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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: Responses to Question 14 of the Referenced RAI

Mr. Wertz:

Here is the response to Question 14, parts 14.1 and 14.2:

14.1 In GSTR SAR Section 13.1, the GSTR SAR states that fuel temperature limits of 1,100 °C for stainless steel clad fuel and 535 °C for aluminum clad fuel were set to preclude the loss of clad integrity. GSTR SAR Technical Specification (TS) Section 14.2.1 lists safety limits for stainless steel clad fuel and aluminum fuel of 1,000 °C and 535 °C, respectively. The NRC has accepted (NUREG–1537 Appendix 14.1) that no fuel damage or cladding failure is expected if the fuel temperature for aluminum clad fuel is maintained at a temperature of less than 500 °C and the stainless steel clad fuel temperature is not to exceed 1,150 °C when the cladding temperature is less than 500 °C and is not to exceed 950 °C if the cladding temperature is greater than 500 °C. Please provide justification for the use of fuel temperature safety limits that are different from the stated limits in NUREG–1537.

The GSTR will accept the stated limits in NUREG—1537 and change GSTR SAR Section 13.1 to read, "Fuel temperature limits of 1,150 °C (if cladding temperature is at or less than 500 °C) or 950 °C (if cladding temperature is greater than 500 °C) for stainless steel clad fuel and 500 °C for aluminum clad fuel have been set to preclude the loss of clad integrity."

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14.2 Within Chapter 13, the assumed power level or trip setpoint for accident analysis is set at 1.0 MW (Maximum Hypothetical Accident, Loss of Coolant Accident) and 1.06 MW (uncontrolled rod withdrawal). However, the Limiting Safety System Setting (LSSS) and SCRAM setpoints are set at 1.1 MW. Please describe how these setpoints ensure that the safety basis is maintained.

The reanalysis of the power level /trip setpoint change for the uncontrolled rod withdrawal accident will be answered at a later date after the neutronic analysis results are available; however, we will change our accident analysis assumptions to be at an assumed power of 1.1 MW in our new neutronic and thermal-hydraulic analyses.

Sincerely,

Tim DeBey

Tim DeBey

USGS Reactor Supervisor

I declare under penalty of perjury that the foregoing is true and correct.
Executed on 5/31/12

Copy to:

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