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

February 8, 2013

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: Follow-up responses from questions of phone conference on 12/20/12

Mr. Wertz:

Responses to questions from the reference phone conference 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 2/8/13

Attachment

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

A020
NRR

GSTR RAI Responses: February 8, 2013

9. Core map showing the contents of the core lattice positions for the LCC and the OC:

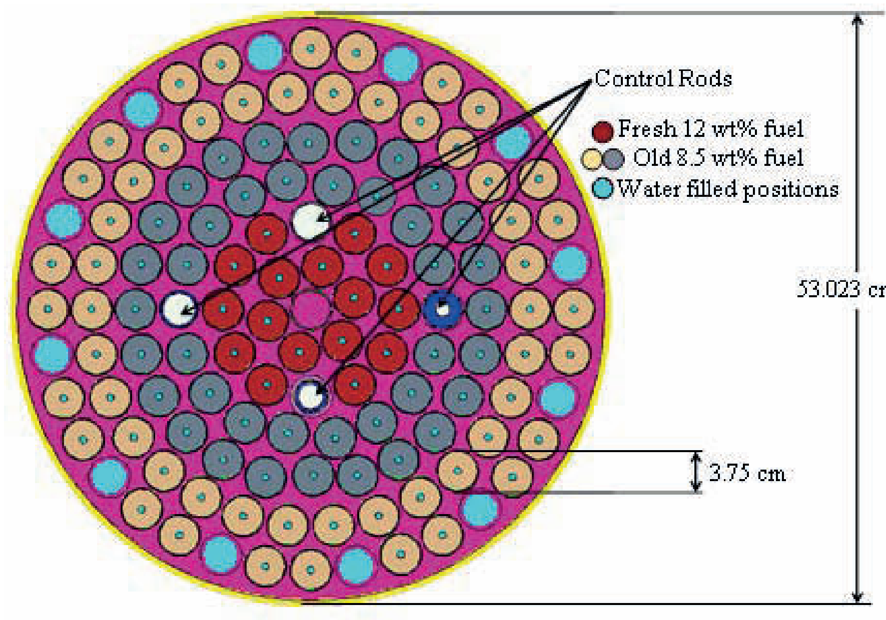


Figure 1, limiting core configuration core layout

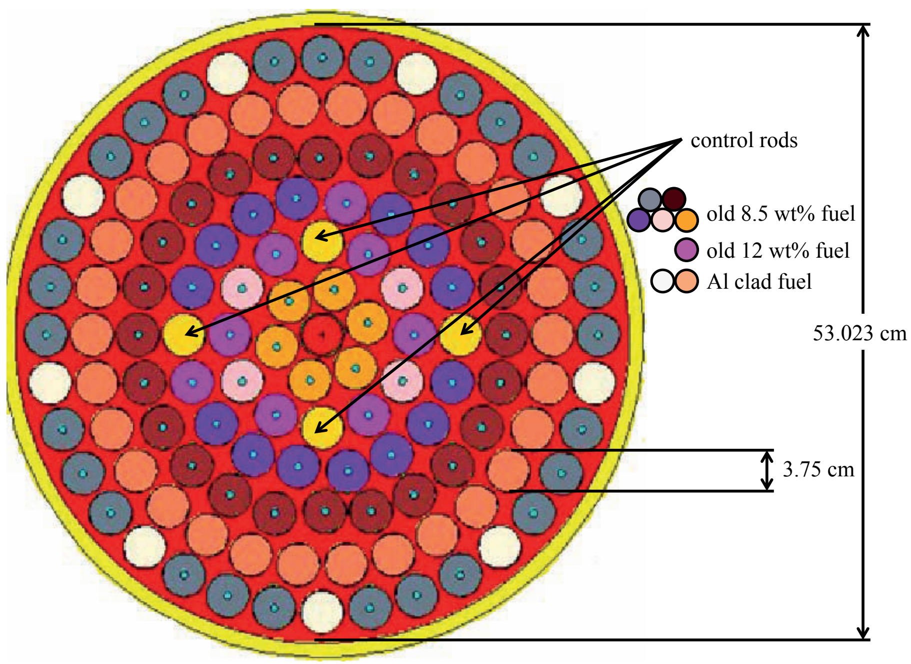
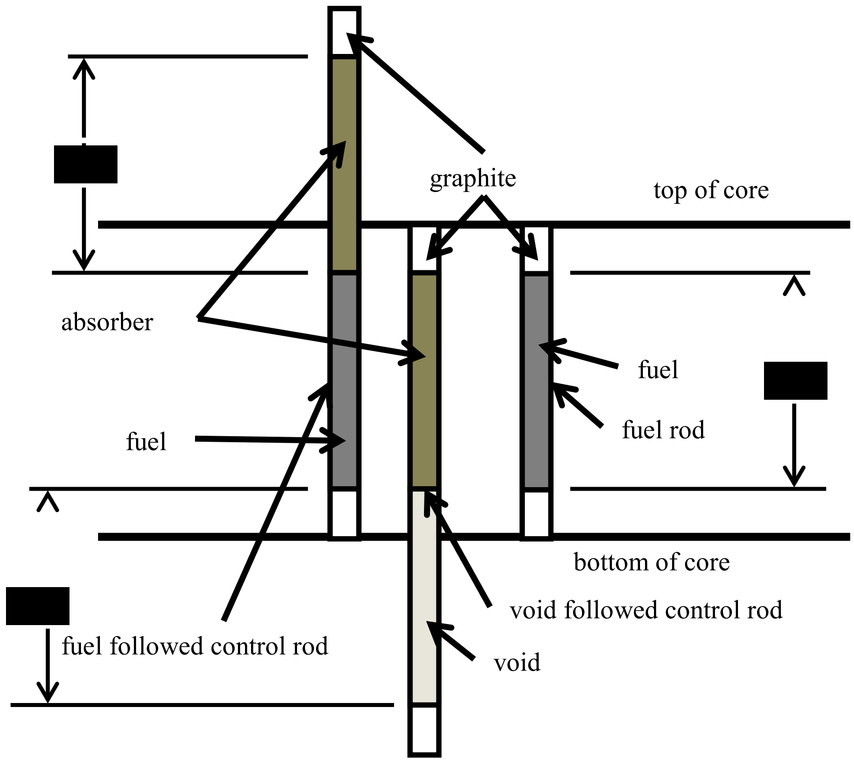
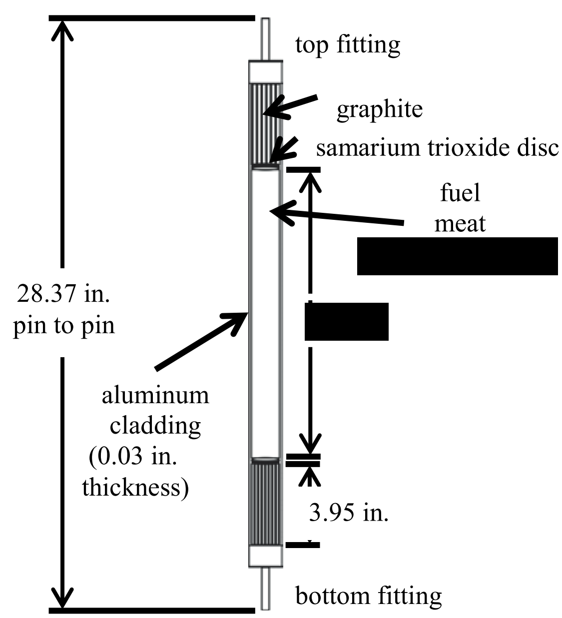
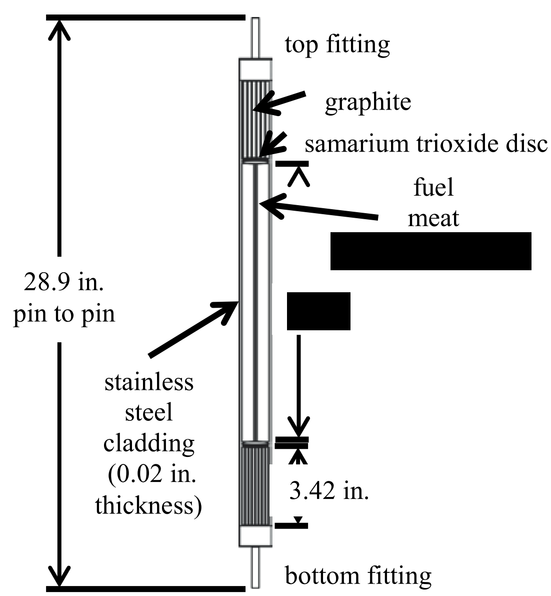


Figure 2, operating core configuration core layout

- 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 analysis is 0.00728 as calculated from MCNP. The difference between the calculated values of the LCC and OC are within the uncertainty bounds calculated by MCNP; the same value is used for both.
- The all-control-rods-out k-effective (k_{eff}) and the excess reactivity (ρ_{excess}) 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 worths for each of the 4 control rods including the k_{eff} values determined for the LCC and the OC. The calculations for the OC were taken with the other control rods at a position of 18.669 cm up from the fully inserted position. For the LCC the rod position was 17.3355 cm up from the fully inserted position.

  - LCC:

	Shim 1	**Shim 2**	**Regulating**	**Transient**
Rod Worth ($)	2.40	2.24	4.25	2.62
k_{eff} (fully inserted)	0.98644	0.98769	0.97735	0.98561
k_{eff} (fully withdrawn)	1.00388	1.0040	1.00832	1.00470

  - OC:

	Shim 1	**Shim 2**	**Regulating**	**Transient**
Rod Worth ($)	2.20	2.29	3.42	2.09
k_{eff} (fully inserted)	0.99944	1.00051	0.99691	1.00056
k_{eff} (fully withdrawn)	1.01542	1.01718	1.02182	1.01578

- 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 ($)	2.20	2.29	3.42	2.09
Experimental Worth ($)	2.305	2.435	3.630	2.142

- Shutdown reactivity of the operating core with the highest-worth control rod (regulating rod) withdrawn is $1.30.

Power Distribution Graphics for the OC & LCC showing power in kW per fuel element

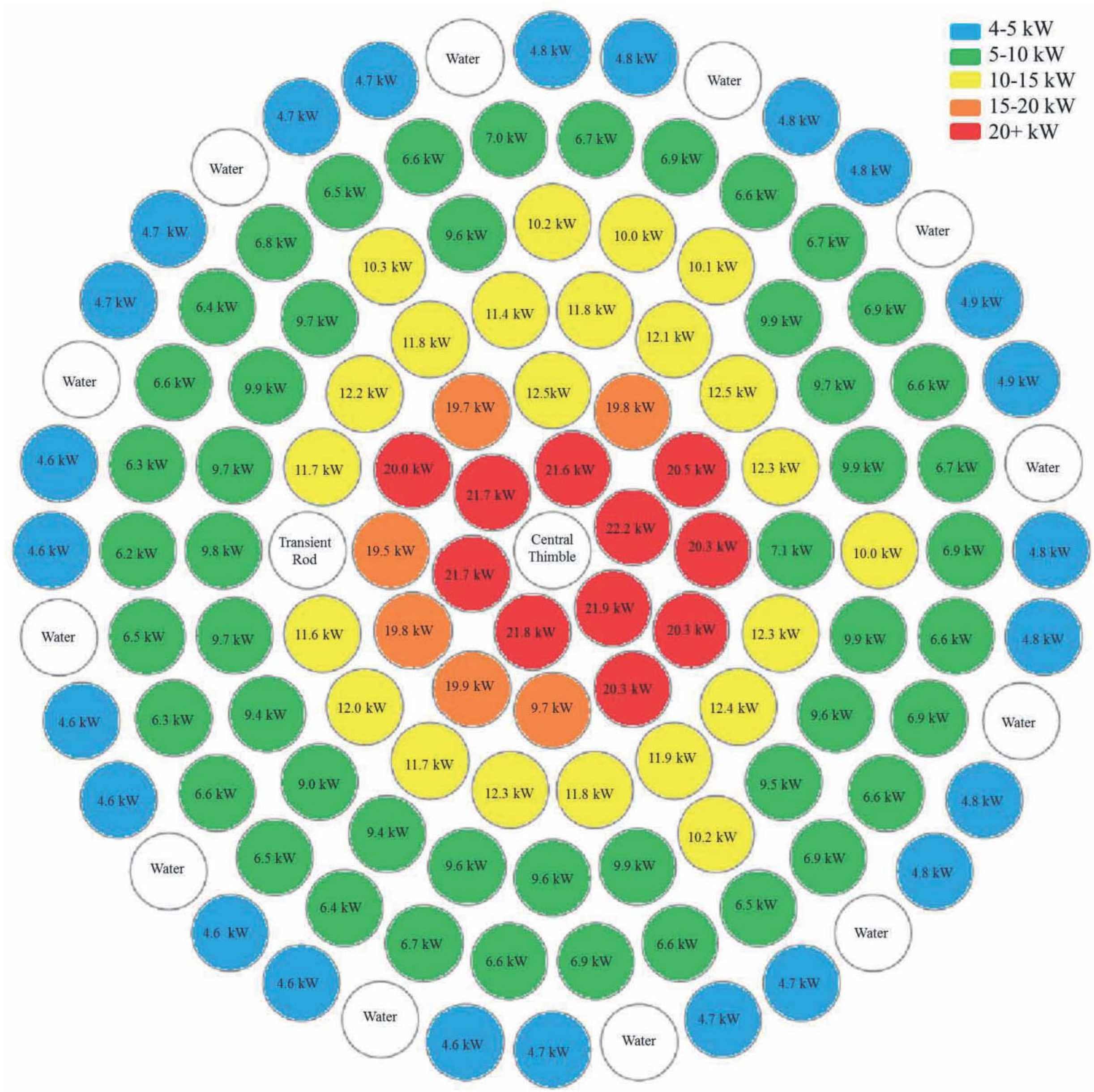


Figure 3, power profile for the limiting core

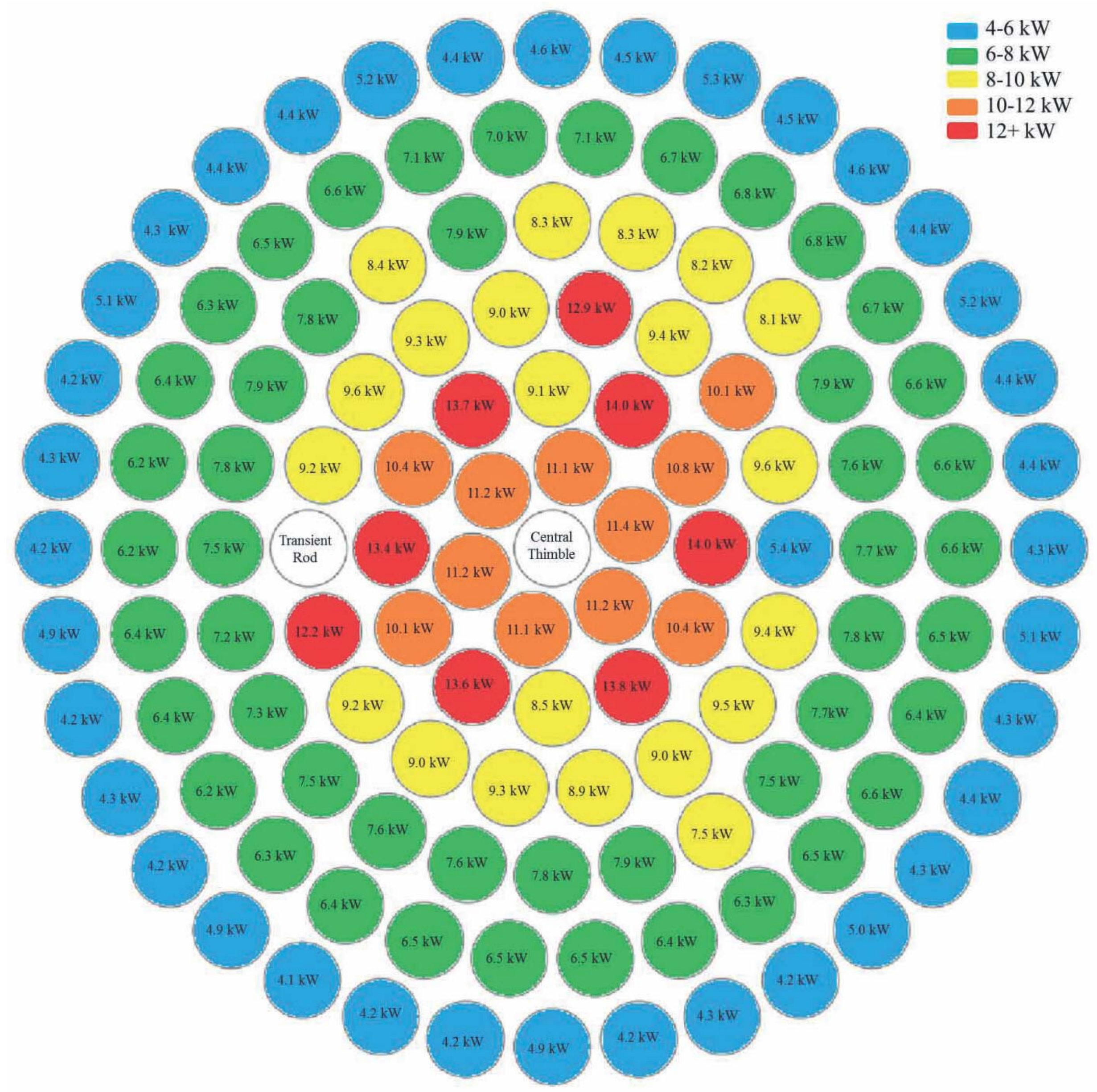
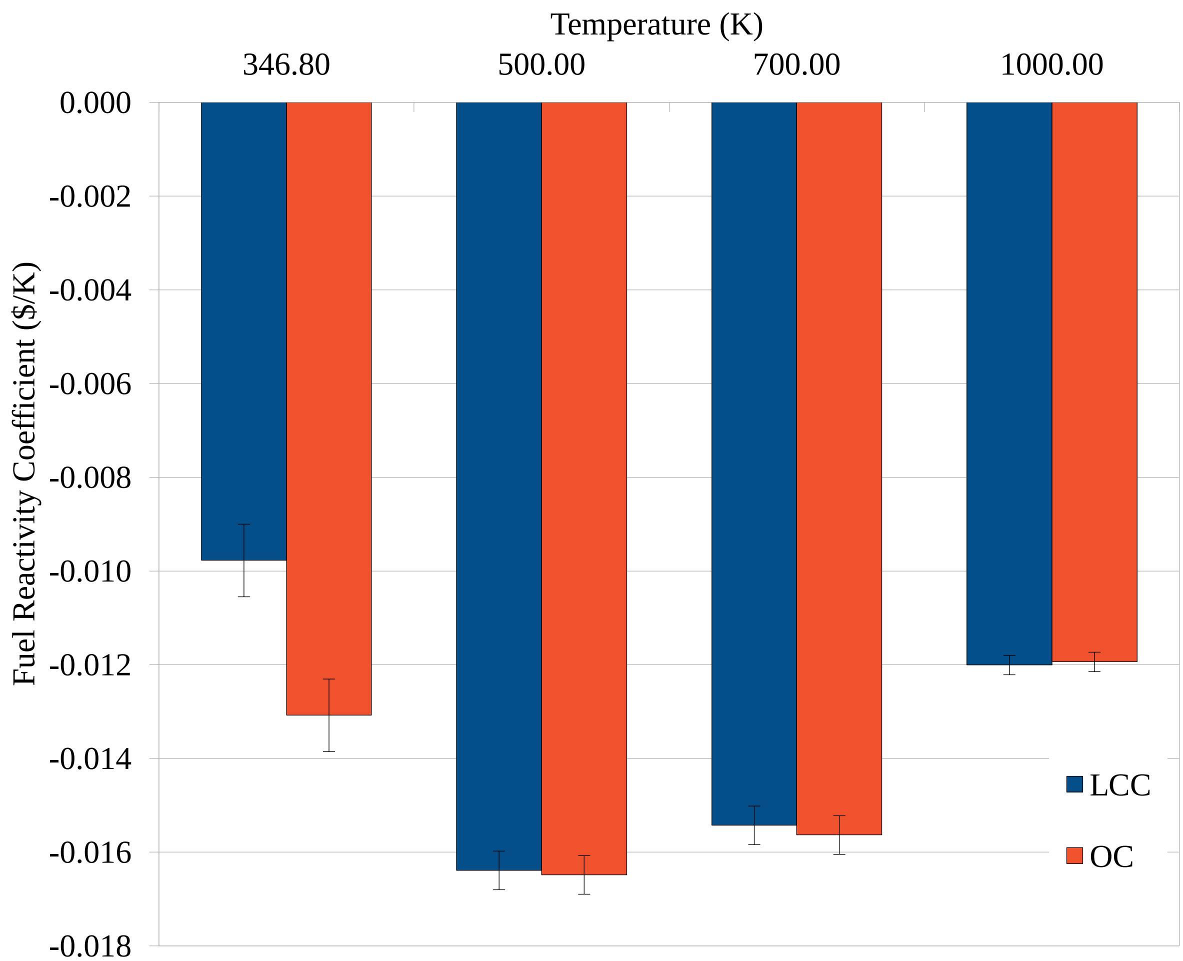


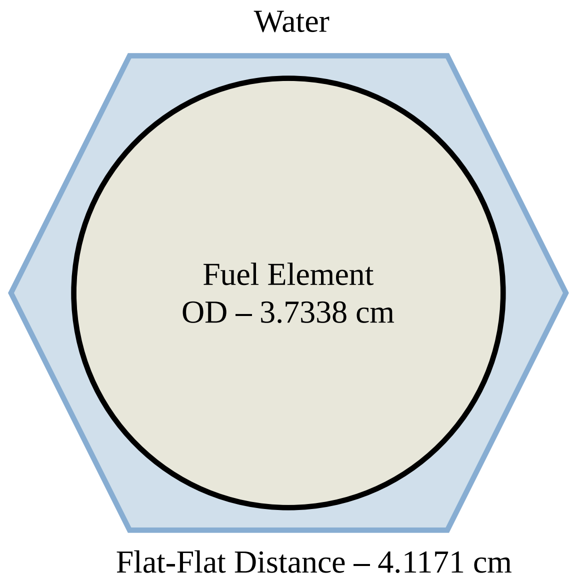
Figure 4, power profile for the operating core

10 – Fuel temperature coefficient for the LCC and the OC as a function of fuel temperature over the temperature range experienced during operation:

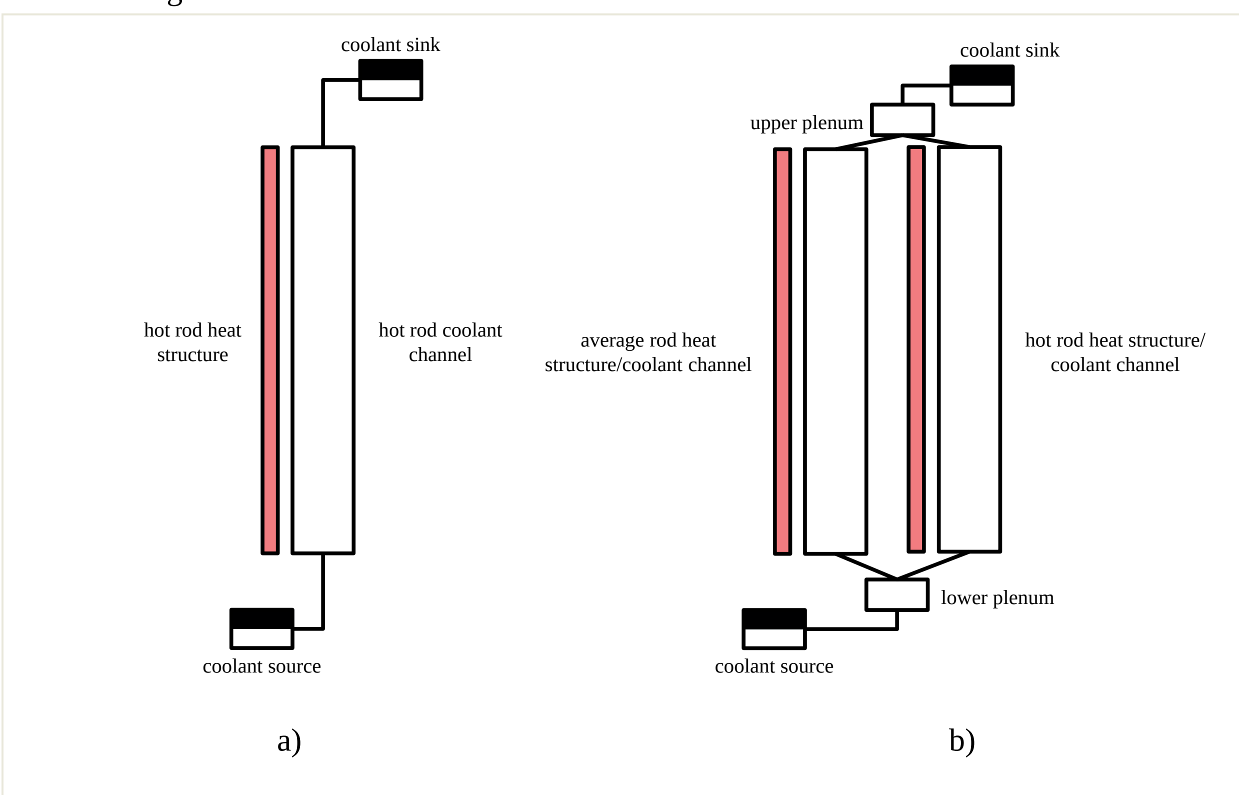


12 –Thermal-hydraulic data for the LCC:

- The unit cell used to define the RELAP model for the DNBR calculations is shown below, graphically with dimensions



- The entry/exit pressure loss coefficients employed in the RELAP model are taken from the OSU model and they are: Inlet: 2.26 Exit: 0.63
- A diagram of the RELAP model is shown below:



- Input assumptions used to analyze DNBR for the LCC:

  Hot rod element power: 22.18 kW
  Peaking Factor: 2.28
  Inlet temperature: 333.15 K
  Inlet Velocity: natural convection, computed by RELAP
  Thermal properties for zirconium-uranium hydride fuel

- RELAP model calculated results such as the core flow rate, peak fuel and cladding temperatures, the location of the minimum DNBR (MDNBR), and the value of the minimum DNBR using the Bernath correlation:

  Core flow rate: ~325.61 kg/ m^2 s
  Peak Fuel Temperature: 829.05 K (556.17 C)
  Peak Cladding Temperature: 409.99 K (136.89 C)
  MDNBR: 1.45 @ 0.41 m from the bottom of the fuel element
  Rod Power: 22.18 kW

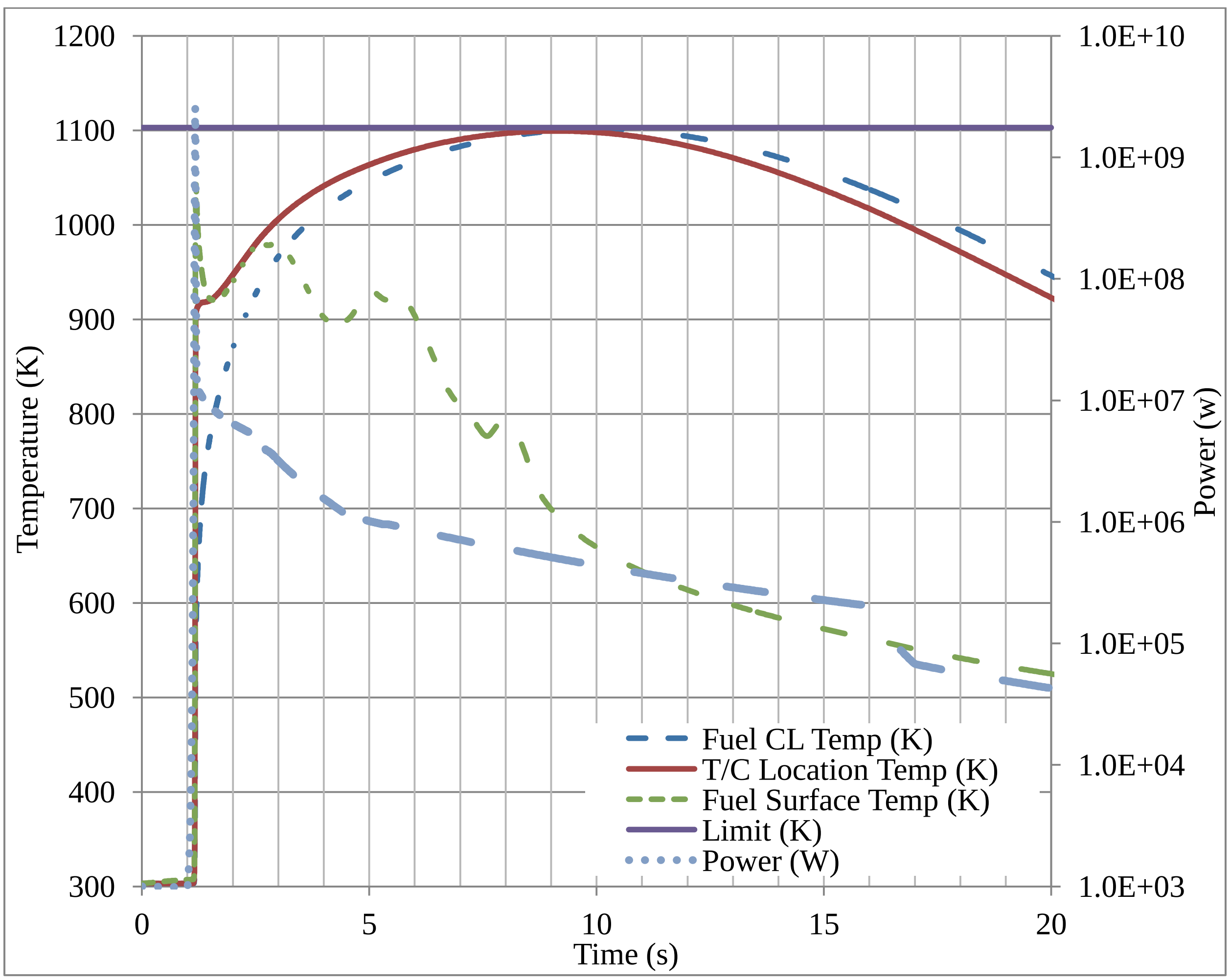
- Characterization of the response of GSTR to a reactivity pulse and an uncontrolled rod withdrawal transient event. Included are the final power achieved in the event, the duration of the event, and the sequence of events.

  The GSTR pulse model sequence begins with the reactor at the Tech Spec limiting conditions of 60 °C water temperature and a steady state power of 1 kW. Since the GSTR uses natural convection for cooling, no initial flow is assumed. These conditions are held for 1 second, after which a reactivity insertion is made that is equal to the requested pulse height ($3.00, $2.75, $2.50, or $2.00) over a 0.2 s period. This reactivity insertion is held for 1.5 seconds from the time of the beginning of the insertion, and then the pulse rod is inserted over a conservative 2 second time. 15 seconds following the initiation of the initial pulse, the reactor scrams, conservatively adding ~$5.00 of negative reactivity into the core over one second. The reactivity chart used in the RELAP simulation is shown with X replacing the pulse amount:

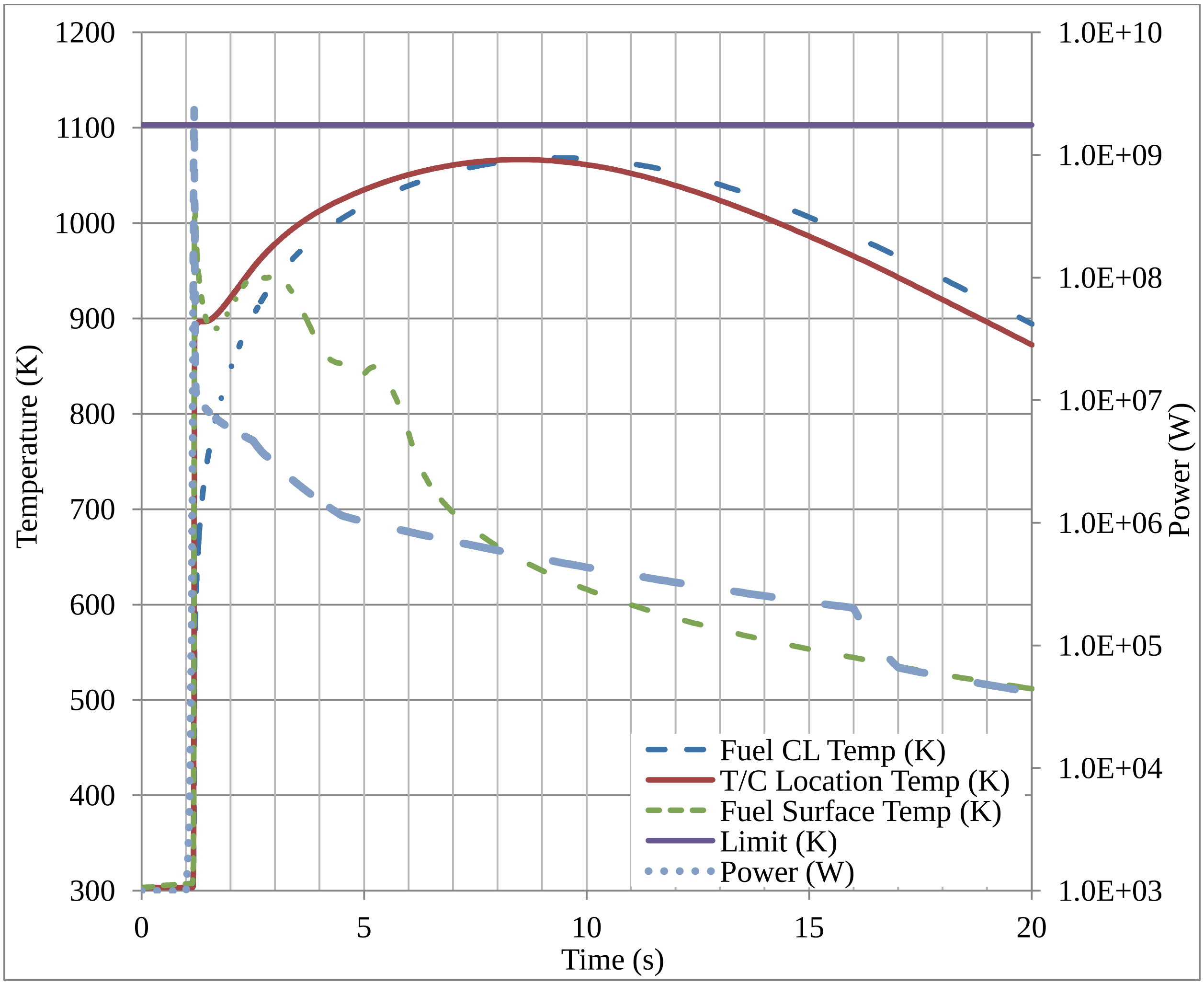
Time (s)	Inserted Reactivity ($)
0	0
1	0 (pulse initiated)
1.2	X (pulse rod fully up)
2.5	X (pulse rod scram signal)
4.5	0 (pulse rod fully down)
16	0 (all rod scram signal)
17	-5 (all rods fully down)

These values are consistent with the Technical Specifications of the GSTR at the time of the analysis, along with information provided from the GSTR staff for values not covered in the technical specifications. In all of the following graphs, the limit line marks an upper bound of the thermocouple temperature at 830 °C as conservatively recommended in report TRD 070.01006.05. It is noted that the TRD report discusses concerns with uranium loading of 20 wt% and higher, although all of the GSTR fuel is less than 20 wt%. The $3.00 pulse analysis shows a peak fuel temperature of 1104 K (831°C) which is one degree above the 830°C recommended level, but the 830°C level has a 44°C safety factor built into it so that the 831°C temperature has no safety significance. The smaller pulse analyses show peak fuel temperatures well below the 830 °C recommended level and even farther below the 1150 °C safety limit.

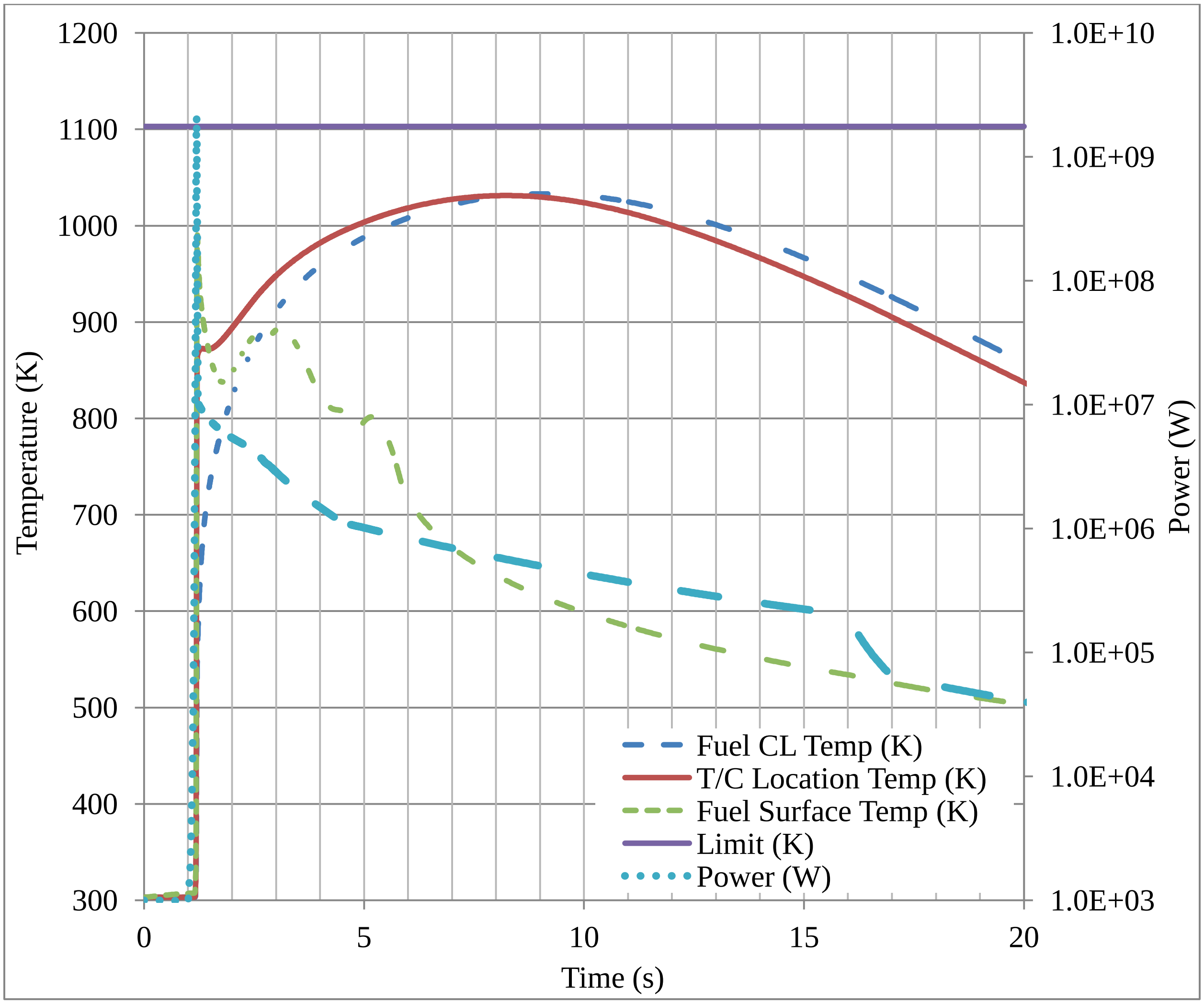
$3.00 Pulse Results – LCC Hot Rod



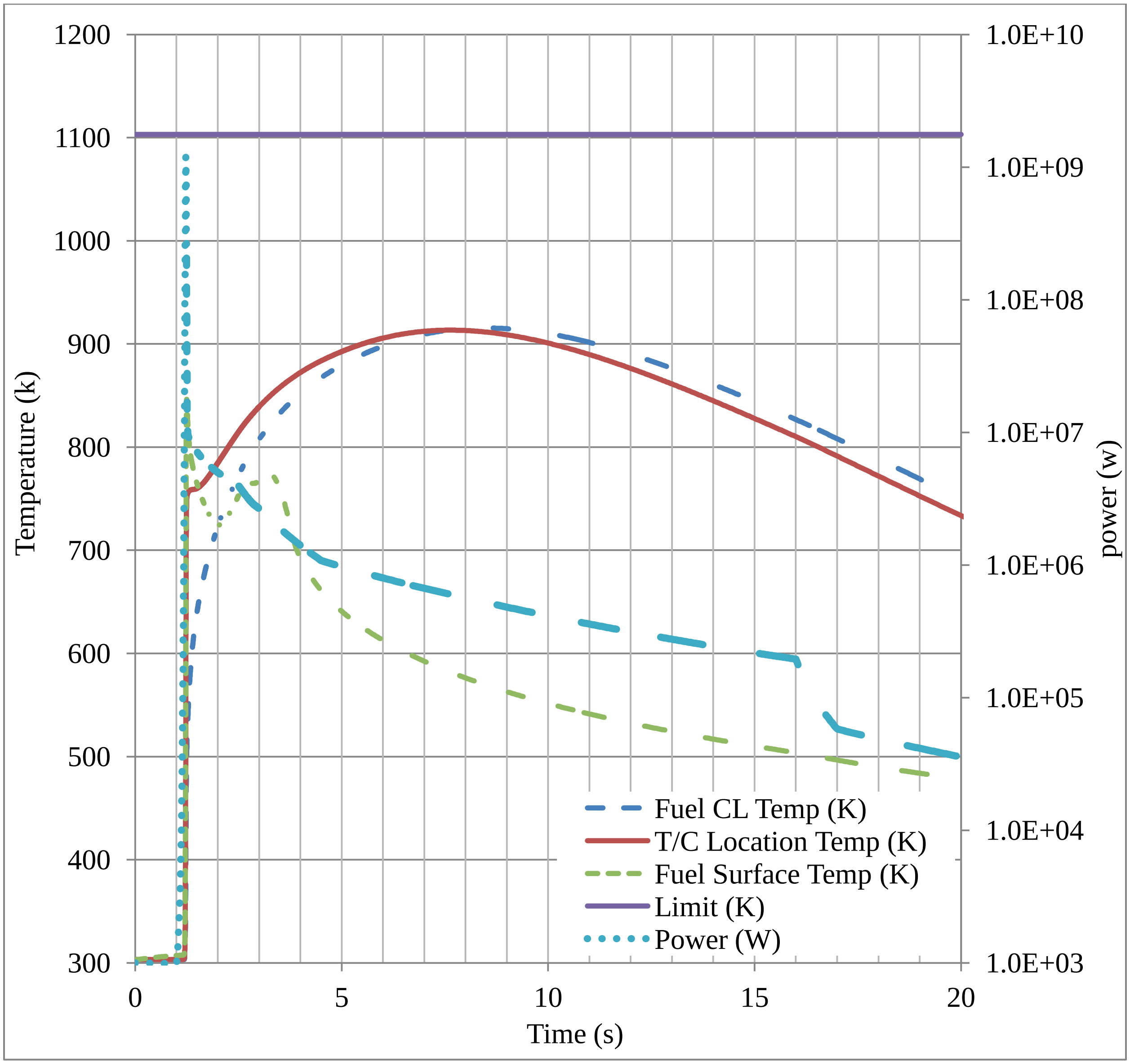
$2.75 Pulse – LCC Hot Rod



$2.50 Pulse – LCC Hot Rod



$2.00 Pulse – LCC Hot Rod



Summary of Pulse Analyses for the Limiting Core Configuration:

Parameter	$3.00	$2.75	$2.50	$2.00
Peak Power (MW)	2455	2244	2037	1210
Peak Fuel Surface Temperature (K)	1034	1010	983	847
Peak T/C Location Temperature (K)	1102	1069	1033	918
Peak CL Temperature	1104	1071	1035	920
Pulse FWHM (s)	0.013	0.013	0.014	0.018

The above table summarizes the safety critical information gathered from the pulse analyses, including the peak pulse power, the pulse full-width half-maximum (FWHM), and the peak temperatures reached at the thermocouple location, element centerline, and at the fuel surface, where the largest proportion of the fissions occur.

14.2 Analysis of the uncontrolled rod withdrawal for the LCC.

The reactivity vs. time table used in the RELAP simulation is provided for the continuous rod withdrawal accident. This simulation starts with the same limiting initial conditions as the pulse simulation (reactor critical at 1 kW, pool water at 60 C).

Time (s)	Reactivity ($)
0.0	0.0
1.0	0.0
4.9	0.07
8.79	0.33
12.69	0.78
13.87	1.09
15.87	-5.57

Reactivity vs. time for Reg rod withdrawal simulation

Hot Rod Parameter	PeakValues
Fuel CL Temp (K)	347
Fuel T/C Location Temp (K)	347
Fuel Surface Temp (K)	345
Peak Power (MW)	1.28
Peak Time (s)	13.9

Peak temperature and power results from the simulation

For this simulation the important safety information is shown in the second table. A SCRAM setpoint of 1.1 MW provides complete safety for the reactor, with an insignificant amount of energy produced and released, leading to a negligible temperature increase.

16.2 The 12 wt% fuel provides the limiting results for fuel used in the GSTR because the 12 wt% fuel contains ~50% more uranium by mass than the 8.5% fuel. When examining the burnup as well, the 8.5% fuel within the GSTR contains much less uranium than a fresh 12wt% rod contains. This causes the 12 wt% fuel to produce a higher power density than the 8.5% fuel.

This density translates to a higher fuel temperature, cladding temperature, heat flux, and reactivity worth.

Accuracy of control rod calibrations:

An estimate of control rod calibration accuracy was made by evaluating the regulating rod calibration that was performed on 12/21/2011. This rod was chosen because it has the highest worth of all GSTR control rods. This calibration was performed using the positive period method, just as all GSTR control rod calibrations are done. The error of not being exactly critical prior to each rod withdrawal step and the error of the measurements made during the calibration are evaluated.

Prior to each rod withdrawal step, the reactor is controlled in the manual mode to ensure that it is exactly critical. The operator waits at least 5 minutes to ensure that power is stabilized and no other confounding activities, such as water temperature change or sample movement, are allowed during this time. The criticality error would be a maximum of a 0.05 w change (at a nominal 2 W power) in 1 minute, giving a period of 2430 s and a reactivity of $0.0053. Assuming a very conservative, worst-case scenario where each of the 12 steps of the calibration was off by this amount and of the same polarity, the total cumulative error in the control rod worth would be $0.0636. This is a 2% error over the full length of the regulating rod worth of $3.104. Since this error is a function of the number of reactivity insertion steps made during the rod calibration, it is highest for the regulating rod (12 steps), and lower for the shim 1 rod (8 steps), shim 2 rod (8 steps), and transient rod (8 steps). The rod calibrations using 8 steps would have a cumulative error over the full rod travel of $0.0424 (= 8 x 0.0053).

The measurements made during the control rod calibration are time intervals. Two stopwatches are used by two persons to independently measure the time required for the reactor to go from 20 W to 400 W for each rod withdrawal step. These times are then averaged, converted to reactor period and then converted to the reactivity associated with that step. For the regulating rod, the statistical error in these timing events is analyzed in the table below, with a resulting standard deviation of $0.0115 over the full length of the regulating rod worth of $3.104. This is a 0.37% standard deviation for the regulating rod. The same error for the other three control rods would have full-withdrawal reactivity values of $0.0073 for Shim 1, $0.0069 for Shim 2, and 0.0082 for the Transient Rod.

USGS Reg Rod Calibration - Timing Error Analysis

	Time 1 (s)	Time 2 (s)	difference
	45.60	45.50	0.10
	142.38	141.60	0.78
	63.80	64.60	-0.80
	63.20	63.34	-0.14
	61.60	61.57	0.03
	80.90	80.57	0.33
	63.20	63.04	0.16
	63.50	63.60	-0.10
	60.00	60.19	-0.19
	102.80	102.85	-0.05
	75.50	75.23	0.27
	84.00	84.46	-0.46
averages	75.54	75.55	
		std dev	0.40
Std dev of avg difference			0.28

A std dev of 0.28 seconds on an average time of 75.54 seconds is a 0.37% std dev. Given an avg reactivity per step of $0.2587, this would result in a std dev of 0.096 cents per step. The reg rod had the most steps (12), so the worst case cumulative std dev over the entire rod would be 1.15 cents. The determined reg rod worth is $3.104.

These errors combine to give errors of approximately $0.0751, $0.0497, $0.0493, and $0.0506 at the fully withdrawn positions of the regulating rod, shim 1 rod, shim 2 rod, and transient rod, respectively.

Since reactivity measurements are done at low power and with rod positions near the mid portion of their motion (sufficient rod worth must be remain to achieve full power and override xenon buildup), the error in reactivity measurements will be much less. Changes in reactivity from events such as sample insertions, xenon transients, and fuel burnup are all made by measuring a change in rod positions while critical at low power. Using a realistic example for the operating core, low power critical rod positions would be ~465 units, with an excess reactivity of $4.43 and a shutdown reactivity of $4.69. If a sample was loaded and measured to have a reactivity worth of -$1.00, the critical rod positions for all four control rods would change to 538 units.

This change of 73 units per rod would represent approximate errors of \$0.00548, \$0.00363, \$0.00360, and \$0.00369 for the regulating rod, shim 1 rod, shim 2 rod, and transient rod, respectively. These combine for a total error of \$0.0164 on the measurement of a \$1.00 sample worth.

The table below summarizes these errors and provides the same analysis for reactivity changes of \$3.00 and \$5.00.

Core status	Control rod positions (units)				Unit change per rod	Reactivity errors (\$)				Cumulative error (\$)	Percent error
	Regulating	Shim 1	Shim 2	Transient		Regulating	Shim 1	Shim 2	Transient		
Cold, clean	465	465	465	465	reference	reference	reference	reference	reference	reference	reference
\$1.00 Δ reactivity	538	538	538	538	73	0.00548	0.00363	0.0036	0.00369	0.0164	1.640
\$2.00 Δ reactivity	616	616	616	616	151	0.01134	0.0075	0.00744	0.00764	0.03392	1.696
\$3.00 Δ reactivity	707	707	707	707	242	0.01817	0.01203	0.01193	0.01225	0.05438	1.813

The result of these analyses is that reactivity changes in the core, that are of the magnitude of those typically seen or postulated, can be measured with an error of several cents or less.

Revised response to 24.9

- Please explain the use of 22 kW per fuel element, and why it was used rather than using the hot rod power determined from the LCC. Clarify the basis for the value used in the accident analysis.

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 22.18 kW ± 0.26 kW. Conservatively using the worst case scenario, and adding the error into the calculated power production factor, the hot rod has a peak power production of 22.44 kW. This value is 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 are provided and the hot rod power production of 22.44 kW, from the limiting core analysis, is 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.
- Analysis done using a ground release as there is no basis for using an elevated release.
- The input data used for the Hot Spot analysis to allow for confirmatory calculations.
- Dose estimates for several locations on the Denver Federal Center near Building 15.
- Use of 295 meters (approximately 968 feet) as 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 sections will replace the sections given in the original 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. The type of accident postulated here has never occurred, even after 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 beyond credible worst-case fuel-failure scenario. Historically, TRIGA fuel failures have shown very small fission product releases.

For TRIGA reactors, the MHA has been defined 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 reactor water. 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 hot rod power density for the non-credible 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 postulated 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.

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 never operated continuously at 1 MW for a period longer than 16 hours, much less for a period of one year. Nevertheless, this non-credible 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 a conservative hot rod power density for a continuous period of one year at 1 MW in the limiting core resulting in 22.44 kW in the element. This results in all of the halogens and noble gases (except Kr-85) reaching their saturated activities. This scenario very conservatively 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^{3}sec^{-1}), through the emergency exhaust stack conservatively assuming no filtration occurs. The air is conservatively 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 conservatively assumed to be 3.1 x 10^8 cubic centimeters. The exhaust system then 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 conservatively assumed stay time of 5 minutes.

13.2.1.2 Accident Analysis and Determination of Consequences

It is conservatively assumed that the GSTR is fueled in the limiting core configuration shown in Figure 13.1, 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 activities. 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 22.44 kW. 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. Table 13.1 was calculated from Oregon State University's (OSU) fission product inventory for the same MHA scenario by multiplying OSU's number by the ratio of the highest power density element at the GSTR (22.44 kW) to the highest power density element at the OSU reactor (15.9 kW).

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

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

At a fuel temperature of 556.17 ^{o}C, this release fraction is 3.60 × 10^{-4}. This assumed fuel temperature (556.17 ^{o}C) is the expected hot rod fuel temperature for our limiting core and will produce a conservative estimate for the fission product release.

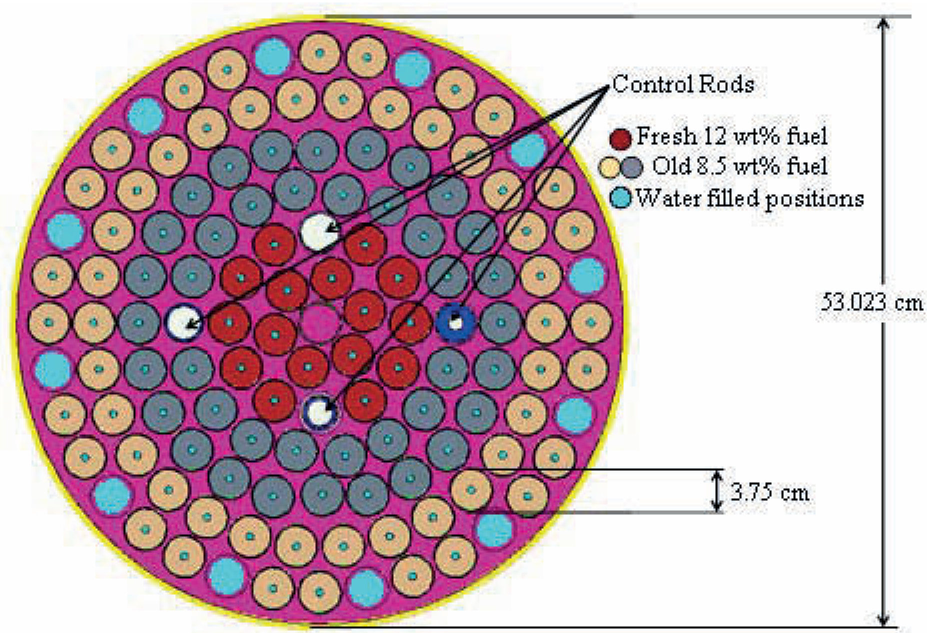


Figure 13.1, Limiting core configuration layout

Once the fission products are released to the cladding gap, this activity is conservatively assumed to be instantly released when the cladding catastrophically fails. The release occurs in the pool water (MHA), and 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 will 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 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 (3.60×10^{-4});

f = the fraction released from the fuel-cladding gap to the pool water;

g = the fraction released from the pool water to the reactor bay air; and

h = the fraction released from the reactor room air to the outside unrestricted environment, due to plateout in the reactor bay.

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		

It is conservatively assumed that 50% of the halogens are released from the gap into the water. This value is based on historical usage and recommendations; however, this value is quite conservative, as some references quote a 1.7% release from the gap rather than 50%. 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 from the gap are released to the reactor room. Due to plateout in the reactor bay, 50% of these halogens are released to the outside environment. Thus 1.25% of the halogens is available for release to the outside environment. For the noble gases released under water, 100% are conservatively assumed to be available for release to the unrestricted environment.

The experiences at Three Mile Island, along with recent experiments, indicate that the 50% halogen release fraction from the cladding gap 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. Experience at the Aerotest TRIGA indicate that severe cladding failure may result in undetectable fission product releases.

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

Table 13.2
Release Fraction Components

Fission product	f	g	h
Noble gas	1.0	1.0	1.0
Halogens	0.5	0.05	0.5

Table 13.3
Total Release Fraction

Fission product	w to the reactor bay	w to the environment
Noble gas	3.60 E -4	3.60 E-4
Halogens	9.00 E -6	4.50 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 (conservatively 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 conservatively assumed to be constant during this exposure period.

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

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

Since a non-stochastic exposure of 1 annual limit on intake (ALI) conservatively 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 estimated by

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

where:

3.33E-4 = the NRC “light work” breathing rate with units of m^3sec^{-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 13.4
Concentrations and Exposures from Gaseous Fission Product Releases

Isotope	Saturated Activity (Ci)	Released Activity to Reactor Room Air (mCi)	Released Activity to environment (mCi)	DAC value of diluted activity in reactor bay (with pool water) (# DACs)	DAC-Hr exposure for 2 minute stay time
Br-82	[redacted]	0.27	0.13	0.43	0.01
Br-83	[redacted]	0.94	0.47	0.10	0.00
Br-84m	[redacted]	0.04	0.02	1.15	0.04
Br-84	[redacted]	1.75	0.88	0.28	0.01
Br-85	[redacted]	2.31	1.16	74.57	2.49
Br-86	[redacted]	3.45	-	-	-
Br-87	[redacted]	4.48	-	-	-
Total Bromine	[redacted]	13.25	2.65	76.54	2.55
I-131	[redacted]	5.07	2.53	817.43	27.25
I-132	[redacted]	7.56	3.78	8.13	0.27
I-133	[redacted]	11.75	5.87	379.01	12.63
I-134	[redacted]	13.79	6.90	2.22	0.07
I-135	[redacted]	11.47	5.73	52.86	1.76
I-136	[redacted]	11.08	-	-	-
Total Iodine	[redacted]	60.72	**24.82**	1259.64	41.99
Kr-83m	[redacted]	37.60	37.60	0.01	0.00
Kr-85m	[redacted]	92.47	92.47	14.91	0.50
Kr-85	[redacted]	5.94	5.94	0.19	0.01
Kr-87	[redacted]	179.35	179.35	115.71	3.86
Kr-88	[redacted]	250.99	250.99	404.82	13.49
Kr-89	[redacted]	331.77	331.77	10702.36	356.75
Total Krypton	[redacted]	898.13	898.13	11238.01	374.60
Xe-131m	[redacted]	3.76	3.76	0.03	0.00
Xe-133m	[redacted]	11.48	11.48	0.37	0.01
Xe-133	[redacted]	456.25	456.25	14.72	0.49
Xe-135m	[redacted]	70.62	70.62	25.31	0.84
Xe-135	[redacted]	458.79	458.79	148.00	4.93
Xe-137	[redacted]	433.90	433.90	13996.66	466.56
Xe-138	[redacted]	474.54	474.54	382.70	12.76
Total Xenon	[redacted]	**1909.35**	**1909.35**	**14567.78**	**485.59**
Total Halogens	[redacted]	**73.96**	**27.47**	**1336.18**	**44.54**
Total Noble Gases	[redacted]	**2807.47**	**2807.47**	**25805.80**	**860.19**
Total Exposure for 2-minute stay time	-	-	-	-	**904.73**

The released amounts of iodine radionuclides in the reactor bay are shown in Table 13.5. A summary of the $CDE_{Thyroid}$ and TEDE for 2- minute and 5-minute stay times in the reactor bay are shown in Table 13.6. As seen in Table 13.6 the analysis with shows a TEDE dose of 5655 mrem in 5 minutes. This dose exceeds the 10 CFR 20 occupational dose limit of 5000 mrem; however, this analysis is highly conservative with assumptions that are not physically possible. Thus any actual dose received would be much lower than 5655 mrem.

Table 13.5
Concentrations and Exposures from Iodine Radionuclides Releases

Isotope	Released Activity to Reactor Room Air (mCi)	Non-stochastic ALI from 10 CFR 20 Appendix B for thyroid (iodine isotopes only) (uCi)	Ingested Iodine in reactor bay WITH POOL WATER, 2 min stay time(uCi)	# ALIs ingested	Very conservative Thyroid dose, 50000 mr per ALI ingested (mR)
I-131	5.07	50	0.65	1.31E-02	653.29
I-132	7.56	8.00E+03	0.97	1.22E-04	6.09
I-133	11.75	3.00E+02	1.51	5.05E-03	252.42
I-134	13.79	5.00E+04	1.78	3.56E-05	1.78
I-135	11.47	2.00E+03	1.48	7.39E-04	36.96
I-136	11.08	2.00E+02	1.43	7.14E-03	356.93
Total Iodine	60.72	**-**	**-**	**-**	**1307.47**

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

Reactor Room Occupancy (minutes)	$CDE_{Thyroid}$ (mrem)	TEDE (mrem)
2	1307	2262
5	3269	5655

The results of the HOTSPOT code version 2.07.2 calculations for the MHA are shown in Table 13.7. 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)	$CDE_{Thyroid}$ (mrem)	TEDE (mrem)
Building 15 south door	11	31	32
Emergency assembly area	32	9.5	5.1
Building 21 east entrance (West of Building 15)	49	32	9.9
Average of eastern intersections	100	30	8.5
Building 16 west entrance	175	13	3.6
-	200	10	2.8
-	250	6.6	1.8
Fence	295	4.8	1.3
Residence	640	1.1	0.27
School	720	0.85	0.21

Response to 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 22.8% 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. The occupancy factor value of 22.8% is a conservative number calculated from 2000 working hours in one year (8760 hours).

Looking at the operation history of the GSTR, a conservative estimate for the dose to personnel is listed in Table 2. The input parameters for this analysis are shown below:

Ar-41 release in Ci/year: 7.75 Ci/year,
Release height: 6 meters,
Building height: 4 meters,
Building width: 30 meters,
Default mean wind speed: 2.0 m/sec.

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 22.8% Occupancy Factor (mrem/yr)
11	135	Building 15 south door	30.78
32	16.7	Emergency assembly area	3.81
49	10.4	Building 21 east entrance	2.37
100	4.1	Average of eastern intersections	0.93
175	1.8	Building 16 west entrance	0.41

Table 2: Yearly dose due to release of 7.75 Ci of Ar-41 at several distances with occupancy factor applied. All yearly doses calculated with EPA's COMPLY code.

Distance (m)	Dose (mrem/yr)	Location	Dose with 22.8% Occupancy Factor (mrem/yr)
11	14.7	Building 15 south door	3.35
32	1.8	Emergency assembly area	0.41
49	1.1	Building 21 east entrance	0.25
100	0.4	Average of eastern intersections	0.09
175	0.2	Building 16 west entrance	0.05
295	0.08	Fence of DFC	0.02

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

Prepared for:
U.S. Environmental Protection Agency
Office of Radiation and Indoor Air
Washington, DC 20460

```
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:
 -------------
                 Release Rate
    Nuclide     (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 **********