

# UNIVERSITY of MISSOURI

## RESEARCH REACTOR CENTER

September 3, 2010

U.S. Nuclear Regulatory Commission  
Attention: Document Control Desk  
Mail Station P1-37  
Washington, DC 20555-0001

REFERENCE: Docket 50-186  
University of Missouri – Columbia Research Reactor  
Amended Facility License R-103

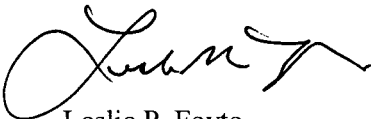
SUBJECT: Written communication as specified by 10 CFR 50.4(b)(1) regarding the response to the “University of Missouri at Columbia – Request for Additional Information Re: License Renewal, Safety Analysis Report, Complex Questions (TAC No. MD3034),” dated May 6, 2010

On August 31, 2006, the University of Missouri-Columbia Research Reactor (MURR) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) to renew Amended Facility Operating License R-103.

On May 6, 2010, the NRC requested additional information and clarification regarding the renewal request in the form of nineteen (19) complex questions. MURR’s responses to seven (7) of those complex questions are attached. As discussed during a conference call on September 1, 2010 with Mr. Alexander Adams, Senior Project Manager, this is just a partial response and a schedule to answer the remaining questions was discussed and is pending approval by the NRC.

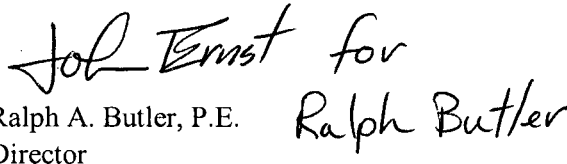
If there are questions regarding this response, please contact me at (573) 882-5276 or [foytol@missouri.edu](mailto:foytol@missouri.edu). I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,



Leslie P. Foyto  
Reactor Manager

ENDORSEMENT:  
Reviewed and Approved,

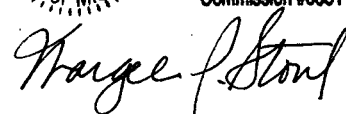


Ralph A. Butler, P.E.  
Director

xc: Reactor Advisory Committee  
Reactor Safety Subcommittee  
Dr. Robert Duncan, Vice Chancellor for Research  
Mr. Alexander Adams, Jr., U.S. NRC  
Mr. Craig Basset, U.S. NRC



MARGEE P. STOUT  
My Commission Expires  
March 24, 2012  
Montgomery County  
Commission #08511436



## CHAPTER 13

### 13.2 Section 13.2.2.1, Accident initiating events and scenarios, Page 13-17.

- a. *From Fig. 13.4 (page 13-22), the DNB limiting power is about 17 megawatt (MW) thermal (t). What is the basis for the “burnout” power of 25.23 MW(t)? What is the corresponding peak fuel temperature?*

The basis for the burnout power level of 25.23 MW(t) can be found in Section 3.2 of Addendum 3 to the MURR Hazards Summary Report (Ref. 13.17). In that section, the steady-state power level at which burnout would occur was calculated. For the analyses, all relevant operating parameters were set at their reactor safety system set point values. Using these values, the burnout heat fluxes and power levels were estimated using various accepted correlations and the correlation providing the lowest value for burnout power was selected.

Figure 13.4 (page 3-22) was generated by an independent consultant [Ref. 4.26 (Vaughan, F.R., “Safety Limit Analysis for the MURR Facility,” NUS Corporation, NUS-TM-EC, May 1973)] using even more restrictive criteria based on tests performed for Advanced Test Reactor (ATR) fuel to set the limiting power values. The limiting power values shown in Figure 13.4 were obtained based on the requirement that the local heat flux at any point in the core shall be less than half of the burnout heat flux given by the Bernath correlation for that point. Hence the discrepancy in the two limiting power values.

Ultimately, safe operation of the reactor is assured by establishing the Limiting Safety System Setting (LSSS) for power based on a value that is even less than the lower of the two burnout power levels calculated by the two different methodologies.

Based on the PARET analysis, the maximum fuel clad surface temperature would be approximately 155 °C (311 °F) for a steady-state power level of about 17.5 MW(t).

### 13.2 Section 13.2.2.1, Accident initiating events and scenarios, Page 13-17.

- b. *The SAR states it was assumed that the MURR core could withstand the prompt power burst associated with the rapid step insertion of positive reactivity. Please provide the bases for this assumption. Please discuss the results of the analysis if parameter initial conditions are at TS or license limits. Please discuss the results for reactor operation in Mode II or III.*

As a general principle during fast reactor transients, a reactor can handle power bursts that are much higher than the steady-state burnout power level, since these power bursts last only for a few tenths of a second. It is only during steady-state operation at power levels above the safety limit curves that melting of the fuel plate may occur. In a reactivity transient, the product of power level and time must yield enough total energy to reach the melting temperature of a fuel plate. Such reactor behavior has been conclusively shown by hundreds of power excursion tests performed at the Special Power Excursion Reactor Test (SPERT) facility at the National Reactor Testing Station in Idaho (Ref. “Experimental Study of Transient Behavior in a Sub-cooled, Water-moderated Reactor,” F. Schroeder et. al.).

The fuel plate surface temperature during such fast transients lags the power trace due to fuel plate heat capacity and the time required for heat transfer from the fuel meat to the plate surface. SPERT tests have shown that the maximum fuel plate surface temperature remains below the saturation temperature as long as the power burst lasts only a few tenths of a second and is

reduced either by the inherent core shut down mechanisms or, as a backup, by the reactor safety system once the short period or the high power limit is exceeded.

For analyzing the reactivity transients for the MURR, the values for the relevant operating parameters were selected in order to obtain conservative results. As an example, the core inlet temperature for the primary coolant was assumed to be 130 °F (54.4 °C). If the Technical Specification (TS) limit of 155 °F (68.3 °C) were selected, the power peak would be lower. The reason for this is that during fast transients the initial power burst is controlled by the inherent self-limiting characteristics of the reactor. These inherent characteristics primarily depend on the fuel meat temperature and, if a higher starting temperature is assumed, the self-limiting characteristics of the reactor will provide the negative reactivity feedback to reduce the peak power earlier than the case where a lower starting temperature is assumed.

A similar argument can be made for assuming the TS value for primary coolant flow rate. The TS value is lower than the assumed value of 3600 gpm (13,627 lpm). A reduction in coolant flow rate will result in quicker negative reactivity feedback due to the higher fuel plate temperature compared to the case analyzed.

These predictions have been confirmed with the help of additional PARET runs where, in the case of a positive 0.006  $\Delta k/k$  step reactivity insertion, the core inlet temperature was increased from 130 °F (54.4 °C) to 155 °F (68.3 °C) and the total core flow rate was reduced from 3,600 gpm (13,627 lpm) to 3,250 gpm (12,303 lpm). In both of these selected cases, the peak of the power burst was reduced from the reported value of 33.19 MW to 32.17 MW and 32.83 MW, respectively.

Reactivity transients during Mode II and Mode III operation should result in lower peak power bursts since the starting power levels are lower than 10 MW. To confirm this, a PARET run was performed for the limiting positive reactivity insertion for MURR, viz., 0.006  $\Delta k/k$  step in Mode II operation. Primary coolant flow rate and core inlet temperature were set at the TS values of 1,625 gpm (6,151 lpm) and 155 °F (68.3 °C), respectively. For a starting power of 5 MW, the peak power obtained following a positive step reactivity insertion of 0.006  $\Delta k/k$  was 17.8 MW and the peak cladding temperature was approximately 163 °C (325 °F). Similar to the Mode I case, this transient also will be terminated by the reactor safety system once the high power scram set point of 6.25 MW or the short period scram set point of 8.0 seconds are exceeded, thus preventing steady-state operation beyond the burnout power limit.

13.2 Section 13.2.2.1, Accident initiating events and scenarios, Page 13-17.

*c. The SAR discusses other step insertions (i.e., up to 0.003  $\Delta k/k$ ). Please discuss the results of the analysis if parameter initial conditions are at TS or license limits. Please discuss the results for reactor operation in Mode II or III.*

As discussed in the answer to question 13.2.b, if the Technical Specification (TS) or license limit values had been used for the parameter initial conditions, the peak power attained during these smaller step reactivity insertions would also have been lower. As stated in the answer to the previous question, the reason for this is that during fast transients, the initial power burst is contained by the inherent self-limiting characteristics of the reactor. These inherent characteristics primarily depend on the fuel meat temperature and whether a higher starting temperature for the fuel meat is assumed. The self-limiting characteristics will start providing the negative reactivity feedback earlier than the case where a lower starting temperature is assumed.

Additionally, the power peak attained during Mode II and Mode III reactivity transients using TS or license limit values would be smaller than the value reached during a Mode I transient.

13.4 Section 13.2.3, Loss of Primary Coolant.

- c. *It is not clear from the SAR which version of RELAP is used for the loss of primary coolant calculations. However, 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 loss of coolant analysis have the fixes implemented? If not, confirm that the analysis model is giving results consistent with the accident, e.g. by checking results as a function of time step or with another stand-alone point kinetics model.*

In the Extended Life Aluminide Fuel (ELAF) analysis, MURR benchmarked both PARET and RELAP5 reactivity transient analyses against one of the SPERT-IV tests (Special Power Excursion Reactor Test). The -12/25 core of the SPERT-IV tests was selected for comparison because the test was performed under conditions very similar to that of MURR - forced coolant circulation in a low pressure and low temperature environment using plate-type fuel. The comparisons of the power and temperature transients for a \$1.14 step reactivity insertion were modeled. The PARET code provided a response to the reactivity transient which was conservative but close to the measured values, while RELAP5 significantly over-predicted the fuel temperatures. The results were presented at the American Nuclear Society (ANS) 1989 Annual Meeting in a paper titled "PARET/ANL and RELAP5/MOD2 Benchmarking Comparison Against the SPERT-IV Test Data" by S. S. Kim and J. C. McKibben (University of Missouri Research Reactor Facility, June 1989). Based on this analysis, MURR has only used PARET for reactivity transient analyses.

The MURR RELAP analysis in the SAR used SCDAP RELAP5 MOD3.3, as noted in Section C.1, Introduction. The use of RELAP5 in the Loss of Flow Accident (LOFA) and Loss of Coolant Accident (LOCA) analyses does not include any point kinetics analysis. RELAP is only used to model the thermal-hydraulic transient. The MURR RELAP5 models handled the reactor decay power as a function of time after a SCRAM is actuated. The modeling is based, conservatively, on a longer operating time than what occurs in the MURR core. This provides a higher than actual decay heat generation.

13.9 Section 13.2.6, Experiment Malfunction.

- b. *Please provide an example of a calculation of an irradiation container that meets the requirement for containing the pressure by at least a factor of two from the detonation of 25 milligrams of TNT-equivalent material.*

The detonation of TNT-equivalent material will release an energy equivalent of 4184 joules (~1000 calories) per gram of material. In the case of 25 milligrams, this would yield 104.6 joules (4184 j x 0.025 g) of energy. The pressure created in a confined space due to this energy release can be calculated using the Ideal Gas Equation  $PV = nRT$ , where:

- P = Pressure (atmospheres)  
V = Volume (cm<sup>3</sup>)  
n = Number of moles gas  
R = Universal Gas Constant = 0.0821 ( $\frac{1 \cdot \text{atm}}{\text{mol} \cdot ^\circ\text{K}}$ )  
T = Temperature (°K)

in this case:  $PV = \text{energy (joules)}^*$

$$\text{From } PV = nRT \quad T = \frac{PV}{nR} \quad \text{therefore by substitution } T = \frac{\text{energy (j)}}{nR}$$

$$\text{From } PV = nRT \quad P = \frac{nRT}{V} \quad \text{therefore by substitution } P = \frac{nR(\frac{j}{nR})}{V} = \frac{j}{V}$$

$\frac{j}{V}$  can be converted to atmospheres by  $j = 9.87 \text{ cm}^3 \cdot \text{atm}^{**}$

\* Pressure (P) is force per unit area (newtons/m<sup>2</sup>) and volume (V) is m<sup>3</sup>, so, PV simplifies to newtons · meters = joules.

$$\begin{aligned} ** \quad 1 \text{ liter} \cdot \text{atm} &= 101.325 \text{ j} \\ 1 \text{ cm}^3 \cdot \text{atm} &= 0.101325 \text{ j} \\ j &= 1 \text{ cm}^3 \cdot \text{atm} / 0.101325 \\ j &= 9.87 \text{ cm}^3 \cdot \text{atm} \end{aligned}$$

A typical standard container that might be used at MURR to encapsulate 25 mg of TNT-equivalent material would be a 4-inch tall thin-walled cylinder manufactured of aluminum alloy 1100 with an inner diameter of 1-inch, a wall thickness of 0.025-inches, and an internal volume of approximately 51.5 cm<sup>3</sup>.

Using the above formula  $P = \frac{j}{V}$ , the pressure created by the detonation of TNT-equivalent material in the confined volume of 51.5 cm<sup>3</sup> can be calculated as:

$$P = \frac{104.6 \text{ j}}{51.5 \text{ cm}^3} \times \frac{9.87 \text{ cm}^3 \cdot \text{atm}}{j}$$

$P = 20.05 \text{ atm}$ , and converted to psi would be:

$$P = 20.05 \text{ atm} \times 14.7 \frac{\text{psi}}{\text{atm}} = 295 \text{ psi}$$

Additional pressure due to the compression of the air volume not occupied by the solid TNT-equivalent material would be negligible.

In order to confine the pressurization due to the detonation of 25 mg of TNT, the stress limit of the confining material cannot be exceeded. As stated earlier, a thin-walled aluminum cylinder of alloy 1100 would be a preferred encapsulation. The yield strength of aluminum alloy 1100 = 15.6 ksi (1 ksi = 1000 psi); therefore  $\sigma_{\text{yield}} = 15,600 \text{ psi}$ .

The hoop stress limit in a cylindrical container with thin walls is represented by one-half the pressure times the ratio of the capsule diameter to wall thickness, or:

$$\sigma = \frac{pd}{2t}$$

where:

- $\sigma$  = maximum wall stress
- $p$  = total pressure in the container
- $d$  = inner diameter of container
- $t$  = wall thickness

In order to determine if a particular container could confine the expected pressure, the maximum stress in the container wall would need to be less than or equal to the yield strength of the material, or:

$$\frac{pd}{2t} \leq \sigma_{\text{yield}}$$

In order to determine if a particular container would meet MURR's more conservative specification, the maximum stress in the container wall would need to be a *factor of two* less than the yield strength of the material, or:

$$\frac{pd}{2t} \leq \frac{\sigma_{\text{yield}}}{2}$$

Solving this equation for  $\frac{d}{t}$  provides a method for evaluating an encapsulation material using this diameter to thickness ratio:

$$\frac{pd}{2t} \leq \frac{\sigma_{\text{yield}}}{2}$$

$$\frac{d}{t} \leq \frac{1}{p} \sigma_{\text{yield}}$$

then:

$$\frac{d}{t} \leq \frac{\sigma_{\text{yield}}}{p}$$

If you were to confine the 295 psi calculated pressure in a 51.5 cm<sup>3</sup> volume using aluminum alloy 1100, the maximum diameter to thickness ratio for the cylinder at this pressure using the *factor of two* model would be:

$$\frac{\sigma_{\text{yield}}}{p} = \frac{15600 \text{ psi}}{295 \text{ psi}} = 53$$

As described above, the typical standard container, which is a 4-inch tall thin-walled cylinder manufactured of aluminum alloy 1100 with an inner diameter of 1-inch and a wall thickness of 0.025-inches, the diameter to thickness ratio would be:

$$\frac{d}{t} = \frac{1 \text{ inch}}{0.025 \text{ inches}} = 40$$

Therefore, this would be an acceptable encapsulation to contain the potential pressurization by a *factor of two* as required by Technical Specification 3.6.i.

## CHAPTER 16

### 16.1 Section 16.1.1, Fuel and Fuel Cladding, TS 3.8, Reactor Fuel, and TS 4.5, Reactor Fuel.

- b. *TS 4.5 requires one of eight fuel elements that have reached their end-of life to be inspected. However, SAR 16.1.1 states that fuel elements are visually inspected during each refueling. Explain this difference between the SAR and proposed TSs. How can unacceptable dimensional changes be found while a fuel element is in service if fuel elements are only inspected at end-of-life?*

The in-service inspection is not intended to locate unacceptable dimensional changes. The in-service and end-of-life inspections have different goals; the in-service inspection ensures that no major changes have occurred to the element, while the end-of-life inspection provides more detailed data useful in trending quality of fuel element fabrication. MURR has long-term experience with monitoring fuel element quality through end-of-life inspections and, based on this experience, a high level of confidence in the current fuel supply.

The visual inspection performed during each refueling, as mentioned in Section 16.1.1, Fuel and Fuel Cladding, is a gross visual inspection to verify that each fuel element removed from the core and each fuel element placed into the core maintains its overall physical integrity. This inspection would reveal major deformation, discoloration, scratches or other anomalies, as well as reveal dimensional changes that become apparent through handling, storage, core placement and height checks. It is the combination of those surveillances mentioned in the 2<sup>nd</sup> paragraph of Section 16.1.1 that provides assurance of proper fuel performance.

The end-of-life inspection is performed with much greater scrutiny and includes detailed dimensional verification, including checking the coolant channel gaps. Due to the logistical nature of a more comprehensive inspection, personnel exposure is reduced by performing this inspection after the end-of-life element has had a sufficient time to decay.

MURR's long-term experience with end-of-life inspections indicates that the fuel supply is of a very high quality. MURR's end-of-life inspections are regulated by the current TS 5.5.a. Proposed TS 4.5.a is unchanged from existing TS 5.5.a. Amendment No. 20 to Facility Operating License No. R-103, issued by the NRC on August 1, 1990, authorized the use of Extended Life Aluminide Fuel (ELAF) in the reactor core. Part of this amendment was the approval of the surveillance requirement for MURR fuel. This surveillance frequency is supported by the demonstrated excellent performance of aluminide UAl<sub>x</sub> fuels, specifically for the past thirty-nine years in test and research reactors such as the Advanced Test Reactor (ATR) and MURR. MURR has used over 800 aluminide UAl<sub>x</sub> fuel elements since August 1971 with no failures. Due to the significantly higher power densities and fuel plate temperatures, the excellent performance of ATR aluminide UAl<sub>x</sub> fuel is even a more extreme test of fuel element quality.

## APPENDIX C

- C.2 *Section C.2.1, RELAP5 Application. 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 thermal-hydraulic analysis have the fixes implemented? If not, confirm that the 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.*

See answer to question 13.4.c.