# U.S. GEOLOGICAL SURVEY
# RESEARCH REACTOR
# LICENSE NO. R-113
# DOCKET NO. 50-274

## U.S GEOLOGICAL SURVEY LETTER
## RESPONSE TO RAI NO 8 AND 9
## NOVEMBER 30, 2011

## REDACTED VERSION*

## SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

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

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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:Response to Questions 7 and 8 of the Referenced RAI

Mr. Wertz:

**Question 7:** NUREG-1537, Part 1 Section 4.4, "Biological Shield," requests a description of the biological shield design basis, design methods, and the conformance of dose consequences to regulatory requirements. The GSTR SAR does not discuss these issues. Please provide information that addresses these subjects.

**Response:** The Geological Survey TRIGA Reactor (GSTR) core is located inside of an aluminum liner that sits at the bottom of an aluminum tank placed roughly 25 feet underground. The GSTR core is surrounded radially by approximately 0.9 ft of graphite and 2 ft of water, vertically by approximately 20 ft of water from the top grid plate, and approximately 1.33 ft of water below the core [1]. The tank rests on top of a 3 ft concrete slab which is on top of approximately 8 ft of concrete [1]. [redacted]. The geometry of the tank is modeled in Figure 7.1. The void between the liner and tank is monitored by a vertical pump tube located at the edge of the tank. The pump tube has a large shield plug with lead and neutron shielding to minimize the dose on top of the reactor tank. Also, the void between the tank and liner has a lead wrap on top of the tank to minimize the dose.

Together the water and concrete act as the biological shield for the reactor. The biological shield supports the reactor tank, prevents soil activation, and minimizes the radiation dose around the reactor tank by shielding neutrons and gamma rays from the core. The effectiveness of the biological shield is supported by years of survey data showing acceptable radiation levels in occupied areas. The Nuclear Regulatory Commission (NRC) routinely reviews the GSTR radiation protection program and has performed independent measurements of radiation levels in the facility.

Based on operational experience, historical data, and reviews from the NRC, the staff concludes that there is reasonable assurance that the biological shield will limit exposures from the reactor and reactor-related sources of radiation so that the limits of 10 CFR 20 will not be exceeded.

[1] Law G.: TRIGA Mark I 1000-KW Pulsing Reactor Mechanical Maintenance and Operating Manual for U.S. Department of the Interior, Geological Survey Denver, Colorado. February 15, 1969

A020
NRR

Figure 7.1 Geometry of the GSTR core, biological shield, and supporting structures

**Question 8:** NUREG 1537, Part 1 Section 4.5, "Nuclear Design," requests a detailed description of analytical methods used in the nuclear design, including computer codes used to characterize technical parameters pertaining to the GSTR. The GSTR SAR has not provided such information. Please provide descriptions of the analytical methods employed in Sections 4.5.1 through 4.5.6 of the application to calculate power distributions, reactivity coefficients, control rod worths, and Departure from Nucleate Boiling Ratio (DNBR).

**Answer:** The analyses that apply to Sections 4.5.1 through 4.5.6 are being redone with two computer codes: MCNP and RELAP.

MCNP is a Monte Carlo particle transport code used extensively in the nuclear research field for its ability to simulate a wide range of particle transport scenarios from reactor design problems to shielding and dosimetry calculations. As a Monte Carlo code, MCNP uses a combination of random numbers pared to different tables and function to simulate the probabilistic behavior of a particle traveling within different materials. Neutrons, electrons and photons can be simulated through MCNP (X-5 Monte Carlo Team, 2003).

All of the neutronic calculations for the GSTR are being done using a MCNP model running on MCNP 5 version 1.51. The program functions through using "decks" (a legacy term from when the program was run on punch-card machines and actual decks of cards were used) that contain the geometric, material, and input parameters for the problem, including the particle source, and any detectors for particle fluxes or reaction rates.

The code begins by creating a particle either from a user-defined source, or through the use of a fission approximation (in the case of a k-code). Which material the particle is currently in determines the distance that particle travels before an interaction occurs with one of the atoms in the material (as defined by material cards and the appropriate cross-section library). MCNP calculates this distance by:

$$l = -\frac{1}{\Sigma_t}\ln(1-\xi) = -\frac{1}{\Sigma_t}\ln(\xi) \quad (1)$$

Where $l$ is the particle track length, $\Sigma_{tot}$ is the total neutron cross-section, and $\xi$ is a random number between 0-1.

After traveling this length, an interaction occurs based on the material the particle is within. If there are multiple nuclides, another random number determines which nuclide the particle interacts with, based on user input. Based on what information the user has provided MCNP with, the interaction between the particle and the nucleus is determined and carried out. At this point the particle will either have been removed from the simulation (due to some form of capture reaction), or a new energy and velocity are determined, and the process begins again (X-5 Monte Carlo Team, 2003).

As MCNP is designed as a generic particle transport problem, the simulation can run in two ways. The primary way uses a generic source that can function using any of the particles found in MCNP. With this source, the geometry, distribution, and energy of the particles can be set in the input deck. Particles are created by the source definition, and run through the described process until they are adsorbed (a setting can also stop MCNP from continuing to track the particle if it exists for too long), at which point the source creates another particle. This continues until a pre-set limit is reached, either in time or number of particles (X-5 Monte Carlo Team, 2003). All of the neutronic simulations run for the GSTR re-licensing used the k-code method (X-5 Monte Carlo Team, 2003).

In each iteration the relative error (R) is calculated through:

$$R = \frac{1}{\bar{x}}\sqrt{\frac{\overline{x^2} - \bar{x}^2}{M-1}} \quad (2)$$

Where:

$$\bar{x} = \frac{1}{M}\sum_m x_m \;\&\; \overline{x^2} = \frac{1}{M}\sum_m x_m^2 \quad (3)$$

$M$ is the number of active cycles, $x_m$ is a quantity from an individual cycle, $m$, $\bar{x}$ is the average value of the $x$ and $\overline{x^2}$ is the average value of $x$ squared. MCNP combines all three of these uncertainties to generate the covariance and correlation for the problem (X-5 Monte Carlo Team, 2003).

Precision in any Monte Carlo simulation is calculated using the Strong Law of Large Numbers (X-5 Monte Carlo Team, 2003). Under this law, the average value will limit the expected value as the number of attempts to find that value approaches infinity. Since an infinite number of runs cannot be done, MCNP calculates the precision of any value given as a function of the number of attempts run. While any individual particle (or even batch of particles) will not represent the overall physical situation, several million particles will provide a reasonable approximation for the physical system. MCNP measures this precision through a standard deviation calculation:

$$S^2 = \frac{\sum_{i=1}^{M}(x_i - \bar{x})^2}{N-1} \cong \overline{x^2} - \bar{x}^2 \qquad (4)$$

Where $S$ is the approximation of the standard deviation (X-5 Monte Carlo Team, 2003).

RELAP5 is a computational fluid dynamics (CFD) suite developed for the Nuclear Regulatory Commission to provide a regulatory thermo-hydraulic code for use in reactor applications. RELAP uses a finite-difference algorithm to determine the thermo-hydraulic properties of a user-defined geometry, and has the capability to represent both steady-state and transient conditions (Information Systems Laboratories, Inc., 2010).

Unlike several more modern codes, RELAP uses a one-dimensional two-fluid model to represent a two-phase system comprised of water, and possibly some non-condensable components in the steam phase, or soluble components in the liquid phase (Information Systems Laboratories, Inc., 2010). This allows the code to represent complex thermal-hydraulic systems (such as nuclear reactor cooling systems) without the need to consume large amounts of computer resources.

A series of eight equations solve eight variables within the thermal-hydraulic system. These variables are pressure, phasic specific internal energies (for both liquid and gas phases), vapor volume faction, phasic velocities (both liquid and gas), non-condensable quality, and boron density (Information Systems Laboratories, Inc., 2010).

Geometry is input into the code as a string of numeric lines in a text file. RELAP has a number of pre-defined geometry types that can be defined to make up the geometry of a system. Each predefined geometry was chosen to represent the different hydraulic components of a light-water reactor's cooling system. Special hydraulic components, referred to as time dependent volumes, represent boundary conditions within the system as their hydraulic properties (temperature, pressure, fluid velocity exc.) are user defined and not affected by the RELAP computation. Solid components, such as pipe walls and fuel rods, are represented as heat structures (Informations Systems Laboratories, Inc., 2010).

Heat structures use a one-dimensional heat-transfer approximation to represent heat flow through solid mediums within a RELAP deck. Either a rectangular or cylindrical geometry can describe the heat structure. A series of nodes represent solid material, as shown in Figure 1. Aside from the nodes for a one-dimensional analysis, multiple sets of nodes can be connected axially (although heat does not transfer from one set of nodes to another) to represent more complex structures. Multiple axial nodes are required when connecting a single heat structure to multiple hydraulic components (such as those in a PIPE) as each heat structure can only connect to a single hydraulic component. Multiple materials can

be represented within a single heat structure as long as the thermal data (thermal conductivity and volumetric heat capacity) for each material is provided by the user. RELAP allows for this data input through tables, equations, or as a constant value (Information Systems Laboratories, Inc., 2010).

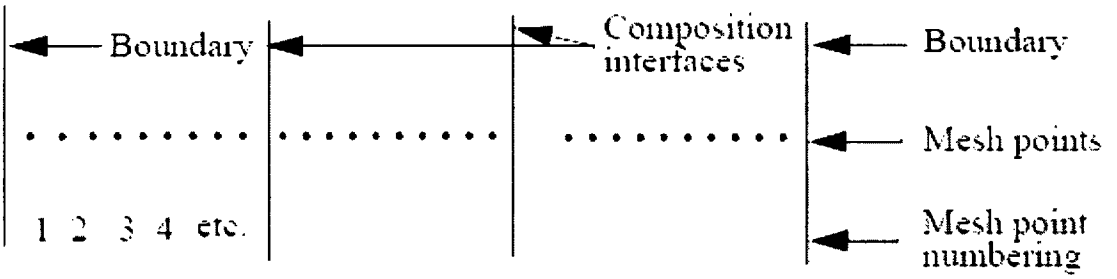


**Figure 1: Example of heat structure nodes in RELAP5**

RELAP calculates the temperature and heat flux at each node of a heat structure. A heat generation term can also be applied to a node, or distributed throughout a heat structure to represent heat generation (such as within a heating coil or fuel rod). Every heat structure has two boundary conditions. These can be set to hydraulic volumes (to represent an interface between the heat structure and fluid), constant power fluxes, constant temperatures, insulated boundaries, or reflecting boundaries (representing the center of a cylinder) (Informations Systems Laboratories, Inc., 2010).

RELAP uses several convergence criteria to determine when a model has moved into a steady-state condition when running a steady-state problem. The steady-state condition for RELAP monitors the change in the thermodynamic density (***ρ***), internal energy (***U***), and pressure (***P***) to monitor the change in the system as a whole. Thus once these three variables reach a constant value (with respect to time), the system has reached steady state (Information Systems Laboratories, Inc., 2010). Within the code this is represented as:

$$\left(\frac{d(\rho\, h)}{dt}\right)_i^n = \frac{\left(\rho_i^{n+1}\, U_i^{n+1} + P_i^{n+1}\right) - \left(\rho_i^n\, U_i^n + P_i^n\right)}{\Delta t^n} \tag{5}$$

However, this function is not well-behaved with respect to time, making a direct measurement difficult, as large fluctuations in the value of the derivative can occur. To compensate, RELAP uses a fitting function which is well behaved that can be solved over a number of time steps to determine steady state. This equation is:

$$\left[\sqrt{\left(\frac{d}{dt}(\rho\, h)\right)^2}\right]^n = y = e^{\alpha+\beta t^n+\gamma(t^n)^2+\varphi(t^n)^3}, (t_o \le t^n \le t)$$

Where ***α, β, γ,*** & ***φ*** are fitting coefficients computed through least squares. This provides a time-smoother root mean square rate of change of the system (Information Systems Laboratories, Inc., 2010).

When doing transient calculations, largely through user-input controls, this method is not always sufficient to reach steady state, or the controls may ensure a steady-state condition is not reached (through rapidly changing conditions). In these cases more extensive approximations are needed to ensure the RELAP model is converging, and still valid (Information Systems Laboratories, Inc., 2010).

The neutronic and thermo-hydraulic analyses are continuing at the Colorado School of Mines, under a contract with the Department of Energy. While these analyses are progressing slower than expected, progress is being made.

Sincerely,

Tim DeBey

USGS Reactor Supervisor

**I declare under penalty of perjury that the foregoing is true and correct.**
**Executed on 9/26/11**

Copy to:

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