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

November 16, 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 October 2, 2012

Subject: Responses are provided to Questions of the subject RAI

Mr. Wertz:

Responses to the above questions are provided in the enclosed pages. This should complete our responses to the RAI dated October 2, 2012.

Sincerely,

Tim DeBey

Tim DeBey

USGS Reactor Supervisor

I declare under penalty of perjury that the foregoing is true and correct.
Executed on 11/16/12

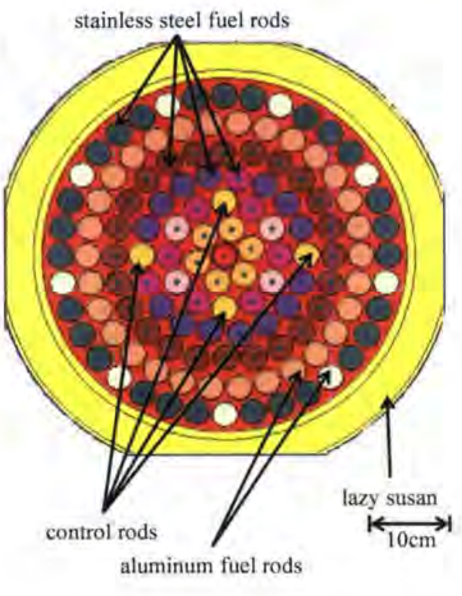
Attachment

Copy to:
Betty Adrian, Reactor Administrator, MS 975
USGS Reactor Operations Committee

A020
NRR

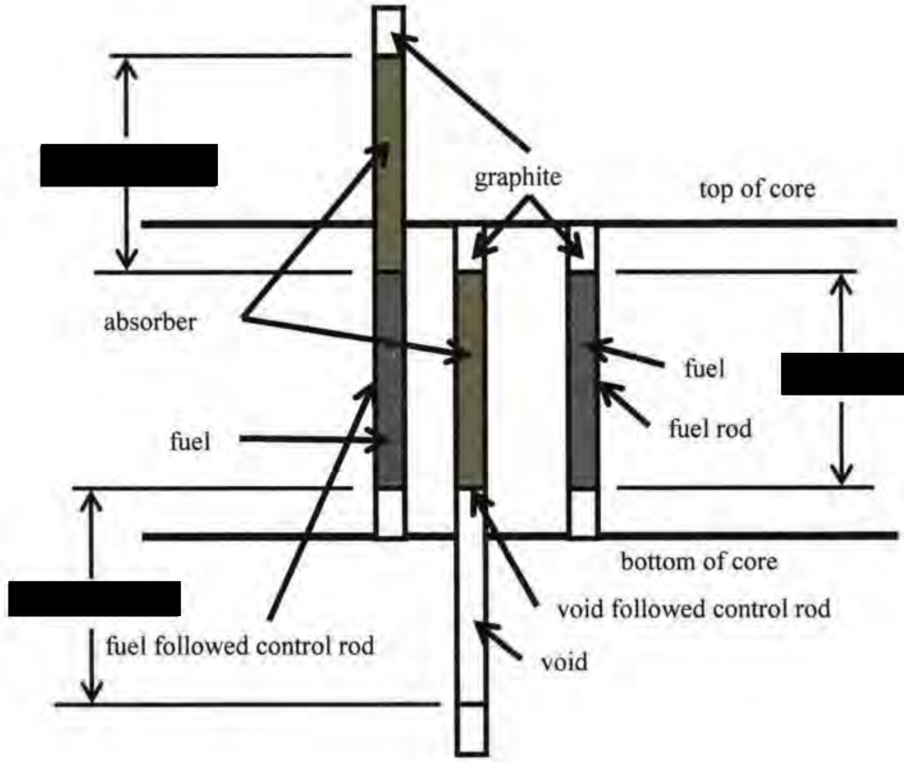
9. **Please provide the results of the GSTR neutronic analysis showing:**
   - **Core map showing the contents of the core lattice positions for the LCC and the OC:**

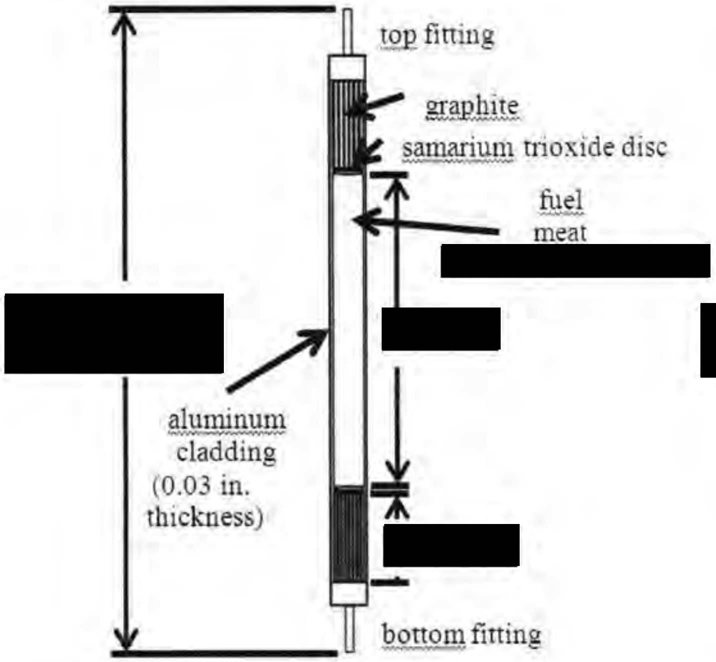
[redacted]

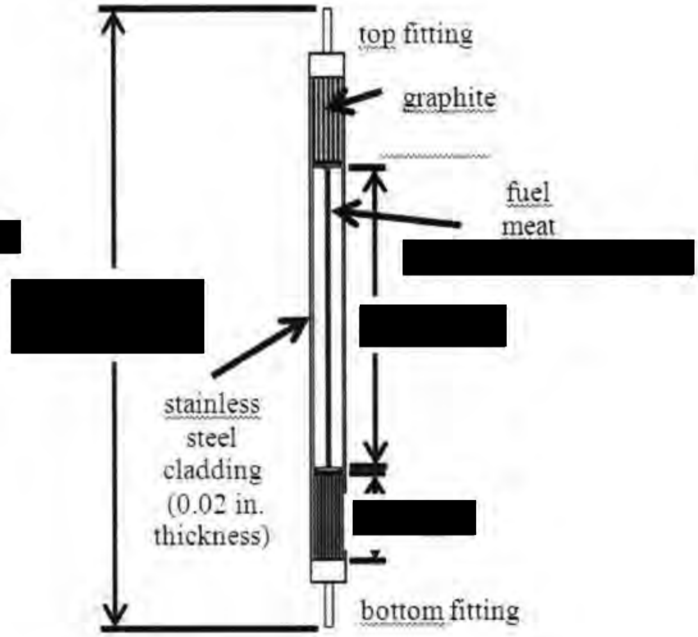


OC

- **The Enrichment and cladding type for fuel elements used at the GSTR**
  - Aluminum & stainless steel clad <20% enriched (average 19.75%)
- **Diagrams and dimensions for fuel elements, control elements and other occupants of lattice positions**







- **The effective delayed neutron fraction (β_{eff}) for the LCC and the OC (0.00728 used for all analysis as most conservative answer).**
  - LCC: 0.00731 ±0.00088
  - OC: 0.00728 ±0.00013

- **The all-control-rods-out k-effective (k_{eff} and the excess reactivity (c) for the LCC and the OC**
  - LCC: k_{eff}: 1.04826, ρ_{excess}: \$6.63
  - OC: k_{eff}: 1.03650, ρ_{excess}: \$5.01
- **The Control Rod worth's for each of the 4 control rods including the k_{eff} values determined for the LCC and the OC**
  - LCC:

	Shim 1	Shim 2	Regulating	Transient
Worth (\$)	2.40	2.30	4.25	2.62
k_{eff}	0.98644-1.00832	0.98769-1.0040	0.97735-1.00832	0.98561-1.00470

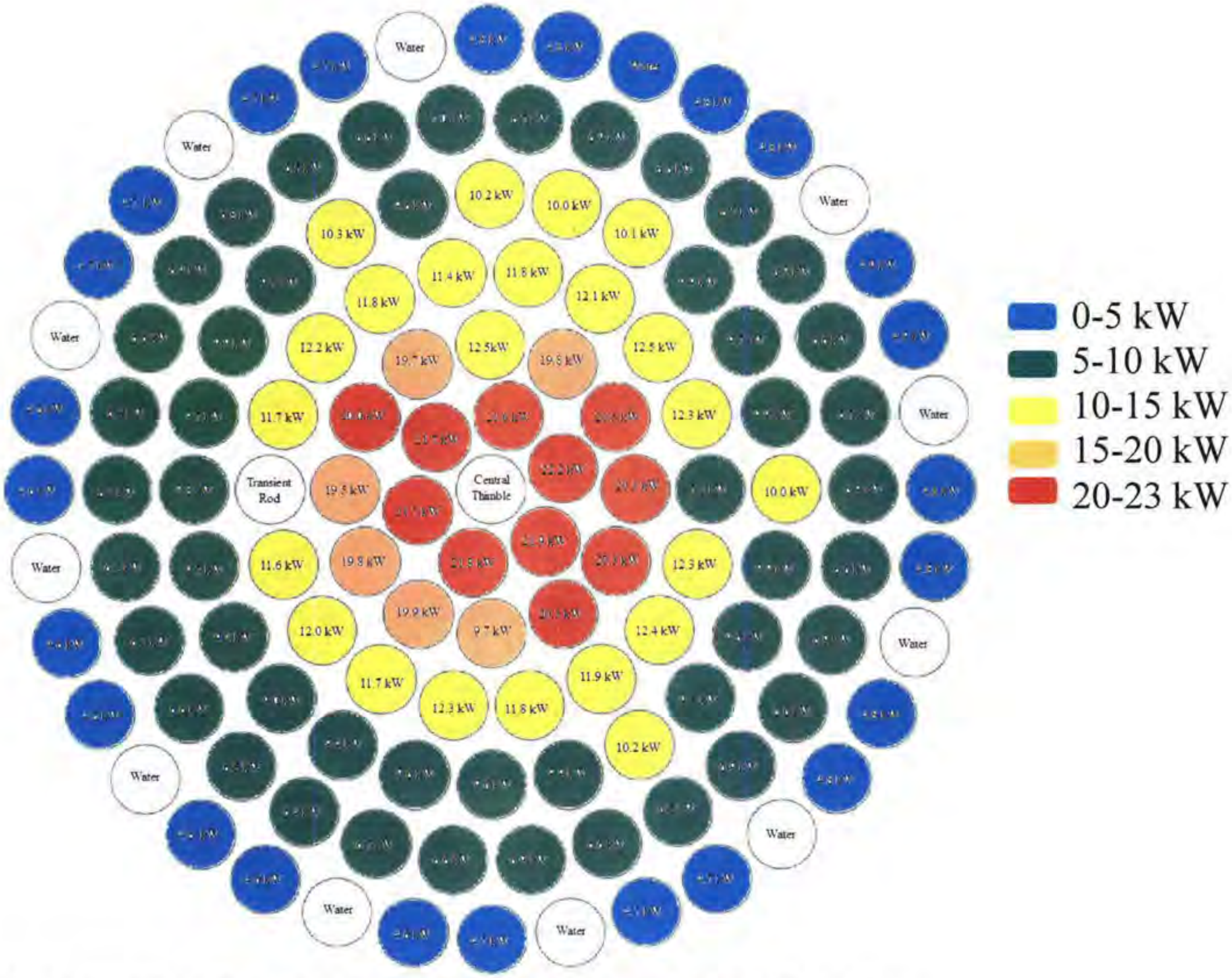
  - OC:

	Shim 1	Shim 2	Regulating	Transient
Worth (\$)	2.28	2.38	3.56	2.17
k_{eff}	0.99944-1.01542	1.00051-1.01718	0.99691-1.02182	0.99691-1.02182

- The comparison of the ρ_{excess} and the control rod worths calculated and measured from the OC

	Shim 1	Shim 2	Regulating	Transient
Calculated Worth (\$)	2.28	2.38	3.56	2.17
Experimental Worth (\$)	2.305	2.435	3.630	2.142

- **Shutdown reactivity of the core with the largest control rod removed for the OC.**
  \$0.51 +/- \$0.04
- **Power Distribution Graphic for the OC showing power in kW per fuel element**



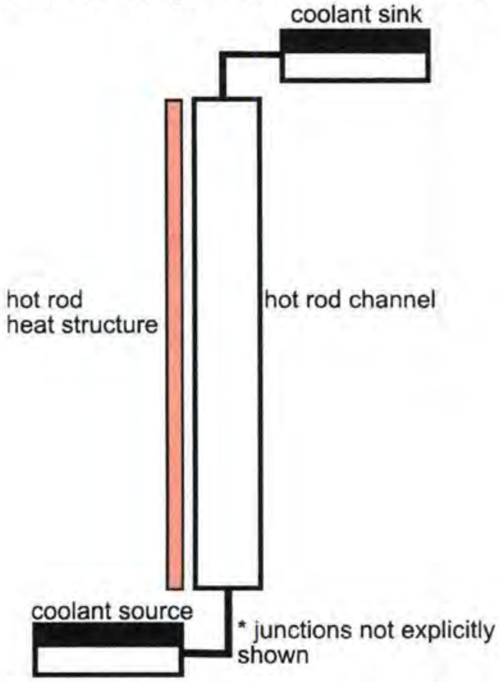
12 – Please provide thermal-hydraulic data for the LCC constant with the following:

- **Identify the unit cell used to define the RELAP model – graphically and with dimensions**

[redacted]

- **Identify and justify any entry/exit loss coefficients employed in the RELAP model**
  Pressure loss coefficients: from OSU model
  Inlet: 2.26
  Exit: 0.63

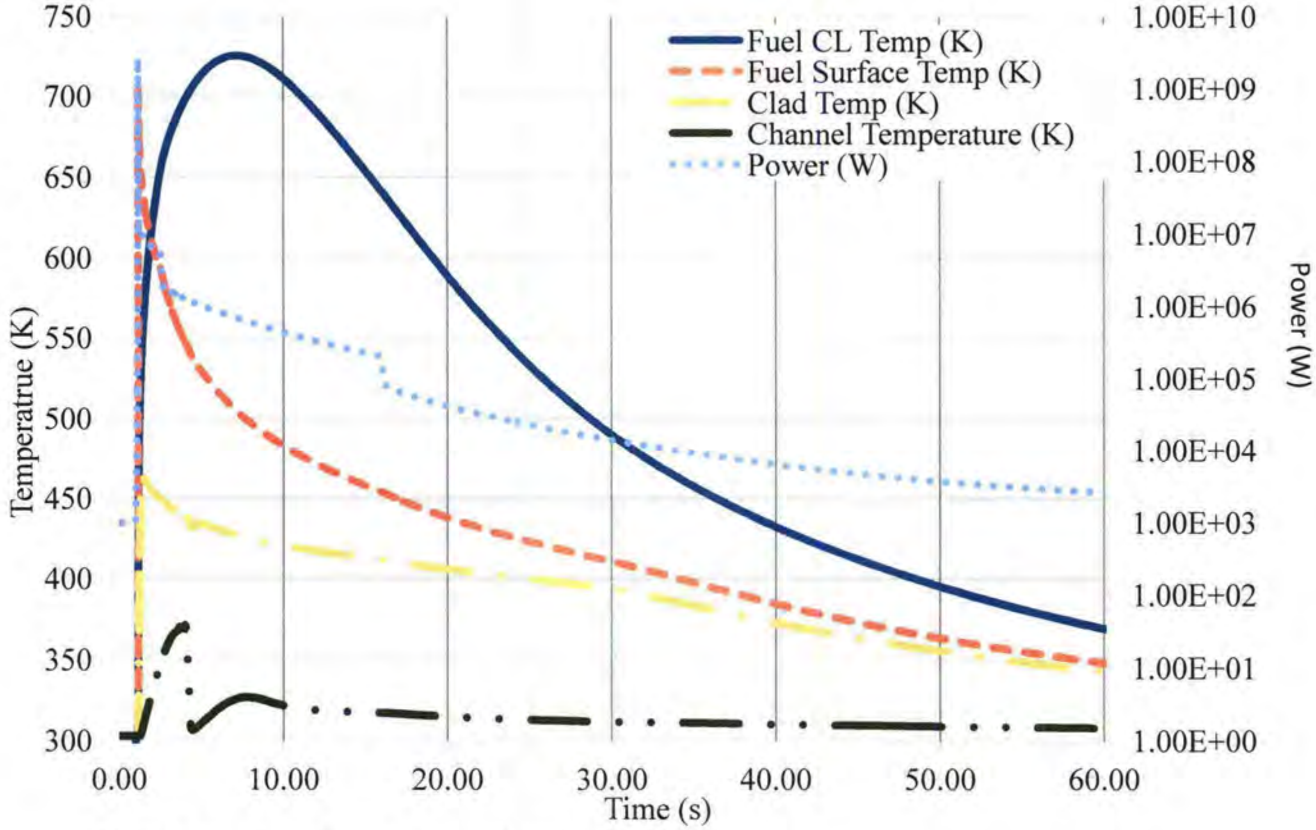
- **Provide a diagram of the RELAP model**



- **Document input assumptions used to analyze DNBR for the LCC such as fuel element power, peaking factors employed, inlet temperatures assumed, etc.**
  Element power: 22.17 kW
  Peaking Factor: 2.28
  Inlet temperature: 303.15K
  Inlet Velocity: natural convection, allowed to be computed. Eventually set to 0.1 m/s based on RELAP results to improve calculation times

- **Document the RELAP model calculated results such as the core flow rate, peak fuel and cladding temperatures, the location of the minimum DNBR, and the value of the minimum DNBR using the Bernath correlation**
  Core flow rate: ~0.10686 m/s
  Peak Fuel Temperature: 829.32 K (556.17 C)
  Peak Cladding Temperature: 410.04 K (136.89 C)
  MDNBR: 2.15 @ 0.392133 from the bottom of the fuel element

- **Characterize the response of GSTR to a reactivity pulse and an uncontrolled rod withdrawal transient event. Please provide similar information as with DNBR results but also include the final power achieved in the event, the duration of the pulse or event, and the sequence of events**



$3.00 Pulse Results – Average Rod

	Average Core	Hot Rod	B-Ring	C-Ring	D-Ring	E-Ring	F-Ring	G-Ring
Full Core Peak Power (MW)	2652.55	2652.55	2652.55	2652.55	2652.55	2652.55	2652.55	2652.55
Peak CL Temp (K)	725.52	1266.15	1249.26	1173.23	822.67	729.74	590.36	510.11
Peak Fuel Surface Temp (K)	708.23	1226.73	1210.53	1137.61	801.40	712.28	578.60	501.64
Peak Clad Temp (K)	463.66	669.11	662.69	633.80	500.58	465.27	412.30	381.80
Peak Water Temp (K)	371.72	459.49	456.75	444.40	387.49	372.41	349.78	336.75
Peak Void (1/100)	0.14	0.31	0.30	0.28	0.17	0.14	0.09	0.07
Power Factor	1.00	2.28	2.24	2.06	1.23	1.01	0.68	0.49

14.2 Please provide the analysis of the uncontrolled rod withdrawal for the LCC; this analysis should be consistent with evaluated control rod worth's and should demonstrate the acceptability of SCRAM setpoints, control rod drop times, and control rod withdrawal rates and speeds in the technical specifications.

The continuous rod withdrawal scenario involved the reactor starting at 5W, with the regulating rod assumed to be fully inserted so that its full worth was available. The regulating rod has the highest rod worth. The regulating rod worth was then scaled up to match the new, limiting core predicted worth of $4.25. Continuous rod withdrawal was done at a constant rate of 0.9535 cm/s (this is actually 6% faster than measured). Power was found to reach the 1.1 MW scram limit at slightly above $1.00 of reactivity inserted. This occurs at 14.46 s after the initiation of rod movement. When the reactor scrams, the rod reactivity is inserted into the core over a 0.2 s time frame, shutting down the reactor with a peak power reached of 1.15 MW.

The table below gives more detailed information.

	Average Core	Hot Rod	B-Ring	C-Ring	D-Ring	E-Ring	F-Ring	G-Ring
Full Core Peak Power (MW)	1.15	1.15	1.15	1.15	1.15	1.15	1.15	1.15
Peak Cl Temp (K)	304.29	305.75	305.70	305.50	304.55	304.30	303.93	303.71
Peak Surface Temp (K)	304.24	305.64	305.59	305.40	304.49	304.25	303.89	303.68
Peak Clad Temp	304.14	305.41	305.37	305.19	304.37	304.15	303.82	303.64
Peak Water Temp (K)	303.22	303.31	303.31	303.29	303.24	303.22	303.20	303.18
Peak Void (1/100)	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Power Factor	1.00	2.28	2.24	2.06	1.23	1.01	0.68	0.49
Peak Water Velocity (m/s)	0.05	0.07	0.07	0.07	0.06	0.05	0.05	0.05

15.3 Please explain the use of 22kW per fuel element and why this is used instead of the hot rod power determined from the LCC.

The analysis for the response to RAI 15.3 was done using a power production of 22 kW because it was the original technical specification power per element. After receiving the analysis from CSM the highest single element power production factor for the limiting core is 22.17 kW ± 0.26 kW. Looking at the worst case scenario, by adding the error into the calculated power production factor, the limiting core has a peak power production of 22.43 kW. This value will now be used in the analysis attached to this RAI.

- The hot rod inventory was calculated using the fission yield factors for uranium-235, and the assumption of saturation conditions for the halogens and noble gases. However, NRC staff calculations using the fission yields in the Chart of the Nuclides, or those of the ENDF/B-VI in Summary Documentation Report (BNL-NCS-17541, ENDF-201, 1991) could not reproduce the licensee's radiological inventory. The major noted differences were in the estimation of Kr-85, where the NRC staff estimate was higher, and Br-82, where the licensee's estimate was higher. Please provide the fission yield data used, or explain how the source term inventory is calculated in sufficient detail to allow independent confirmation. Please explain whether GSTR is using 1 year of operation or saturated results.

The USGS is unable to confirm how the original hot rod inventory was calculated. Therefore the analysis has been redone and attached to this RAI. All pertinent assumptions and are be provided and the hot rod power production of 22.43 kW, from the limiting core analysis, will be used in the calculations.

- The calculations of offsite dose were based on an elevated release with the ventilation working. This analysis does not include a scenario that could lead to ground release which is typically included in TRIGA MHA dose calculations. In addition, an elevated release can only be used if the release point is 2½ times greater than the height of the adjacent solid structures, or higher (see RG 1.1.45); no statement is made concerning the applicability of this assumption to GSTR. GSTR is requested to provide the following:
  - There is no explanation of the HVAC system in SAR Section 9.1.3 including differentiating between normal exhaust and emergency exhaust. SAR Figure 9.1 refers to a "filtered exhaust" that employs a HEPA filter. However, in the MHA analysis, the release is assumed to be instantaneous with no HEPA filtration, or decay of fission product gases that were released into the reactor bay. Please clarify the assumptions used in the accident analysis regarding the HVAC system (e.g., normal ventilation or emergency ventilation mode of operation).
  - Please include in your revised response the public dose estimates assuming a ground release, or clarify why such estimates are not required.
  - Please include in your revised response a justification for using the assumption of elevated release.
  - Because the results of HotSpot calculations are input dependent, please provide the complete input scenario along with the source term used for all HotSpot calculations.
  - Please provide doses estimates for adjacent or nearby offices, where non-involved workers could be present or clarify why such estimates are not required.
  - If a decision is made to use all possible modes of HVAC operation, then evaluate corresponding occupational and public doses for all such modes and demonstrate that regulatory requirements of 10 CFR Part 20 are satisfied. Please state clearly all assumptions such as actuation speed, manual activities required, flow rates, damper conditions, fan conditions, etc.
  - RAI 17.1 response provides a distance to the fence line of 968 feet, as opposed to 350 meters in RAI 15.3 response. Please provide dose calculations based on consistent distances, or explain the differences.

To address the above concerns the USGS will provide the following in the attached analysis:

- All assumptions used in the calculation, including all information about the ventilation system operating during the MHA analysis.
- The analysis will be redone using a ground release as there are no bases for using an elevated release.
- The entire input code used for the Hot Spot analysis to allow for confirmatory calculations.
- Dose estimates for several areas on the Denver Federal Center near Building 15.
- The decision has been made that 295 meters (approximately 968 feet) is the distance from the reactor bay exhaust to the fence line, which is the nearest unrestricted access location for a member of the public. This distance has been updated in the RAI responses.

The following response will take the place of the sections numbered in the SAR for the USGS.

13.2 Accident Initiating Events and Scenarios, Accident Analysis, and Determination of Consequences

13.2.1 Maximum Hypothetical Accident (MHA)

13.2.1.1 Accident Initiating Events and Scenarios

A single fuel element could fail at any time during normal reactor operation or while the reactor is shutdown due to a manufacturing defect, corrosion, or handling damage. This type of accident is very infrequent, based on many years of operating experience with TRIGA® fuel, and such a failure would not normally incorporate all of the necessary operating assumptions required to obtain a worst-case fuel-failure scenario. Historically, TRIGA fuel failures have shown very small fission product releases.

For the GSTR, the MHA has been determined to be the cladding rupture of one highly irradiated fuel element with no radioactive decay followed by the instantaneous release of the noble gas and halogen fission products into the air. For the GSTR, with three different possible fuel types, a 12 wt% fuel element was chosen as the irradiated element since it contains the most ^{235}U and, hence, the highest inventory of fission products (50 g U, 19.75% enriched). The failed fuel element was assumed to have been operated at the highest core power density for the extremely conservative continuous period of one year at 1 MW. This results in all of the halogens and noble gases (except Kr-85) reaching their saturated activities.

This is the most severe accident and is analyzed to determine the limiting or bounding potential radiation doses to the reactor staff and to the general public in the unrestricted area. A less severe, but more credible accident, involving this same single element having a cladding failure in water will also be analyzed. This latter accident more correctly falls into the mishandling or malfunction of fuel accident category and will be addressed there.

During the lifetime of the GSTR, fuel within the core may be moved to new positions or removed. Fuel elements are moved only during periods when the reactor is shutdown. Also, the GSTR is very rarely operated continuously at 1 MW for a period longer than 12 hours, and never for a period of one year. Nevertheless, this extremely conservative MHA has been analyzed for the GSTR.

The following scenario has been chosen for analysis:

- A 12 wt% fuel element was chosen as the irradiated element since it contains the most ^{235}U and, hence, the highest inventory of fission products (50 g U, 19.75% enriched). The failed fuel element was assumed to have been operated at the highest core power density for a continuous period of one year at 1 MW in the limiting core resulting in 22.43 kW in the element. This results in all of the halogens and noble gases (except Kr-85) reaching their saturated activities. This scenario assumes that the noble gas and halogen fission products instantly and uniformly mix with the reactor room air. The fission products that have been released to the reactor room air are then exhausted at the stack ventilation rate of 800 cfm (3.78×10^5 cm^3sec^{-1}), through the emergency exhaust stack with no filtration taken into account. The air is assumed to be discharged at 6 meters (19.69 feet) above ground, at the exit of the exhaust stack. The reactor room free volume is assumed to be 3.1×10^8 cubic centimeters. The exhaust system takes 15.6 minutes to expel one reactor room volume of air (3.84 room changes per hour). The time to discharge 95% of the fission product gases from the reactor room is 47 minutes, but this analysis conservatively assumes that all fission product gases are released instantaneously in a single pulse discharge. Similarly, it is conservatively assumed that the gas concentration in the reactor bay undergoes no dilution during the maximum assumed stay time of 5 minutes.

13.2.1.2 Accident Analysis and Determination of Consequences

It is assumed that the GSTR is fueled with 12 wt% fuel elements, 100 fuel elements in the core, and that the reactor has operated continuously at 1 MW for a period of one year. Thus, all halogens and all noble gases (except Kr-85) are at their saturation activity. The highest-power density fuel element fails and releases the noble gases and halogens to the gap between the cladding and the fuel. This highest-power-density element has a conservative power density of 22.43 kW (Ref. 13.4). The fission product inventory of halogen and noble gases are given in Table 13.1 for this element. The inventory assumes a saturated activity is present and is based upon the fission yield for each isotope.

Considerable effort has been expended to measure and define the fission product release fractions for TRIGA® fuels. Data on this aspect of fuel performance are reported. Using these data, GA developed a conservative correlation for fission product release to be

$$e = 1.5\times 10^{-5} + 3.6\times 10^{3}\exp\left\{\frac{-1.34\times 10^{4}}{(T+273)}\right\}. \tag{13.1}$$

At an average fuel temperature of 526.37 °C, this release fraction is 2.04×10^{-4}. This assumed fuel temperature (526.37 °C) is the expected hot rod fuel temperature for our limiting core and will produce a conservative estimate for the fission product release.

Once the fission products are released to the cladding gap, this activity is released when the cladding catastrophically fails. If the release is in air (MHA), then this activity is released directly into the reactor bay air. If the release occurs in the pool water, then the fission products must migrate through the water before being released to the reactor bay air. Once released into the reactor bay air, a further reduction of the halogen activity is expected to occur due to plateout on the surfaces of the bay.

The fraction (w) of the fission product inventory released from a single fuel element that reaches the reactor room air and, subsequently, the atmosphere in the unrestricted environment is:

$$w = e\,f\,g\,h, \tag{13.2}$$

where:

- e = the fraction released from the fuel to the fuel-cladding gap (2.04×10^{-4});
- f = the fraction released from the fuel-cladding gap to the reactor bay air (if no water is present), or to the pool water (if water is present);
- g = the fraction released from the pool water to the reactor bay air (g=1.0 when no water is in the pool); and
- h = the fraction released from the reactor room air to the outside unrestricted environment, due to plateout in the reactor bay.

For the accident where the cladding failure occurs in air, it is very conservatively assumed that 25% of the halogens released to the cladding gap are eventually available for release from the reactor bay to the outside environment. This value is based on historical usage and recommendations. It uses a 50%

release of the halogens from the gap to the air with a natural reduction factor of 50% due to plateout in the reactor bay. Combining the 50% release from the gap with the 50% plateout results in the 25% total release. However, this value appears to be quite conservative, as some references quote a 1.7% release from the gap rather than 50%. In the reactor bay it is conservatively assumed that 50% of the halogens released to the cladding gap are released into the reactor bay.

For the accident in air, 100% of the noble gases are assumed to be available for release to the reactor bay and later the unrestricted environment.

For the accident in water, it is assumed that 95% of the halogens released from the cladding gap remain in the water and are removed by the demineralizer. A small fraction, 5%, of the halogens is assumed to escape from the water to the reactor room air. Combining this with the 50% release from the gap to the water, the result is that 2.5% of the halogens in the gap are released to the reactor room. Again, 50% of these plateout in the reactor bay before release to the outside environment. Thus a total of 1.25% of the halogens is available for release to the outside environment. For the noble gases released under water, 100% are assumed to be available for release to the unrestricted environment.
The experience at Three Mile Island, along with recent experiments, indicate that the 50% halogen release fraction is much too large. Possibly as little at 0.06% of the iodine reaching the cladding gap may be released into the reactor bay due in part to a large amount of the elemental iodine reacting with cesium to form CsI, a compound much less volatile and more water soluble than elemental iodine.

The very conservative values for these various release fractions (see Equation 13.2) are given in Tables 13.2 and 13.3.

Table 13.1
Saturated Activities for Highest Power Density 12 wt% Fuel Element

Isotope	Half Life	Saturated Activity (Ci)
Br-82	35.3 h	[redacted]
Br-83	2.4 h	
Br-84m	6.0 min	
Br-84	31.8 min	
Br-85	2.87 min	
Br-86	55.5 sec	
Br-87	55.9 sec	
Total Bromine		
I-131	8.02 d	
I-132	2.28h	
I-133	20.8 h	
I-134	52.6 min	
I-135	6.57 h	
I-136	83.4 sec	
Total Iodine		
Kr-83m	1.86 h	
Kr-85m	4.48 h	
Kr-85	10.76 yr	
Kr-87	76.2 min	
Kr-88	2.84 h	
Kr-89	3.15 min	
Total Krypton		
Xe-131m	11.9 d	
Xe-133m	2.19 d	
Xe-133	5.24 d	
Xe-135m	15.3 min	
Xe-135	9.1 h	
Xe-137	3.82 min	
Xe-138	14.1 min	
Total Xenon		
Total Halogens		
Total Noble Gases		

Table 13.2
Release Fraction Components

Fission product	f No pool water	F With pool Water	G No pool water	g With pool water	h
Noble gas	1.0	1.0	1.0	1.0	1.0
Halogens	0.5	0.5	1.0	0.05	0.5

Table 13.3
Total Release Fraction

Fission product	w to the reactor bay No pool water	w to the reactor bay With pool water	w to the environment No pool water	w to the environment With pool water
Noble gas	2.04 E -4	2.04 E -4	2.04 E-4	2.04 E-4
Halogens	1.02 E -4	5.10 E -6	5.10 E-5	2.55 E-6

For the GSTR, the prevailing wind is from the west, blowing to the east. The minimum distance to the unrestricted environment (295 m) is to the east, the minimum distance to the nearest public residence (640 m) is to the north, and a public school is about 720 m to the east. For this accident, therefore, it was assumed that the wind is blowing from west to east and all recipients are east.

The DOE HOTSPOT computer code version 2.07.2 was used for areas outside of the reactor bay, assuming uniform dispersion with ICRP 30 dose conversion factors. The HotSpot Health Physics Code was created for use for safety-analysis of DOE facilities handling nuclear material. Additionally, HotSpot provides emergency response personnel and emergency planners with a fast, field-portable set of software tools for evaluating incidents involving atmospheric releases of mixed isotopes of radioactive material. HotSpot incorporates Federal Guidance Reports 11, 12, and 13 (FGR-11, FGR-12, FGR-13) Dose Conversion Factors (DCFs) for inhalation, submersion, and ground shine. The results of the Hotspot analyses are provided in Table 13.7.

Furthermore, for calculations beyond the reactor bay, it was conservatively assumed that all of the fission products were released to the unrestricted area by a discharge pulse, which would maximize the dose rate to persons exposed to the plume during the accident. Calculations inside the reactor bay assumed uniform distribution of the released fission products within the $\geq 3.1 \times 10^8$ cc volume of the bay.

It was also assumed that the receptor breathing rate was 3.33 E-4 m^3sec^{-1} (NRC "light work" rate) and that the longest isotope retention category was applicable.

Calculations for personnel inside the reactor bay conservatively assumed that all of the fission product gases released were instantly and uniformly distributed within the reactor bay. The exposures for personnel in the reactor room for short stay-times (up to 5 minutes) were calculated by conservatively assuming that the fission product concentration was constant for that time period. The isotope concentrations in terms of DAC values and DAC-Hr exposures during a 2-minute stay time are given in Table 13.4 below. Values for 5 minute stay times are 2.5 times higher than the 2 minute stay time values since the fission product gas concentration is assumed to be constant during this exposure period.

Table 13.4
Concentrations and Exposures from Gaseous Fission Product Releases

Isotope	Saturated Activity (Ci)	Released Activity to reactor bay Air NO POOL WATER (mCi)	Released Activity to environment NO POOL WATER (mCi)	DAC from 10 CFR 20 Appendix B(uCi/ml)	DAC value of diluted activity in reactor bay no pool water (# DACs)	DAC-Hr exposure for 2 minute stay time	Released Activity to Reactor Room Air WITH POOL WATER (mCi)	Released Activity to environment WITH POOL WATER (mCi)	DAC value of diluted activity in reactor bay (with pool water) (# DACs)	DAC-Hr exposure for 2 minute stay time
Br-82	[redacted]	3.05	1.53	2.00E-06	4.92	0.16	0.15	0.08	0.25	0.01
Br-83	[redacted]	10.65	5.32	3.00E-05	1.14	0.04	0.53	0.27	0.06	0.00
Br-84m	[redacted]	0.40	0.20	1.00E-07	13.00	0.43	0.02	0.01	0.65	0.02
Br-84	[redacted]	19.86	9.93	2.00E-05	3.20	0.11	0.99	0.50	0.16	0.01
Br-85	[redacted]	26.19	13.09	1.00E-07	844.78	28.16	1.31	0.65	42.24	1.41
Br-86	[redacted]	39.14	-	1.00E-07	1262.52	42.08	1.96	-		
Br-87	[redacted]	50.79	-	1.00E-07	1638.50	54.62	2.54	-		
Total Bromine	[redacted]	150.08	30.07		3768.06	125.60	7.50	1.50	43.35	1.45
I-131	[redacted]	57.41	28.71	2.00E-08	9260.05	308.67	2.87	1.44	463.00	15.43
I-132	[redacted]	85.61	42.81	3.00E-06	92.06	3.07	4.28	2.14	4.60	0.15
I-133	[redacted]	133.10	66.55	1.00E-07	4293.51	143.12	6.65	3.33	214.68	7.16
I-134	[redacted]	156.27	78.13	2.00E-05	25.20	0.84	7.81	3.91	1.26	0.04
I-135	[redacted]	129.93	64.97	7.00E-07	598.77	19.96	6.50	3.25	29.94	1.00
I-136	[redacted]	125.47	-	1.00E-07	4047.50	134.92	6.27	-		
Total Iodine	[redacted]	687.80	281.16		18317.10	610.57	34.39	14.06	713.48	23.78
Kr-83m	[redacted]	21.30	21.30	1.00E-02	0.01	0.00	21.30	21.30	0.01	0.00
Kr-85m	[redacted]	52.38	52.38	2.00E-05	8.45	0.28	52.38	52.38	8.45	0.28
Kr-85	[redacted]	3.37	3.37	1.00E-04	0.11	0.00	3.37	3.37	0.11	0.00
Kr-87	[redacted]	101.59	101.59	5.00E-06	65.54	2.18	101.59	101.59	65.54	2.18
Kr-88	[redacted]	142.16	142.16	2.00E-06	229.30	7.64	142.16	142.16	229.30	7.64
Kr-89	[redacted]	187.92	187.92	1.00E-07	6061.97	202.07	187.92	187.92	6061.97	202.07
Total Krypton	[redacted]	508.71	508.71	-	6365.37	212.18	508.71	508.71	6365.37	212.18
Xe-131m	[redacted]	2.13	2.13	4.00E-04	0.02	0.00	2.13	2.13	0.02	0.00
Xe-133m	[redacted]	6.50	6.50	1.00E-04	0.21	0.01	6.50	6.50	0.21	0.01
Xe-133	[redacted]	258.43	258.43	1.00E-04	8.34	0.28	258.43	258.43	8.34	0.28
Xe-135m	[redacted]	40.00	40.00	9.00E-06	14.34	0.48	40.00	40.00	14.34	0.48
Xe-135	[redacted]	259.87	259.87	1.00E-05	83.83	2.79	259.87	259.87	83.83	2.79
Xe-137	[redacted]	245.77	245.77	1.00E-07	7927.91	264.26	245.77	245.77	7927.91	264.26
Xe-138	[redacted]	268.79	268.79	4.00E-06	216.76	7.23	268.79	268.79	216.76	7.23
Total Xenon	[redacted]	1081.48	1081.48		8251.40	275.05	1081.48	1081.48	8251.40	275.05
Total Halogens	[redacted]	837.87	311.24		22085.16	736.17	41.89	15.56	756.83	25.23
Total Noble Gases	[redacted]	1590.19	1590.19		14616.77	487.23	1590.19	1590.19	14616.77	487.23
Total Exposure for 2-minute stay time	[redacted]					1223.40				512.45

Since a stochastic exposure of 2000 DAC-Hr results in a TEDE of 5000 mrem, the TEDE in mrem can be calculated by

$$\text{TEDE} = (\text{DAC-Hr})*5000/2000. \qquad (13.3)$$

Since a non-stochastic exposure of 1 annual limit on intake (ALI) gives a CDE of 50,000 mrem for the target organ (thyroid for radioiodine) the dose received to the thyroid of a person standing in the reactor room can be calculated by

$$\text{CDE} = 3.33\text{E-}4*t*C/\text{ALI}*50000, \qquad (13.4)$$

where:

3.33E-4 = the NRC "light work" breathing rate with units of $m^3 sec^{-1}$;

t = the time exposed to the radionuclide;

ALI = the occupational inhalation limit for the specified isotope from 10 CFR 20 Appendix B;

and C = the concentration of the radionuclide in $\mu Ci/m^3$.

Table 13.5
Concentrations and Exposures from Iodine Radionuclides Releases

Isotope	Released Activity to reactor bay Air NO POOL WATER (mCi)	Non-stochastic ALI from 10 CFR 20 Appendix B for thyroid (iodine isotopes only) (uCi)	Ingested Iodine in reactor bay NO POOL WATER, 2 min stay time(uCi)	# ALIs ingested	Thryoid dose, 50000 mr per ALI ingested (mR)
I-131	57.41	50	7.40	1.48E-01	7400.64
I-132	85.61	8.00E+03	11.04	1.38E-03	68.98
I-133	133.10	3.00E+02	17.16	5.72E-02	2859.48
I-134	156.27	5.00E+04	20.14	4.03E-04	20.14
I-135	129.93	2.00E+03	16.75	8.37E-03	418.72
I-136	125.47	2.00E+02	16.17	8.09E-02	4043.45
Total Iodine	**687.80**	-	-	-	**14811.41**

The released amounts of iodine radionuclides in the reactor bay are shown in Table 13.5. A summary of the the $CDE_{Thyroid}$ and TEDE for 2- minute and 5-minute stay times in the reactor bay are shown in Table 13.6.

Table 13.6
Occupational $CDE_{Thyroid}$ and TEDE in the Reactor Room Following a Single Element Failure in Air and Water

Reactor Room Occupancy (minutes)	$CDE_{Thyroid}$ (no water) (mrem)	TEDE (no water) (mrem)	$CDE_{Thyroid}$ (water) (mrem)	TEDE (water) (mrem)
2	14811	3058	741	1281
5	37029	7646	1851	3203

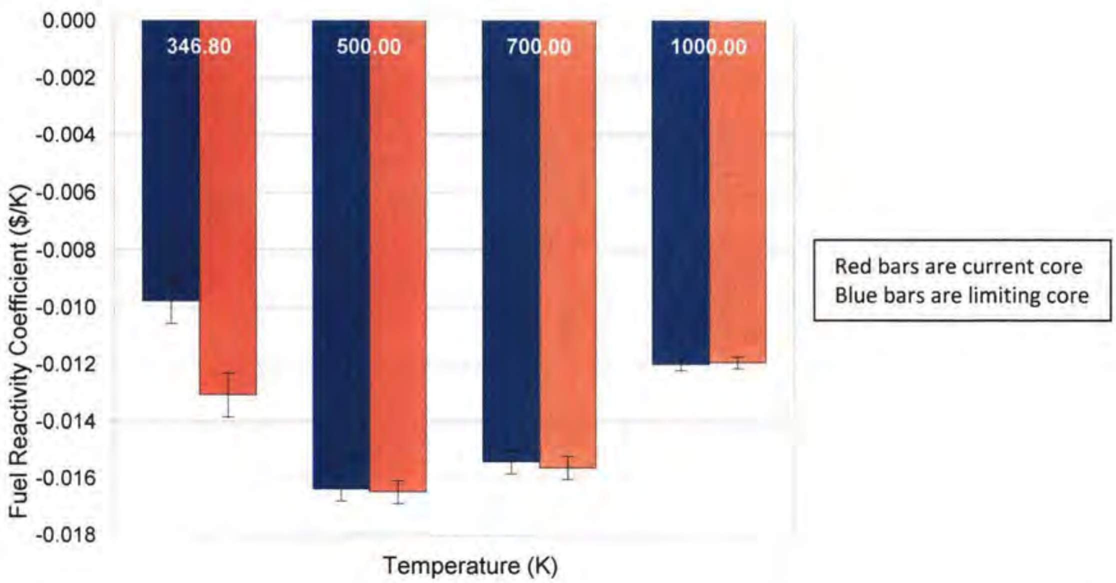
The results of the HOTSPOT code version 2.07.2 calculations for the two scenarios (no water vs water in reactor tank) are shown in Table 13.7. As seen from the tables, no water in the reactor gives the highest doses to the general public at any distance, as is expected since there is no capture of fission products by the water. The scenario with water in the reactor tank gives the lowest doses at any given distance since the capture and retention of fission products in the water is significant. In all cases, doses for the general public and occupational workers were all well below the annual dose limits specified by 10 CFR 20. For our model we used the following inputs:

- Atmospheric Dispersion Models: General plume model,
- Mixture of isotopes from Table 13.4, when requested the D categorization for the Br isotopes was used. Br-86, Br-87, and I-136 were not used in the calculation. It was assumed that those isotopes would not cause a significant dose as their half lives are too short (<84 sec) compared to the relative time it would take to travel out of the reactor bay and into the environment.
- Release height of 0 m for a ground release,
- A 10-meter wind speed of 3.84 m/s (average from Chapter 2 of the Safety Analysis Report),
- Wind is blowing from the west to the east,
- The ambient environment is moderately stable (F),
- Terrain is standard,
- Wind reference height is 10 m,
- Sample time is 10 min,
- Source geometry is simple,
- Include ground shine,
- The non-respirable deposition velocity is 8 cm/sec,
- The holdup time is 0 min,
- DCF library used was the FGR-11 corresponding to ICRP 30 series,
- The breathing rate is 3.33e-4 m^3/s,
- And all distances are on the plum center line for a conservative dose estimate at each location.

Table 13.7
Radiation Doses to Members of the General Public Following a Single Element Failure

Location	Distance (m)	$CDE_{Thyroid}$ (no water) (mrem)	TEDE (no water) (mrem)	$CDE_{Thyroid}$ (water) (mrem)	TEDE (water) (mrem)
Building 15 south door	11	350	360	18	18
Emergency assembly area	32	92	41	5.4	2.9
Building 21 east entrance (West of Building 15)	49	280	30	18	5.6
Average of eastern intersections	100	260	17	17	4.8
Building 16 west entrance	175	110	6.9	7.2	2.0
-	200	87	5.4	5.7	1.6
-	250	58	3.6	3.8	1.0
Fence	295	42	2.6	2.7	0.75
Residence	640	9.6	0.57	0.61	0.15
School	720	7.7	0.45	0.48	0.12

16.1 The value of the fuel temperature coefficient cited in SAR Table 13.7 was a linear function. As can be seen in Figure 1 below, General Atomics (GA) and NRC staff confirmatory analysis shows that this function was not linear. The GSTR linear function provides additional negative reactivity feedback at elevated temperatures that is not consistent with the GA or NRC staff confirmatory analysis. Please justify the use of the GSTR linear fuel temperature coefficient or provide a revised fuel temperature coefficient.



16.2 Please provide an explanation of why the 12 wt% fuel provides the limiting results for fuel used in the GSTR.

The 12 wt% fuel contains 41.1% more uranium by mass than the 8.5% fuel. When examining the burnup as well, the 8.5% fuel within the GSTR contains 41.1% less uranium than fresh 12wt% rod contains. This results in the 12 wt% fuel producing a higher power density than 8.5% fuel. This density translates to a higher fuel temperature, cladding temperature, heat flux, and reactivity worth.

17.1 Please provide the following:

- **The parameters used in determining the scattered dose at the fence line location (about 259 meters from the center of reactor bay). This should include all data and calculations with and without the credit for the 1 ft concrete wall of the reactor bay.**
- **In response to RAI 15.3, the fence line distance to the reactor bay is identified as 350 meters, as opposed to 968 feet ini the response to RAI 17.1. Provide the dose calculations based on a consistent value of the distance to a member of the public.**
- **The analysis of offsite dose was limited to one location beyond the GSTR fence line, and did not consider locations within the owner controlled area between the fence line and the reactor bay (e.g., parking lot, office locations, etc.) where individuals**

(members of the public) could be exposed. Please provide an analysis of the potential radiation exposure to individuals between the fence line and the reactor bay. Incorporate any assumptions, as described in the GSTR emergency plan, concerning evacuation of individuals from the owner controlled area.

- **The NRC staff notes what seems to be a typo in the definition of μ as 0/cm.**

Response to Question 17.1:

- The assumptions and parameters used in determining the scattered dose at multiple locations will be placed into the original response to the first RAI 17.1. This new response will be attached below.
- The decision has been made that 295 meters (approximately 968 feet) is the distance from the reactor bay exhaust to the fence line, which is the nearest unrestricted access location for a member of the public, even though this area is not generally occupied. This distance has been updated in the RAI responses.
- The response to this RAI (attached below) will include the dose at additional locations. All assumptions and parameters used in determining the scattered dose will be made available.
- There is not a typo in the definition of μ as 0/cm in Equation (1). The definition of μ is 0/cm because the analysis conservatively assumes that there is no shielding from the reactor components to yield a conservative flux of gamma rays at the scattering position.

The following will replace sections 13.2.3.2.2.3 through 12.2.3.2.2.5 in the SAR and includes changes made to the first RAI 17.1 to answer the second RAI 17.1:

13.2.3.2.2.3 Dose Rate Directly Above Core, Following a LOCA

During a Loss of Cooling Accident (LOCA) radiation from the reactor will scatter off the roof of the reactor bay and increase the dose rate at various locations. The first point of interest for calculating the dose rate is located at a point on the axis of the core cylinder at a distance of 746.8 cm from the top grid plate of the core. This is the distance from the top of the core to a point about 3 feet above the tank cover grating. The basic assumption for the calculations is that the reactor has been operating at a maximum power level of 1 MW for one continuous year, and then the cooling water is instantly lost. This is a very conservative assumption, since there is no conceivable way the GSTR could be operated continuously, 24 hours per day, at 1 MW for one year, nor is there any way all of the cooling water could be instantly lost. The GSTR normally operates on an 8-hour-per-day shift for 5 days per week.

The reactor core, shutdown and drained of water, was treated as a point source of 1-MeV photons. No accounting was made of sources other than fission product decay gamma rays, and no credit was taken for attenuation through the fuel, fuel element end pieces, and the upper grid plate. The first of these assumptions (point source of 1-MeVphotons) is optimistic, the second conservative (no attenuation), and the net effect is conservative. The equation to calculate the flux of gamma rays at the dose point is [1]

$$I = \frac{S*e^{-\mu h}}{4\pi x^2}, \quad (1)$$

where:
I = gamma flux intensity in $\gamma/cm^2/s$;
S = source strength in γ/s;
μ = core attenuation coefficient (0/cm, not accounted for in model to yield conservative calculation);
h = core height (28.1 cm); and
x = distance from top of core to dose point (746.8 cm).

The source strength is calculated from [2]

$$S = A*3.7e10 = 1.4e6*P(t^{-0.2}-(t+T)^{-0.2})*3.7e10, \quad (2)$$

where:

A = total fission product activity as a function of time (Ci);
P = reactor thermal power (1 MW);
t = time after shutdown (days); and
T = operating time (365 days).

The flux is calculated at 5 different times after shutdown: 10 seconds, 1 hour, 1 day, 1 week, and 1 month. After the gamma flux is calculated we can use a conversion factor to determine the effective dose equivalent rate. The dose conversion factor, K, for effective dose equivalent per unit photon fluence was obtained from ICRP 51, Table 2 [3]. This has been calculated for photons incident on an anthropomorphic phantom from various geometries. The worst case (highest dose factor) was for the anteroposterior geometry. For 1-MeV photons or gammas, the anteroposterior value of K is 4.60×10^{-12} Sv cm^2. The dose conversion factor, K, is energy dependent and the value was interpolated from the table. The effective dose equivalent rate is then calculated by multiplying the gamma flux times this K value, converting to rem (factor of 100 Rem/Sv) and converting to a time base of one hour by multiplying by a factor of 3600 s/hr. Using equations (1) and (2) the total fission product activity, source strength, flux, and effective dose equivalent are shown in Table 17.1.

Table 17.1: Total fission product activity, source strength, flux, and dose after shutdown

Time After Shutdown	Fission Product Activity (Ci)	Source Strength (γ/s)	Flux of gamma rays ($\gamma/cm^2/s$)	Dose 3 ft above grates (R/hr)
10 sec	8.15E+06	3.02E+17	4.30E+10	7.12E+04
1 hour	2.21E+06	8.19E+16	1.17E+10	1.93E+04

1 day	9.70E+05	3.59E+16	5.12E+09	8.48E+03
1 week	5.20E+05	1.92E+16	2.75E+09	4.55E+03
1 month	2.86E+05	1.06E+16	1.51E+09	2.50E+03

13.2.3.2.2.4 Dose Rate from Scattered Radiation in Reactor Bay.

The second point of interest for calculating the effective dose equivalent rate is located in the SE corner of the reactor bay, 3 ft above the floor, and 16 ft away from the vertical line intersecting the center of the core. This point is the furthest distance a person can get from the edge of the reactor and remain in the reactor bay. The ceiling immediately over the reactor tank is a staggered ceiling with a steel access hatch and concrete support. To yield a conservative dose calculation it will be modeled as a concrete slab located 18.3 ft above the floor of the reactor bay. In reality the scattering will not be as great as calculated because the radiation from the unshielded core will undergo less interaction with the roof. A representation of this model is shown in Figure 17.1. The initial assumptions from Section 13.2.3.2.2.3 will be used once again: the basic assumption for the calculations is that the reactor has been operating at a maximum power level of 1 MW for one continuous year, and then the cooling water is instantly lost. This is a very conservative assumption; since there is no conceivable way the GSTR could be operated continuously, 24 hours per day, at 1 MW for one year, nor is there any way all of the cooling water could be instantly lost. The GSTR normally operates on an 8-hour-per-day shift for 5 days per week. The reactor core, shutdown and drained of water, was treated as a point source of 1-MeV photons. No accounting was made of sources other than fission product decay gamma rays, and no credit was taken for attenuation through the fuel, fuel element end pieces, and the upper grid plate. The first of these assumptions is optimistic, the second conservative, and the net effect is conservative. The calculations for the flux of gamma rays, Equation 1, and the source term strength, Equation 2, from section 13.2.3.2.2.3 will be used again in this analysis.

For the dose position of interest, we are looking at gamma rays that are scattered at an angle of 46.28°, and travel a total distance of 22.1 ft (673.6 cm) from the scatter point to the position. A representation of this geometry is shown in Figure 17.2. Gammas that have an initial energy of 1 MeV and are scattered according to Figure 17.2 have a scattered energy calculated by [1]

$$E = \frac{Eo}{1+\frac{Eo(1-\cos(\beta))}{0.51}}, \tag{3}$$

where:
Eo = the initial energy of the gamma ray (1 MeV); and
β = the scattering angle of the gamma ray relative to the initial vector of travel (180° - 46.28° = 133.72°).

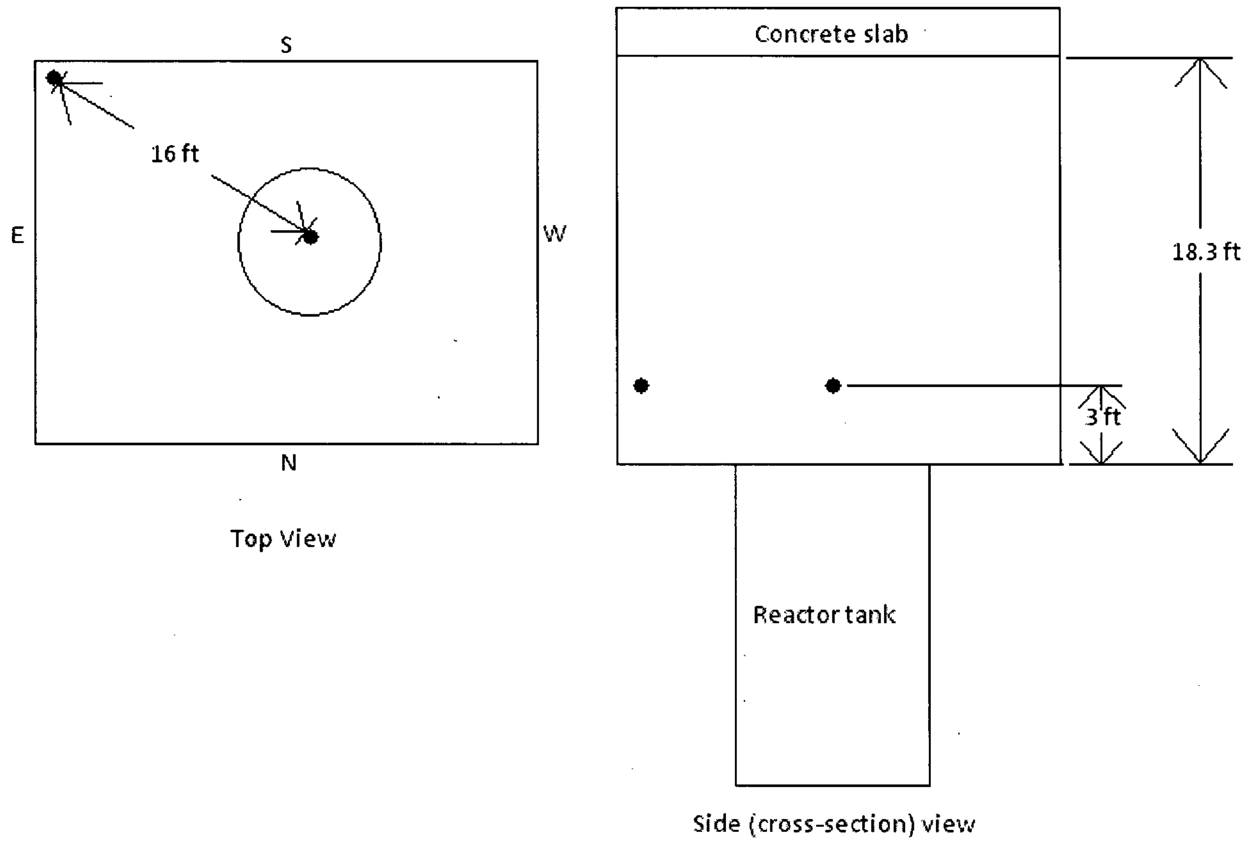


Figure 17.1 Top and side views of the model for calculating the scattered dose in the SE corner after a LOCA

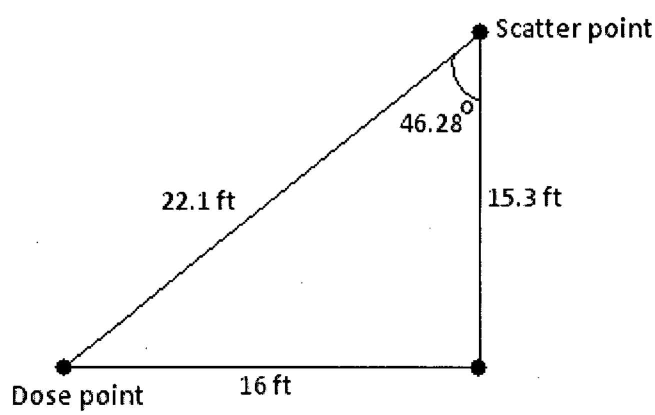


Figure 17.2 Representation of the geometry for the calculations

Using Equation 3, the resulting energy of the scattered gamma rays is 0.232 MeV. The flux of gamma rays that are scattered from the interaction with the concrete slab is calculated by [1]

$$I = \frac{6.03e23*\rho*Z*Io*C}{\frac{A*d^2*(\mu o + \mu 1*\cos(\theta o))}{\cos(\theta 1)}} * \frac{d\sigma}{d\Omega}, \tag{4}$$

where:
ρ = the density of the scattering material (concrete ρ = 2.35 g/cm^3) [4];
Io = the flux of gamma rays at the scatter point determined by Equation (1) with x = 1213.1 cm (746.8 cm + 466.3 cm);
C = cross sectional area of the incident beam (cm^2);
Z/A = ratio of the average atomic number to the atomic mass (~0.5 for light elements such as concrete);
d = distance from scatter point to dose point (22.1 ft = 673.6 cm);
μo = attenuation coefficient in scattering material for incident gamma rays (0.150/cm) [5] ;
μ1 = attenuation coefficient in scattering material for scattered gamma rays (0.284/cm) [5];
θo = incident angle, measured from normal to incident gamma rays on the concrete (0°);
θ1 =incident angle, measured from normal of the concrete to the dose point (46.28°); and
$d\sigma/d\Omega$ = is the Klein-Nishina formula for scattering cross section from a single electron (cm^2).

For Equation (4) the incident gamma beam is conservatively assumed to be collimated by the reactor tank and equal to the cross sectional area of the reactor tank. Therefore, C equals 41764.6 cm^2 and is calculated by

$$C = \pi*R^2; \tag{5}$$

where R = the tank radius (115.3 cm).

The Klein-Nishina formula is calculated by [1]

$$\frac{d\sigma}{d\Omega} = \frac{r^2}{2}*\left(\frac{E}{Eo} - \frac{E^2*\sin^2\beta}{Eo^2} + \frac{E^3}{Eo^3}\right), \tag{6}$$

where:
r = the classical electron radius (2.82e-13 cm);
E = scattered gamma energy from equation (3) (0.232 MeV);
Eo = incident gamma energy (1 MeV); and
β = the scattering angle of the gamma ray relative to the initial vector of travel (180° - 46.28° = 133.72°).

Using Equations (1), (2), (3), (4), (5), and (6) the flux of gamma rays incident upon the SE corner dose point can be calculated. Then as before, the ICRP 51, Table 2 dose factors were used to calculate the effective dose equivalent at that position. The anteroposterior geometry was used as it gave the largest dose factor for 0.232-MeV scattered photons, of 1.20 × 10^{-12} Sv cm^2. Unit conversion factors of 100 Rem/Sv and 3600 s/hr were also applied. The flux and effective dose equivalent for the SE corner position from Figure 17.1 are shown in Table 17.2.

Table 17.2: Flux and dose after shutdown for SE corner

Time After Shutdown	Flux of gamma rays in SE corner ($\gamma/cm^2/s$)	Effective dose equivalent for SE corner (R/hr)
10 sec	1.63E+07	7.01
1 hour	4.42E+06	1.90
1 day	1.94E+06	0.83
1 week	1.04E+06	0.45
1 month	5.70E+05	0.25

13.2.3.2.2.5 Dose Rate from Scattered Radiation at the Eastern Federal Center Fence and Other Areas within the Denver Federal Center

The third point of interest for calculating the effective dose equivalent rate is located at the eastern DFC fence, not in the direct beam from the exposed core, but subject to scattered radiation from the reactor bay ceiling, as in section 13.2.3.2.2.4. The dose point is chosen to be 3 feet above the ground at the closest location along the fence line, where a member of the public could stand. The distance to this point from the center of the reactor bay ceiling above the reactor tank is roughly 968 ft. A representation of this model is shown in Figure 17.3.

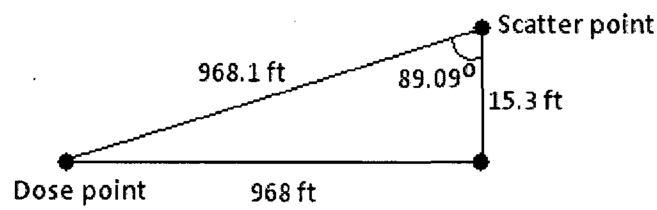


Figure 17.3 Representation of the geometry for the dose calculations at the eastern Federal Center fence.

All assumptions from section 13.2.3.2.2.4 will be utilized. The ceiling immediately over the reactor tank is a staggered ceiling with a steel access hatch and concrete support. To yield a conservative dose calculation it will be modeled as a concrete slab located 18.3 ft above the floor of the reactor bay. In reality the scattering will not be as great as calculated because the radiation from the unshielded core will undergo less interaction with the roof. The initial assumptions from Section 13.2.3.2.2.3 will be used once again: the basic assumption for the calculations is that the reactor has been operating at a maximum power level of 1 MW for one continuous year, and then the cooling water is instantly lost. This is a very conservative assumption, since there is no conceivable way the GSTR could be operated continuously, 24 hours per day, at 1 MW for one year, nor is there any way all of the cooling water could be instantly lost. The GSTR normally operates on a 8-hour-per-day shift for 5 days per week. The reactor core, shutdown and drained of water, was treated as a point source of 1-MeV photons. No accounting was made of sources other than fission product decay gamma rays, and no credit was taken for attenuation through the fuel, fuel element end pieces, and the upper grid plate. The first of these assumptions is optimistic, the second conservative, and the net effect is conservative. The calculations for the flux of gamma rays, Equation 1, and the source term strength, Equation 2, from section 13.2.3.2.2.3 will be used again in this analysis. Also, the Equations (3), (4), (5), and (6) from section 13.2.3.2.2.4 will be used again in this analysis with the different values listed below:

x = distance from top of core to scattering point which is 1213.1 cm;
θ1 =incident angle, measured from normal to dose point which is 89.09°;
β = the scattering angle of the gamma ray relative to the initial vector of travel which is 90.91° (180° - 89.09° = 90.91°);
E = resulting energy of the scattered gamma rays which is 0.334 MeV, from Equation (3);
μ1 = attenuation coefficient in scattering material for scattered gamma rays which is 0.244/cm [5];
dσ/dΩ = is the Klein-Nishina formula for scattering cross section from a single electron which is 1.032e-26 cm^2, from Equation (6); and
K = is the dose factor for 0.334 MeV gamma rays which is 1.73×10^{-12} Sv cm^2, from ICRP 51, Table 2.

Also, for the eastern DFC fence dose point, there is an attenuation factor applied, due to the attenuation of the gamma rays through the 1 ft thick concrete wall of the reactor bay. Attenuation from the remainder of the building structure, the air, and environmental components between the building and fence are conservatively ignored. The attenuation factor is applied according to [4]

$$I = I_o * e^{(-\mu 1 * x)} \qquad (7)$$

where:
I = flux of gamma rays at fence with attenuation;
I_o = flux of gamma rays at fence without attenuation;
μ1 = attenuation coefficient in concrete for the scattered gamma rays (0.244/cm) [5]; and
x = thickness of the concrete wall (1 ft = 30.48 cm).

Table 17.3 shows the flux and effective dose equivalent for the eastern Federal Center fence position.

Table 17.3: Flux and dose after shutdown for eastern fence

Time After Shutdown	Flux of gammas at fence (γ/cm^2/s)	Effective dose equivalent at fence (mR/hr)
10 sec	0.21643	1.35e-4
1 hour	0.05878	3.66e-5
1 day	0.02576	1.60e-5
1 week	0.01381	8.60e-6
1 month	0.00759	4.72e-6

The same procedure as listed above can be recreated for multiple positions within the Denver Federal Center. Several locations have been picked to be analyzed for dose rate due to scattered radiation: Building 15 front door, emergency assembly area, Building 21, average distance of the eastern intersections, and Building 16. Table 17.4 lists the location, distance, and other variables that have changed during the calculation. It is conservatively assumed that no member of the public could stand more than 24 hours in any given location during an emergency, as they would be escorted from the premises by FPS or reactor staff. Thus, using equations (1) through (7) and looking at the dose rate at the front door of Building 15 per each second in a 24 hour period the total dose a member of the public could receive standing in that location for 24 hours, following a LOCA, would be 2.01 mrem.

Table 17.4: Dose rate at various positions on the Denver Federal Center due to scattered radiation 1 second after a LOCA

Distance (m)	Location	E (MeV)	β (180-θ1)	d (cm)	θ1	I (γ/cm²/s)	μ1 (1/cm)	dσ/dΩ	K (Sv cm²)	Iatt (γ/cm²/s)	Dose (mR/hr)
11	Building 15 south door	0.268	112.974	1194.77	67.026	3640705.58	0.268	9.012E-27	1.39E-12	1031.78755	5.16E-01
32	Emergency assembly area	0.308	98.291	3233.80	81.709	236405.77	0.252	9.723E-27	1.60E-12	109.10874	6.28E-02
49	Building 21 east entrance	0.318	95.437	4922.14	84.563	71198.35	0.249	9.933E-27	1.65E-12	36.00670	2.14E-02
100	Average of eastern intersections	0.328	92.670	10010.87	87.330	8976.81	0.247	1.017E-26	1.70E-12	4.82514	2.95E-03
175	Building 16 west entrance	0.332	91.526	17506.21	88.474	1737.23	0.244	1.027E-26	1.72E-12	1.02319	6.34E-04

References

[1] Introduction to Nuclear Engineering, Richard Stephenson, McGraw-Hill, 1954, pp. 182-200.

[2 Introduction to Nuclear Engineering, 2nd Edition, John R. Lamarsh, Addison-Wesley, 1983, p. 72.

[3] "Data for Use in Protection Against External Radiation," International Commission on Radiation Protection, ICRP Report No. 51, Pergamon Press, March 1987.

[4] Introduction to Health Physics, Herman Cember, Thomas E. Johnson, Fourth Edition, McGraw-Hill, 2009, pp. 167-167.

[5] Radiological Health Handbook, Revised Edition, U.S. Department of Health, Education, and Welfare, U.S. Government Printing Office, 1970, pp.137-140.

23.1 The response to this RAI does not provide criteria for determining what constitutes a significant change in core configuration. Please provide and justify the criteria chosen.

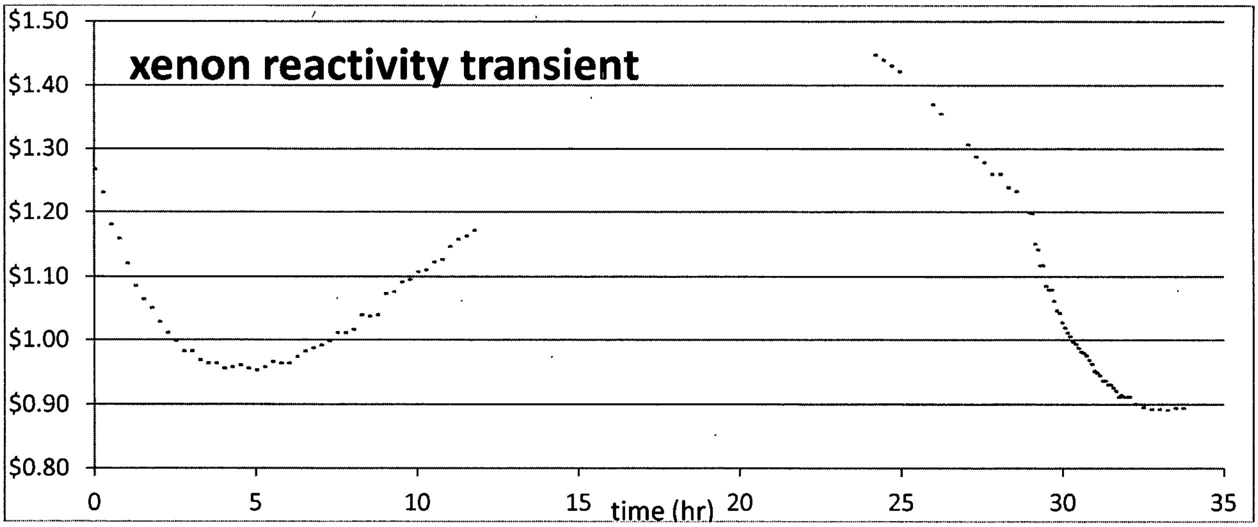
Significance is determined by the potential for violation of Technical Specifications. This could occur, for example, when core fuel loading is changed or an in-core experimental facility is moved, especially when that change is near a control rod location. Loading or unloading of experiment samples is not normally considered sufficient reason for requiring control rod recalibration. When a relatively large shutdown margin exists, the potential for Technical Specification violation is significantly reduced. Therefore, if a numeric value must be assigned to "significance", we propose that a significant change be defined as a positive reactivity change of at least one-half of the shutdown margin that exists prior to the change.

For example, if the shutdown margin is $1.30, then a positive reactivity change of $0.65 or more would constitute a significant change.

24.3 The proposed SDM of -$0.30 is less than the guidance provided in NUREG-1537 (-$0.50). That guidance is predicated on the licensee's ability to be capable of accurately measuring reactivity +$0.50. To justify the SDM of -$0.30, please explain or demonstrate USGS's ability to consistently discern a reactivity change of this magnitude.

The GSTR performs several reactivity measurement activities that demonstrate its ability to consistently measure a reactivity change of much less than $0.30. The graph below is excess reactivity measurements over the first 9 months of CY2012. The measurements were taken on days when there was minimal (<$0.10) of xenon reactivity, minimal (<$0.05) sample reactivity, and the pool temperature was ~20C. You can very clearly see the reactivity loss due to fuel burnup as the year progressed.

A second demonstration of our reactivity measurement abilities is the xenon transient curve shown below. These data show reactivity measurements made in October, 2012, over a 34-hour period of high-power operation, shutdown, and low-power operation. Visual inspection of the plot shows a precision of ~ $0.02 in the reactivity measurements.



These data clearly demonstrate our ability to measure a reactivity change of less than $0.30.

24.9 The NRC staff observed that the input to COMPLY has resulted in a failed calculation for screening level 1, and that for screening level 2, the RAI response provided an input of 2.266e-6 Ci/sec. The NRC staff cannot reproduce this number using the technical specification values for allowed release concentration, volume, and flow rates. Furthermore, even if this numerical value is correct, the NRC staff notes that the calculated exposure from COMPLY is 0.5 mrem, and not 5 mrem, the value in the technical specifications. Please provide an RAI response that demonstrates that the technical specification limit on release concentrations is justified by the statements in the basis.

The Basis section under Section 14.3.7.2 from the original submitted SAR was replaced with the answer to RAI #24.9. The submission to RAI 24.9 will be now modified as shown below to respond to the second RAI #24.9. The response below will take the place of the Basis section under Section 14.3.7.2 in the SAR.

Basis. If ^{41}Ar is continuously discharged at 4.8 x 10^{-6} µCi/ml, measurements and calculations show that ^{41}Ar released to publicly accessible areas under the worst-case weather conditions would result in an annual TEDE of 0.7 mrem. This is only 7% of the applicable limit of 10 mrem. The calculation was performed with the Environmental Protection Agency's COMPLY code. The following input parameters were used:

Nominal exhaust flow: 1000 cfm,
Ar-41 release in Ci/s: ((4.8 x 10^{-6} µCi/ml)(1000 cfm)(1/60 min/sec)(1/(1 x 10^{6})Ci/ µCi)(28316.85 ml/ft^{3}))= 2.266e-6 Ci/s,
Release height: 6 meters,
Building height: 4 meters,
Distance from source to the receptor: 295 meters,
Building width: 30 meters,
Default mean wind speed: 2.0 m/sec.

Using the above input parameters the USGS passes the EPA's Comply code at level 2. This is shown in the COMPLY code report included below. Using level 2, EPA's COMPLY code, and the above input parameters, the dose from the Ar-41 exhaust was also calculated at various distances from the exhaust stack. The calculated dose is shown in Table 1 and in the far right column an occupancy factor of 10%

has been applied to the dose. The occupancy factor comes from the fact that the Denver Federal Center (DFC) is not occupied all week long and it is constantly monitored by the Federal Protective Service. Anyone loitering in an area would be questioned and asked to leave; therefore, the occupancy factor is estimated to be 10%.

Table 1: Yearly dose due to Ar-41 release limit at several distances with occupancy factor applied. All yearly doses calculated with EPA's COMPLY code.

Distance (m)	Dose (mrem/yr)	Location	Dose with 10% Occupancy Factor (mrem/yr)
11	135	Building 15 south door	13.5
32	16.7	Emergency assembly area	1.67
49	10.4	Building 21 east entrance	1.04
100	4.1	Average of eastern intersections	0.41
175	1.8	Building 16 west entrance	0.18

COMPLY: V1.6. 11/ 5/2012 2:30

40 CFR Part 61
National Emission Standards
for Hazardous Air Pollutants

REPORT ON COMPLIANCE WITH
THE CLEAN AIR ACT LIMITS FOR RADIONUCLIDE EMISSIONS
FROM THE COMPLY CODE - V1.6.

Prepared by:
USGS GSTR
PO Box 25046, DFC MS-974
Alex Buehrle
303-236-4726

COMPLY: V1.6. 11/ 5/2012 2:30

Ar-41 release 4.8e-6 uCi/ml for 1 year

SCREENING LEVEL 1

DATA ENTERED:

Effluent concentration limits used.

```
         CONCENTRATION
Nuclide  (curies/cu m)
-------- --------------
AR-41    4.80E-06
```

NOTES:

Input parameters outside the "normal" range:
None.

RESULTS:

You are emitting 706.0 times the allowable amount
given in the concentration table.
*** Failed at level 1.

COMPLY: V1.6. 11/ 5/2012 2:30

Ar-41 release 4.8e-6 uCi/ml for 1 year

SCREENING LEVEL 2

DATA ENTERED:

Nuclide	Release Rate (curies/SECOND)
AR-41	2.266E-06

Release height 6 meters.
Building height 4 meters.
The source and receptor are not on the same building.
Distance from the source to the receptor is 295 meters.
Building width 30 meters.
Default mean wind speed used (2.0 m/sec).

NOTES:

Input parameters outside the "normal" range:
None.

RESULTS:

Effective dose equivalent: 0.7 mrem/yr.

*** Comply at level 2.

This facility is in COMPLIANCE.

It may or may not be EXEMPT from reporting to the EPA.

You may contact your regional EPA office for more information.
********** END OF COMPLIANCE REPORT **********

26.d & e The use of an acute airborne release model (HotSpot) for chronic airborne release analysis does not appear applicable as HotSpot Version 2.07 was intended to estimate doses from short-term airborne releases of less than a few hours following nuclear accidents and significant releases, and was not designed to estimate doses from annual releases. It does not provide conservative best estimates of the annual dose to individuals. Further, the input parameters shown in the response include deposition velocities for large and small particle material, while Ar-41 (as a noble gas) is not associated with particles. If computational analysis is desired then CAP-88 PC (The U.S. EPA COMPLY software) that is commonly used for clean air act compliance calculations is appropriate. Alternatively, Regulatory Guide 1.111 provides another acceptable method. Please provide dose calculations using acceptable code.

GSTR staff questioned the LLNL HotSpot experts on the use of HotSpot for the Ar-41 release dose estimate and received the following reply, *"Both MACCS and CAP88 along with HotSpot's percentile dose capability will divide an annual dose based on meteorological conditions, and report fractional doses based either on wind sector or percentiles. Given that you have already run an estimate of your entire inventory, i.e., an extreme worst case, and still did not reach significant health effect levels there does not seem to be any point in re-running your scenario in either of the other two models or through HotSpot's percentile dose feature as your resulting doses will be much lower than what you have already generated. We recommend using a respirable deposition velocity of 0 cm/s (not 0.3 cm/s) for Ar-41. However, for "B" stability and your stack parameters, there is no difference in the TEDE values even out to 80 km. The 8 cm/s respirable deposition velocity does not impact your calculations because you assumed (correctly) that 100% of the release is respirable. The non-respirable deposition velocity only impacts the non-respirable component of the AF."*

Based on this response, I believe the HotSpot code is a valid code that provides results at least as good as the EPA's codes. The HotSpot code was rerun with the distance to the public receptor changed to 0.295 km (to be consistent with other RAI responses) and the respirable deposition velocity changed to zero. The input parameters and output value are given below.

HotSpot Version 2.07.1 General Plume
Nov 16, 2012 10:49 AM
Source Material : Ar-41 1.0961E+02 m
Material-at-Risk (MAR) : 7.7500E+00 Ci
Damage Ratio (DR) : 1.000
Airborne Fraction (ARF) : 1.000
Respirable Fraction (RF) : 1.000
Leakpath Factor (LPF) : 1.000
Respirable Source Term : 7.75E+00 Ci
Non-respirable Source Term : 0.00E+00 Ci
Effective Release Height : 6.40 m
Wind Speed (h=10 m) : 3.84 m/s
Distance Coordinates : All distances are on the Plume Centerline
Wind Speed (h=H-eff) : 3.72 m/s
Stability Class : B
Respirable Dep. Vel. : 0.00 cm/s
Non-respirable Dep. Vel. : 8.00 cm/s
Receptor Height : 1.5 m
Inversion Layer Height : None

Sample Time : 10.000 min
Breathing Rate : 3.33E-04 m3/sec

Maximum Dose Distance : 0.034 km
MAXIMUM TEDE : 2.30E-03 rem
Inner Contour Dose : 1.0 rem
Middle Contour Dose : 0.500 rem
Outer Contour Dose : 0.100 rem
Exceeds Inner Dose Out To : Not Exceeded
Exceeds Middle Dose Out To : Not Exceeded
Exceeds Outer Dose Out To : Not Exceeded

FGR-11 Dose Conversion Data - Total Effective Dose Equivalent (TEDE)

DISTANCE	T E D E	RESPIRABLE TIME-INTEGRATED AIR CONCENTRATION	ARRIVAL TIME
km	(rem)	(Ci-sec)/m3	(hour:min)
0.034	2.3E-03	9.6E-03	<00:01
0.100	7.2E-04	3.0E-03	<00:01
0.200	2.0E-04	8.3E-04	<00:01
0.295	9.4E-05	3.9E-04	00:01
0.640	2.0E-05	8.5E-05	00:02