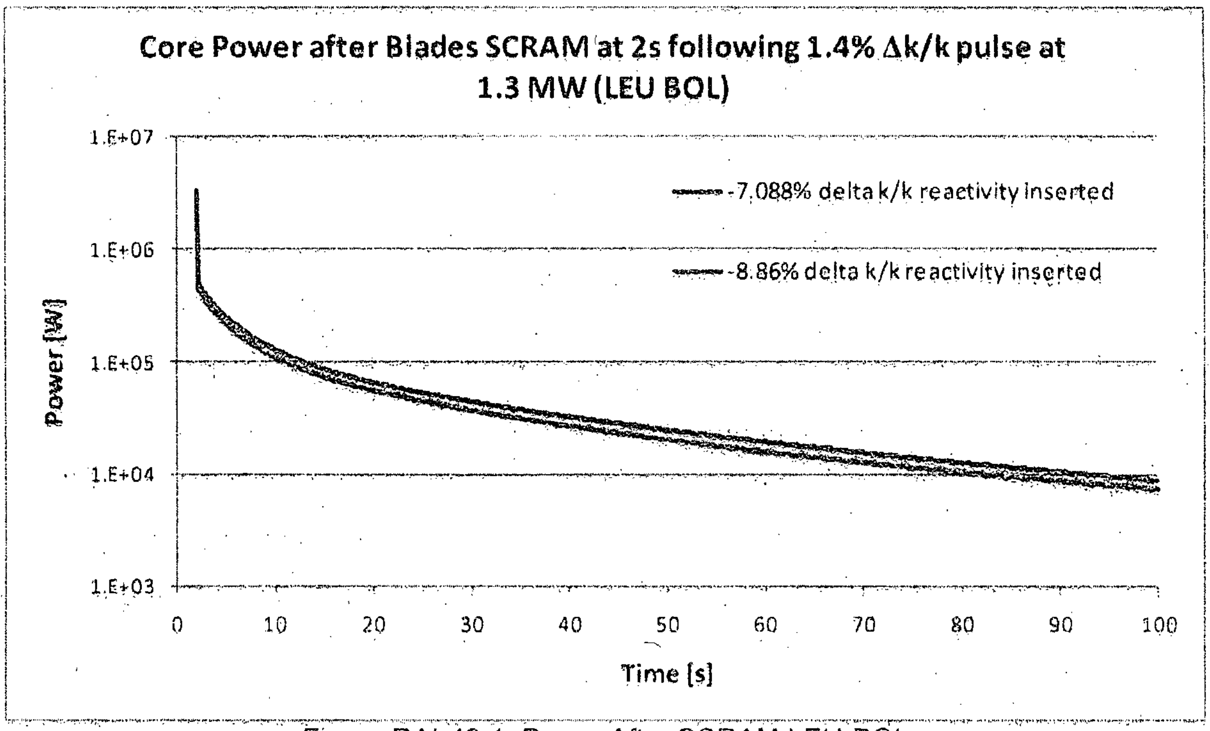

Figure RAI-49-1, Power After SCRAM LEU-BOL

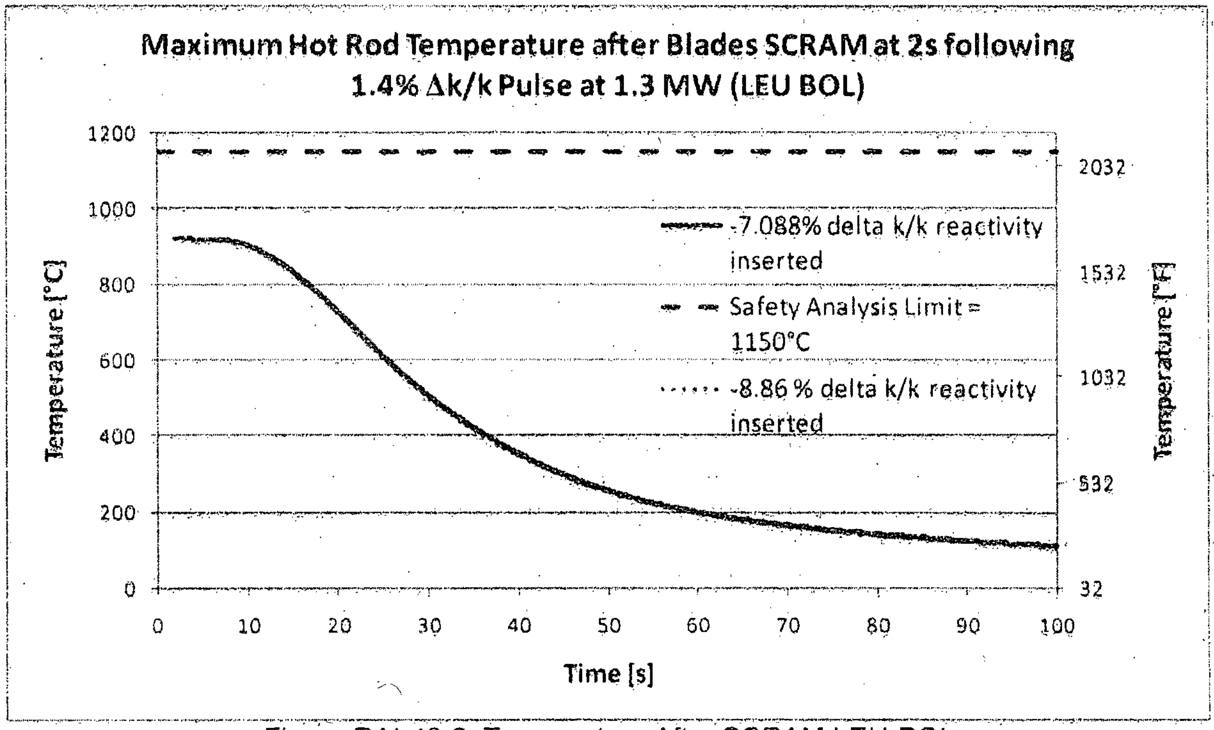

Figure RAI-49-2, Temperature After SCRAM LEU-BOL

50. Section 13.2.1 The total scrammable blade worth used in the analysis of the rapid addition of reactivity accident included the worth of the transient rod. However, the Technical Specifications (TS 3.3.3 Table 1) require that in the pulse mode a transient rod scram occurs 15 seconds or less after the pulse. Is the 2-second scram delay assumption (including the transient rod) conservative?

Licensee's Response:

Typically, for TRIGA reactors, the rapid addition of reactivity accident is often analyzed as a pulse from full power. However, pulsing from full power requires willful violation of procedure and failure of the pulse mode interlocks and is therefore not considered credible. Therefore, the rapid addition of reactivity accident analysis assumes a failure of an experiment with a total worth of 1.4 %Δk/k. Therefore the total worth of the shutdown margin is available, including the transient rod, at 2 seconds, because the transient rod must comply with TS 3.3.1 when not in pulse mode.

51. Section 13.3.2. In reference to Table 13.3.1, what are the estimated doses to the staff and people in the Mechanical Engineering Building from direct radiation following a LOCA?

Licensee's Response:

From Table 13.3.1, the calculated dose rate to the 3rd floor non-restricted classroom is significant, but in the event of a loss of coolant accident the building evacuation alarm would alert people to evacuate these classrooms before the core was completely uncovered. In order to estimate the integrated dose received by a member of the public during the evacuation, the MCNP5 model of the unshielded core was modified to include partial water shielding at several time steps. The core gamma source term was also modified to simulate an appropriate level of decay from full power. The integrated dose to the 3rd floor classroom was calculated at various times during the pool water loss and is shown in the figure below.

Figure RAI-51-1, 3rd Floor Integrated Dose During LOCA

The pool water would drain to approximately 7.4 ft above the core in at which point it would trip the bridge area radiation monitor, which in turn would automatically initiate the building evacuation alarm. A 5 minute evacuation time from the sounding of the evacuation alarm is assumed. Therefore, the hypothetical member of the public that remains in the 3rd floor classroom for 5 minutes following the automatic initiation of the building evacuation alarm (after start of the LOCA) would receive a dose of during the first and an integrated dose of before evacuating at . Realistic doses would be far less than this, because the preceding analysis does not take into account time spent in hallways and stairwells (where the dose rate is much lower) during the evacuation. Because the time spent in the high dose rate field in the 3rd floor classroom would be far less than 5 minutes, the integrated dose would be substantially lower due to the majority of the dose being received in the final minute as shown in the figure above.

Using the same model, the maximum dose rate at the site boundary, which is the area evacuated. was calculated to be immediately after the core is uncovered (after the start of the LOCA). This dose rate would remain for no more than 24 hours for a total dose of by which time the emergency procedures would refill the pool with water.

52. Section 13.3.3. The estimated time required to drain the pool was calculated using Equation 13.3.1. Does the calculated drain time of 836 sec. represent the time to lower the pool water level to the top of the core or to the core mid-plane where the beam port is located?

Licensee's Response:

The drain time of represents the time .

53. Section 13.3.3. Do the calculations of the dose rates shown in Table 13.3.1 assume the fuel is completely uncovered or half-covered with water?

Licensee's Response:

The calculated dose rates in Table 13.3.1 assume the core is completely uncovered. However, in order to determine the impact of the competing effects of increased shielding and increased reflected scatter from the water in the event the core was partially covered, a modified case with the water level at core mid-plane was analyzed and found to have no statistical difference from the uncovered case.

54. Table 13.3.3. According to Table 13.3.3, the starting temperatures for the LEU-BOL and LEU-MOL cores are the same. Please discuss.

Licensee's Response:

The wrong initial starting temperature was reported for LEU-BOL. Using the correct axial power shape for LEU-BOL, the starting temperature is 506.19°C, end of water transient temperature is 75.29°C, and the maximum temperature in hot rod is 652.59°C 8,350 seconds after start of transient. The new LEU-BOL complete LOCA curve can be seen below:

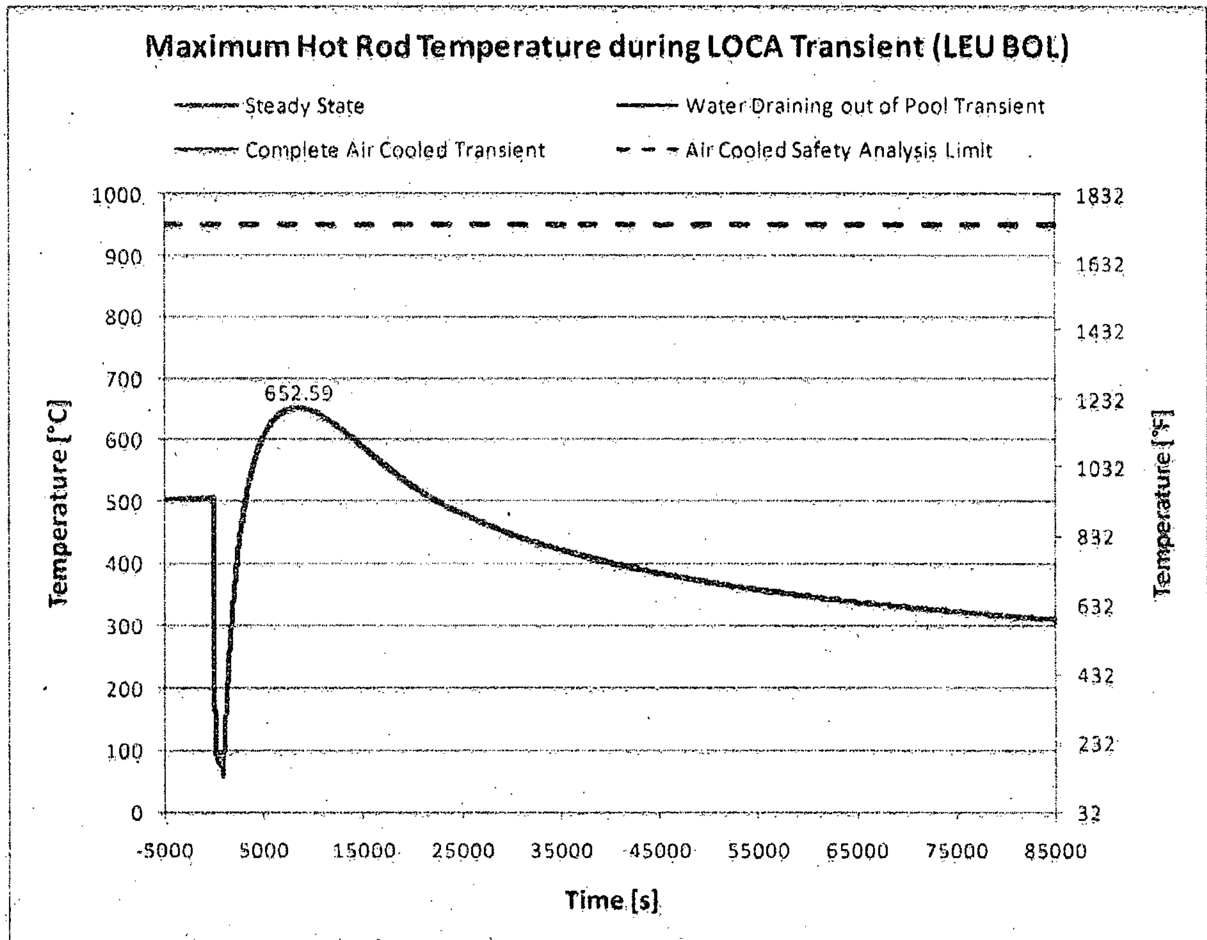

Figure RAI-54-1, Maximum Hot Rod Temperature during LOCA (LEU-BOL)

55. Section 13.3.3. Your application states analysis has been performed that demonstrates a complete LOCA is more limiting than a partial LOCA. Please discuss or provide references 33 and 34.

Licensee's Response:

An excerpt from Reference 33 is provided below detailing the LOCA. The model is detailed in sections 2.1.3 and 2.2.

2.1.3 LOCA Model

The RELAP5 LOCA model uses the same 2-channel model used in the pulsing analysis, with a few exceptions. First, since the power is input manually, the point reactor kinetic equations were not used. Secondly, it was necessary to split the problem into three parts:

1. 2-channel steady state model at 1.02 MW, 5.7912m (19 feet) of water above the core, and inlet water temperature of 54.44°C (130°F)
2. 2-channel transient model where the loss of water is modeled by losing water pressure until 101.3 kPa achieved at to simulate losing 5.7912m (19 feet) of water.
3. 2-channel transient model where the water coolant has been replaced with air. It is assumed the inlet air temperature is 25°C (77°F).

The entire LOCA transient was run for a total of 86,400 seconds or 1 day in order to ensure the peak fuel temperature was captured during the analysis. In addition, since the decay heat corresponds with the steady state operational history, the pin power, axial, and radial peaking factors are identical to the steady state analysis. The delayed neutron power after the rods drop into the core is not substantial enough to change the decay heat power profile from the steady state power profile.

2.2 EES

EES[1] was used to analyze both the complete and partial LOCA for the UWNR. EES allows the user to enter in equations and use the embedded fluid libraries to calculate specific parameters at a particular set of conditions. This is very helpful so that the user does not need to constantly interpolate tables to find the specific property for his/her particular problem. By entering in the governing equations for the gravitational pressure gain and the frictional pressure loss, a natural convection loop could be calculated in order to determine the maximum fuel temperature during a LOCA.

2.2.1 Complete LOCA Model

In order to perform the complete LOCA analysis by hand, it is necessary to understand the governing equations used in the analysis. The main governing equation is the gravitational buoyancy pressure gain set equal to the frictional pressure loss. The gravitational buoyancy pressure gain is:

$$\Delta P_{grav} = g \cdot node \cdot \sum_{i=1}^{15} \left(\rho_i \cdot \beta_i \cdot (T_i - T_\infty)\right) + g \cdot L_{non-fuel} \cdot \rho_{15} \cdot \beta_{15} \cdot (T_{15} - T_\infty)$$

Where:

- g = 9.81 m/s^2
- node = 0.0254 m (15 axial nodes)
- ρ_i = density of the air at the local air temperature
- β_i = local volume expansion coeffiicnet of air at the local air temperature
- T_i = local air temperature
- T_∞ = 298 K
- $L_{non\text{-}fuel}$ = 0.1905 m (length of fuel above / below active fuel region)

The frictional pressure loss is determined by the following equation:

$$\Delta P_{fric} = \left(f_{in} L_{non-fuel} \rho_{in} \frac{v_{in}^2}{2} + f_{out} L_{non-fuel} \rho_{out} \frac{v_{out}^2}{2} + \sum_{i=1}^{15} node \cdot f_i \rho_i \frac{v_i^2}{2}\right) \cdot \frac{1}{2D_H} + K_{in} \rho_{in} \frac{v_{in}^2}{2} + K_{out} \rho_{out} \frac{v_{out}^2}{2}$$

Where:

- $f = {}^{16}/_{Re}$ Assuming Re < 2000
- D_H = 0.0154703 m (hydraulic diameter of the entire core)
- v = local air velocity
- "in" corresponds to inlet core conditions before reaching heated fuel
- "out" corresponds to outlet core conditions after being heated by fuel
- K_{in} = 2.02 (inlet pressure loss coefficient)
- K_{out} = 1.38 (outlet pressure loss coefficient)

The local Reynolds number is calculated by, where μ is the local viscosity of the air:

$$Re = \frac{\rho v D_H}{\mu}$$

In order to determine the properties of air, it is necessary to determine the energy released per node, air temperature per node, and the mass flow rate of the air by the following equations:

$$q_i = \dot{m} Cp_i \Delta T_i$$

$$q_i = \frac{ppf_i \cdot \dot{Q}}{15}$$

$$T_i = \Delta T_i + T_{i-1}$$

$$\dot{m} = \rho_i A v_i$$

Where:

q_i =	local energy released per node across the core
$\dot{m}$ =	total mass flow rate of air in the core
Cp_i =	local specific heat at the local air temperature
ΔT_i =	local change in temperature across the node
ppf_i =	local axial power peaking factor
$\dot{Q}$ =	total core power as a function of time
T_{i-1} =	previous local air temperature. When i=1, T_o = 298 K
A =	0.046598868 m^2 is the flow area (with rods) or 0.138943 m^2 (without rods)

After determining these conditions, it is possible to find the mass flow rate of air and the temperature of the air. In addition to this, it is necessary to calculate the maximum fuel temperature of the rods. This is done by using the Dittus-Boelter equation:

$$Nu_i = 0.023 Re_i^{0.8} Pr_i^{0.4}$$

$$Nu_i = \frac{h_i D_H}{k_i}$$

Where:

Pr_i =	local Prandtl number at the local air temperature
h_i =	local heat transfer coefficient
k_i =	local air thermal conductivity

Having determined the heat transfer coefficient, the temperature of the fuel rod can be determined via the following equations:

$$q''_i = \frac{q_i}{N_{rod} \pi D_{rod} node}$$

$$T_{outerclad,i} = T_i + {q''_i}/{h_i}$$

Where:

q''_i =	local heat flux out of the rod
N_{rod} =	91 (number of rods in the HEU core)
D_{rod} =	0.0353894 m (diameter of the rod)
$T_{outerclad,i}$ =	local temperature of the outer clad

While there should not be a significant difference between the outer clad temperature and the maximum fuel centerline temperature, additional calculations were performed in order to determine the maximum fuel centerline temperature. This is necessary because the RELAP5 results for the LOCA used the maximum fuel temperature and not the inner cladding temperature in which fuel cladding interactions would become an issue. Thus, the following equations were constructed:

$$T_{gapclad,i} = T_{outerclad,i} + q''_i \pi r_{clad} \frac{\ln\left(r_{clad}/r_{gap}\right)}{2\pi k_{clad,i}}$$

$$T_{fuelgap,i} = T_{gapclad,i} + 2\pi r_{clad} q''_i \frac{\ln\left(\left(r_{fuel} + t_{gap}\right)/r_{fuel}\right)}{2\pi k_{gap,i}}$$

$$T_{fuelmax,i} = T_{fuelgap,i} + \frac{2\pi r_{clad} q''_i}{4\pi k_{fuel}}$$

Where:

$T_{gapclad,i}$ =	local fuel temperature of the gap/clad region
$T_{fuelgap,i}$ =	local fuel temperature of the fuel/gap region
$T_{fuelmax,i}$ =	local fuel temperature of the fuel centerline regio
r_{clad} =	0.0179197 m (radius of the clad)
r_{gap} =	0.01740154 m (radius of fuel + thickness of gap)
t_{gap} =	2.54E-06 m (assumed thickness of gap)
r_{fuel} =	0.017399 m (radius of fuel)
k_{fuel} =	18 W/m-K (thermal conductivity of the fuel)

The thermal conductivity of the cladding is determined by the following equation in W/m-K:

$$k_{clad,i} = 0.01466 T_{clad,i} + 10.84697$$

The thermal conductivity of the gap is determined by the following equation in W/m-K:

$$k_{gap,i} = 8.58773E-15 \cdot \left(T_{gapclad,i}\right)^4 - 3.86727E-11 \cdot \left(T_{gapclad,i}\right)^3 + 5.83945E-8 \cdot \left(T_{gapclad,i}\right)^2 - 9.87506E-6 \cdot \left(T_{gapclad,i}\right) - 1.01597E-2$$

The equations for the thermal conductivity of the stainless steel cladding and the gap were determined using a best fit line to the known data points provided by ANL. The known data points were incorporated into the RELAP5 script as seen in the appendix. Interestingly, the difference in temperature between the maximum fuel centerline and the outer clad is approximately 2°C.

After entering these equations into EES, it was possible to find the maximum temperature of the fuel rods as a function of time if the initial core power was given as a function of time. For the HEU core, the air-cooled portion of the LOCA power transient can be given by the following power function:

$$\dot{Q} = 2.27992E + 05 \cdot (time)^{-0.365981}$$

To determine the maximum temperature of the hot rod using the hand calculation model, the power function was multiplied by the pin power peaking factor of the hot rod. This gave a first approximation of what the maximum fuel temperature in the hot rod would be.

2.2.2 Partial LOCA Models

In conjunction with the total LOCA hand calculation, analysis was also performed looking at the case in which the water does not completely drain from the core but partially covers the core. Kevin Austin calculated the water would cover the bottom 4.5 inches (11.43 cm) of active fuel and the remaining 10.5 inches (26.67 cm) of active fuel would be air cooled. For simplicity with the 15 axial nodes employed in the model, this analysis assumed the water would cover the bottom 5 inches. It is not anticipated this difference would drastically change the results.

At first, an air-only model was constructed looking at the case in which axial conduction was ignored and air cooling was only supplied. Since the beam ports do not make direct contact with the fuel to provide fresh air, only air coming down the empty slots of the grid box could be used to create a driving flow of air. The most limiting case was looked at where the area of interest was rods next to blade 3 and the regulating blade. Air flow would come down the two empty grid boxes and then go across 5 rods before going up. If the mass flow rate were assumed to be the same at the top of each fuel channel, a pressure loss scenario similar to the total LOCA could be created. Since it would have been very cumbersome to create a model based on 10 axial nodes for 10 rods (5 x 2), it was assumed the rods were heated uniformly axially. The maximum predicted temperature from this model was well over any safety limit, but this calculation produced a mass flow rate of air of 2.272×10^{-3} kg/s for 10 rods. This number would be used in subsequent analysis.

After constructing this model, it was then suggested to look at the mass flux of water vapor being provided by the portion of the rods submerged in water. This model, called the air water vapor model, is identical to the complete LOCA model with a few exceptions. First, it was assumed there were no inlet frictional losses since the mass flow rate of steam is not interacting with rods at the top of the core. Second, the mass flux of steam was determined by the following equation:

$$\dot{m}_{steam} = \frac{Q_{sub}}{h_{fg}}$$

Where:

Q_{sub} = total power produced by the lower portion of the rods
h_{fg} = heat of vaporization of water at 54.44°C

Then, the total mass flow rate was determined to be the sum of the mass flow rate of air calculated in the previous calculation averaged over the core plus the mass flow rate of steam. The humidity of the air was then determined assuming the initial air had a relative humidity of 50% at 298 K. Then the humidity ratio of the air was determined via the following equations[2]:

$$\varphi = \frac{P_v}{P_g} = \frac{y_v}{y_{v,sat}}$$

$$P_g = y_{v,sat} \cdot P_\infty$$

$$\omega_{intical} = 0.622 \frac{P_v}{P_\infty - P_v} = \frac{\dot{m}_v}{\dot{m}_a}$$

$$\omega = \frac{\dot{m}_v + \dot{m}_{steam}}{\dot{m}_a}$$

Where:

φ = relative humidity (50%)
P_v = vapor pressure
P_g = saturated vapor pressure at the temperature of the air
y_v = mole fraction of water
$y_{v,sat}$ = saturated mole fraction of water
P_∞ = 101.3 kPa
ω = humidity ratio
m_v = initial mass flow rate of vapor in the air
m_a = initial core average mass flow rate of air
m_{steam} = calculated core average mass flow rate of steam

The remaining difference is that the fluid properties are based off of an air-water mixture and not air-only as calculated in the total LOCA model.

While the air-water vapor model did reduce the maximum temperature calculated, the core average temperature was close to 900°C. It was then decided to create a third model, called the axial conduction model, based off of axial conduction as seen in Figure RAI-55-1 with a uniform heat generation along the entire length of the fuel rod, where L = 0.127 m (5 in).

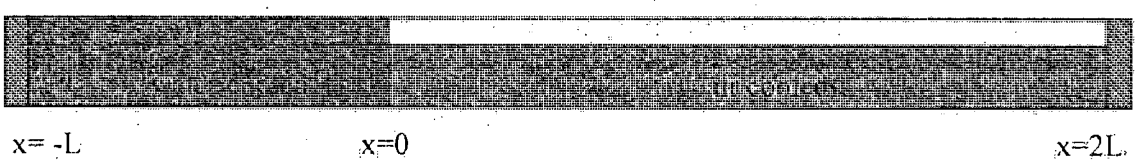

Figure RAI-55-1, Axial conduction model used for partial LOCA

The governing equations for this model are derived from El Wakil's nuclear heat transport for a fin.[3] The general equation for the water portion is:

$$\frac{d^2T}{dx^2} - \frac{h_w P}{kA}(T - T_w) = \frac{-q'''}{k}$$

$$\theta_w = T - T_w$$

$$m_w^2 = \frac{h_w Perimeter}{kArea}$$

$$\frac{d^2\theta_w}{dx^2} - m_w^2\theta_w = \frac{-q'''}{k}$$

The particular solution to the second order differential equation is:

$$\theta_{w,p} = \frac{-q'''}{m_w^2 k}$$

Thus the solution is, where A and B are constants to be solved:

$$\theta_w = Ae^{(-m_w x)} + Be^{(m_w x)} + \theta_{w,p}$$

The corresponding solution for the air cooled section of the rod have had all 'w' subscripts replaced with 'a'. Additionally, the constants C and D are to be solved as well:

$$\theta_a = Ce^{(-m_a x)} + De^{(m_a x)} + \theta_{a,p}$$

In order to solve the four unknown constants, the following four boundary conditions were used:

$$T(x=0) - T_w = T(x=0) - T_a$$

$$-kA\frac{dT_w}{dx}\bigg]_{x=0} = -kA\frac{dT_a}{dx}\bigg]_{x=0}$$

$$-kA\frac{dT_w}{dx}\bigg]_{x=-L} = -h_w A(T - T_w]_{x=-L})$$

$$-kA\frac{dT_a}{dx}\bigg]_{x=2L} = h_a A(T - T_a]_{x=2L})$$

Thus four equations and four boundary conditions can be used to solve the four unknowns:

$$A + B + T_w + \theta_{w,p} = C + D + T_a + \theta_{a,p}$$

$$-m_w A + m_w B = -m_a C + m_a D$$

$$-k\left(-m_w A e^{(Lm_w)} + m_w B e^\wedge(-Lm_w)\right) = -h_w \theta_w]_{x=-L}$$

$$-k\left(-m_a C e^{(-2Lm_a)} + m_a D e^\wedge(2Lm_w)\right) = h_a \theta_a]_{x=2L}$$

The constants are defined as:

k =	18 W/m-K
L =	0.127 m
Radius =	0.0179197 m
Perimeter =	2π·Radius
Area =	π·Radius^2

The remaining terms, T_w, T_a, h_w, and h_a are the bulk fluid temperature and heat transfer coefficient in air or water. The properties of the air are determined via an interpolating scheme with the previous model. In order to model conduction, the slope of the air temperature line at the air-water inter phase (x=0) is used to determine the heat conducted to the water by the following equation:

$$q_{conduction} = -kA\frac{dT_a}{dx}\bigg]_{x=0} = -kA(m_a D - m_a C)$$

This conduction heat is then added to the air-water vapor model as additional heat generation in the water-submerged portion of the rods. The heat lost in the air portion of the rods is subtracted out of the rod power emitted to the air. The air-water vapor model determines the fluid properties of the air while the axial conduction model determines the maximum fuel temperature of the rod and the heat conducted down the rod. Then an interpolation can be performed between the two models to determine the correct fluid properties, heat conduction, and maximum fuel temperature. The only parameters that have not been explicitly incorporated into either model are the temperature of the water and the heat transfer coefficient of the water. By doing a sensitivity study, a tolerance band can be created to see how the maximum temperature is dependent upon these parameters.

With the partial LOCA models created, it is then possible to see which accident is more limiting, the complete LOCA or the partial LOCA, by comparing the maximum fuel temperature of the core average position.

Section 6 from reference 33 is excerpted below, which details the LOCA analysis.

6.0 Loss of Coolant Accident

After analyzing a pulse at full power, the other accident considered is a loss of coolant accident (LOCA). A LOCA occurs due to a sheared and open beam port, a very unlikely event. The time in which it took the fuel to be uncovered from the start of the LOCA transient was determined to be by Kevin Austin in the HEU to LEU conversion analysis. Since the beam ports are in the mid-plane of the core, water will only flow out of the core to the bottom of the beam port leaving 11.43 cm (4.5 in) of active fuel still covered by water.

GA had previously calculated a LOCA using complete air cooling and did not present any analysis for reactors, such as the UWNR, to have a case in which water is still cooling the bottom third of the active fuel. The reason to be concerned about having the bottom third with water is that a natural circulation loop of air is much harder to form when the water would act as an insulating layer against air trying to go down into the core and up along the fuel rods to cool the fuel rods. However, the presence of water will allow for axial conduction from the fuel rod to the water and the generation of water vapor that would carry away the heat to portions of the fuel that are air cooled. Thus, the LOCA analysis presented here will look at two different cases. One in which there is a complete LOCA with the other being a partial LOCA where water is still in contact with the bottom third of the active fuel.

Another issue that has been brought to our attention by ANL is the fuel temperature limit. When the rods are water cooled, the Safety Analysis Limit (SAL) fuel temperature limit is 1150°C (2100°F). This number is a function of the gap size, hydrogen content in the fuel, and the cladding being at a much lower temperature than the centerline temperature in the fuel. However in air, the cladding temperature is at roughly the same temperature as the fuel centerline, and thus the fuel temperature SAL was determined to be 950°C (1740°F) in NUREG-1282. In addition to the analysis presented by TRIGA® International, the methodology for calculating the cladding strength as a function of temperature was performed in GA-9064,[4] and the fuel temperature SAL in air of 950°C (1740°F) was determined in the TAMU 1979 SAR.

In order to determine the fuel temperature during both LOCA calculations, it is necessary to make a few appropriate assumptions. It is assumed the reactor is operating at 1.02 MW (1 MW nominal power + 2% uncertainty) for 50 days of continuous operation. While previous accident analysis, such as the pulse at full power was performed at 1.3 MW, it is unreasonable to believe the reactor operators would operate the reactor beyond the scram set point for 50 days straight. The UWNR has never operated continuously since the UWNR is a research reactor and not a power reactor, and thus continuous operation at 1.02 MW is still a very conservative assumption.

In addition, the hot rod channel thermal hydraulic parameters were changed from previous analysis to be more physically representative of the actual hot rod channel. Previously, the hot rod channel assumed the channel had a limiting flow area due to the presence of the transient rod and still assumed the transient rod was producing power. In order to make the assumption more accurate, the flow area was changed from 4.7429 cm^2 (0.73516 in^2) to 5.0144 cm^2 (0.77723 in^2) and the hydraulic diameter was changed from 1.66318 cm (0.65479 in) to

1.78143 cm (0.70135 in). This analysis still assumed the four quarter rod segments were powered by a rod with the same pin power peaking factor of D5 SW. The assumptions made for all previous analysis are still valid so that the predicted maximum fuel temperatures and CHF values are bounding. For the LOCA, the change in assumption was made so that the actual conditions in the core would be modeled and then the accident scenario would occur.

In the event of massive water loss, the reactor would be shut down after receiving the pool high/low alarm. Kevin Austin has previously calculated it would take at least to uncover the fuel due to the LOCA. During the first the fuel would still be water cooled and the power of the core would be decreasing from the power of delayed neutron fission and decay heat. At and beyond, it is assumed that all water cooling is lost, and the fuel only has air cooling.

To determine the maximum fuel temperature during the LOCA transient, it is assumed that blades 1, 2, 3, and the transient rod SCRAM into the core 2 seconds after the pool level alarm is activated. The regulating blade is not inserted during a SCRAM and is assumed to not be manually inserted into the core. This causes a prompt drop in power, and the dominant source of power for the transient is delayed neutron fission power and decay heat. Using an 80 second delayed neutron period, the power from fission after the blades have dropped in can be calculated. At , the power in the core is determined predominately from the decay heat. Since the decay heat is determined from the previous steady state operation, the axial and radial power distributions are identical to those used in steady state analysis. In addition, the core power peaking factors are also identical to those used in the steady state analysis.

Since no benchmark can be performed with measured results for the LOCA, a hand calculation or analytic solution was created in order to determine the fuel temperature from first principals. By comparing the RELAP5 results to the analytic solution performed with the assistance of EES, confidence in the model can be gained.

6.1 Decay Heat during LOCA transient

To determine the overall power transient during the LOCA, the ORIGEN2 data used in the conversion report to determine the radiation levels in an unshielded core was also used to determine decay heat. Since the ORIGEN2 data was only constructed for the hot pin, the decay heat for the hot pin was multiplied by the number of rods in the core and divided by the hot rod power peaking factor at the respective time of core life.

In addition to the decay heat, there is additional fission power that needs to be added for the first part of the transient due to the influence of delayed neutrons. The prompt negative jump due to the effect of control blades falling in is:

$$P_{after\ blades\ drop} = P_o \frac{\beta}{(\beta - \rho_{shutdown})}$$

For HEU BOL these parameters are where $\rho_{shutdown}$ is computed by adding the shutdown margin and the worth of blade 3 together:

Po = 1.02 MW

β = 0.753%

$\rho_{shutdown}$ = -3.691%Δ k/K

$P_{after\ blades\ drop}$ = 172830.7831 W

Following the prompt jump, the fission power decreases as a function of time as follows:

$$P_{fission}(t) = P_{after\ blades\ drop} \cdot e^{\left(\Delta t_{shutdown}/_{\tau}\right)}$$

For HEU BOL the parameters are:

$P_{after\ blades\ drop}$ = 172830.7831 W

$\Delta t_{shutdown}$ is the time since SCRAM in seconds

τ = 80s

The summation of the fission power and the decay heat gives the final total core power curve entered into the 2-channel RELAP5/MOD3.3 model. The LOCA power curves did not incorporate the positive reactivity effects of the rods cooling down. While this would increase the fission power, it is not anticipated to effect the total power curves computed for the air cooled transient. The total power curve for HEU BOL is shown The entire total core power transient is shown in Figure RAI-55-2, and the air cooled portion of the transient is shown in Figure RAI-55-3. Further analysis will follow for LEU BOL, MOL, and EOL cases. The core exposure for LEU was 50 MWd for BOL, 800 MWd for MOL, and 1800 MWd for EOL. The input conditions for all stages of core life are shown in Table RAI-55-1.

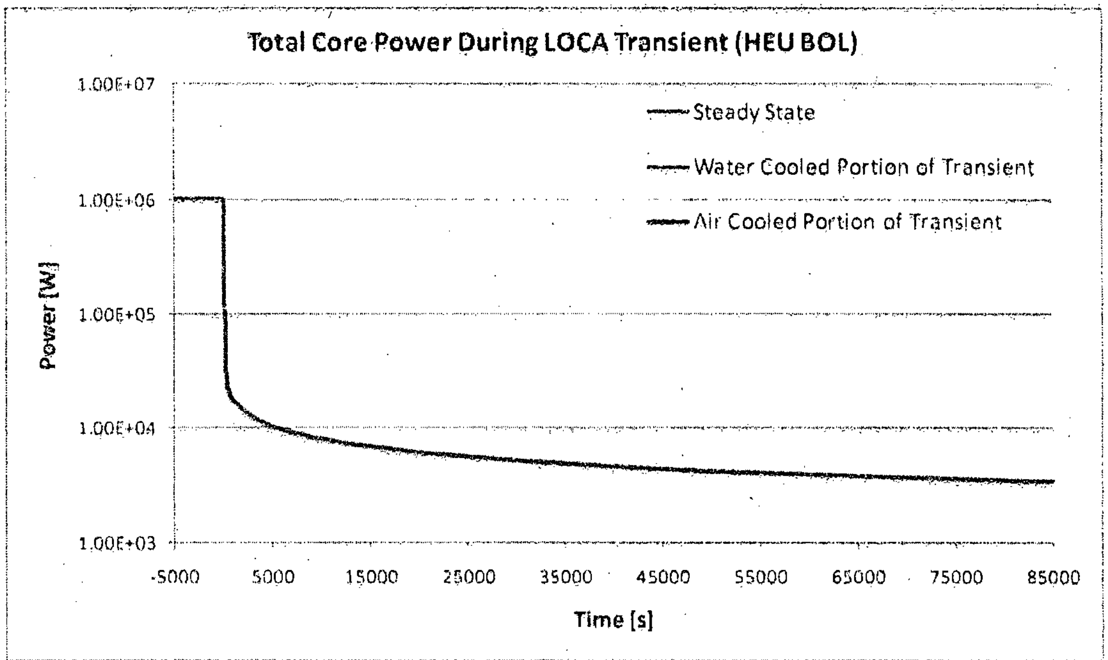

Figure RAI-55-2, Total core power used in LOCA starting at 1.02 MW (HEU BOL)

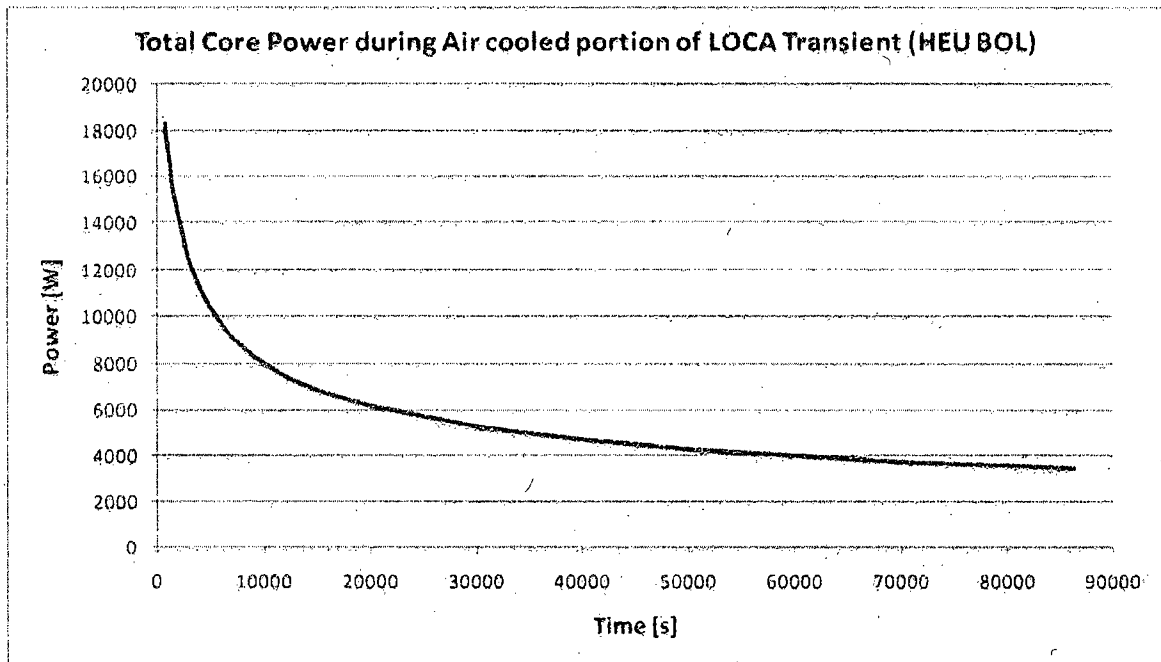

Figure RAI-55-3. Total core power used in LOCA during air transient (HEU BOL)

Table RAI-55-1, Input conditions to determine the power profile for the LOCA transient

Core Configuration	HEU BOL	LEU BOL	LEU MOL	LEU EOL
Steady State Power	1.02 MW	1.02 MW	1.02 MW	1.02 MW
Infinite Operation Time	50 days	50 days	784.3 days	1764.7 days
β	0.753 %	0.782%	0.774%	0.7389%
$\rho_{shutdown}$	-3.691%Δk/k	-3.593%Δk/k	-3.902%Δk/k	-5.114%Δk/k
$P_{after\ blades\ drop}$	172,830.78 W	182,317.71 W	168,853.09 W	128,769.58 W
τ	80 s	80 s	80 s	80 s
Power at the start of the air cooled transient, t=836s	18,369.86 W	18,883.59 W	19,833.55 W	19,719.38 W

Using the results shown in Table RAI-55-1, the power profile for the LEU core at BOL, MOL, and EOL can also be constructed. These power profiles are then put into the RELAP5 input decks with the same methodology employed for the HEU BOL case. The maximum fuel temperature can then be calculated depending upon whether a total LOCA or partial LOCA, occurs.

6.2 Complete water drainage from core during LOCA transient

After inputting this power profile into each component of the transient, the maximum temperature during the transient can be calculated. During the water cooled portion of the transient, the temperature falls from the steady state temperature of 483.00°C (901.40°F) to 73.46°C (164.23°F) after for the HEU BOL case. At this point in time the water level would reach the top of the fuel, and the remaining water is assumed to vacate the core.

In order to run the RELAP5 case with air cooling, the initial mass flow rate of air is set to nearly zero (0.0001 kg/s) to simulate the buildup of the natural circulation of air. With this condition, and the decay heat curve as shown by the red line in Figure RAI-55-2, RELAP5 would predict a maximum fuel temperature to jump by over 800 degrees centigrade in the first iteration. Due to this non-physical effect, it was necessary to alter the first few seconds of the power transient to allow RELAP5 the ability to converge on the correct mass flow rate and heat transfer coefficients. By starting at a low power level and increasing the power to the correct power level after 4 seconds, RELAP5 was able to converge on the mass flow rate and give realistic results. For HEU BOL, 18,369.86 W is the calculated decay heat power at the start of the air cooled transient. This power perturbation is not expected to significantly alter the maximum fuel temperature calculated as seen in Figure RAI-55-4.

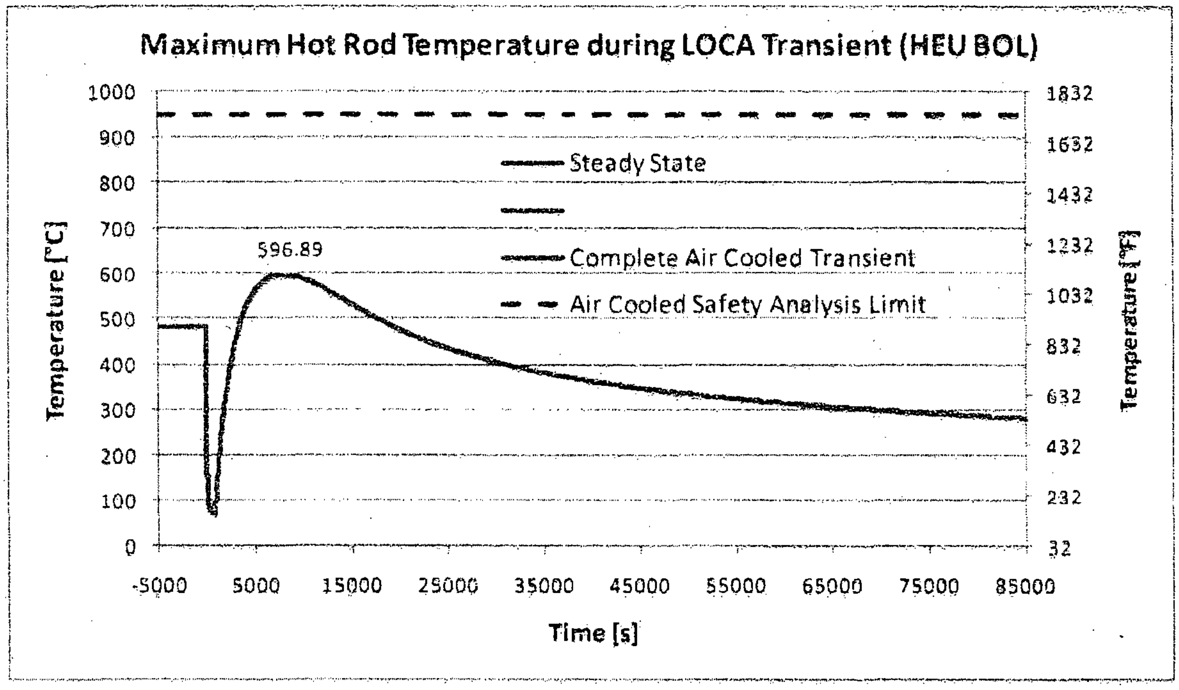

Figure RAI-55-4, Temperature profile during LOCA transient (HEU BOL)

The maximum temperature for HEU BOL during the air cooled portion of the transient was calculated to be 596.89°C (1106.40°F). This temperature occurs 7,750 seconds from the initiation of the accident. While clearly a LOCA is a significant accident, no damage to the fuel is predicted since the fuel temperature does not exceed the SAL.

While RELAP5 is predicting a particular temperature, it is unclear at first whether the predicted maximum temperature during a LOCA is accurate without having another calculation to compare it with. Thus, a hand calculation was performed with the assistance of EES. This analytic solution was constructed using first principals of balancing the buoyancy pressure gains due to the heated air with the frictional pressure drop over the entire core. By determining the mass flow rate, heat transfer coefficients, and air temperature, it was possible to determine the maximum fuel temperature in the core.

The analytic model was designed for the entire core, and thus it is necessary to compare with the 2nd channel of the 2-channel model. In addition, since the hand calculation model assumes that the power level is at steady state, the first portion of the air cooled transient where the fuel rod is heating up is not modeled. Thus it was assumed that steady state results would start at around 5000 seconds when the total core power is about 10,000 W. The maximum core channel result comparison between the analytic solution and the RELAP5 2nd channel is shown in Figure RAI-55-5.

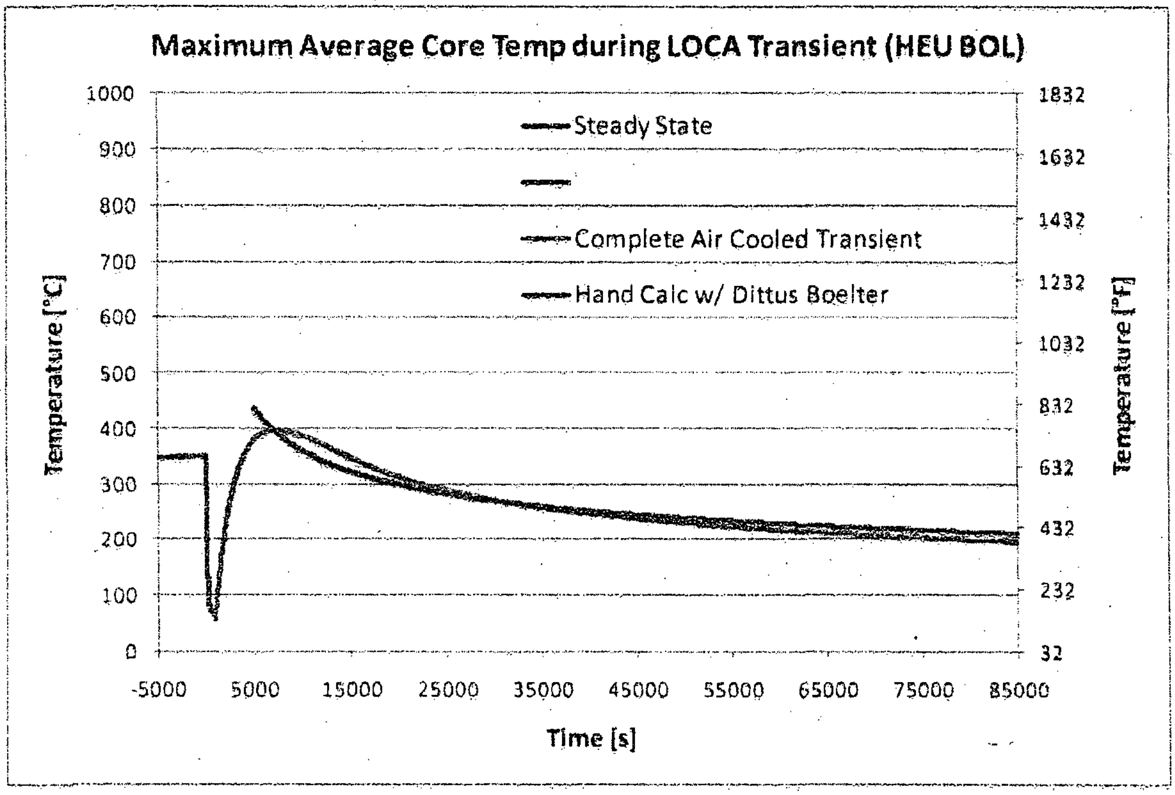

Figure RAI-55-5, Core Channel temperature comparison during LOCA (HEU BOL)

Figure RAI-55-5 shows that the analytic solution and the RELAP5 solution for the core averaged channel give very similar results from about 25,000 seconds onwards. The analytic and RELAP5 solutions diverge during the peak portion of the transient since this is still in the transient region. The maximum difference between these two lines is approximately 45°C around 12,500 seconds. In order to compare with the hot rod temperature, the total core power was artificially increased by the hot rod power peaking factor, 1.6. This produced a maximum temperature profile as shown in Figure RAI-55-6.

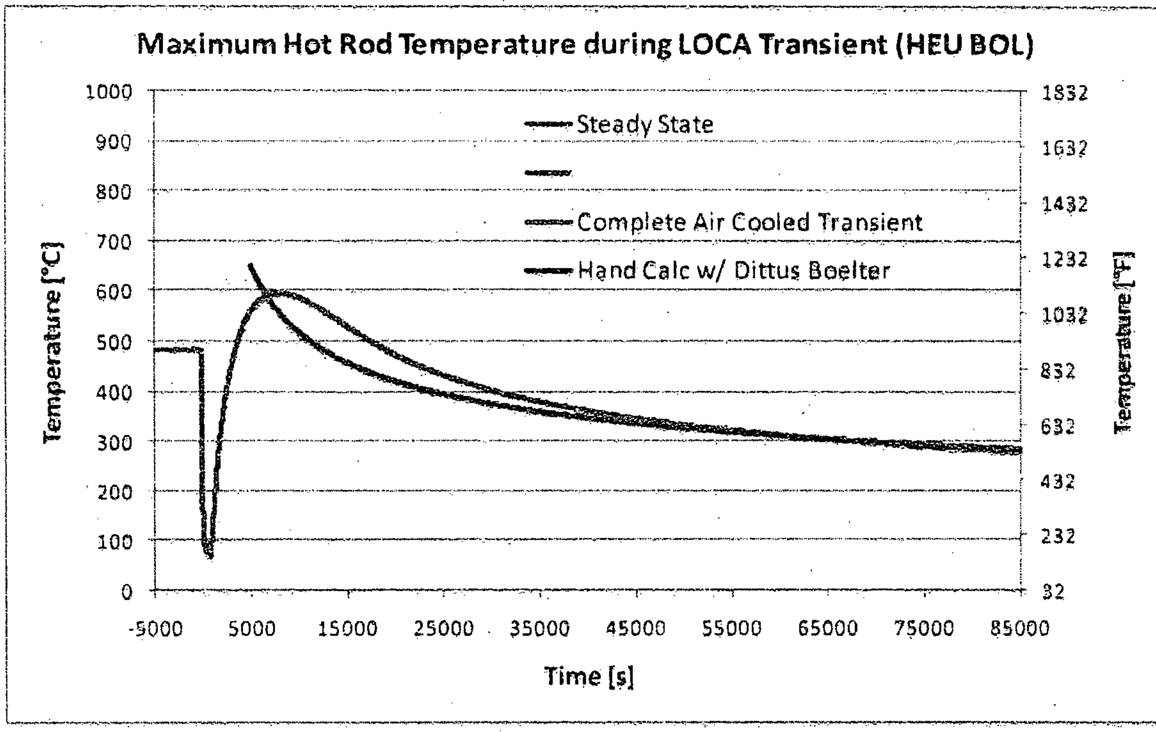

Figure RAI-55-6, Hot channel temperature comparison during LOCA (HEU BOL)

While the core channel analytic solution compared very well with the RELAP5 results, the hot channel analytic solution does not compare as well with the RELAP5 results as seen in Figure RAI-55-6. The maximum difference between these hot channel results is approximately 90°C at 12,000 seconds, or about double the difference of the core channel results. To look at why the results are different, the mass flow rate, heat transfer coefficient, exit air temperature, and exit air velocity for the analytic solution and the RELAP5 results are shown for the core channel in Figures RAI-55-7 through RAI-55-10.

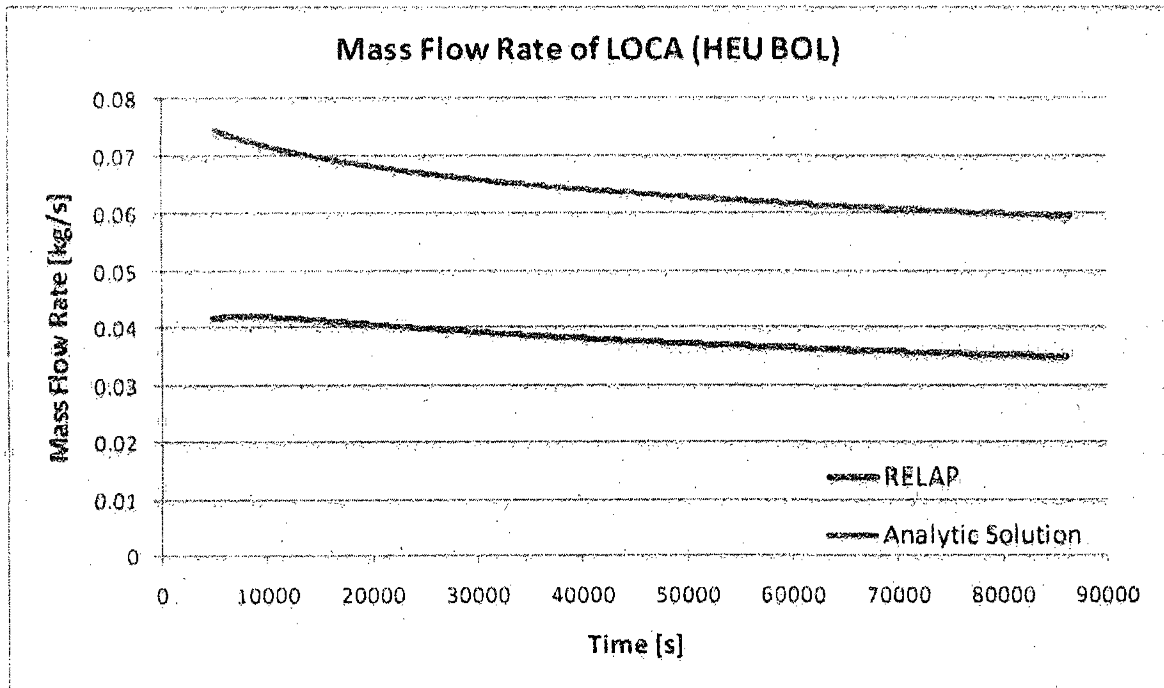

Figure RAI-55-7, Mass flow rate comparison during LOCA (HEU BOL)

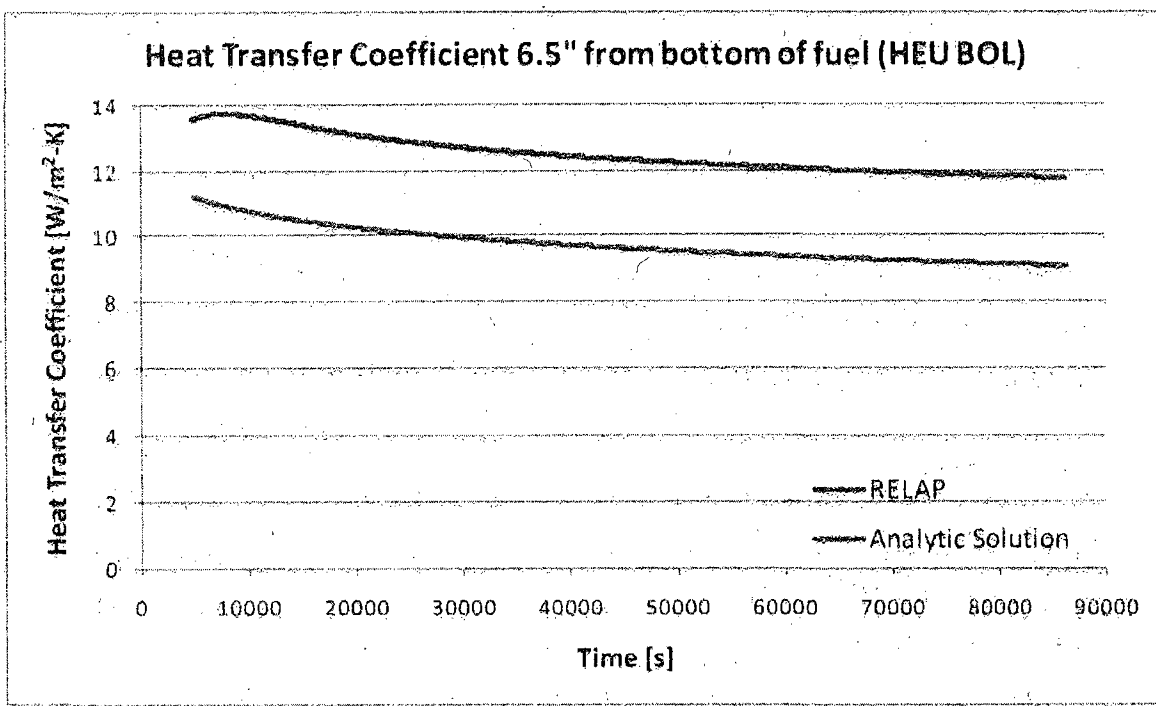

Figure RAI-55-8, Heat transfer coefficient comparison during LOCA (HEU BOL)

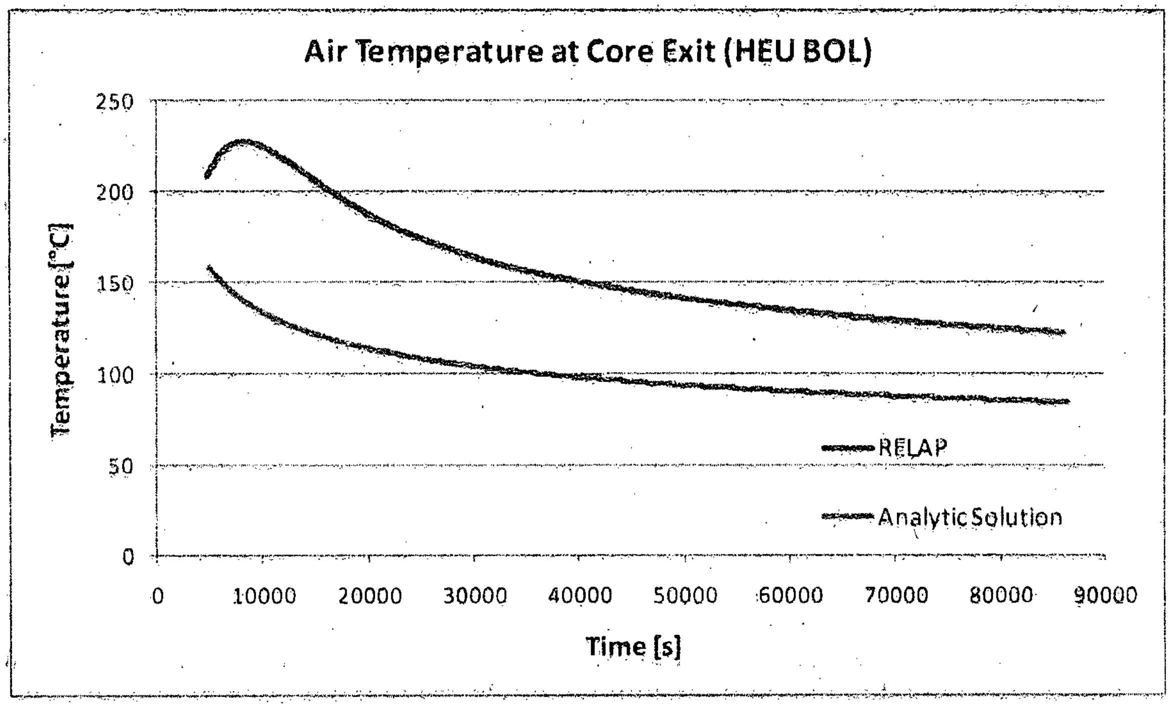

Figure RAI-55-9, Air temperature at core exit during LOCA (HEU BOL)

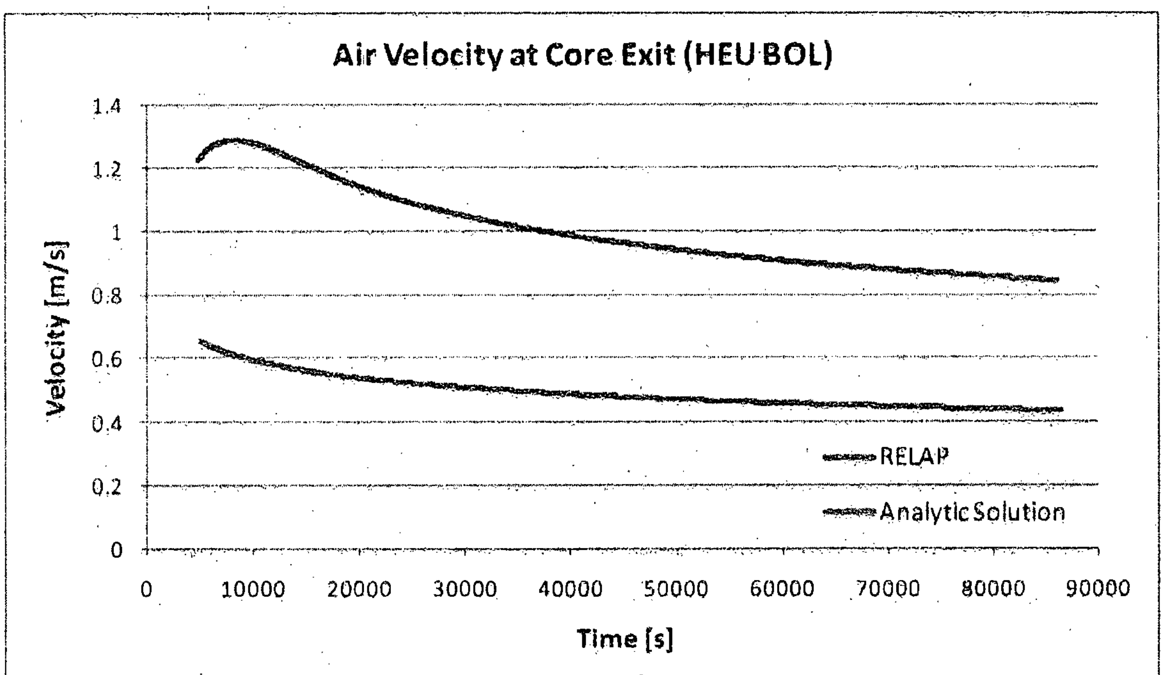

Figure RAI-55-10, Air velocity comparison at core exit during LOCA (HEU BOL)

What is apparent from these comparison figures is that while the temperature profile of the hot channel matches up very well between the two models, they do not compare nearly as well when looking at the fundamental constituents. There are significant differences between each of the constituent terms, especially the mass flow rate and the air velocity at the core exit. The mass flow rate RELAP5 calculates is nearly half the mass flow rate of the analytic solution, and the air temperature RELAP5 predicts at the top of the core is about 100°C higher than the analytic solution. Interestingly, RELAP5 is calculating a heat transfer coefficient higher than the analytic solution, making up for the lower mass flow rate and higher air temperature. In addition, a comparison between the axial temperature profile of the analytic solution and RELAP5 is shown in Figure RAI-55-11.

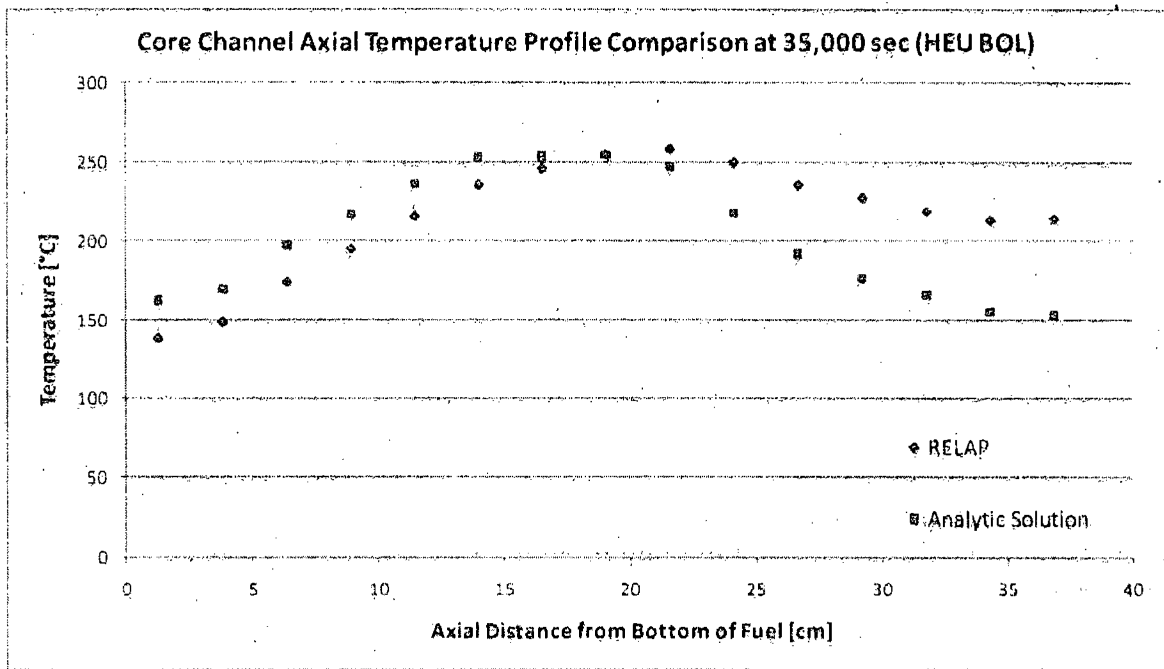

Figure RAI-55-11, Core channel axial temperature comparison at 35,000 seconds (HEU BOL)

While the maximum temperature is approximately the same, the axial location shifts between RELAP5 and the analytic solution by 2 nodes (2 inches). This difference is probably coming from the difference in the air temperature in the two models and the latent heat of the fuel rod accounted for in the RELAP5 transient model and not modeled in the analytic solution. Notwithstanding these differences, the full LOCA RELAP5 model has been compared with an analytic model, and the RELAP5 model gives reasonable results. Therefore, only the RELAP5 results will be presented. As shown in Figure RAI-55-4, the maximum temperature calculated by RELAP5 for the HEU BOL during the LOCA transient is 596.89°C (1106.40°F). Using the same methodology used for the HEU core, the LEU core LOCA results can be calculated. The most limiting stage of core life for the LOCA transient is LEU MOL. The maximum hot rod temperature during the LOCA transient is shown in Figure RAI-55-12.

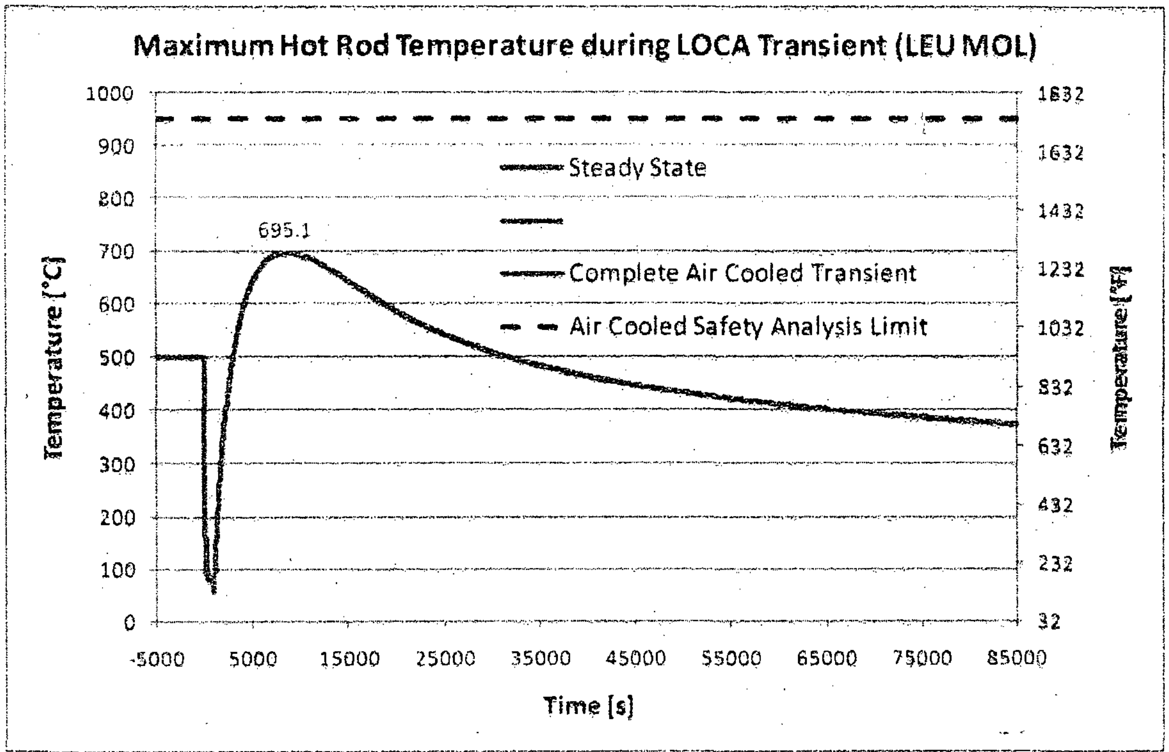

Figure RAI-55-12, Temperature profile during LOCA transient (LEU MOL)

The maximum temperature predicted by RELAP5 is 695.10°C (1283.18°F). This temperature is below the SAL by 254.9°C. Many of the very conservative and limiting assumptions could be dropped to recover even more margin. For example, it could be assumed that it takes   to drain the pool as opposed to   as currently calculated if the water level was at the true operating level. In addition, a more realistic operational schedule could be used to determine the total power level. Also, if such an accident were to occur, the operators also have the option of using the emergency city water pump to slow the rate of water draining from the pool. Furthermore, the most likely scenario for a beam port rupturing is if a large heavy object were dropped on the beam port from the pool top. Handling of such objects is not anticipated until several hours after reactor shutdown.

Table RAI-55-2, Summary of LOCA temperature results

Core Configuration	HEU BOL	LEU BOL	LEU MOL	LEU EOL
Starting temp during steady state operation	483.00°C (901.40°F)	498.75°C (929.75°F)	498.75°C (929.75°F)	482.71°C (900.88°F)
Temperature at end of water cooled transient	73.46°C (164.23°F)	74.87°C (166.77°F)	75.73°C (168.31°F)	74.81°C (166.66°F)
Change in temperature from steady state to	409.54°C (737.17°F)	423.88°C (762.88°F)	423.02°C (761.44°F)	407.90°C (734.22°F)
Core Power at	18,369.86 W	18,883.59 W	19,833.55 W	19,719.38 W
Maximum temperature in hot rod during LOCA	**596.89°C** (1106.40°F)	**648.37 °C** (1199.07 °F)	**695.10°C** (1283.18°F)	**679.15C** (1254.57°F)
Time of maximum temperature in hot rod	7,750 sec	7,775 sec	8,450 sec	9,300 sec

In summary, all of the LOCA temperature results are shown in Table RAI-55-2. As can be seen, at no time is the predicted temperature greater than fuel temperature safety analysis limit in air.

6.3 Partial water drainage from core during LOCA transient

In conjunction with the complete LOCA, an analysis was conducted on the partial LOCA. As stated in section 2.2.2, three different models were created in order to analyze the partial LOCA. The mass flow rate of air in the air-only model was computed to be 2.272×10^{-3} kg/s for 10 rods or 0.0206752 kg/s for the entire 91 elements in the HEU core. Then, using this mass flow rate, the fluid properties of the air were calculated in the air-water vapor model. Water vapor was generated due to the heat of the rods submerged in water and axial conduction calculated in the axial conduction model. By interpolating between the air water vapor model and the axial conduction model, the maximum fuel temperature can be calculated using the axial conduction model.

Since the fluid properties of the water were never modeled explicitly, a sensitivity study was done when performing the analysis. The first assumption was to assume the water temperature was 100°C, and the heat transfer coefficient of the water was 1,000 W/m^2-K. This produced the axial heat generation curve at 5,000 seconds seen in Figure RAI-55-13. This produced a maximum fuel temperature in the average core position to be 417.9°C (784.22°F). Further analysis on having the water temperature be either 60°C or 100°C and the water heat transfer coefficient being either 500 W/m^2-K or 1,000 W/m^2-K is shown in Table RAI-55-3. The temperature of the air was taken to be the highest air temperature calculated in the air-water vapor model.

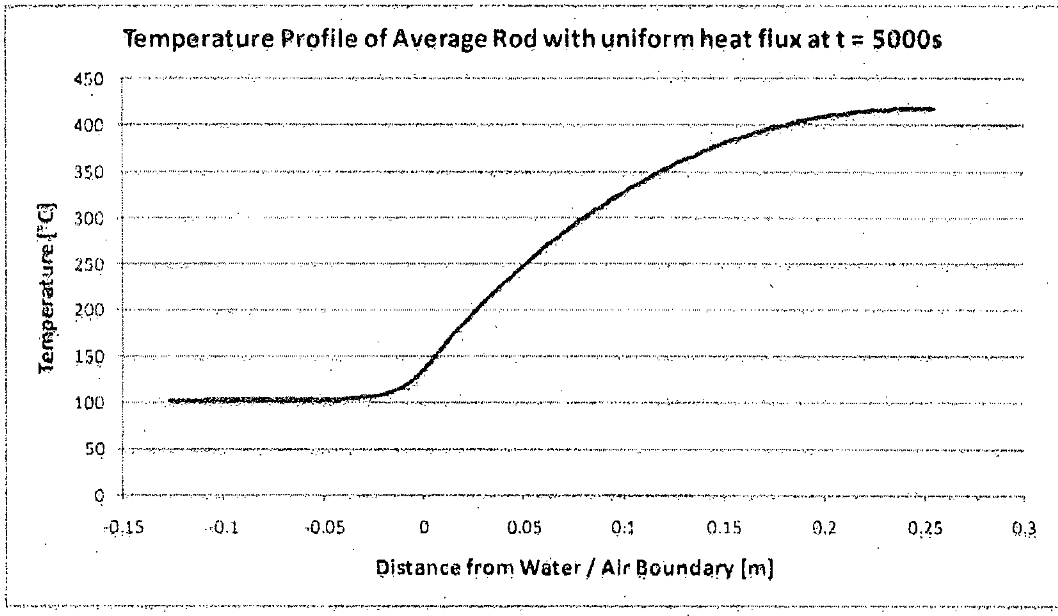

Figure RAI-55-13, Axial temperature profile during partial LOCA (HEU BOL)

Table RAI-55-3, Summary of partial LOCA results without weighted average h_a

$q_{conduction}$ [W]	T_a [°C]	T_w [°C]	h_a [W/m^2-K]	h_w [W/m^2-K]	Max Temp [°C]
4,585	166.6	60	5.105	500	406.3
4,698	163.9	60	5.109	1000	397.6
4,324	172.7	100	5.096	500	426.1
4,431	170.2	100	5.100	1000	417.9

When determining the heat transfer coefficient of air with the air-water vapor model, the average was taken for every node except the node directly above the water. This node had a very large heat transfer coefficient and thus was thrown out for conservatism. However, if this node was added, and a weighted average on the heat transfer coefficients was performed, the following results would result as seen in Table RAI-55-4.

Table RAI-55-4, Summary of partial LOCA results with weighted average h_a

$q_{conduction}$ [W]	T_a [°C]	T_w [°C]	h_a [W/m^2-K]	h_w [W/m^2-K]	Max Temp [°C]
4,181	176.15	60	8.592	500	351.7
4,317	172.9	60	8.575	1,000	345.4
3,846	184.3	100	8.631	500	367.5
3,972	181.15	100	8.617	1,000	361.5

As can be seen in these tables, incorporating the first axial node above the water line makes a large difference in the heat transfer coefficient and also the maximum fuel temperature calculated. For conservatism, no benefit will be assumed for the first axial node. Further refinement of the model could be done to look at the effect of how the power shape would impact the analysis. However, this would either take incorporating a power shape in the volumetric heat generation term, complicating the solution, or solving an equation for each axial node.

Additionally, it is important to determine whether the partial or complete LOCA is more limiting. The maximum fuel temperature of the average rod in the complete LOCA has been calculated to be 431.9°C (809.42°F) at 5,000 seconds. Even with the highest water temperature and the lowest heat transfer coefficient, the fuel temperature in the partial LOCA never exceeds this temperature. Thus, it is concluded the complete LOCA is more limiting than the partial LOCA.

In addition to the analysis just at 5,000 seconds, the partial LOCA analysis can be performed at other time intervals to create a transient curve. This required iterating at each time step, so fewer time steps were taken than the 1,000 points created automatically for complete LOCA analysis. The analysis for the partial LOCA used a water temperature of 100°C, and the heat transfer coefficient of the water was 1,000 W/m^2-K throughout the iteration process. The two transient solutions are presented in Figure RAI-55-14.

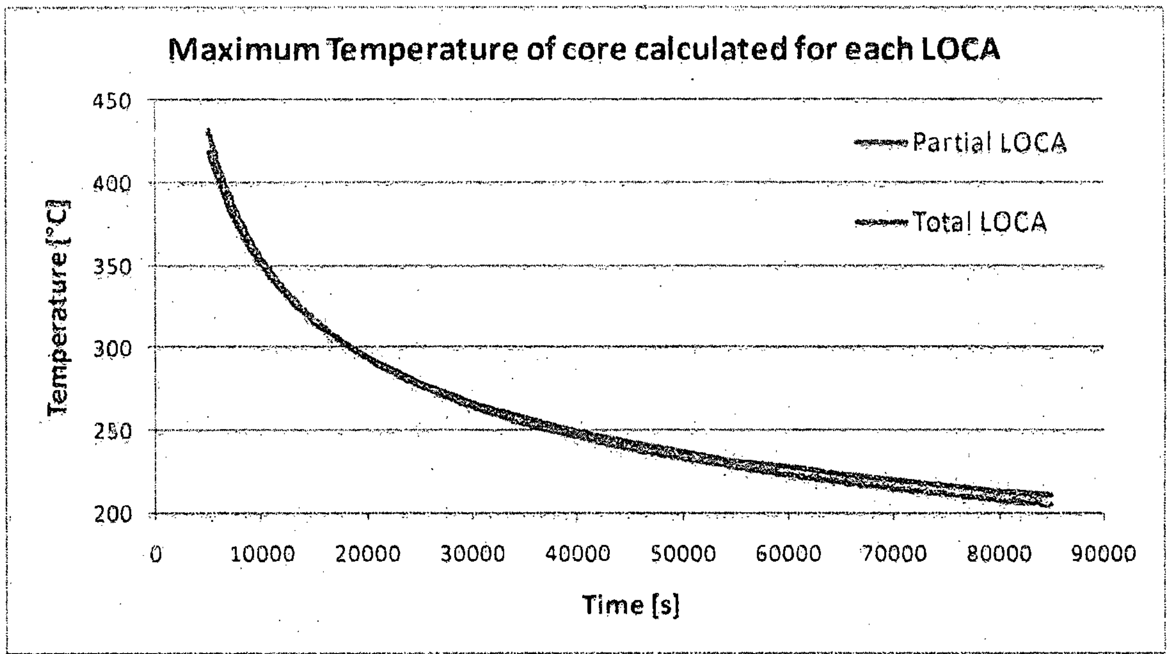

Figure RAI-55-14, Temperature comparison between total and partial LOCA (HEU BOL)

This plot shows that the maximum temperature of the core average rod during the partial LOCA is lower than the total LOCA until their intersection point around 17,500 seconds. After this point the partial LOCA has a higher calculated temperature. By that time step, the peak fuel temperature has already been reached as seen in the previous total LOCA analysis. Therefore, the total LOCA is still expected to be more limiting that the partial LOCA.

References:

1. Klein, S.A, "Engineering Equation Solver Version 7.968", Madison, WI, September 2007.
2. Moran, Michael J and Shapiro, Howard N., *Fundamentals of Engineering Thermodynamics: 5^{th} Edition.* John Wiley and Sons: 2004.
3. El-Wakil, M. M. *Nuclear Heat Transport.* The American Nuclear Society: La Grange Park, IL, 1993.
4. GA-9064, "Safety Analysis Report for the Torrey Pines TRIGA Mark II Rector." General Atomics, Inc., January, 5, 1970.

In addition to the preceding analysis taken from reference 33 of the analysis report, an independent analysis was conducted by ANL which confirms the conclusion that the complete LOCA is more limiting than the partial LOCA. Applicable excerpts from reference 34 of the analysis report are included below.

5.2 Partial LOCA

The centerlines of the four UWNR beam tubes are aligned with the core mid-plane, which is located at 7.5 inches above the bottom of the fuel. Since the beam tubes are 6 inches in diameter, the lowest initial water level for the partial LOCA analysis is 4.5 inches above the bottom of the fuel. For a fuel rod power of 19.7 kW and a drain time of , the analysis in Appendix B predicted a peak fuel temperature of 578°C for the partial LOCA, compared to 585°C for the complete LOCA. Since this temperature is 372° C below a maximum fuel temperature in air of 950°C, a LOCA initiated by a failure of one of the UWNR beam ports will not result in failure of the hottest fuel rod.

B.2 COMPUTATIONAL FLUID DYNAMICS (CFD) MODELING

A computational fluid dynamics (CFD) model was developed with the STAR-CD code[1] and used to do the computations. The geometry was kept simple so that the problem would run quickly and be easy to understand while enabling the concept to be tested and easily understood. An axisymmetric wedge of a single rod was analyzed. The coolant channel geometry was assumed to be annular rather than the shape of a cusp between four adjacent rods on a square pitch. The channel flow area associated with a single rod in the UWNR was preserved. The assumed axial power shape in the rod is shown in Figure B1. The decay heat curves for infinite operation and 120 hours per week of operation are shown in Figure B2 and are based on Reference 2. The one for infinite operation is based on 10^{13} seconds of continuous operation and includes the $G_{max}(t)$ factor, as given in Table 13 of the Reference 2, to account for neutron capture in the fission products. The curve for 120 hours of operation does not include this factor and is based on 40 years of operation.

Figure B3 shows the CFD model geometry, including the mesh used in the CFD analysis. The computational volume is a 3° axisymmetric wedge of a cylindrical region. In the figure, the wedge is viewed from a skewed angle that makes the fuel rod appear very short and very large in diameter. The total fuel rod length is 30 inches. Some of the key dimensions are provided in Table B1 for the UWNR. The zirconium rod is 0.25 inches in diameter. The fuel and the upper and lower reflector outer diameters all are assumed to be 8.80 mils (0.0088 inches) less than 1.371 inches. The clad thickness is 0.020 inches and the fuel rod outer diameter is 1.411 inches. Since it was not practical to represent the very small typical gap thickness between the fuel and the clad of 0.1 mils in the CFD mesh, a larger radial gap of 0.01117 cm (4.40 mils) was used. In order to compensate for the thicker gap in the model, the gas thermal conductivity was increased to 0.699 W/m-K. This resulted in a gap conductance of 6260 W/m^2-K, which is a representative value for a 0.1-mil gap in a TRIGA fuel rod.

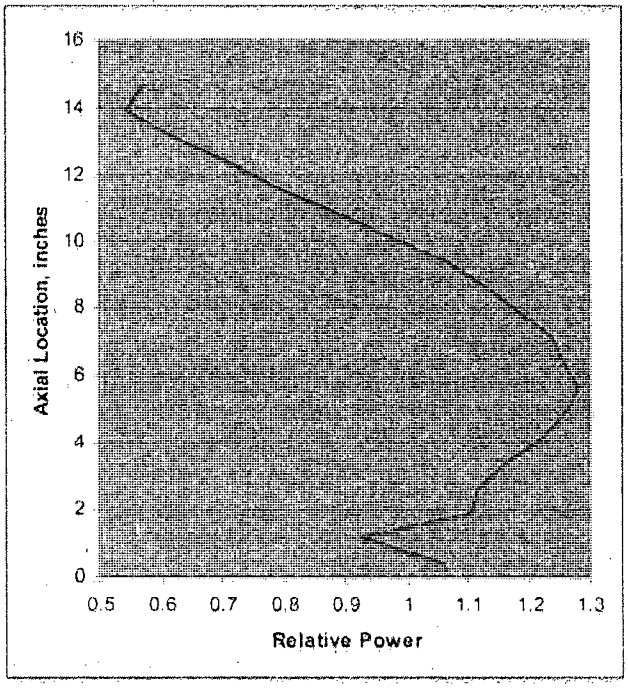

Figure B1. Fuel Rod Axial Power Shape

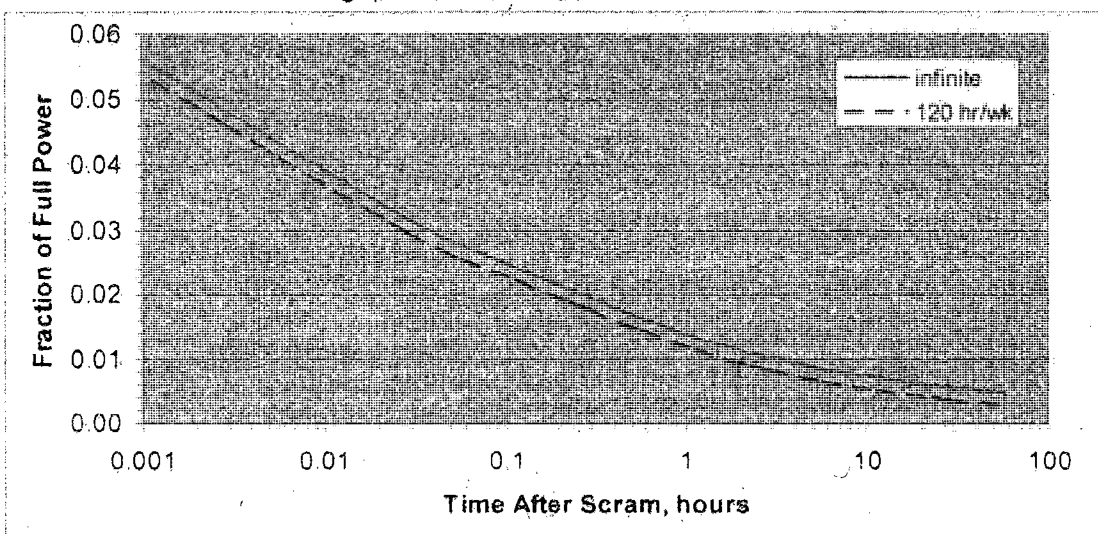

Figure B2. Decay Heat Power Levels for Infinite Operation and 120 hr/wk Operation

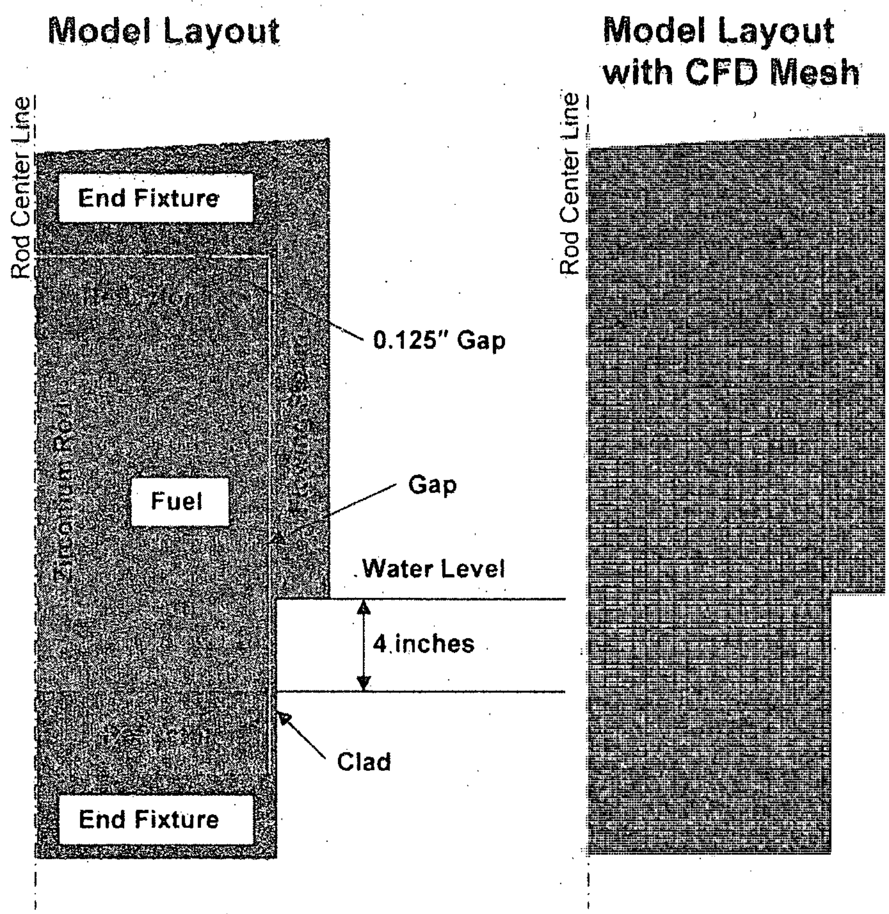

Figure B3. CFD Model Geometry
(3° wedge viewed from a skewed angle)

The fuel rod upper and lower end fittings are represented as solid stainless steel cylinders. The turquoise-colored region in Figure B3 represents the flow channel for the steam. It starts at the water level, which in Figure B3 is 4 inches above the bottom of the fuel, and extends along the exposed lateral fuel rod surface to the top of the rod. This dimension was adjusted as needed, but was fixed within any given transient solution. Hence, several CFD mesh models similar to the one shown in Figure B3 were used so that the appropriate water level was used in each analysis. The gas in the 1/8th inch gap above the upper reflector was assumed to be a mixture of 86% xenon and 14% krypton by mole fraction. This gas mixture determines its thermal conductivity. The thermal conductivity of this gap is expected to have very little influence on the peak temperatures predicted. Therefore, a very limiting assumption was used. The top surface of the rod portion of the model is assumed to be insulated. The steam enters the channel at 100°C. The outer lateral cylindrical boundary of the steam is modeled as a "symmetric boundary" in that it is thermally insulated and provides no viscous shear forces to the flowing steam. The channel flow area and channel hydraulic diameter of the UWNR were preserved in this axisymmetric representation. The material properties used in the CFD analysis for the fuel, the stainless steel clad and end fittings, the zirconium rod, and the graphite reflectors are listed in Table B2.

Table B1. Model Parameters

Item	UWNR
Maximum Licensed Reactor Power, MW	1.0[1]
Peak Rod Power, kW	19.7[2]
Rod OD, in. (cm)	1.411 (3.58394)
Rod Arrangement in Limiting Channel	Square
Pitch, in.	1.630
Flow Area per rod, in^2	0.77723
Channel OD for CFD model, in. (cm)	1.7264 (4.3851)
Clad Thickness, in.	0.020
Radial Gap Conductance, W/m^2-K	6260.
Radial Gap Thickness in CFD Model, cm	0.01117
Fuel Pellet OD in CFD model, cm.	3.71396
Total Rod Length in CFD Model, in. (cm)	30.0 (76.2)
Length of Upper End Fitting, in. (cm)	4.387 (11.1430)
Thickness of Upper Gap, in. (cm)	0.125 (0.3175)
Length of Upper Reflector, in. (cm)	3.45 (8.763)
Length of Fuel, in. (cm)	15.0 (38.1)
Length of Lower Reflector, in. (cm)	3.45 (8.763)
Length of Lower End Fitting, in. (cm)	3.588 (9.1135)
Beam Tube ID Outside of Pool, in.	6.0
Beam Tube ID thru Wall, in.	6.0
Beam Tube Centerline below Core Centerline, in.	0.0
Initial Water Level (0=bottom of fuel), in.	4.5
Time after Scram to reach water level, min.	
Effective Pool Surface Area, ft^2	85

[1]The analysis uses a power level of 1.02 MW to account for a 2% uncertainty in power level measurement. [2]Peak rod power including 2% uncertainty (19.3 x 1.02 = 19.7 kW).

The exterior surface of the fuel rod that is immersed in the water is assumed to have a 110°C constant temperature boundary condition. The 110° C temperature is based on the assumption that the surface temperature is 10°C above the 100°C water saturation temperature. If the surface were adjacent to a flowing subcooled liquid, then the McAdams, Jens and Lottes, and Thom et al. correlations would indicate that 10° C above the water saturation temperature would be a reasonably conservative estimate of the rod surface temperature. Since it is the agitation of the liquid caused by nucleate boiling, rather than the flowing of the liquid, that keeps the temperature rise relatively small, the 110° C value is judged to be a reasonable upper bound, although the liquid is essentially stagnant.

Table B2. Material Properties of Solid Regions in CFD Solutions

Fuel Meat

$k = 18$ W/m-K
$\rho = 7150$ kg/m^3
$c_p = (132.67 + 0.565\ T)$ J/kg-K, where T = [k]

Clad and End Fittings (stainless steel)

$k = (9.038 + 2.182 \times 10^{-2}\ T - 8.048 \times 10^{-6}\ T^2 + 2.491 \times 10^{-9}\ T^3)$ W/m-k, where T= [k]
$\rho = 8000$ kg/m^3
$c_p = (308.3 + 0.7890\ T - 8.245 \times 10^{-4}\ T^2 + 3.345 \times 10^{-7}\ T^3)$ J/kg-k, where T = [k]

Zirconium Rod

$k = 20$ W/m-K
$\rho = 6500$ kg/m^3
$c_p = 300$ J/kg-K

Reflector (graphite)

$k = 40$ W/m-K
ρ 2000 kg/m^3
$c_p = 700$ J/kg-K

The pool water temperature is assumed to be 25°C, typical of the value that would exist during normal operation. Water at this temperature can be thought of as being supplied to a coolant channel that extends the length of the fuel rod and is filled with liquid below the water level and vapor above. Seventy-five calories are required to raise 1 gram of 25°C water 75°C to the 100°C boiling point. An additional 539 calories per gram (2257 kJ/kg) are required to convert the saturated water to saturated steam. Thus, 539 + 75, or 614, calories are required to convert 1 gram of 25°C water to saturated steam. An equivalent perspective is that 614 calories/gram (2571 kJ/kg) is needed to increase the specific enthalpy of 25°C liquid water to that of saturated steam. In the model of a single rod and its associate coolant channel, this heat is supplied by the submerged end of the fuel rod.

At each instance in time the heat flux integrated over the 110°C surface of the submerged end of the rod provides the power that converts liquid to steam. It is assumed that for every 614 calories of energy provided, 1 gram of saturated steam is sent up the rod coolant channel. In the model the coolant channel is assumed to originate at the surface of the water. Ideally, the STAR-CD code would calculate the power being delivered to the water at each time step and would determine the steam flow rate to be used for the succeeding time step. However, it appears that the required heat fluxes are available only during post processing after all of the time steps have been solved. Therefore, an iterative approach was used. A guess was made of the steam production as a function of time. Then the results were post-processed to deduce the required steam production function. The deduced function was assumed and the CFD solution was repeated. After about two or three iterations, a converged solution was obtained. The analysis considered only steam flow and includes no air recirculation. The analysis assumes that the top of the tank is completely open so that the superheat steam can freely flow into the large room where the reactor is located.

The initial condition for the rod is that the temperature is 100°C everywhere except on the submerged boundary, which is 110°C. This is slightly conservative because when the fuel rod is completely submerged all of the temperatures will be essentially that of the of the pool water, which is assumed to be 25°C. Perhaps, a less pessimistic approach would be to assume the pool water temperature for the initial condition. However, this may be too optimistic since the fuel heats up as the level drops from the top of the fuel to the bottom of the beam port. Also, 30°C is a more typical pool water temperature. The 5°C higher pool temperature will increase the steam production by about a factor of 614/609, since each degree corresponds to 1 calorie per gram. This represents only a 0.8% increase in steam production and will have only a very small effect on the peak fuel temperature.

The fuel meat heats up very slowly due to its relatively low (decay heat) power and its very large heat capacity (2.46×10^6 J/m^3-K at 100°C). For the volume of the fuel meat in one rod (3.51×10^{-4} m^3), the heat capacity of one rod (evaluated at 100°C) is 863 J/K. Some of the transients are assumed to start at after the scram. Integration of the (infinite) decay heat curve from , i.e., the first of the transient, indicates that 9.36 full power seconds of energy are generated in the fuel. For a 20 kW rod, this corresponds to 20,000 W × 9.36 s, or 1.87×10^5 J. This energy divided by the fuel meat heat capacity, 863 J/K, yields a temperature rise of 217°C. This implies that if all of the power generated in fuel meat during the first of the transient stayed in the fuel meat rather than be transferred away, the fuel meat temperature on average would rise only 217°C, or less than an average of a half of a degree per second.

B.3 CFD RESULTS AND DISCUSSION OF CFD RESULTS

B.3.1 University of Wisconsin

The maximum licensed power for the UWNR is 1.0 MW. The uncertainty in the measured power level is 2%. It is reasonable to assume that the reactor will not be operated for an extended period of time above 1.0 MW (1.02 MW, including the uncertainty). Thus, the shutdown decay heat in this analysis is based on operation at 1.02 MW. The calculated[3] peak rod power of 19.3 kW at 1.0 MW was increased by 2% to 19.7 kW for the UWNR partial LOCA analysis.

The centerlines of the beam tubes are aligned with the core mid-plane, which is located at 7.5 inches above the bottom of the fuel. Since the beam tubes are 6 inches in diameter, the lowest initial water level for the partial LOCA analysis is 4.5 inches above the bottom of the fuel. It is assumed that the water drains down to its initial level after the reactor is scrammed due to low water level.

Figure B4 shows the maximum temperature of each region within the 19.7 kW fuel rod as a function of time. The peak fuel temperature, 558°C, occurs at 4.07 hours after the start of the transient (4.32 hours after the scram).

The top (dark blue) curve in Figure B5 shows the total decay power history in watts for the 19.7 kW UWNR fuel rod. The solid pink curve indicates how may of those watts at each instance are coming out of the submerged portion of the rod. This was obtained by post processing the CFD results to find the integral of the heat flux over the submerged portion of the rod. Similarly, the yellow curve represents the watts coming out of the exposed portion of the rod and superheating the steam that is flowing in the channel. The turquoise curve was obtained by subtracting the total the power transferred from the rod, which is the sum of the pink and yellow curves, from the power generated, which is the dark blue curve. This difference must be the power that is stored in the fuel pin. A positive value indicates that the rod is heating up. A negative value indicates that the rod is cooling off. The zero value between the heat-up and the cool-down, which is reached at 4.07 hours into the transient, is the time of the peak temperature.

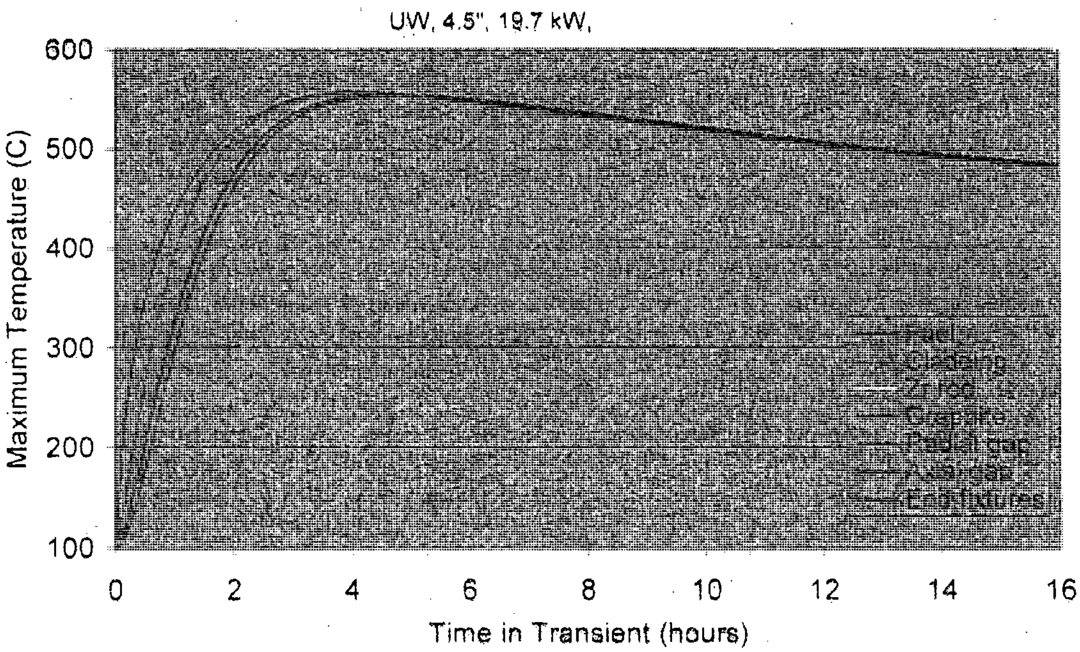

Figure B4. UWNR Maximum Temperature by Region
(4.5-inch water level; 19.7 kW/rod; drain)

The red dashed curve, which coincides with the solid pink curve, is the guessed value (that was used in the CFD analysis) of the amount of heat that is used to generate steam. This should closely match the heat that is transferred from the submerged surface of the rod to the water, the solid pink curve. It took a few iterations in which the pink curve from one iteration became the red dashed curve of the next before the pink and red curves became essentially coincident.

The ratio of the pink curve to the dark blue curve is shown in Figure B6. Thus, at very early times much of the heat is stored in the rod as the rod is heating up, leaving only a relatively small amount (<50%) for steam production. At an hour, more than 60% is going to steam production and after 3 hours more than 70% is going to steam production. At the time of the peak temperature, 4.07 hours, about 72% of the power is going to steam production. These relatively large percentages are to be expected because 4.5 inches, or 30% of the total length of the fuel meat, is submerged. If only about 60% of the power produced by the exposed 70% of the fuel meat length is conducted down the length of the rod to the water, then the portion going to the water would be about 72% of the total decay power.

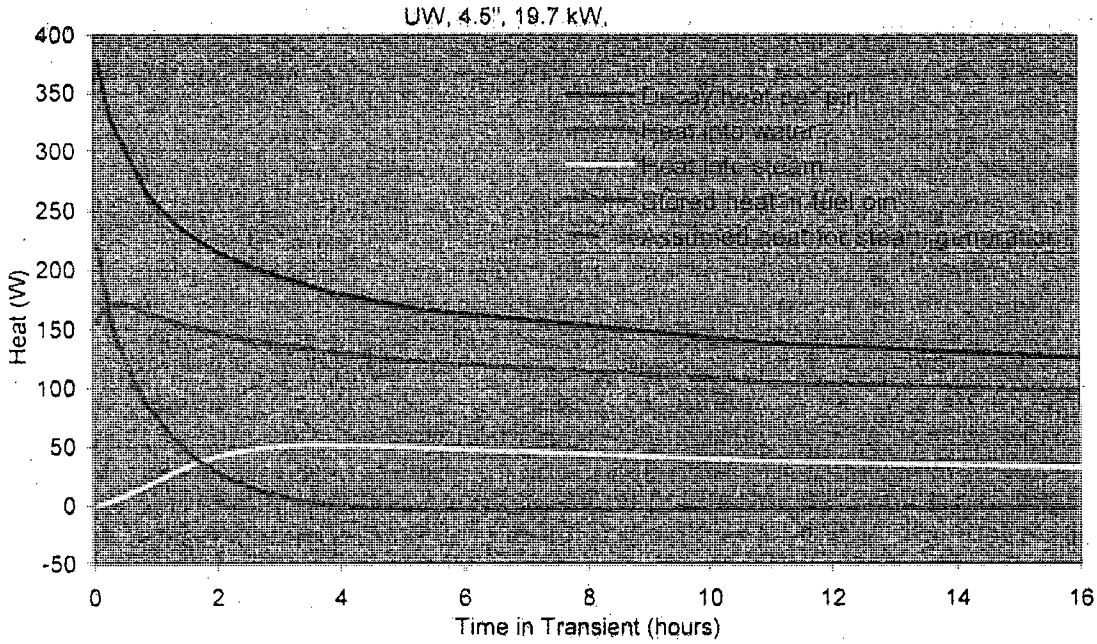

Figure B5. UWNR Distribution of Decay Power
(4.5-inch water level; 19.7 kW/rod;   )

Figure B7 shows the spatial distribution of temperature throughout the model at 14,640 s (4.07 hours) into the transient (4.22 hours after the scram), which is the time that the peak fuel temperature of 557.5°C is reached. The peak steam temperature of 555.8°C is within 2°C of this value. This small difference is to be expected. There is very little radial temperature variation in the fuel rod and the temperature of the steam that is in contact with the rod should be the same as the rod. The horizontal multicolored stripes in the figure clearly demonstrated that the temperature gradient is predominantly in the axial direction, as is the heat flow.

The portion of the fuel rod that is below the water level (where the steam inlet in the model cross section occurs) appears in Figure B7 as a solid blue color because the 110° C boundary condition tends to keeps this region at the boundary temperature. The minimum temperature shown in the legend, 103.4°C, is less than this value because the steam enters at 100°C.

The peak temperature in the fuel rod occurs very close to the top of the fuel column. The temperature distribution in the figure along the axial length of the fuel column approximates that of steady-state one-dimensional heat transfer in the axial direction in a solid that has a uniform heat generation rate. Therefore, the rod temperature essentially increases with the square of the distance from the peak rod temperature. This is shown more clearly in Figure B8, where the temperatures along a vertical line at the inner edge of the fuel (outer edge of the zirconium rod) are plotted for the same time as in Figure B7. Both figures show the increase in the magnitude of the temperature gradient with distance downward from the peak temperature, as is expected in heat-generating solids.

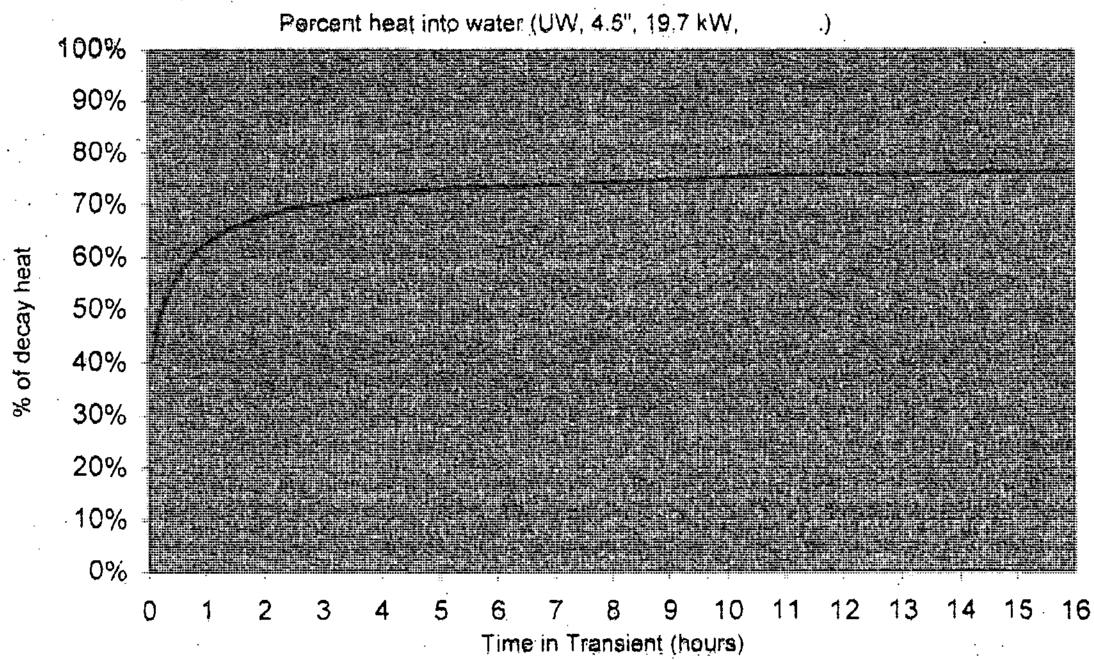

Figure B6. UWNR % of Decay Power to Water
(4.5-inch water level; 19.7 kW/rod;)

The horizontal colored stripes bend upward and look like halves of parabolas in the fuel region. This is because the steam temperature decreases from the clad surface to the outer fluid boundary. This decrease in temperature becomes obvious if one follows a thin imaginary horizontal line from the clad to the outer edge of the fluid region. Both the rod and the steam temperatures increase with axial distance from the water level and are nearly the same at the channel exit.

Figure B6, which was developed for the channel surrounding the highest power rod, is assumed to be representative of the entire core. The reduction in water level with time is obtained by integrating the decay power going into the liquid with respect to time. The integration starts at the start of the transient, which is after the scram. Thus, for any time during the transient, the total energy that was used to convert 25°C water to 100°C saturated steam can be determined. Since for every 614 calories of energy, 1 gram of steam is produced, it is a simple matter to determine the mass and the volume of water that has been boiled from the pool. The reduction in level is simply this volume divided by the surface area of the pool in the core region. The surface area of the (empty) UWNR pool is given as 89.13 ft^2. The fuel rods and the reflector and other structures in the core region will occupy some of this surface area. Thus, the surface area in the core region is estimated to be 85 ft^2. Figure B9 shows the reduction in water level in inches and the decay power history in percent. Thus, the water level goes down 0.25 inches in 4.04 hours, 0.50 inches in 7.98 hours, 0.75 inches in about 14 hours, 1.00 inches in 21.95 hours, and 2.00 inches in about 50 hours. The corresponding decay power percentages can be read from the figure.

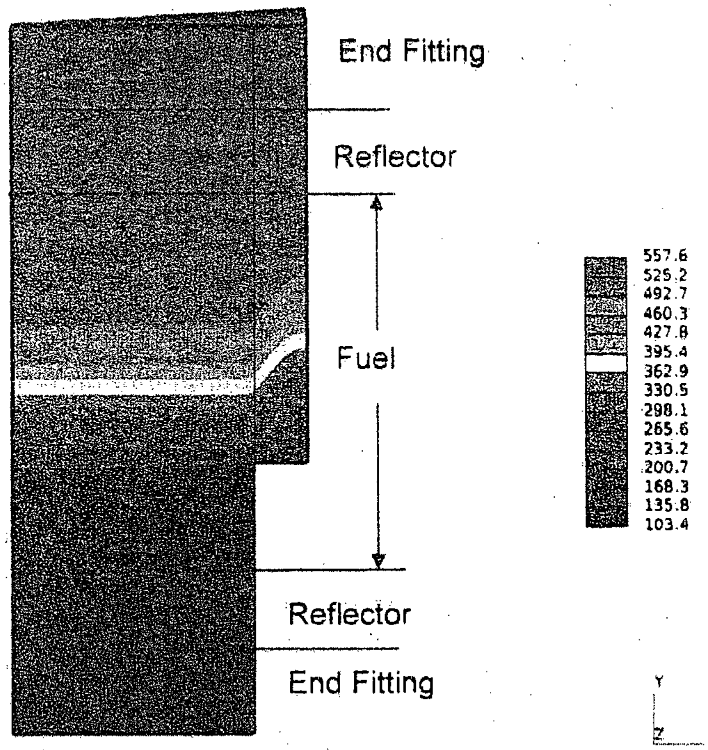

Figure B7. UWNR Temperature Distribution (C) at 4.07 Hours
(4.5-inch water level; 19.7 kW/rod,)

When the transient analysis is performed with a fixed water level of 4.5 inches, the peak fuel temperature, as shown in Figure B4, is predicted to be 554.7°C and to occur at 4.07 hours. As Figure B9 indicates, the water level drops by 0.25 inches by about the time the peak is reached. Therefore, the peak fuel temperature could be significantly higher than the predicted 554.7°C. Since at this time the behavior of the system is quasi steady state, a steady-state solution was obtained with the decay power level corresponding to 4.04 hours (4.29 hours after the scram) and with the water level reduced to 4.25 inches. This steady-state solution produced a peak fuel temperature of 577.5°C. Additional steady-state analyses corresponding to points later in time along Figure B9 were also considered. As the results shown in Table B3 indicate, later times with lower water levels and decay powers produce lower temperatures.

Some may question the notion that for times near or beyond the peak temperature, a steady-state solution can be used in place of a transient one. It was a simple matter to demonstrate the degree of validity of this theory with the aid of additional steady-state solutions. Therefore, in Table B4 the transient solution results at 4.04 seconds and an initial water level of 4.5 inches are compared with its steady-state solution counterpart. The differences between the two sets of results are extremely small.

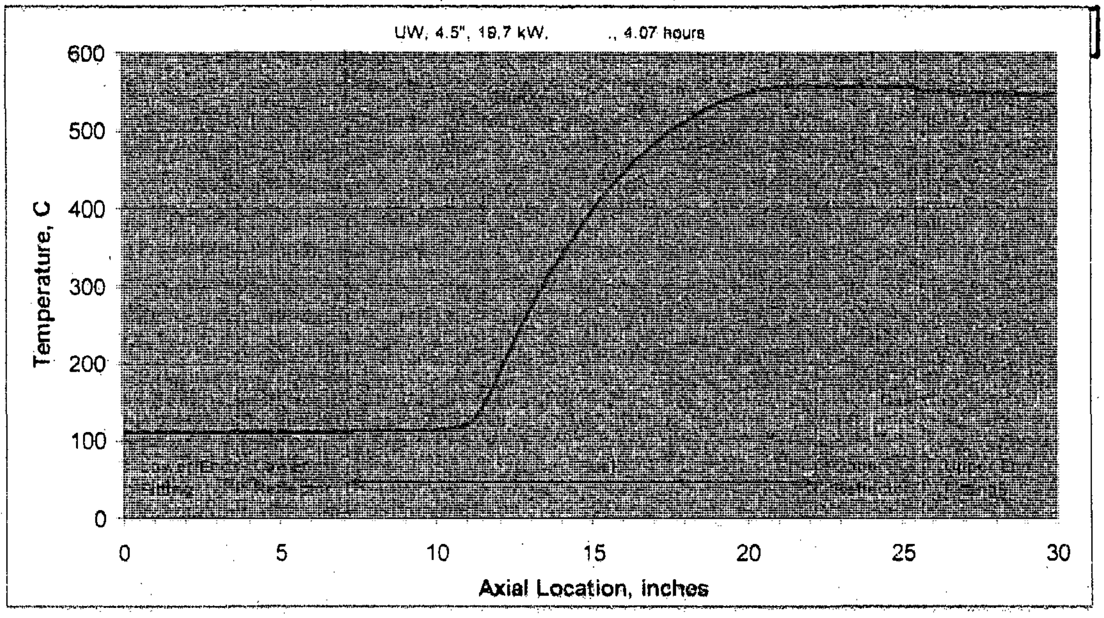

Figure B8. UWNR Axial Temperature Distribution (C) at 4.07 Hours along Vertical Line at Inner Edge of Fuel (4.5-inch water level; 19.7 kW/rod,)

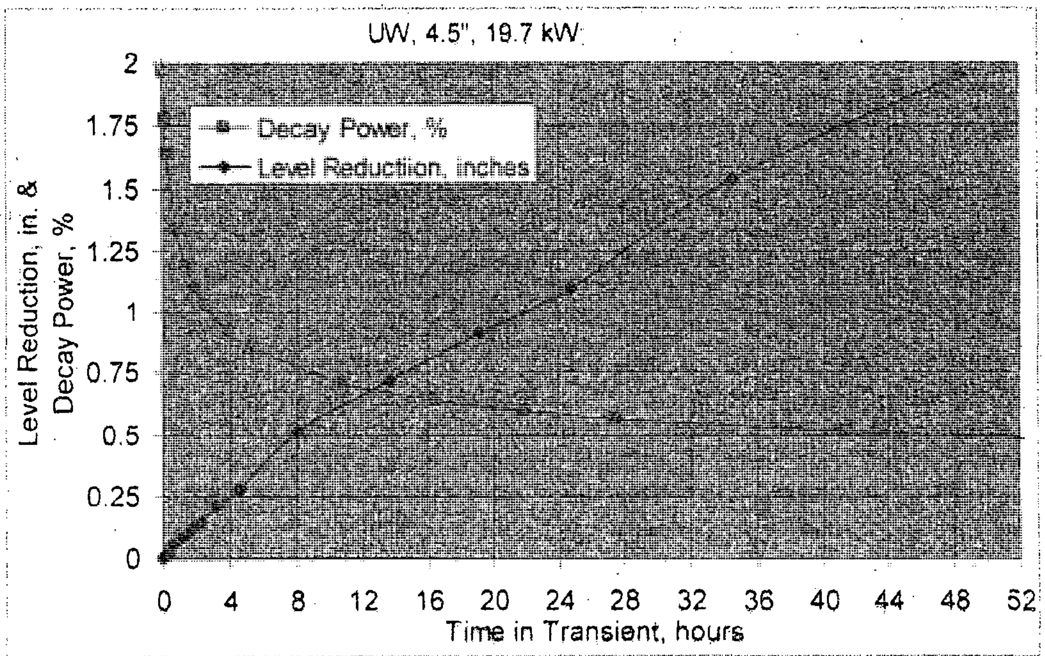

Figure B9. UWNR Decay Power and Level Reduction
(4.5-inch water level; 19.7 kW/rod,)

Table B3. Maximum Fuel Temperature for the
University of Wisconsin Nuclear Reactor

Case or Point in Time	Time, hours*	Water Level above Fuel Bottom, in.	Decay Power, W	Decay Power into Water, %	Maximum Fuel Temp., C
UW-1	4.04	4.25	178.6	70.00	577.5
UW-2	7.98	4.00	152.7	70.90	560.4
UW-3	21.95	3.50	115.7	72.59	528.4

*Time = 0 is when the water level has just reached the bottom of the beam port, which is assumed to be after the scram. All of these results are based on steady-state analyses.

Table B4. Comparison of Transient and Steady-State Results
for the University of Wisconsin Nuclear Reactor

Case or Point in Time	Type of Solution	Time hours*	Water Level above Fuel Bottom, in.	Decay Power, W	Decay Power into Water, %	Maximum Fuel Temp., C
UW-4	Transient	4.04	4.50	178.6	72.05	557.5
UW-5	Steady State				71.27	554.7

*Time = 0 is when the water level has just reached the bottom of the beam port, which is assumed to be after the scram.

References:

1. *STAR-CD*, CD-adapco, Plymouth, MI, USA.
2. *American National Standard Decay Heat Power in Light Water Reactors*, ANSI/ANS-5.1-2005, American Nuclear Society, La Grange Park, Illinois, USA, 2005.
3. Safety Analysis Report for the Conversion of the University of Wisconsin – Madison TRIGA Reactor from HEU to LEU Fuel, University of Wisconsin – Madison, August 2008.

The results of the analyses performed in references 33 and 34 are summarized in the table below. It is evident that in both analyses, the complete LOCA is more limiting than the partial LOCA.

Table RAI-55-5, Summary of Complete vs. Partial LOCA Results

	Complete LOCA Max Temp (°C)	Partial LOCA Max Temp (°C)
Reference 33	432*	418*
Reference 34	585	578

* Reference 33 compares the maximum temperature in a core-averaged location instead of the hot-rod.

56. Section 14. Please provide replacement TS pages with the proposed changes to the TS.

Licensee's Response:

Replacement Technical Specification pages with the proposed changes are attached. See attachment 8.

57. Section 14. For each proposed change to the TS, provide a justification.

Licensee's Response:

Justifications for each proposed change to the Technical Specifications are attached. See attachment 8.

58. Please refer to questions 30 and 58 when responding to the following question:

Section 14.2.2. What are the predicted temperatures of the IFE in their existing core positions when the limiting safety system setting (LSSS) for power is reached?

Licensee's Response:

The predicted IFE temperatures in D4 SW and E3 NE at 1.3MW are:

Table RAI-58-1, IFE Temperatures at 1.3MW

IFE Summary Table at 1.3 MW							
IFE Location		0.1 mil gap		0.05 mil gap		0.15 mil gap	
		°C	°F	°C	°F	°C	°F
D4 SW	Bottom	535.23	995.41	476.65	889.96	589.16	1092.48
	Center	516.39	961.49	460.09	860.15	568.30	1054.93
	Top	494.89	922.80	441.24	826.22	544.45	1012.01
E3 NE	Bottom	348.95	660.11	313.48	596.26	382.38	720.28
	Center	338.97	642.15	304.89	580.79	371.16	700.08
	Top	324.47	616.05	292.42	558.36	354.81	670.66

Note that the revised LEU-BOL axial power shape was used for D4 SW.

59. Section 14.2.2. The discussion of the basis for the LSSS for the fuel temperature refers to a 25°C margin to the fuel temperature safety limit. Provide the analysis for the development of this margin to the safety limit.

Licensee's Response:

The basis in section 14.2.2 is incorrect. The 25°C margin is applied to the LSSS and not the safety limit. Section 4.7.6 of the conversion analysis does not predict that if the IFE thermocouple reaches 400°C then the maximum fuel temperature would be no greater than 1125°C.

The basis for the 25°C margin, as applied to the LSSS, and not the safety limit, was derived by noting the average maximum difference between two thermocouples within an IFE, as shown in Table 4.7.14, page 116, to be 20.625°C. Additionally, the uncertainty of the measurement from a thermocouple embedded in an IFE was determined to be ± 3.72°C[1,2] and the uncertainty of the calibrated fuel temperature safety channel was determined to be ± 0.3°C[3], for an overall measurement uncertainty of ± 3.73°C. The uncertainty in the measurement was applied to the thermocouple range, to derive 24.355°C using error propagation (square root of sum of squares). Therefore, if the hottest thermocouple was reading 425°C, then the coldest thermocouple would be reading no less than the LSSS of 400°C.

The analysis of section 4.7.6 applied this margin of 25°C to the LSSS, to determine that any pin with a peaking factor of at least 0.866 will have a maximum thermocouple temperature of 425°C, and from the foregoing analysis, the coldest thermocouple reading of no less than 400°C.

The relationship between the LSSS and the safety limit is analyzed in section 4.7.6. The analysis shows that when D5 SW measures 678°C, a pin with a peaking factor of at least 0.866 will measure 425°C under the following conservative assumptions. First, because the Groeneveld 2006 and Bernath correlations were not developed for use in TRIGA analysis, the more limiting Bernath correlation was used. However, Anderson, et al[4] from the University of Wisconsin has proposed to ANL to precisely determine CHF for the three TRIGA fuel assembly types (hexagonal, circular and rectangular). Second, the flow rate is assumed to be constant from the point in which RELAP5/MOD3.3 predicts flow oscillations to occur. Under these assumptions, CHF is predicted to occur in D5 SW at a rod power of 35.6kW BOL, 35.9kW MOL, and 35.5kW EOL. However, assuming the flow rate continues to increase linearly in the extrapolated region of Figure 4.7.43, page 113, a more realistic rod power would be 41kW to achieve CHF in D5 SW with the Bernath correlation. This provides a 15% margin to DNB. This assumption is consistent with the results predicted in the response to question 23. Finally, the axial power shape of the thermocouple rod used the axial power shape of the cold rod. The cold rod at B3 NE has a smaller axial peaking factor than the hot rod located at D5 SW. The lower axial peaking factor translates into less power being generated near the mid-plane of the element and therefore lower thermocouple temperatures would be calculated. Any element with a pin power peaking factor of at least 0.87 would have a higher axial peaking factor than the cold rod. A lower predicted thermocouple temperature means the necessary pin power peaking factor must be higher to get to the temperature trip set-point. With these assumptions, the margin to the fuel temperature limit is calculated as 472°C and not 25°C.

Finally, it is important to note that to achieve a rod power of 35.6kW or 41kW in D5 SW would require a core power of 1.8MW and 2.1MW, respectively, which is in excess of the power level safety limit of 1.5MW.

References:

1. Sandia Report SAND2004-1023, April 2004, Uncertainty Analysis of thermocouple measurements used in normal and abnormal thermal environments: experiments at Sandia's Radiant Heat Facility and Lurance Canyon Burn Site. By James T. Nakos.
2. Manual on the Use of Thermocouples in Temperature Measurement, Fourth Edition, ASTM Manual Series: MNL12, Revision of Special Technical Publications (STP) 470B, 1993.
3. Operators Manual, "DP81/DP82 Digital Process Indicators," Omega.
4. Critical Heat Flux in TRIGA Research Reactors, M. Anderson proposal to Argonne National Laboratory.

60. Please refer to questions 30 and 56 when responding to the following question: Section 14.2.2. The basis for the TS discusses a relationship between power peaking of 1.16 and a LSSS of 500°C. However, this is not discussed in section 4.7.6 of the SAR. Please address.

Licensee's Response:

The original basis for the LSSS of 400°C was based on the 1973 SAR estimate of peak fuel temperatures at the UWNR from the Torrey Pines TRIGA Mark III reactor analysis, despite the fact that these two reactors are geometrically dissimilar. During the refueling of the UWNR to the TRIGA core, measured temperatures for D4 SW were reported to exceed 400°C at 1MW, as reported in the startup program and included in the HEU 2000 license renewal SAR (page 4-45). Therefore, the IFE connected to the fuel temperature safety channel was placed in a location that would not exceed 400°C at 1MW, specifically E3 NE. It is fully expected that fuel temperatures in the interior of the core will be greater than 400°C. Since the analysis for the proposed LEU core shows that the central region of the core would exceed 400°C at 1.0MW, the proposed alternate LSSS of 500°C for the central region of the core allows greater flexibility if it is desired to place the IFE closer to the hot rod. The calculation of the relationship between the peaking factors and the LSSS is detailed below.

Currently the technical specification for the IFE allows the IFE to be placed anywhere in the core with FLIP fuel. The trip set point is when a thermocouple in the IFE hits the LSSS of 400°C providing margin for the fuel temperature limit of 1150°C. However, depending upon which portion of the core the IFE is operating at, it would be necessary for the reactor to operate in excess of the 1.25MW trip set point to even approach 400°C. If the IFE were placed in the coldest location of the reactor (B3 NE) the hot rod location would experience CHF before the IFE would experience a fuel temperature greater than 400°C. Therefore it was necessary to determine a range of core locations that would still protect the core from possible fuel damage.

The power when CHF would occur in the hot rod is calculated with the Bernath correlation to be 35.6307 kW/rod, 35.8807 kW/rod, and 35.4920 kW/rod for LEU-BOL, LEU-MOL, and LEU-EOL respectively using the pseudo-transient results to extend the RELAP5 predicted non-oscillatory flow rate for rod powers up to 29 kW/rod. This correlates to a predicted core power to reach CHF of 1.837 MW, 1.863 MW, and 1.879 MW for LEU-BOL, LEU-MOL, and LEU-EOL respectively with their respective pin power peaking factors. This number makes the limiting assumption the mass flow rate of water will not increase as core power increases once the RELAP5 model predicts the flow will oscillate. Further calculations have shown if one assumes the flow rate continues on the same linear trend as shown by the dashed line in Figure 4.7.38, the power to reach CHF would be approximately 41 kW/rod with the Bernath correlation or a total core power of 2.113 MW as seen in Figure 4.7.40. This gives a limiting assumption of the power to reach CHF by 15%. All analysis used the predicted core power to reach CHF using the last known RELAP5 calculated flow rate in order to determine the most limiting thermocouple temperature.

Furthermore, instead of using the hot rod axial and radial power profiles, the cold rod axial and radial power profile was used to calculate the temperatures of the thermocouples. The thermocouples are located 0.3 in (0.762 cm) from the fuel centerline, 6.5, 7.5, and 8.5 in (16.51, 19.05, 21.59 cm) above the bottom of the active fuel. The maximum axial peaking factor for B3NE is 1.2943, 1.2428 and 1.2466 for LEU-BOL, LEU-MOL, and LEU-EOL respectively. The maximum radial peaking factors for B3 NE is 1.7154, 1.6414 and 1.602 for LEU-BOL, LEU-MOL, and LEU-EOL respectively. Thus, it is expected that by using the cold rod axial peaking factors for all rods, the maximum predicted thermocouple is bounding.

All analysis conducted used a hot gap size of 0.1 mils as previously assumed for the hot rod. Then, using the pin power peaking factors of the core, the maximum thermal couple reading for each rod in the core was calculated. It was noticed during the course of the analysis the predicted maximum thermal couple reading had a very linear shape as a function of the pin power peaking factor. Thus a least squares regression line was found to see what the maximum thermocouple temperature is as a function of the pin power peaking factor. For all cases of core life this function had an R^2 value very close to 1, and thus was deemed to be acceptable for predicting the thermocouple temperature without having to run 83 different cases.

While 400°C is the LSSS for the thermocouple, acceptable thermocouples may not read exactly 400°C at the predicted lower bound of power to reach CHF. That does not leave adequate margin, based on analysis detailed in question 59, and it was decided to give a 25°C of further margin to the LSSS of 400°C. With the predicted temperatures, and the acceptance criteria for the thermocouple locations, maps of acceptable IFE locations for all times of core life can be seen in Figures RAI-60-1 to RAI-60-6. The red (X) signifies the predicted temperature is less than 400 (500 for inner core positions), a yellow exclamation (!) signifies the temperature is between 400 and 425 (500 – 525 for inner core positions) and a green check mark (✓) signifies the temperature is greater than 425 (525 for inner core positions) at the predicted core power to reach CHF. It is recognized that with burnup the peaking factors in the core will change and therefore acceptable IFE locations will also change. To account for this, Figures RAI-60-7 and RAI-60-8 show acceptable IFE locations for all phases of core burnup.

Figure RAI-60-1, Locations for IFE where IFE would hit at least 425°C (LEU-BOL)

Figure RAI-60-2, Locations for IFE where IFE would hit at least 525°C (LEU-BOL)

Figure RAI-60-3, Locations for IFE where IFE would hit at least 425°C (LEU-MOL)

Figure RAI-60-4, Locations for IFE where IFE would hit at least 525°C (LEU-MOL)

Figure RAI-60-5, Locations for IFE where IFE would hit at least 425°C (LEU-EOL)

Figure RAI-60-6, Locations for IFE where IFE would hit at least 525°C (LEU-EOL)

After finding the locations where the IFE would hit at least 425 or 525 for all times of core life, 2 summary figures were constructed to show where the IFE could be placed for all times of core life. If the IFE predicted at least 425 or 525 throughout all times of core life, then the IFE was deemed to be acceptable to be placed there throughout core life as seen in Figures RAI-60-7 and RAI-60-8. The IFE locations will be D4 SW and E3 NE for the LEU core which are in the same location as the HEU core. If an IFE were moved from these positions a 50.59 review would be performed.

Figure RAI-60-7, Where IFE would hit at least 425°C for all times of LEU core life

Figure RAI-60-8, Where IFE would hit at least 525°C for all times of LEU core life

61. Section 14.3.2 (p. 235). Is a separate Technical Specification needed for a limit to prevent exceeding 830°C? Does the 1.4 %Δk/k reactivity insertion limit in TS 3.2 prevent reaching 830°C for all conditions of operation and all times in core life? Please discuss.

Licensee's Response:

Existing Technical Specification 3.2 limits reactivity insertions to 1.4 %Δk/k. Sections 4.7.5 and 4.7.10 show that the prompt peak fuel temperature after a 1.4 %Δk/k pulse is 726.95°C (1340.51°F) for LEU-MOL. The response to RAI question 27 shows that the maximum temperature within 15 seconds after the pulse is 826°C, although question 27 clarifies that the 830°C limit only applies to the prompt peak temperature, not the maximum temperature within 15 seconds after the pulse. Therefore, a 1.4 %Δk/k reactivity insertion does prevent the reactor from reaching 830°C for all normal conditions of operation at all times in core life and no separate technical specification is needed to prevent exceeding 830°C.

62. Section 14.3.3.4 (p. 236). This proposed technical specification appears to be identical to the LSSS. Please explain why it is needed.

Licensee's Response:

It is agreed that the proposed technical specification 14.3.3.4 is redundant and is not needed. This specification has been removed from the proposed Technical Specifications. See question 56.

63. UWNR TS 3.3.3, Reactor Safety System. In Table 1, it appears the function for the Fuel Element Temperature may be inconsistent with the proposed LSSS of 500°C as measured in an instrumented fuel element with a pin power peaking factor of at least 1.61. Please discuss.

Licensee's Response:

Note that the proposed LSSS of 500°C is for an IFE pin power peaking factor of at least 1.16. It is agreed that the function for the fuel element temperature channel is inconsistent with the proposed Technical Specification 2.2. A revision to specification 3.3.3 is included in the proposed Technical Specifications. See question 56.

64. UWNR TS 5.6, Reactor Building. Please verify that the bases for the current TS are consistent with the conversion safety analysis report.

Licensee's Response:

It is agreed that the bases for TS 5.6 are inconsistent with the conversion safety analysis report. Revised bases are included in the proposed Technical Specifications. See question 56.

65. Section 15.2. The NRC suggests restating license conditions to make the conditions clearer to understand. Also, the exempt status table in the license condition is not needed with the conversion to LEU. The requirements of 10 CFR 73.6 applies to your special nuclear material the table not withstanding. An example of a restated license condition based on your proposed possession limits is as follows:

2.B.(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material,"

a. to receive, possess and use, in connection with operation of the facility, up to 15.0 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

b. to receive, possess and use, in connection with operation of the facility, up to 150 grams of contained uranium-235 of any enrichment in the form of neutron detectors;

c. to receive, possess and use, in connection with operation of the facility, up to 16 grams of contained plutonium in the form of plutonium-beryllium neutron source;

d. to receive, possess, use, but not separate, in connection with operation of the facility, such special nuclear material as may be produced by operation of the facility; and

e. to possess, but not use, up to 18.0 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility.

Please provide a bases for your proposed LEU possess limit of 15 kilograms of contained uranium-235.

Licensee's Response:

The possession limit of 15 kg U-235 is based on [redacted] for [redacted] which is 14.55 kg. This number is rounded up to allow for some pins which may exceed [redacted] as built. Furthermore, it is agreed to restate the license conditions as proposed to make them clearer to understand, and to remove the exempt status table.

UNIVERSITY OF WISCONSIN
NUCLEAR REACTOR
LICENSE NO. R-74
DOCKET NO. 50-156

RESPONSE TO RAI REGARDING HEU/LEU CONVERSION

ATTACHMENT 1 - REDACTED

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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RESPONSE TO RAI REGARDING HEU/LEU CONVERSION

ATTACHMENT 2 - REDACTED

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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RESPONSE TO RAI REGARDING HEU/LEU CONVERSION

ATTACHMENT 3 - REDACTED

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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RESPONSE TO RAI REGARDING HEU/LEU CONVERSION

ATTACHMENT 4 - REDACTED

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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RESPONSE TO RAI REGARDING HEU/LEU CONVERSION

ATTACHMENT 5 - REDACTED

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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RESPONSE TO RAI REGARDING HEU/LEU CONVERSION

ATTACHMENT 6 - REDACTED

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

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RESPONSE TO RAI REGARDING HEU/LEU CONVERSION

ATTACHMENT 7 - REDACTED

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

LEU Conversion Changes to Technical Specifications: Item-By-Item Justification

Note: Text from the Technical Specifications appears in `fixed-width Courier New` font.

Table of Contents

The following entries are deleted:

Furthermore, the following entry is added:

Numbering of later entries is left unchanged. See changes to Technical Specifications Page 4 for further justification.

TABLE OF CONTENTS

Technical Specifications Page 3

TS 1.14, which says:

A fuel element is a single TRIGA fuel rod of either standard or FLIP type

Is changed as follows:

A fuel element is a single TRIGA fuel rod of ~~either standard or FLIP~~ LEU 30/20 type

To read:

A fuel element is a single TRIGA fuel rod of LEU 30/20 type

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore the definition of "Fuel Element" is revised to add LEU 30/20 fuel type and remove standard and FLIP types.

1.10 EXPERIMENTAL FACILITIES

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and in-pool irradiation facilities.

REACTOR COMPONENTS

1.11 SHIM-SAFETY BLADE

A shim-safety blade is a control blade having an electric motor drive and scram capabilities. It may have a fueled follower section.

1.12 TRANSIENT ROD

The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It may have a voided follower.

1.13 REGULATING BLADE

The regulating blade is a low worth control blade that need not have scram capability and may have a fueled follower. Its position may be varied manually or by the servo-controller.

1.14 FUEL ELEMENT

A fuel element is a single TRIGA fuel rod of LEU 30/20 type.

1.15 FUEL BUNDLE

A fuel bundle is a cluster of three or four fuel elements secured in a square array by a top handle and a bottom grid plate adaptor.

1.16 CORE LATTICE POSITION

The core lattice position is that region in the core (approximately 3" by 3") over a grid plug hole. It may be occupied by a fuel bundle, an experiment or experimental facility, or a reflector element.

TS 1.18, 1.19, and 1.20, which say:

1.18 STANDARD CORE
A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

1.19 MIXED CORE
A mixed core is an arrangement of standard TRIGA fuel elements with at least 35 TRIGA-FLIP fuel elements located in a central region of the core.

1.20 FLIP CORE
A FLIP core is an arrangement of TRIGA-FLIP fuel in the reactor grid plate.

Are deleted. Furthermore, the following definition is added:

1.18 LEU 30/20 CORE
A LEU 30/20 core is an arrangement of TRIGA LEU 30/20 fuel in the reactor grid plate.

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore, the definitions of "Standard Core," "Mixed Core," and "Flip Core" are deleted because following conversion to LEU 30/20 fuel there is only one valid operational core. Similarly, a LEU 30/20 core is defined to be an arrangement of TRIGA LEU 30/20 fuel in the reactor grid plate.

Technical Specifications Page 4, Continued

TS 1.21, which says:

An operational core may be a standard core, mixed core, or FLIP core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

Is changed as follows:

An operational core ~~may be a standard core, mixed core, or FLIP core~~ is an LEU 30/20 core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

To read:

An operational core is an LEU 30/20 core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore, the definition of "Operational Core" is changed to identify only an LEU 30/20 core as being a valid operational core.

1.17 INSTRUMENTED ELEMENT

An instrumented element is a special fuel element in which a sheathed chromel-alumel or equivalent thermocouple is embedded in the fuel near the horizontal center plane of the fuel element at a point approximately 0.3 inch from the center of the fuel body.

1.18 LEU 30/20 CORE

A LEU 30/20 core is an arrangement of TRIGA LEU 30/20 fuel in the reactor grid plate.

1.21 OPERATIONAL CORE

An operational core is an LEU 30/20 core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

REACTOR INSTRUMENTATION

1.22 SAFETY LIMITS

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

1.23 LIMITING SAFETY SYSTEM SETTINGS

Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions.

TS 2.1, which says:

a. The temperature in a TRIGA-FLIP fuel element shall not exceed 1150°C under any conditions of operation.
b. The temperature of a standard TRIGA fuel element shall not exceed 1000°C under any conditions of operation.
c. The reactor power level shall not exceed 1500 kW under any conditions of operation.

Is changed as follows:

a. The temperature in a TRIGA ~~FLIP~~ LEU 30/20 fuel element shall not exceed 1150°C under any conditions of operation.
~~b. The temperature of a standard TRIGA fuel element shall not exceed 1000°C under any conditions of operation.~~
~~c~~b. The steady-state reactor power level shall not exceed 1500kW under any conditions of operation.

To read:

a. The temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1150°C under any conditions of operation.
b. The steady-state reactor power level shall not exceed 1500kW under any conditions of operation.

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore, the fuel temperature safety limit for LEU 30/20 fuel is added, and the fuel temperature safety limits for FLIP and standard fuels are removed because after the conversion only LEU 30/20 fuel is used. The outline numbering for the power level safety limit is updated, and clarified to be steady-state power since TRIGA LEU 30/20 fuel is designed for pulse operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to fuel element temperature and steady-state reactor power level.

Objective

The objective is to define the maximum fuel element temperature and reactor power level that can be permitted with confidence that no fuel element cladding failure will result.

Specifications

a. The temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1150°C under any conditions of operation.

b. The steady-state reactor power level shall not exceed 1500 kW under any conditions of operation.

Bases

A loss of integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by air, fission produce gases, and hydrogen from dissociation of the fuel moderator. The magnitude of this pressure is determined by the fuel moderator temperature and the ratio of hydrogen to zirconium in the alloy.

Technical Specifications Page 8

TS 2.1 bases, which say:

It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500 kW. It has been shown by analysis and by measurements on other TRIGA reactors that a power level of 1500 kW corresponds to a peak fuel temperature of approximately 500°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

Are changed as follows:

It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500 kW. ~~It has been shown by analysis and by measurements on other TRIGA reactors~~ The LEU Conversion SAR section 4.7.8 shows by analysis that a power level of 1500 kW corresponds to a peak fuel temperature of approximately ~~500°C~~ 665°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

To read:

It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500 kW. The LEU Conversion SAR section 4.7.8 shows by analysis that a power level of 1500 kW corresponds to a peak fuel temperature of 665°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

Justification:

Calculations performed as part of the conversion analysis show a peak fuel temperature of 665°C at 1.5MW. These calculations are based on the proposed specific TRIGA LEU 30/20 core design at the University of Wisconsin and are not based on analyses of other TRIGA reactors. The basis for Technical Specification 2.1 is therefore updated to reference this calculation in the LEU Conversion SAR.

Technical Specifications Page 8, Continued

TS 2.2(1), which says:

The limiting safety system setting for fuel temperature shall be 400°C (750°F) as measured in an instrumented fuel element. For a mixed core, the instrumented element shall be located in the region of the core containing FLIP type elements.

Is changed as follows:

The limiting safety system setting for fuel temperature shall be 400°C ~~(750°F)~~ as measured in an instrumented fuel element with a pin power peaking factor between 0.87 and 1.16, or 500°C as measured in an instrumented fuel element with a pin power peaking factor of at least 1.16. ~~For a mixed core, the instrumented element shall be located in the region of the core containing FLIP type elements.~~

To read:

The limiting safety system setting for fuel temperature shall be 400°C as measured in an instrumented fuel element with a pin power peaking factor between 0.87 and 1.16, or 500°C as measured in an instrumented fuel element with a pin power peaking factor of at least 1.16.

Furthermore, the outline headings for TS 2.2(1) and TS 2.2(2) are rewritten as 2.2(a) and 2.2(b) to be consistent with the rest of the Technical Specifications.

Justification:

Calculations performed as part of the conversion analysis show that it is possible that an IFE located in a core position with a pin power peaking factor of less than 0.87 will not protect the fuel temperature safety limit from being reached at the LSSS of 400°C. The technical specification is modified to impose this limit. Additionally the analysis for the proposed LEU core shows that the central region of the core would exceed 400°C at 1.0MW. The proposed alternate LSSS of 500°C for the central region of the core allows greater flexibility if it is desired to place the IFE closer to the hot rod. However, the pin power peaking factor at the core location of the IFE must be at least 1.16 to prevent the fuel temperature safety limit from being reached at the LSSS of 500°C.

Additionally, the technical specification references to mixed cores, which are no longer approved for use, are removed.

Technical Specifications Page 8, Continued (continuing on page TS-9)

TS 2.2(1) bases, which say:

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 400°C provides a safety margin of 750°C for FLIP type fuel elements and a margin of 600°C for standard TRIGA fuel elements. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. If the thermocouple element is located in a region of lower temperature, such as on the periphery of the core, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than a factor of two. Thus, when the temperature in the thermocouple elements reaches the trip setting of 400°C, the true temperature at the hottest location would be no greater than 800°C providing a margin to the safety limit of at least 200°C for standard fuel elements and 350°C for FLIP type elements. These margins are ample to account for the remaining uncertainty in the accuracy of the fuel temperature measurement channel and any overshoot in reactor power resulting from a reactor transient during steady state mode operation. For a mixed core (i.e., one containing both standard and FLIP type elements), the requirement that the instrumented element be located in the FLIP region of the core provides an even greater margin of safety since the peak to average power ratio within that region will be smaller than over an entire core composed of elements of the same type.

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting of the "tail" of the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

Technical Specifications Page 8, Continued (continuing on page TS-9)

Are replaced to read (2nd paragraph does not change):

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. Analyses performed in section 4.7.6 of the LEU Conversion Analysis show that with the IFE in a core location with a pin power peaking factor of at least 0.87, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 400°C providing a margin of 472°C to the safety limit. The same analyses also show that with the IFE in a core location with a pin power peaking factor of at least 1.16, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 500°C providing a margin of 472°C to the safety limit.

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting off the "tail" of the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

Furthermore, the TS 2.2(1) bases are rewritten as TS 2.2(a) bases to be consistent with the rest of the Technical Specifications.

Justification:

Calculations performed as part of the conversion analysis show that it is possible that an IFE located in a core position with a pin power peaking factor of less than 0.87 will not protect the fuel temperature safety limit from being reached at the LSSS of 400°C. The technical specification is modified to impose this limit. Additionally the analysis for the proposed LEU core shows that the central region of the core would exceed 400°C at 1.0MW. The proposed alternate LSSS of 500°C for the central region of the core allows greater flexibility if it is desired to place the IFE closer to the hot rod. However, the pin power peaking factor at the core location of the IFE must be at least 1.16 to prevent the fuel temperature safety limit from being reached at the LSSS of 500°C.

Additionally, the technical specification references to mixed cores, which are no longer approved for use, are removed.

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TS 2.2(2) bases, which say:

Calculations and measurements for similar TRIGA reactors indicate at 1.25MW, the peak fuel temperature in the core will be approximately 400°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

Are changed as follows:

~~Calculations and measurements for similar TRIGA reactors indicate at 1.25MW~~ Analysis in section 4.7 of the Conversion Analysis SAR shows that at 1.3 MW, the peak fuel temperature in the core will be approximately ~~400°C~~ 604°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

To read:

Analysis in section 4.7 of the Conversion Analysis SAR shows that at 1.3 MW, the peak fuel temperature in the core will be approximately 604°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

Furthermore, the TS 2.2(2) bases are rewritten as TS 2.2(b) to be consistent with the rest of the Technical Specifications.

Justification:

Calculations performed as part of the conversion analysis show a peak fuel temperature of 604°C at 1.3MW, providing margin to the departure of nucleate boiling and additional protection of the fuel temperature safety limit at the reactor power LSSS. These calculations are based on the proposed specific TRIGA LEU 30/20 core design at the University of Wisconsin and are not based on calculations and measurements of similar TRIGA reactors. Therefore, the calculation of the maximum fuel temperature at the power level LSSS is updated.

The safety limit for the TRIGA-FLIP fuel element is based on data which indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided the temperature does not exceed 1150°C and the fuel cladding is water cooled (pages 3-1 to 3-23 of GA-9064).

The safety limit for the standard TRIGA fuel is based on data including the large amount of experimental evidence obtained during high performance reactor tests of this fuel. These data indicate that the stress in the cladding (due to hydrogen pressure from the dissociation of zirconium hydride) will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1000°C and the fuel cladding is water cooled (GA-9064, pages 3-1 to 3-23).

It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500kW. The LEU Conversion SAR section 4.7.8 shows by analysis that a power level of 1500 kW corresponds to a peak fuel temperature of 665°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

2.2 LIMITING SAFETY SYSTEM SETTING

Applicability

This specification applies to the scram setting which prevents the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being reached.

Specifications

a. The limiting safety system setting for fuel temperature shall be 400°C as measured in an instrumented fuel element with a pin power peaking factor between 0.87 and 1.16, or 500°C as measured in an instrumented fuel element with a pin power peaking factor of at least 1.16.

b. The limiting safety system setting for reactor power level shall be 1.25 MW.

Bases

a. The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. Analyses performed in section 4.7.6 of the LEU Conversion Analysis show that with the IFE in a core location with a pin power peaking factor of at least 0.87, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 400°C providing a margin of 472°C to the safety limit. The same analyses also show that with the IFE in a core location with a pin power peaking factor of at least 1.16, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 500°C providing a margin of 472°C to the safety limit.

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting of the "tail" of the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

b. Analysis in section 4.7 of the Conversion Analysis SAR shows that at 1.3 MW, the peak fuel temperature in the core will be approximately 604°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

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TS 3.2 bases, which say:

Measurements performed on the Puerto Rico Nuclear Center TRIGA-FLIP reactor indicated that a pulse insertion of reactivity of 1.4 %Δk/k resulted in a maximum temperature rise of approximately 400°C. With an ambient water temperature of approximately 100°C, the maximum fuel temperature would be approximately 500°C resulting in a safety margin of 500°C for standard fuel and 650°C for FLIP type fuel. These margins allow amply for uncertainties due to the accuracy of measurement or location of the instrumented fuel element or due to the extrapolation of data from the PRNC reactor.

Are replaced in their entirety to read:

The LEU Conversion SAR section 4.7.10 shows by analysis that a 1.4 %Δk/k limitation on pulse reactivity will result in a maximum fuel temperature of 790°C. This leaves a margin to the 1150°C Safety Limit of 360°C, and a margin of 40°C to the 830°C operational limit recommended by General Atomics, "Pulsing Temperature Limit for TRIGA LEU Fuel," GA-C26017 (December, 2007).

Justification:

Calculations performed as part of the conversion analysis show a peak fuel temperature of 727°C following a 1.4 %Δk/k pulse reactivity insertion. These calculations are based on the proposed specific TRIGA LEU 30/20 core design at the University of Wisconsin and are not based on measurements of the Puerto Rico Nuclear Center reactor.

Specification

The reactivity to be inserted for pulse operation shall be determined and mechanically limited such that the reactivity insertion will not exceed 1.4% Δ k/k.

Basis

The LEU Conversion SAR section 4.7.10 shows by analysis that a 1.4% $\Delta k/k$ limitation on pulse reactivity will result in a maximum fuel temperature of 790°C. This leaves a margin to the 1150°C Safety Limit of 360°C, and a margin of 40°C to the 830°C operational limit recommended by General Atomics, "Pulsing Temperature Limit for TRIGA LEU Fuel," GA-C26017 (December, 2007).

3.3 CONTROL AND SAFETY SYSTEM

3.3.1 Scram Time

Applicability

This specification applies to the time required for the scrammable control elements to be fully inserted from the instant that a safety channel variable reaches the Safety System Setting.

Objective

The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification

The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control element reaches its fully inserted position shall not exceed 2 seconds.

Basis

This specification assures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor.

TS 3.3.3(a), which says (headings reproduced for clarity):

	Safety System Or Measuring Channel	Minimum No. Operable	Function & Operating Mode in Which Required
a.	Fuel Element Temperature	1	Scram at 400°C. All modes.

Is changed as follows:

	Safety System Or Measuring Channel	Minimum No. Operable	Function & Operating Mode in Which Required
a.	Fuel Element Temperature	1	Scram at 400°C for IFE peaking factors 0.87-1.16 or 500°C for IFE peaking factors >1.16. All modes.

To read:

	Safety System Or Measuring Channel	Minimum No. Operable	Function & Operating Mode in Which Required
a.	Fuel Element Temperature	1	Scram at 400°C for IFE peaking factors 0.87-1.16 or 500°C for IFE peaking factors >1.16. All modes.

Justification:

Calculations performed as part of the conversion analysis show that it is possible that an IFE located in a core position with a pin power peaking factor of less than 0.87 will not protect the fuel temperature safety limit from being reached at the LSSS. The technical specification is modified to impose this limit. Additionally the analysis for the proposed LEU core shows that the central region of the core would exceed 400°C at 1.0MW. The proposed alternate LSSS of 500°C for the central region of the core allows greater flexibility if it is desired to place the IFE closer to the hot rod. The pin power peaking factor requirements for the IFE ensure that the reactor will scram before reaching the fuel temperature safety limit. This revision is in response to RAI question 63.

Technical Specifications Page 13, Continued

The following is added to the end of Table 1 of TS 3.3.3 (headings reproduced for clarity):

	Safety System Or Measuring Channel	Minimum No. Operable	Function & Operating Mode in Which Required
j.	Reactor Pool-water Temperature	1	Scram if water temperature is greater than 130°F; All modes.

Justification:

Calculations performed as part of the conversion analysis are based on a maximum core inlet temperature of 130°F as originally assumed in the SAR. In order to remain within the design basis of this analysis, a new technical specification 3.3.3(j) is added to limit pool water temperature. The limit already exists as an administrative limit, but it is now added to the Technical Specifications in response to RAI question 15.

TABLE 1

	Safety System Or Measuring Channel	Minimum No. Operable	Function & Operating Mode in Which Required
a.	Fuel Element Temperature	1	Scram at 400°C for IFE peaking factors 0.87-1.16 or 500°C for IFE peaking factors >1.16. All modes.
b.	Reactor Power Level	2	Scram at 125% of full licensed power level; Square Wave & Steady State Modes.
c.	Manual Pushbutton	1	Scram; All modes.
d.	Reactor Pool-water Level	1	Scram if water level is less than 19 feet above top of core; All modes.
e.	Log N	1	Prevent firing transient rod when drive is not full in and power level is above 1 kW in all modes.
f.	Log Count Rate	1	Prevent control element withdrawal when neutron count rate is less than 2 per second; All modes.
g.	Preset Timer	1	Transient rod scram 15 seconds or less after pulse; Pulse mode.
h.	High Voltage Monitor	1	Scram on loss of high voltage supply to neutron and gamma ray power level instrumentation detectors; All modes.
i.	Pulse Mode Control Blade Withdrawal Interlock.	1	Prevents withdrawal of control blades while in pulse mode.
j.	Reactor Pool-water Temperature	1	Scram if water temperature is greater than 130°F; All modes.

Technical Specifications Page 14

The following is added to the end of TS 3.3.3 bases:

```
The thermal-hydraulic analysis in the SAR assumes a pool water
temperature of 130°F.  If the temperature exceeds 130°F then the
scram will prevent continued operation in an un-analyzed
condition.
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Justification:

Calculations performed as part of the conversion analysis are based on a maximum core inlet temperature of 130°F as originally assumed in the SAR. In order to remain within the design basis of this analysis, a new technical specification 3.3.3(j) is added to limit pool water temperature. The limit already exists as an administrative limit, but it is now added to the Technical Specifications in response to RAI question 15. The bases are added for the new technical specification 3.3.3(j) to explain that the specification will prevent operation in an un-analyzed condition, since the LEU Conversion SAR assumes a pool water temperature of 130°F.

Bases

The fuel temperature scram provides the protection to assure that if a condition occurs in which the limiting safety system setting is exceeded, an immediate shutdown will occur to keep the fuel temperature below the safety limit.

The reactor power level scrams are provided in steady state and square wave modes as added protection against abnormally high fuel temperatures and to assure that reactor operation stays within the licensed limits.

The manual scram allows the operator a means of rapid shutdown in the event of unsafe or abnormal conditions.

The reactor pool water level scram assures shutdown of the reactor in the event of a serious leak in the primary system or pool.

The Log N interlock prevents firing of the transient rod at power levels above 1.0 kW if the transient rod drive is not in the full down position. This effectively prevents inadvertent pulses which might cause fuel temperature to exceed the safety limit on fuel temperature.

The Log N interlock does not allow control element withdrawal unless the neutron count rate is high enough to assure proper instrument response during reactor startup.

The preset timer assures reduction of reactor power to a low level after a pulse.

The high voltage monitor prevents operation of the reactor with other systems inoperable due to failure of the detector high voltage supplies.

The pulse mode control blade withdrawal interlock prevents reactivity addition in pulse mode other than by firing the transient rod.

The thermal-hydraulic analysis in the SAR assumes a pool water temperature of 130°F. If the temperature exceeds 130°F then the scram will prevent continued operation in an un-analyzed condition.

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TS 5.1, which says:

a. TRIGA-FLIP Fuel
The individual unirradiated FLIP fuel elements shall have the following characteristics:
(1) Uranium content: maximum of 9 Wt-% enriched to nominal 70% Uranium 235.
(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms.
(3) Natural erbium content (homogeneously distributed): nominal 1.5 Wt-%.
(4) Cladding: 304 stainless steel, nominal 0.020 inch thick.
(5) Identification: Top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores.

b. Standard TRIGA fuel
The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:
(1) Uranium content: maximum of 9.0 Wt-% enriched to a nominal 20% Uranium 235.
(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.7 H atoms to 1.0 Zr atoms.
(3) Cladding: 304 stainless steel, nominal 0.020 inch thick.

Is changed as follows:

a. ~~TRIGA-FLIP Fuel~~ TRIGA LEU 30/20 Fuel
The individual unirradiated ~~FLIP~~ TRIGA LEU 30/20 fuel elements shall have the following characteristics:
(1) Uranium content: maximum of ~~9~~ 30 Wt-% enriched to maximum of 19.95 Wt-% with nominal enrichment of ~~70%~~ 19.75 Wt-% Uranium 235.
(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.
(3) Natural erbium content (homogeneously distributed): nominal ~~1.5~~ 0.9 Wt-%.
(4) Cladding: 304 stainless steel, nominal 0.020 inch thick.
~~(5) Identification: Top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores.~~

~~b. Standard TRIGA fuel~~
~~The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:~~
~~(1) Uranium content: maximum of 9.0 Wt-% enriched to a nominal 20% Uranium 235.~~

Technical Specifications Page 26 (continuing on page TS-27)

~~(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.7 H atoms to 1.0 Zr atoms.~~
~~(3) Cladding: 304 stainless steel, nominal 0.020 inch thick.~~

To read:

a. TRIGA LEU 30/20 Fuel
The individual unirradiated TRIGA LEU 30/20 fuel elements shall have the following characteristics:
(1) Uranium content: maximum of 30 Wt-% enriched to maximum of 19.95 Wt-% with nominal enrichment of 19.75 Wt-% Uranium 235.
(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.
(3) Natural erbium content (homogeneously distributed): nominal 0.9 Wt-%.
(4) Cladding: 304 stainless steel, nominal 0.020 inch thick.

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore, the design features of Standard and FLIP fuel are removed and replaced with the design features of LEU 30/20 fuel since it is the only type of fuel used after the conversion. NUREG-1282 documents the LEU 30/20 fuel design features.

5.0 **DESIGN FEATURES**

5.1 REACTOR FUEL

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

a. TRIGA LEU 30/20 Fuel

The individual unirradiated TRIGA LEU 30/20 fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 30 Wt-% enriched to maximum of 19.95 Wt-% with nominal enrichment of 19.75 Wt-% Uranium 235.

(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.

(3) Natural erbium content (homogeneously distributed): nominal 0.9 Wt-%.

(4) Cladding: 304 stainless steel, nominal 0.020 inch thick.

Technical Specifications Page 27

TS 5.1 bases, which say:

a. A maximum uranium content of 9 Wt-% in a TRIGA-FLIP element is about 6% greater than the design value of 8.5 Wt-%. Such an increase in loading would result in an increase in power density of about 2%. Similarly, a minimum erbium content of 1.1% in an element is about 30% less than the design value. This variation would result in an increase in power density of only about 6%. An increase in local power density of 6% reduces the safety margin by at most ten percent. The maximum hydrogen-to-zirconium ration of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zirconium ration of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

   When standard and FLIP fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings. The accidental rotation of fuel bundles containing standard and FLIP elements can be detected by visual inspection. Should this occur, however, studies of a single FLIP element accidentally rotated into a standard fuel region indicate an insubstantial increase in power generation in the FLIP element.

b. A maximum uranium content of 9 Wt-% in a standard TRIGA element is about 6% greater than the design value of 8.5 Wt-%. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% reduces the safety margin by at most 10%. The maximum hydrogen-to-zirconium ratio of 1.8 will produce a maximum pressure within the clad during an accident well below the rupture strength of the clad.

Are replaced in their entirety to read:

The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

Technical Specifications Page 27, Continued

The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 Wt-%. (However, the quantity of erbium in the full core must not deviate from the design value by more than -3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zorconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad (M.T. Simnad, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," General Atomics Report E-117-833, February, 1980).

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore, the bases of TS 5.1 for the design features of the fuel are revised for the new LEU 30/20 fuel. The effects of the uranium and erbium design limits have already been estimated at Texas A&M using LEU 30/20 fuel. The effects of the hydrogen-to-zirconium design limit has been reported by General Atomics in GA report E-117-833.

Bases

The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 Wt-%. (However, the quantity of erbium in the full core must not deviate from the design value by more than -3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad (M.T. Simnad, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," General Atomics Report E-117-833, February, 1980).

5.2 REACTOR CORE

Applicability

This specification applies to the configuration of fuel and in-core experiments.

Technical Specifications Page 28

TS 5.2(a), which says:

a. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

Is changed as follows:

a. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

To read:

a. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore, the core arrangement is clarified to be exclusively LEU 30/20 fuel to preclude any operation with other TRIGA fuel.

Technical Specifications Page 28, Continued

TS 5.2(b), which says:

b. The Triga core assembly may be standard, FLIP, or a combination, thereof (mixed core) provided that any FLIP fuel be comprised of at least thirty-five (35) fuel elements, located in a contiguous, central region.

And TS 5.2(b) bases, which say:

b. In mixed cores, it is necessary to arrange FLIP elements in a contiguous, central region of the core to control flux peaking and power generation peak values in individual elements.

Are deleted.

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. Therefore, the design specification for mixed cores is removed because only cores using LEU 30/20 fuel are used after conversion.

Technical Specifications Page 28, Continued

TS 5.2(a) bases, which say:

Standard TRIGA cores have been in use for years and their characteristics are well documented. The Puerto Rico Nuclear Center and the Gulf Mark III all-FLIP cores have operated and their characteristics are available. Gulf has also performed a series of experiments using standard and Flip fuel in mixed cores and a mixed core has been used successfully in the Texas A&M University TRIGA reactor. In addition, studies performed at Wisconsin for a variety of mixed core arrangements indicate that such cores with mixed loadings would safely satisfy all operational requirements (SAR Chapters 4 and 6).

Are changed as follows:

~~Standard~~ TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and Texas A&M ~~The Puerto Rico Nuclear Center and the Gulf Mark III all-FLIP cores have operated~~ and their successful operational characteristics are available. ~~Gulf has also performed a series of experiments using standard and Flip fuel in mixed cores and a mixed core has been used successfully in the Texas A&M University TRIGA reactor.~~ In addition, ~~studies~~ the analysis performed at Wisconsin ~~for a variety of mixed core arrangements~~ indicates that ~~such cores with mixed loadings would~~ the LEU 30/20 core will safely satisfy all operational requirements ~~(SAR Chapters 4 and 6)~~. See chapters 4 and 13 of the LEU Conversion Analysis SAR.

To read:

TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and Texas A&M and their successful operational characteristics are available. In addition, the analysis performed at Wisconsin indicates that the LEU 30/20 core will safely satisfy all operational requirements. See chapters 4 and 13 of the LEU Conversion Analysis SAR.

Justification:

The only type of fuel approved for use is TRIGA LEU 30/20 type. The bases are updated to reference other current facilities successfully operating with LEU 30/20 fuel, and to refer to detailed calculations in the LEU Conversion Analysis SAR.

Technical Specifications Page 28, Continued

TS 5.2(c and d), which say:

c. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

d. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

Are changed as follows:

~~c~~b. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

~~d~~c. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

To read:

b. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

c. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

Justification:

The outline numbering is revised because a previous entry referring to mixed cores, which are no longer approved, was deleted. No wording is changed.

TS 5.2(c and d) bases, which say:

c. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

d. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

Are changed as follows:

~~c~~b. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

~~d~~c. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

To read:

b. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

c. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

Justification:

The outline numbering is revised because a previous entry referring to mixed cores, which are no longer approved, was deleted. No wording is changed.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

a. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

b. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

c. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

Bases

a. TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and Texas A&M and their successful operational characteristics are available. In addition, the analysis performed at Wisconsin indicates that the LEU 30/20 core will safely satisfy all operational requirements. See chapters 4 and 13 of the LEU Conversion Analysis SAR.

b. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

c. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.3 Control Elements

Applicability

These specifications apply to the control blades and transient control rod.

Objective

The objective is to assure that control elements are fabricated to reliably perform their intended control and safety function.

a. The safety blades shall be constructed of boral plate and shall have scram capability.

b. The regulating blade shall be constructed of stainless steel.

c. The transient rod shall contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient control rod shall have scram capability and may incorporate an aluminum or air follower.

Bases

The boral safety blades and stainless steel regulating blade used in the reactor have been shown to provide adequate reactivity worth, structural rigidity, and reliability to assure reliable operation and long life under operating conditions. The transient control rod materials and fabrication techniques have been used in many TRIGA reactors and have demonstrated reliable operation and long life.

5.4 Radiation Monitoring Systems

Applicability

These specifications describe the functional performance and essential components of the radiation monitoring systems.

Objective

The objective is to describe those systems which provide information on radiation levels and effluent radioactivity.

Specifications

a. The area radiation monitoring system shall provide gamma radiation level information at the control console for at least three locations in the Laboratory. It shall cause an alarm at the control console and initiation of an evacuation alarm if high radiation levels occur and prompt remedial action is not taken.

Technical Specifications Page 31

TS 5.6(b), which says:

All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 17 meters above ground level.

Is changed as follows:

All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of ~~17~~ 30.5 meters above ground level.

To read:

All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 30.5 meters above ground level.

Justification:

Calculations performed in the conversion analysis are based on a minimum stack exhaust height of 30.5m above ground level. In order to remain within the design basis, the specifications for ventilation stack height are revised to reflect current stack design and to be consistent with the methodology of calculations in the LEU Conversion SAR.

Technical Specifications Page 31, Continued

TS 5.6 bases, which say:

Calculations in Chapter 6 of the Safety Analysis Report show that exposure of occupants of the Laboratory can be kept below 10 CFR part 20 limits for occupational exposure under accident conditions if the room volume is 2,000 m^3. Calculations in Chapter 6 of the SAR based on release of radioactive effluent at ground level show that concentrations of radioactive materials are within limits of 10 CFR Part 20 for non-restricted areas during the accidents considered. Further calculations based on release at the stack height show a further reduction by a factor of 10 due to operation of the ventilation system and release of effluent at a height of 17m.

Are changed as follows:

Calculations in Chapter ~~6~~ 13 of the ~~Safety Analysis Report~~ SAR ~~show that exposure of occupants of the Laboratory can be kept below 10 CFR part 20 limits for occupational exposure under accident conditions~~ demonstrate that the occupational doses in the event of the maximum hypothetical accident do not exceed limits if the ~~room~~ lab volume is at least 2,000 m^3. ~~Calculations in Chapter 6 of the SAR based on release of radioactive effluent at ground level show that concentrations of radioactive materials are within limits of 10 CFR Part 20 for non-restricted areas during the accidents considered. Further calculations based on release at the stack height show a further reduction by a factor of 10 due to operation of the ventilation system and release of effluent at a height of 17m.~~ Furthermore, calculations in chapter 13 that assume operation of the ventilation system assume a stack height of 30.5m.

To read:

Calculations in Chapter 13 of the SAR demonstrate that the occupational doses in the event of the maximum hypothetical accident do not exceed limits if the lab volume is at least 2000m^3. Furthermore, calculations in chapter 13 that assume operation of the ventilation system assume a stack height of 30.5m.

Justification:

Calculations performed in the conversion analysis are based on a minimum stack exhaust height of 30.5m above ground level. In order to remain within the design basis, the specifications for ventilation stack height are revised to reflect current stack design and to be consistent with the methodology of calculations in the conversion analysis. The bases are modified to reference the methodology of calculations in the LEU Conversion SAR.

5.6 Reactor Building

Applicability

These specifications apply to the room housing the reactor and the ventilation system controlling that room.

Objective

The objective is to provide restrictions on release of airborne radioactive materials to the environs.

Specifications

a. The reactor shall be housed in a closed room designed to restrict leakage. the minimum free volume shall be 2,000 cubic meters.

b. All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 30.5 meters above ground level.

Bases

Calculations in Chapter 13 of the SAR demonstrate that the occupational doses in the event of the maximum hypothetical accident do not exceed limits if the lab volume is at least $2000m^3$. Furthermore, calculations in chapter 13 that assume operation of the ventilation system assume a stack height of 30.5m.

5.7 REACTOR POOL WATER SYSTEMS

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Technical Specifications Page 32

The following is added to the end of TS 5.7:

f. A pool water temperature alarm shall indicate if water temperature reaches 130°F.

Furthermore, the following is added to the end of TS 5.7 bases:

f. The thermal-hydraulic analysis in the SAR assumes a pool water temperature of 130°F. If the temperature exceeds 130°F then the alarm will prevent continued operation in an un-analyzed condition.

Justification:

Calculations performed as part of the conversion analysis are based on a maximum core inlet temperature of 130°F. In order to remain within the design basis, a new specification 5.7(f) for pool water temperature is added in response to RAI question 15.

Technical Specifications Page 32, Continued

TS 5.7 bases, which say:

a. This specification is based on thermal and hydraulic calculations which show that the TRIGA-FLIP core can operate in a safe manner at power levels up to 2,700 kW with natural convection flow of the coolant water. A comparison of operation of the TRIGA-FLIP and standard TRIGA Mark III has shown operation to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as that for TRIGA-FLIP and standard cores.

Are replaced in their entirety to read:

a. The LEU Conversion SAR section 4.7.8 shows by analysis that the natural convective cooling of the reactor core is sufficient to maintain the fuel in a safe condition up to at least a power level of 1500 kW (the power Safety Limit).

Justification:

Calculations performed as part of the conversion analysis show a peak fuel temperature of 665°C at 1.5MW under natural circulation conditions. These calculations are based on the proposed specific TRIGA LEU 30/20 core design at the University of Wisconsin and are not based on calculations and measurements of similar TRIGA reactors. Therefore, the bases for natural convection cooling are revised to refer to current calculations for the LEU core in the LEU Conversion SAR.

Specifications

a. The reactor core shall be cooled by natural convective water flow.

b. The pool water inlet and outlet pipe to the demineralizer shall not extend more than 15 feet into the top of the reactor pool when fuel is in the core. The outlet pipe from the demineralizer shall be equipped with a check valve to prevent inadvertent draining of the pool.

c. Diffuser and other auxiliary systems pumps shall be located no more than 15 feet below the top of the reactor pool.

d. All other piping and pneumatic tube systems entering the pool shall have siphon breakers and valves or blind flanges which will prevent draining more than 15 feet of water from the pool.

e. A pool level alarm shall indicate loss of coolant if the pool level drops approximately one foot below normal level.

f. A pool water temperature alarm shall indicate if water temperature reaches 130°F.

Bases

a. The LEU Conversion SAR section 4.7.8 shows by analysis that the natural convective cooling of the reactor core is sufficient to maintain the fuel in a safe condition up to at least a power level of 1500 kW (the power Safety Limit).

b. The inlet pipe to the demineralizer is positioned so that a siphon action will drain less than 15 feet of water. The outlet pipe from the demineralizer penetrates the pool below core level and a check valve prevents leakage from the pool by reverse flow from pipe ruptures or improper operation of the demineralizer valve manifold.

c. In the event of pipe failure and siphoning of pool water, the pool water level will drop no more than 15 feet from the top of the pool.

d. Other pipes which enter the pool have siphon breakers which prevent pool drainage. Valves are provided for pneumatic tube system lines and primary cooling system pipe. Other piping installed in the pool has blind flanges permanently installed.

e. Loss of coolant alarm, after one foot of loss, requires corrective action. This alarm is observed in the reactor control room and outside the reactor building.

f. The thermal-hydraulic analysis in the SAR assumes a pool water temperature of 130°F. If the temperature exceeds 130°F then the alarm will prevent continued operation in an un-analyzed condition.