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US Geological Survey
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January 22, 2016

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 10, 2015.

Subject: Responses to RAI questions 1a, 1b, and 1c.

Mr. Wertz:

Responses to RAI questions 1a, 1b, and 1c are provided in the enclosed pages. Please contact me if further details, or corrections, are needed.

Sincerely,

Tim DeBey

Tim DeBey

USGS Reactor Supervisor

**I declare under penalty of perjury that the foregoing is true and correct.**
**Executed on 1/22/2016**

Attachment

Copy to:
Vito Nuccio, Reactor Administrator, MS 911
USGS Reactor Operations Committee

A020
NRR

RAI dated 9/10/2015

Question:

1. The NRC staff performed confirmatory calculations based on the GSTR thermal-hydraulic analysis (T-HA), provided by letters dated May 17, and October 31, 2013 (ADAMS Accession Nos. ML13162A662, and ML13311A047 (redacted version), respectively). The NRC staff's confirmatory calculations indicated that the limiting core configuration (LCC) provided did not result in stable power and flow conditions. As such, a determination of the departure from nucleate boiling ratio (DNBR) could not be established. The NRC staff requests that USGS provide a T-HA for their proposed LCC that results in stable power and flow conditions (stability) with a DNBR of 2.0 or greater. The NRC staff has identified the following items that USGS should consider in the analysis:

    a. The outer dimension (OD) of the fuel element diameter used in the LCC was 1.47 inches (in.). Information from the fuel vendor (General Atomics) indicates that the fuel OD may be as large as 1.478 in. Verify the correct OD for the TRIGA fuel elements used at GSTR and adjust the hydraulic diameters or other affected parameters from the USGS LCC T-HA, if necessary.

Response: The 1.47" OD of the fuel elements was taken from both the USGS TRIGA Mk I Mechanical Maintenance and Operating Manual and the Oregon State University (OSU) TRIGA relicensing documentation. In fact, the NRC accepted the OSU Safety Analysis Report that lists a value of 3.7 cm (1.457 inches). The NRC staff told the GSTR staff that submission of a relicensing request that was consistent with the OSU relicensing documentation would be adequate. Also, in earlier discussions with the NRC staff regarding the fuel OD, the NRC staff found 1.47" acceptable. This change, along with other changes in requirements from the NRC staff is prolonging the relicensing actions for the GSTR.

Despite the inconsistent guidance from the NRC staff, we have reanalyzed our LCC for a conservative fuel element OD of 1.478 inches. The change in thermal-hydraulic analyses results is insignificant.

Question:

b. The parameters chosen that defined the flow area of the limiting hot channel do not appear to be consistent for performing the limiting sub-channel analysis. The flow area appears to be approximately 20 percent larger than the limiting channel flow area, as determined by the NRC staff in their confirmatory analysis, due to non-conservative assumptions in the selection of the hydraulic channel orientation relative to the limiting fuel elements. Determine a limiting hot channel flow area conservative for the LCC T-HA.

Response:

The question of defining an appropriate hot channel for a TRIGA reactor with complicated flow paths and steep rod-to-rod power gradients is a challenging one. Based on previous NRC guidance, the originally submitted GSTR thermal-hydraulics analyses were based on the minimum average per-ring flow area. This area is determined by calculating the area between the inner and outer boundaries of each ring (defined as the mid-point between fuel element centers in adjacent rings), subtracting the cross sectional area of the fuel elements in the ring, and dividing by the total number of fuel elements in the ring. This is referred to as the "average channel" in subsequent discussions and it defines a reasonable flow area to use in the thermal-hydraulic analyses.

Once again, the guidance received from the NRC has changed. We are now being required to use an unreasonably conservative hot channel that is based on a minimum possible fuel element distance between the B and C rings. With this very small, nonexistent channel ("minimum" in Table 1 below),

RELAP shows flow instabilities at power as low as 20 kW/rod. However, several factors argue against the validity of this minimum flow channel. The minimum flow channel model assumes that all of the rods surrounding the channel are producing an equal amount of power and that there is no flow between channels. In reality, the rod power varies by approximately 10% between the B and C rings of the limiting core. More significantly, the minimum channel calculations assume that six identical channels surround the hot rod, which prevents flow between channels and forces 1/6$^{th}$ of the power produced by the hot rod to be removed by the hot channel. Since the hot rod is directly adjacent to the central thimble, which produces no power, heat from the hot rod will preferentially flow away from the minimum channel and toward the central thimble. Therefore the hot channel will be responsible for removing significantly less than 1/6$^{th}$ of the rod power. The 1/6$^{th}$ power assumption (supported by the NRC) for power supplied to the minimum channel implies that heat is actually moving from cooler to hotter regions of the fuel rod – a behavior that is not physically possible.

Detailed 2-dimensional and 3-dimensional simulations of the GSTR operating core with supporting benchmark experiments were performed (Alkaabi, 2015). These indicate that two-dimensional single-channel models such as the minimum channel model underestimate actual channel flow by at least 35% for the B, C, and D rings (see Figure 4). In-core cooling water measurements were made to benchmark the models. Detailed information can be found in the reference. The best way to analyze the DNBR for the additional flow (cross-flow) is to increase the flow area in the model. We have done that in increments of 10%, 20%, 30%, and 35% (see Table 1). Additionally, the Bernath equation represents a very conservative estimate of the minimum departure from nuclear boiling ratio (MDNBR). Figure 3 displays the MDNBR predicted by the Bernath correlation, the 1987 Groenveld lookup tables (internal to RELAP5/Mod 3.3), and the 2006 Groenveld lookup tables for the "Minimum + 30%" case in Table 1. Based on the 2006 Groenveld lookup tables, the MDNBR is over 2.45 for a rod power of 22.5 kW.

**Table 1. Flow Channel Dimensions**

| Area Case | Fuel Rod OD (m) | Flow Area (m^2) | Wetted Perimeter (m^2) | Hydraulic Diameter (m) |
|---|---|---|---|---|
| Minimum | 0.03754 | 4.184E-04 | 0.11794 | 0.01419 |
| Minimum +10% | 0.03754 | 4.602E-04 | 0.11794 | 0.01561 |
| Minimum +20% | 0.03754 | 5.020E-04 | 0.11794 | 0.01703 |
| Minimum +30% | 0.03754 | 5.439E-04 | 0.11794 | 0.01845 |
| Minimum +35% | 0.03754 | 5.648E-04 | 0.11794 | 0.01916 |
| Average Channel | 0.03734 | 5.855E-04 | 0.11730 | 0.01997 |

The key question in the limiting core analysis is whether or not operation of a fuel rod at a power level of 22.2 kW (the peak rod power from the neutronics analysis of the limiting core) will result in fuel damage leading to fission product release. A consideration of several cases in which the flow area is increased from the unrealistic minimum flow area demonstrates that operation at 22.2 kW will not result in flow instabilities in the actual core and provides a sufficient MDNBR to protect against fuel damage. These cases are summarized in Table 1. The "minimum" case in Table 1 represents the minimum channel case discussed above. The "average channel" case corresponds to the average flow channel used in the previous analyses. The "minimum + X%" cases are those in which the minimum channel area is increased by 10-35%.

Figure 1 displays the MDNBR predicted by the conservative Bernath correlation as a function of rod power, based on data provided by the Colorado School of Mines RELAP5/Mod 3.3 model of the GSTR flow channel for each case in Table 1. The far right data point for each line corresponds to the last calculation without flow instabilities. Thus, every point on this graph represents stable flow predicted by the RELAP model. Figure 2 shows the mass flow rates as a function of rod power predicted by the RELAP5 model for each case in Table 1. An examination of Figures 1 and 2 show that the 35% greater flow area and the average channel cases predict stable flow at rod powers in excess of 25 kW with a MDNBR in excess of 2 at a rod power of 22.5 kW. Even a very modest increase of 10% in the flow area results in stable flow at a rod power of 22.5 kW with a MDNBR in excess of 1.8. In conclusion, the LCC has sufficient cooling even with a very conservative thermal-hydraulics analysis.

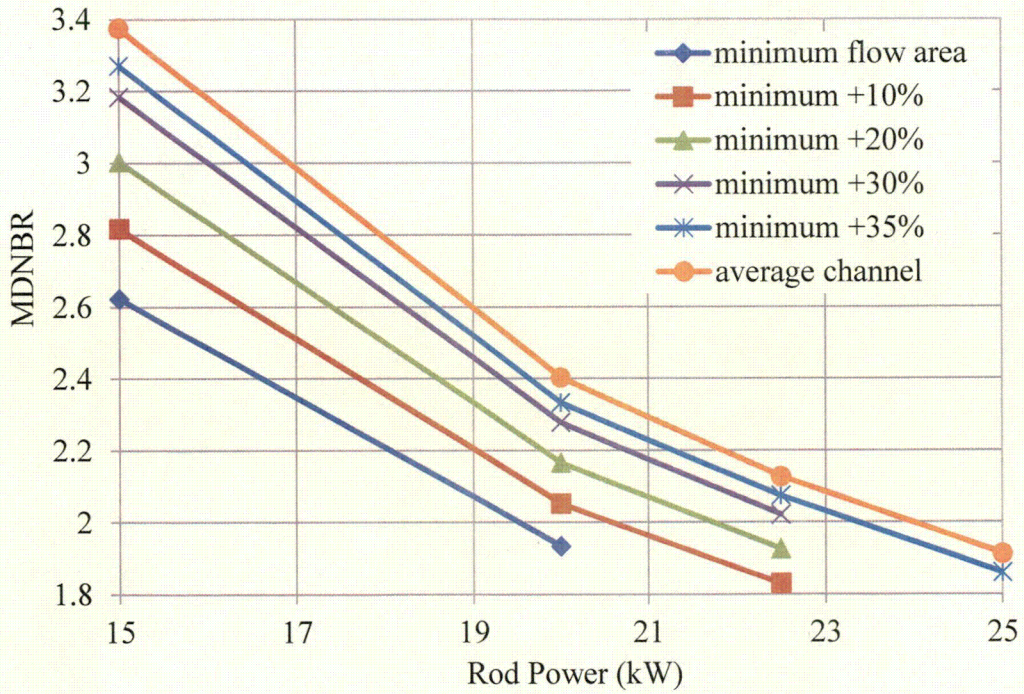


**Figure 1. Minimum Departure from Nucleate Boiling Ratio as a function of rod power for the six flow channel cases in Table 1, using conservative Bernath correlations.**

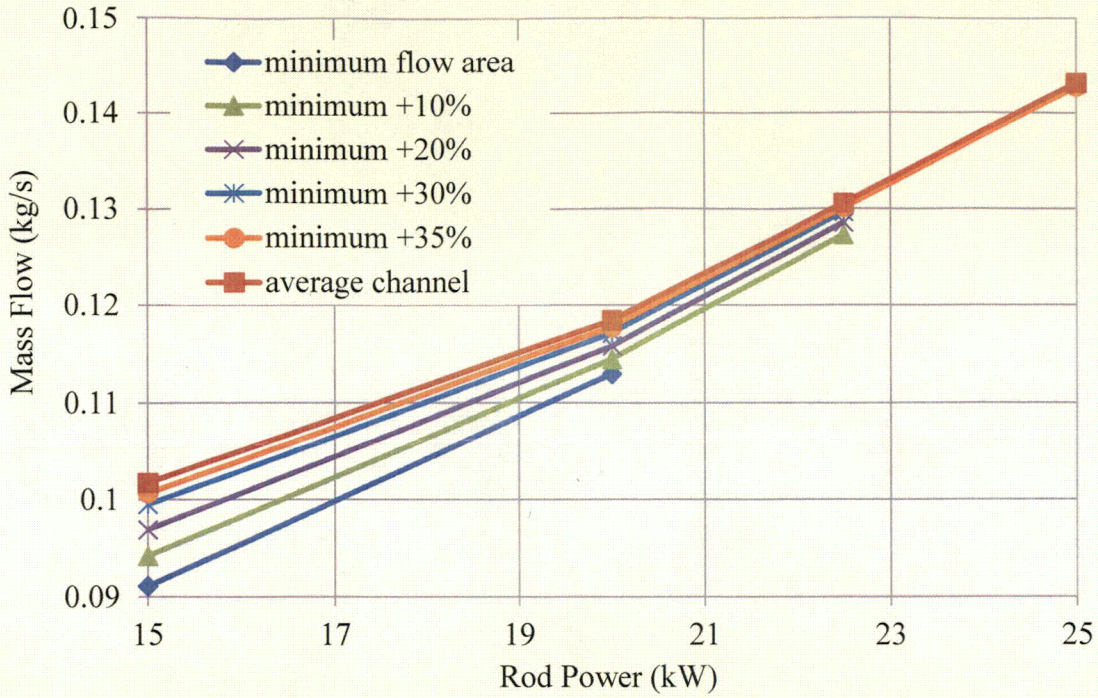


**Figure 2. Channel mass flow rate as a function of rod power for the six flow channel cases in Table 1**

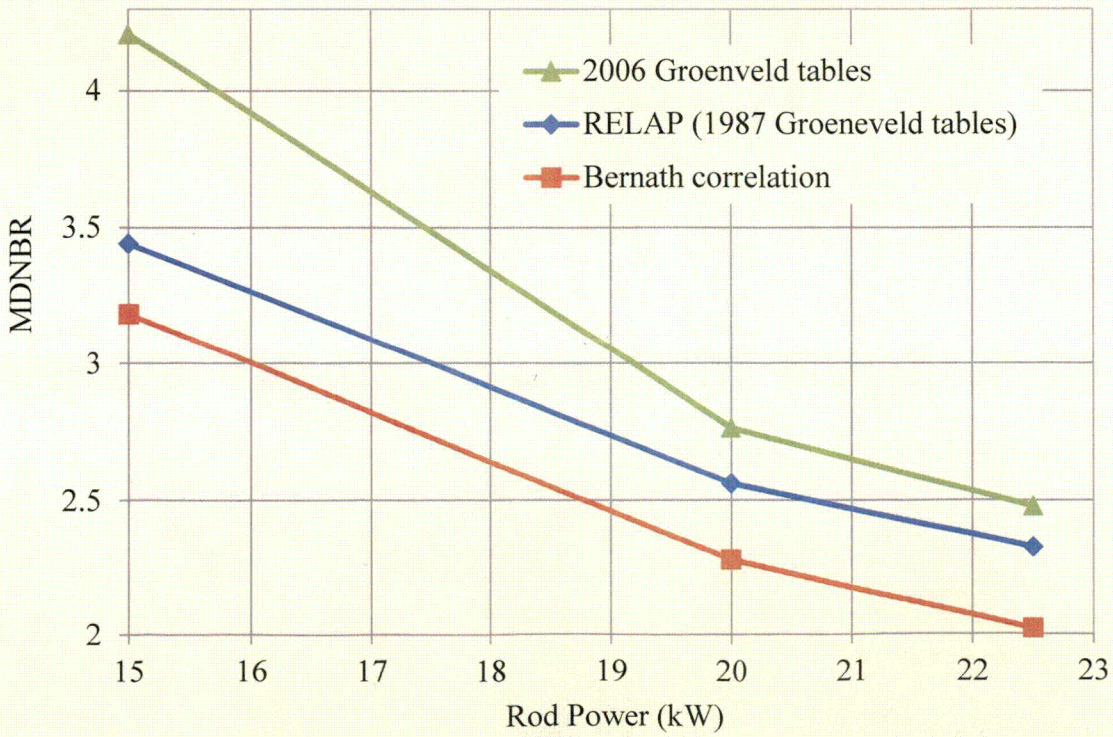


**Figure 3. Minimum Departure from Nucleate Boiling Ratio as a function of rod power for the "minimum + 30%" case in Table 1.**

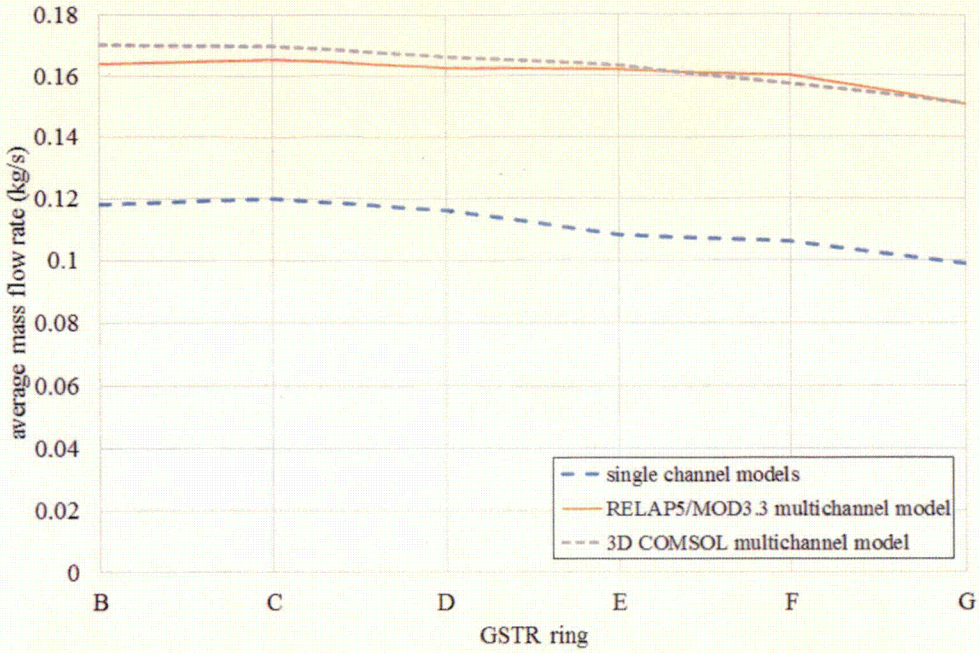


**Figure 4. Average channel mass flow rates as a function of fuel ring for single- and multi-channel models of the GSTR (Alkaabi, 2015).**

Question:

c. If any assumptions were used in the updated LCC T-HA with regard to any difference between the temperature of the bulk pool water versus the temperature of the core inlet (at the lower core support (grid) plate entrance to the fuel array), provide supporting validation information to substantiate any assumptions used, as necessary.

Response:

The calculations presented in 1.b., above, are all based on a core inlet temperature of $60^0$ C, based on the maximum allowed coolant temperature at the top of the pool. Experiments at the GSTR reveal that the core inlet temperature may be $>7^0$ cooler than the upper pool temperature. Figure 5 displays the MDNBR at 20 kW per rod for the minimum channel case as a function of core inlet temperature. Figure 6 shows the empirical data for the delta T experiment. Reducing the inlet temperature by $7^0$ C provides a small (~5%) increase in the MDNBR based on the Bernath correlation. The larger increase (~12.7) in the MDNBR predicted by the 2006 Groenveld tables between $60^0$ C and $55^0$ C results from a boundary in the look-up table data regions.

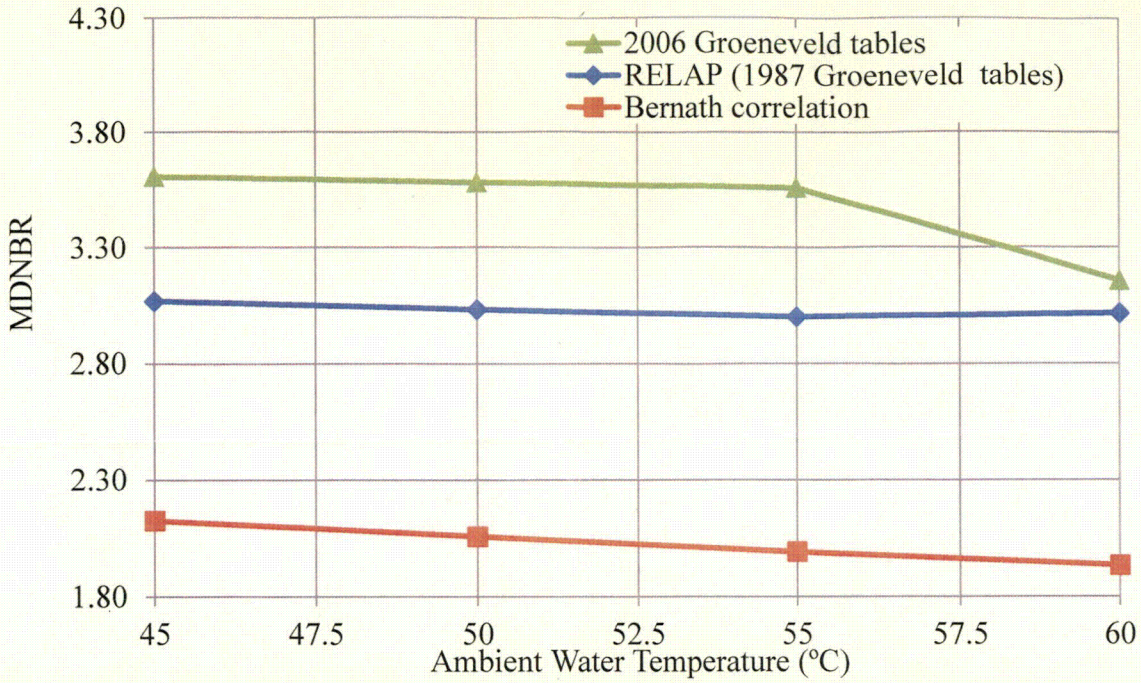


**Figure 5. Minimum Departure from Nucleate Boiling Ratio as a function of coolant inlet temperature for the "minimum" case in Table 1 at a rod power of 20 kW.**

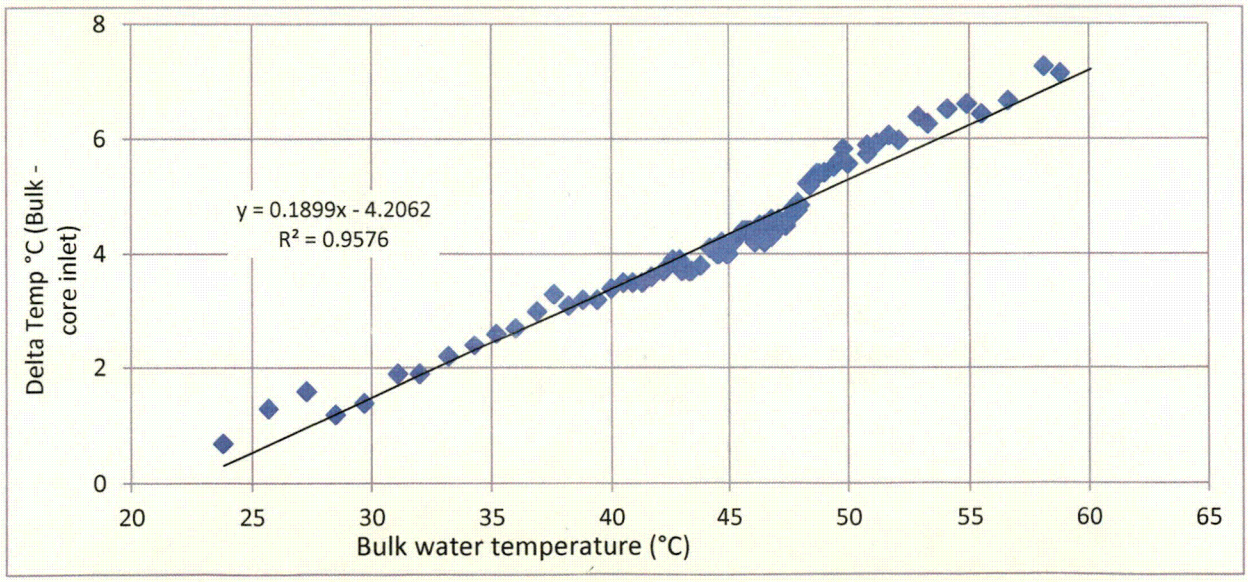


**Figure 6. Delta T in reactor cooling water, between core inlet and measured tank water temperature.**

**Reference:** Alkaabi, A.K., "Thermal Hydraulics Modeling of the US Geological Survey TRIGA Reactor," Ph.D. Dissertation, Colorado School of Mines, 2015, http://hdl.handle.net/11124/20148