Department of the Interior
US Geological Survey
PO Box 25046 MS 974
Denver, CO 80225-0046

April 27, 2012

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: U.S. Geological Survey TRIGA Reactor (GSTR), Docket 50-274, License R-113
Request for Additional Information (RAI) dated September 29, 2010

Subject: Response to Question 16 of the Referenced RAI

Mr. Wertz:

16.1 GSTR SAR Table 13.7 provides maximum reactivity insertion and related quantities for 12 wt% fuel. The table cites a peak fuel temperature of 1100 °C but then cites reactivity at 1000 °C. Please explain this table and describe how the calculated values were obtained is sufficient detail to allow confirmatory analysis.

Response: GSTR SAR Section 13.2.2.2.1 will be updated throughout to reflect performing calculations utilizing a fuel temperature limit of 1150 °C as per NUREG 1537 Appendix 14.1. Table 13.7 will be changed to reflect a maximum reactivity insertion of $3.83 (which is the reactivity insertion yielding a peak temperature of 1150 °C) and a core average temperature of 314 °C. These two parameters will be updated throughout GSTR SAR Chapter 13. In addition, there was an error in the description for the input value for α in Equation 13.8. The description will be changed to read "α = the prompt negative temperature coefficient of the core." This correction should allow you to perform confirmatory calculations for a peak fuel temperature limit of 1150 °C.

16.2 GSTR SAR Table 13.7 provides maximum reactivity insertion and related quantities for 12 wt% fuel but not for the 8 wt% or 8.5 wt% fuel in the GSTR inventory. Please provide this information for the 8 and 8.5 wt% fuel.

Response: The information for 8 and 8.5 wt% fuel is unnecessary to analyze excess reactivity events. The maximum reactivity insertion for 12 wt% fuel provides the worst case scenario or bounding condition for the core. In addition, detailed descriptions of the fuel are given in section 4.5 of the SAR.

A020
NRR

16.3 GSTR SAR Subsection 13.2.2.2.1 states that "the maximum peak temperature measured for 12/20 fuel element was 345 °C from a $2.31 pulse. The theory presented predicts this pulse would have a peak fuel temperature of 514 °C. Please provide additional text explaining how the measured value was determined and show the assumptions and parameters used in calculating the 514 °C temperature.

Response: The measured temperature value of 345 °C from a $2.31 pulse was measured using a standard 12/20 instrumented fuel element. The actual calculated peak fuel temperature is 592 °C, and the GSTR SAR will be changed to reflect this correction. The assumptions and parameters for calculating the temperature of 592 °C are the same as for calculating the maximum reactivity insertion and core average temperature, i.e. Equations 13.6 and 13.8, using the same correction for the description for the input value for α explained in our response to RAI Question 16.1 (see above).

16.3 cont:

GSTR SAR Subsection 13.2.2.2.2 provides information regarding analysis of the rod withdrawal accident. The analysis assumes an initial power of 100 watts and describes moving all 4 rods together until the reactor power reaches the trip setpoint. NUREG-1537, Section 4.5.3, "Operating Limits" requests that only the maximum worth rod is withdrawn in the most reactive region Please provide an analysis showing the limiting initiating power, the position of the non-moving rods during the analysis, and their response to the accident.

Response: The following is a direct quote from GSTR SAR Subsection 13.2.2.2.2: "Operator error or failure of the automatic power level control system could cause one of the control rods to be driven out, starting either at low power or high power levels. The most reactive rod is the regulating rod and this rod also has one of the fastest withdrawal times. This rod was considered in this accident." The initiating power at 100 W is the limiting power and the positions of the non-moving rods would be near mid-core, but their exact positions have no bearing on the minimal accident consequence. The non-moving rods respond to the event by scramming (gravity falling) to their "down" positions.

The neutronic/thermal hydraulic analyses update is provided as attachment 1 to this document.

Sincerely,

Tim DeBey
USGS Reactor Supervisor

I declare under penalty of perjury that the foregoing is true and correct. Executed on 4/27/12

Attachment

Copies to:

Betty Adrian, Reactor Administrator, MS 975
USGS Reactor Operations Committee

Safety Analysis Support for the USGS Research Reactor
Battelle Energy Alliance Award #00114253 (CSM # 400305)
March 2012 Progress Report – Revision 1
Nicolas Shugart and Jeffrey King

GSTR Reactivity Coefficients

The Geological Survey TRIGA Reactor (GSTR) is currently seeking a renewal of its operating license from the United States Nuclear Regulatory Commission (NRC). As part of the re-licensing procedure, the NRC requested that the facility re-calculate the reactivity coefficients for the new proposed GSTR limiting core. This report describes the analysis involved in calculating the GSTR reactivity coefficients and gives the preliminary results for the GSTR limiting core.

1. Description of the GSTR Limiting Core

Analysis of the limiting core configuration is key to the GSTR relicensing evaluation. A limiting core represents the most compact critical assembly available to the operators, and usually consists completely of fresh fuel. It is unlikely that the GSTR will be able to acquire a full core of fresh fuel in the future. Thus, the limiting core consists of the current fuel with a number of new 12 wt% fuel rods replacing some of the oldest rods in the core.

The limiting core provides a limiting case for the operating conditions of the reactor. The limiting core must safely operate under federal guidelines and the GSTR's technical specifications, and will provide both the regulators and the operators an upper bound on the acceptable operating conditions for the core under the new license. These include the limits on the core's excess reactivity, shutdown reactivity, and transient rod worth's.

The limiting core for the GSTR is based on the criteria provided by the GSTR staff. This core represents the most compact core with the greatest fuel density that could be conceivably loaded in to the GSTR in the future. In the limiting core, the B and C rings include fresh 12 wt% fuel, while the current 12 wt% fuel for the GSTR is placed in the D ring. The E and F rings match are fully loaded with the fuel currently in the GSTR core. In order to meet the reactivity limits, the outermost ring (G) was emptied of fuel, and these fuel locations contain only water. Eight fuel elements have been removed from the F ring in a symmetric manner. This core, while compact, still meets the GSTR Technical Specifications requirements for shutdown, excess, and transient rod worth.

2. Neutronics Analysis Methodology

The neutronics analysis of the GSTR involved calculating the fuel temperature, moderator temperature, and void coefficients of reactivity for the GSTR limiting core. In every case, the model calculates the coefficients for two conditions: the bulk of the reactor components at room temperature (293.6 K) and the bulk of the reactor at 600K to provide bounding cases for the analysis. At full power, the GSTR core will operate between these two temperatures, so this bounds conditions between the zero-power and full-power cases.

In the case of the temperature reactivity coefficients, the MCNP model specifies the temperature of a material through four variables: the specified neutron cross-section library, the specified S(α, β) data library, the specified thermal scattering (TMP) card, and the specified material density. Table 1 described these changes for each case considered in this analysis.

In each case MCNP predicted an effective multiplication factor (k_{eff}) for that situation (see Table 1) providing the basis for each reactivity analysis. The plots of k_{eff} as a function of the dependent variable in the simulation (temperature of void fraction) tend to behave linearly. The derivative of each line gives an average, constant reactivity coefficient for that variable in terms of \$/ K or \$/% void. The effective delayed neutron fraction for this analysis is an assumed value of 0.007.

To provide more detail, the linear derivative between each pair of data points is calculated and plotted individually. This provides a piecewise expression for the reactivity coefficient between each of the calculated data points. Since previous work indicates these coefficients are not constant over the ranges examined, this method provides a more accurate picture of the reactor's behavior (General Atomics, 1967).

Table 1. Parameters examined for each reactivity coefficient

Reactivity Coefficient	Temperature Range (K)	Fuel Temperature	Core Water Temperature	Fuel Density	Core Water Density	Other Component Temperature (K)	Other Component Density
fuel temperature	293.6-1200	corrected to temperature	unchanged	unchanged	unchanged	293.6 or 600	unchanged
core water temperature	293.6-380	unchanged	corrected to temperature	unchanged	corrected to temperature	293.6 or 600	unchanged
void	unchanged	unchanged	unchanged	unchanged	altered to represent voids	293.6 or 600	unchanged

All recorded runs used MCNP5 version 1.60 with 1000 active cycles including 100,000 neutrons per cycle unless otherwise noted. The average 1-σ uncertainty for all the runs was ~0.0008. This uncertainty is too small to yield meaningful error-bars in most figures when plotted the errors represent the 3-σ (~99%) uncertainty bounds for that data.

2.1. Fuel Temperature Reactivity Coefficient

Temperature dependent reactivity coefficients can be calculated by varying the necessary parameters in the model across a range of operating temperatures. For the fuel this comes in three parts related to each fuel rod: the specified cross-section data, the specified $S(\alpha,\beta)$ data. and the specified TMP card value. The makxsf utility included in MCNP5 is capable of doppler broadening the neutron cross-section libraries to a requested temperature, as well as linearly interpolating the $S(\alpha,\beta)$ data between available temperatures (Brown, 2006).

Adjusting the cross-section, $S(\alpha,\beta)$, and TMP card data alters the input deck to represent new temperatures. Running a series of MCNP model with identical physical geometries but different temperature data results in a plot of reactivity as a function of temperature (see Table 1).

2.2. Core Water Reactivity Coefficient

Calculating the moderator temperature reactivity coefficient involves altering the density and temperature of the water within the core of the reactor (where it serves as a neutron moderator as well as the primary reactor coolant). As density is coupled with temperature, both variables require adjustment over a number of operating temperatures. The makxsf program can alter the cross-section data to represent temperature changes in the water within the core, while density is a input value in the MCNP model. The remaining parameters of the reactor are held constant. The MCNP model predicts reactivity at each moderator temperature. Plotting this data and then calculating the derivative yields the moderator temperature reactivity coefficient.

2.3. Void Reactivity Coefficient

The reactor void reactivity coefficient represents the effect that voids within the reactor core will have on the total reactivity of the reactor. The MCNP model can approximate voids in two ways. In the first method, the in-core regions currently filled with water can be replaced with voids (air), and a relationship between the void/water ratio and reactivity can be calculated. In the second, simpler, method, the density of the water can be artificially decreased at a constant temperature to represent voids (steam bubbles) forming within the coolant. In this case, the given relationship is in terms of decreasing density.

The second method will provide the required feedback coefficient, while the first method will serve as a validation case for the model. A numerical fit of the reactivity versus coolant density curve yields the void reactivity coefficient.

3. Results

The reactivity analysis reveals several general trends.. The higher temperature calculations (600 K) consistently resulted in more positive reactivity coefficients than the room temperature (293.5 K) cases.

3.1. Fuel Temperature Reactivity Coefficient

Figure 1 displays multiplication factor on the limiting core as a function of fuel temperature. The clear negative trend displays the prompt negative temperature feedback inherent in TRIGA fuel (Simnad, 1980). From a simple linear fit, the average calculated reactivity coefficient over the examined temperature range is -$0.014/K for the 300 K case, and -$0.013/K for the 600 K case.

Figure 2 presents a more detailed examination, looking at the change between each set of data points. This data matches well with the published numbers from General Atomics (General Atomics, 1967), with a peak in the reactivity coefficient at around 500K and a general increase in the reactivity factor as the temperature increases.

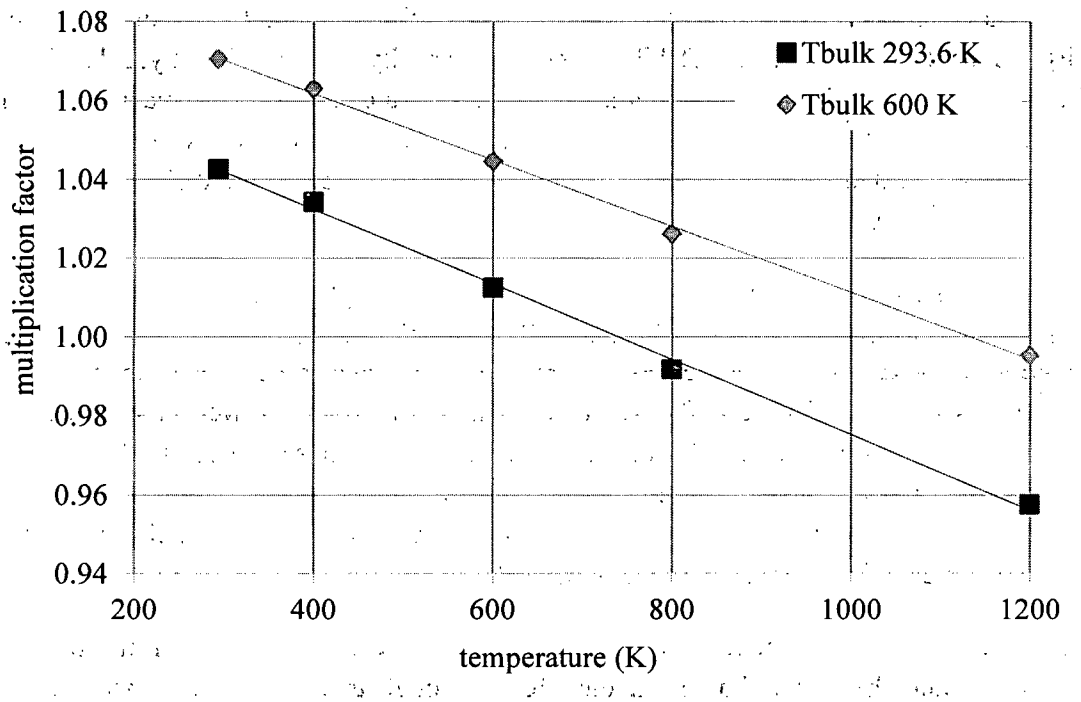

Figure 1. MCNP predicted multiplication factor as a function of the fuel temperature.

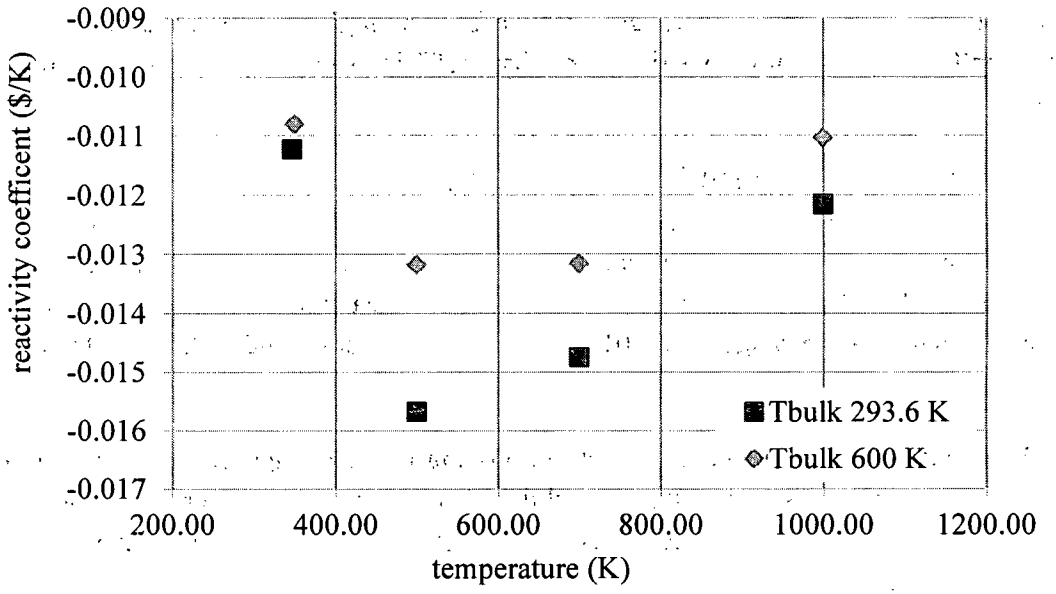

Figure 2. Calculated fuel temperature reactivity coefficient as a function of temperature.

3.2. Core Water Reactivity Coefficient

Figure 3 details the effect of the water temperature within the GSTR core on the reactor's reactivity. Since the moderator in a TRIGA reactor is primarily the zirconium hydride within the fuel, the water is a less significant moderator. The core water temperature reactivity analysis indicates that the GSTR has a positive core water temperature reactivity coefficient. Previous research has documented this effect, but it has not been extensively studied (Zagar, Ravnik, & Trkov, 2002) (Saftey Analysis Working Group, 2009).

To further examine this effect, Figure 4 breaks down the multiplication factor for the 293.6 K case as a function of variables altered in the MCNP model. The $S(\alpha,\beta)$ data seems to be the primary contributor to the increase in core reactivity as a function of core water temperature within the MCNP model of the GSTR. The exact reason for this is not well understood, as the total reactor temperature feedback is strongly negative as a consequence of the significant negative feedback from the fuel (which provides the majority of the moderation). The data in Figure 4 is based on 50,000 neutrons per cycle to increase run speed. This increased the uncertainty in the calculation to a $1\text{-}\sigma$ value of 0.00033.

Figure 5 presents the core water temperature reactivity coefficient as a function of temperature. Taking into account the uncertainty from the MCNP model, there does not appear to be any significant change to the coefficient over the observed temperature range. The average coefficient for the 293.6 K case is \$0.011/K while the coefficient for the 600 K case is \$0.015/K (based on a linear fit in both cases).

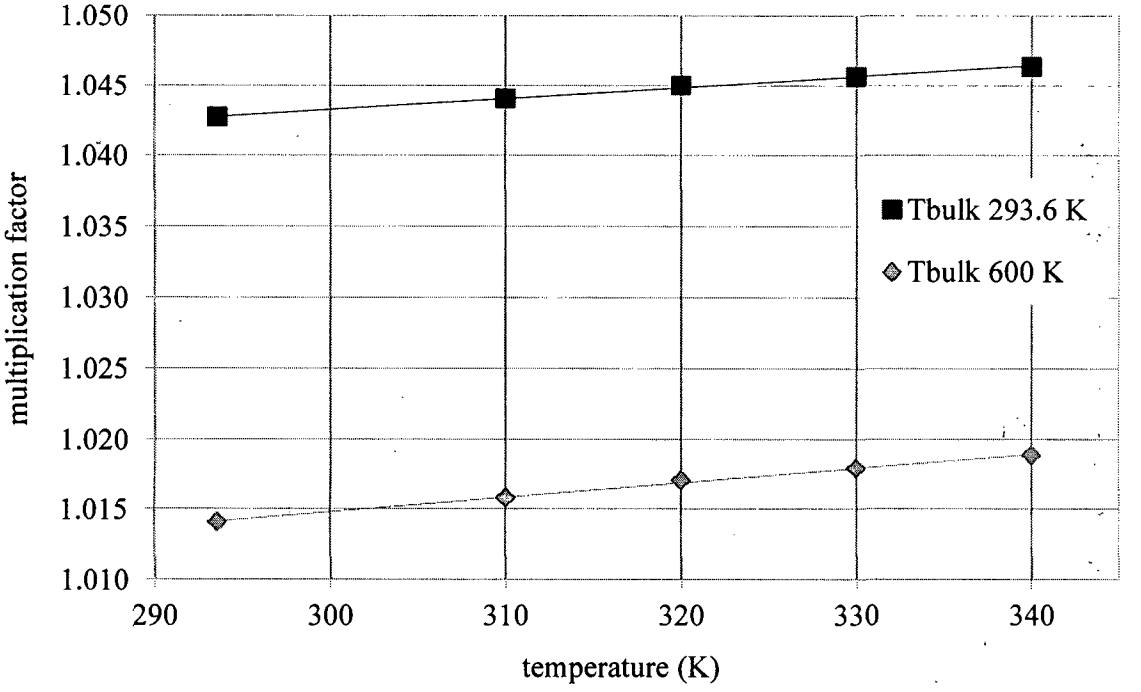

Figure 3. MCNP predicted multiplication factor as a function of core water temperature.

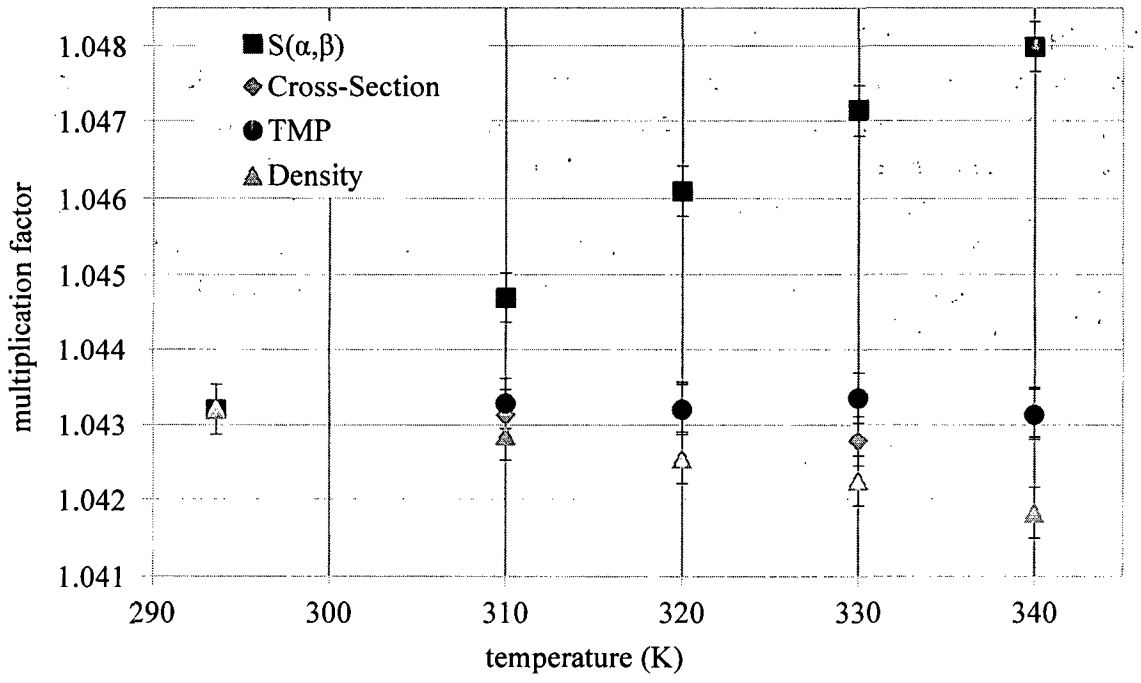

Figure 4. Breakdown of the factors contributing to the core water temperature reactivity coefficient.

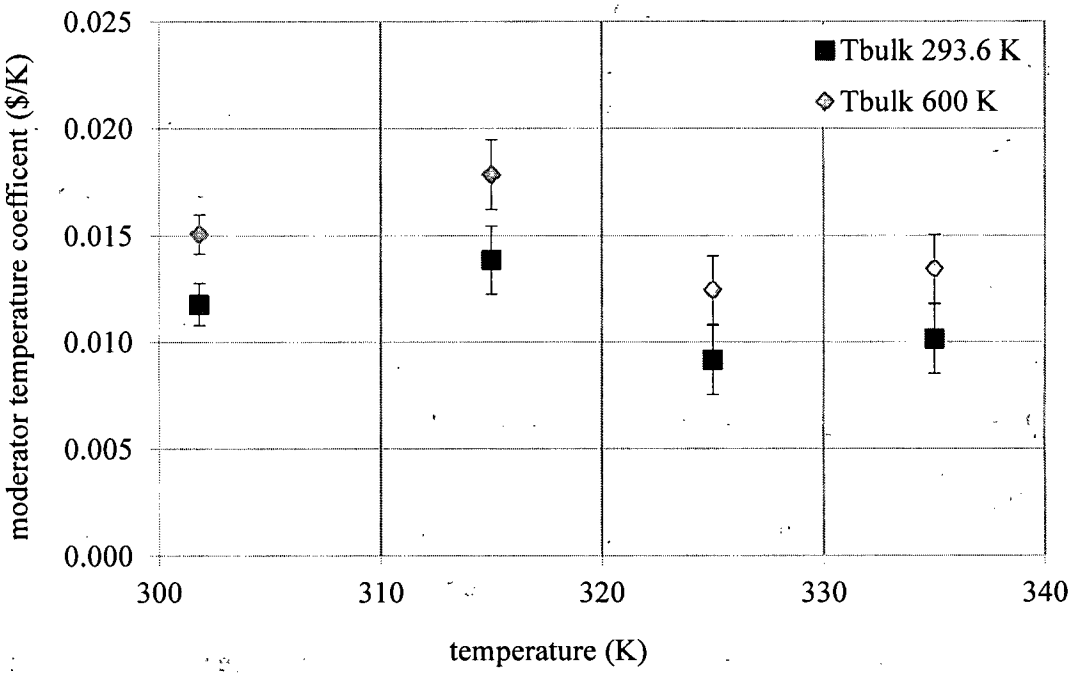

Figure 5. Calculated core water temperature reactivity coefficient as a function of temperature.

3.3. Void Reactivity Coefficient

Figure 6 shows the multiplication factor within the core as a function of coolant void fraction. As expected from the analysis of the core water temperature coefficient (Figure 4), the void coefficient is highly negative. Figure 7 shows the calculated of the void reactivity coefficients. While a linear fit appears valid over the range of 0-20% void fraction, there is a discernible non-linear trend to both datasets above 20%. Assuming a linear trend between 0 and 20% void fraction, the average void reactivity coefficient is -$0.10/% void fraction at 293.6 K and -$0.086/ % void fraction at 600 K.

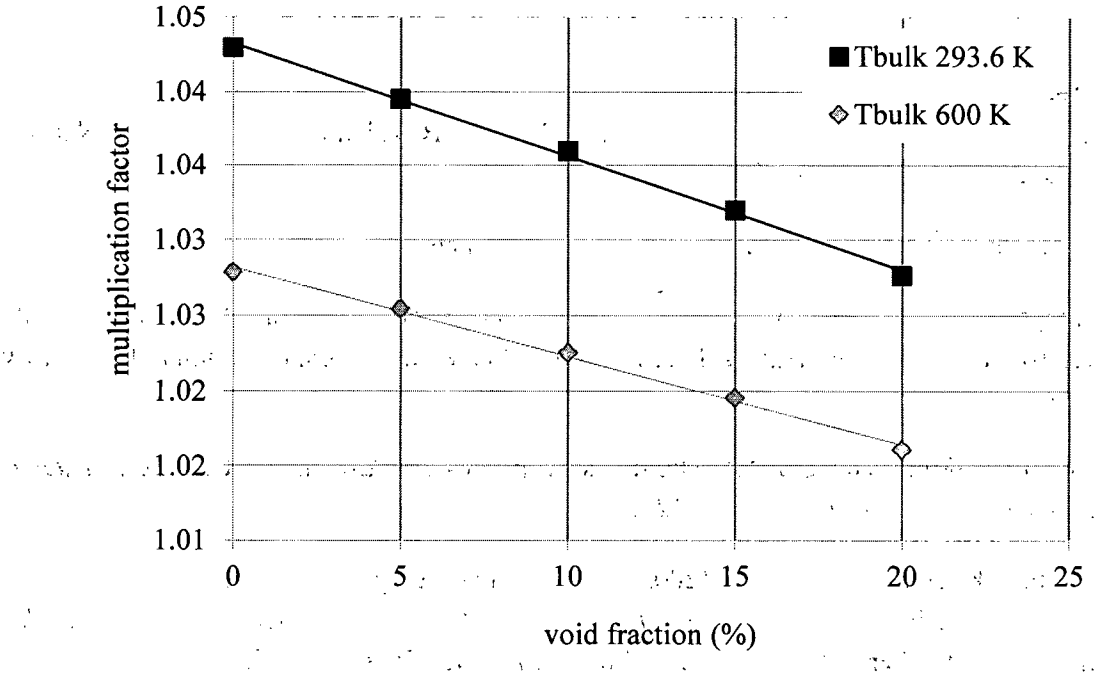

Figure 6. Multiplication factor as a function of core void fraction

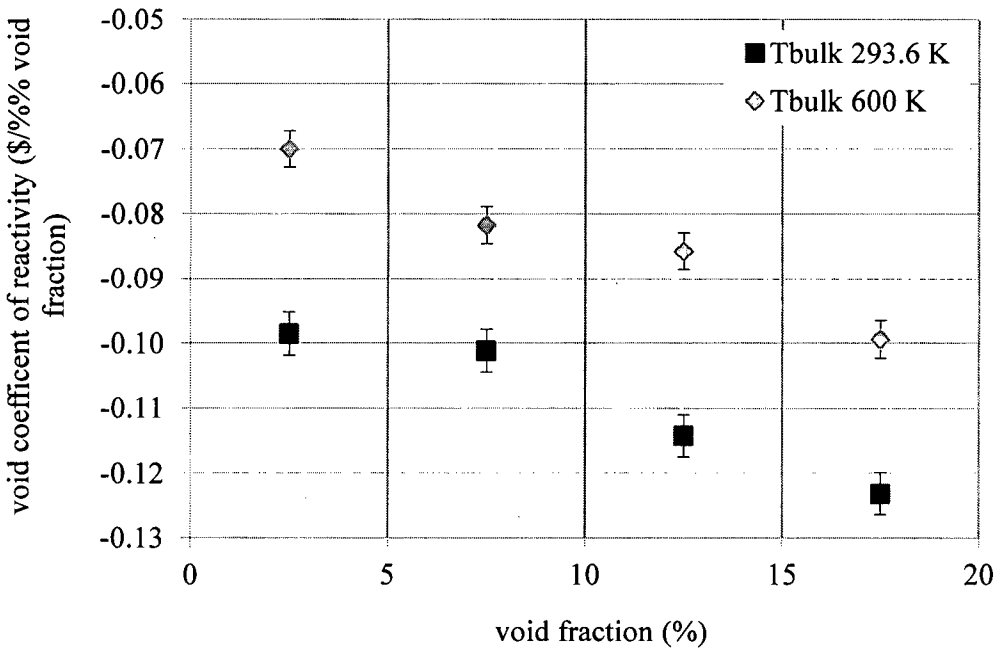

Figure 7. Void reactivity coefficient as a function of core void fraction

Works Cited

Brown, F. B. (2006). "The makxsf Code with Doppler Broadenin," Los Alamos National Laboratory, LA-UR-06-7002

General Atomics. (1967). "Kinetic Behavior of TRIGA Reactors," GA-7882.

Saftey Analysis Working Group. (2009). "Documented Saftey Analysis Addendum for the Neutron Radiography Reactor Facility Core Conversion," Idaho National Laboratory, INL/CON-09-15854.

Simnad, M. T. (1980). "The U-ZRH Alloy: It's properties and use in TRIGA fuel," *Nuclear Engineering and Design, 64*, 402-422.

Zagar, T., Ravnik, M., & Trkov, A. (2002). "Isothermal Temperature Reactivity Coefficient Measurement in TRIGA Reactor," *International Conference Nuclear Energy for New Europe,* September 9-12, pp. 0302.1-0302.6, Kranjska Gora: Slovenia.