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

U. S. GEOLOGICAL SURVEY
RAI 15.3 RESPONSE
JANUARY 3, 2012

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

***REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS**

USGS
science for a changing world

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

January 3, 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 September 29, 2010

Subject:Response to Question 15.3 of the Referenced RAI

Mr. Wertz:

15.3 GSTR SAR Table 13.5 provides the occupational committed dose equivalent (CDE) for the thyroid and total effective dose equivalent (TEDE) for a 2- and a 5-minute exposure in the reactor bay. The NRC staff was unable to reproduce the TEDE doses using the derived air concentration (DAC)-hour method. Please explain the methods used to determine the thyroid doses in sufficient detail to permit confirmatory calculations.

Response:

The Geological Survey TRIGA Reactor (GSTR) Safety Analysis Report (SAR) section 13.2.1 calculates the dose to members of the public and GSTR staff in the reactor bay during the MHA. Table 13.5 provides the occupational committed dose equivalent (CDE) for the thyroid and total effective dose equivalent (TEDE) for a 2 and a 5 minute exposure in the reactor bay using the DAC-hour method. There were three errors in the document, one typo in Table 13.2 which propagated through the analysis, an accidental inclusion of I-130m in the analysis, and an error in the calculation of the fission product fraction released into the reactor bay. The corresponding data and explanation of the calculations are given below in the newly revised section 13.2.1.

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

A020
NRR

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.

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. 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. This results in all of the halogens and noble gases (except Kr-85) reaching their saturated activities.

This most severe accident 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, let alone a period of one year. Nevertheless, this extremely conservative MHA has been analyzed for the GSTR.

The following scenario has been chosen for analysis:

- 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. The air is assumed to be discharged at 21 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 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 350 °C, this release fraction is 1.66×10^{-5}. This assumed fuel temperature (350 °C) is much higher than the actual expected average fuel temperature of 240 °C, and results in a release fraction about 11% higher.

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 (1.66×10^{-5});

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.

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

Isotope	Half Life	[redacted]
Br-82	35.3 h	
Br-83	2.4 h	
Br-84	31.8 min	
Br-85	2.87 min	
Total Bromine		
I-129	1.7e7 yrs	
I-131	8.02 d	
I-132	2.28h	
I-133	20.8 h	
I-134	52.6 min	
I-135	6.57 h	
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		

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 13.2
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	1.66 E -5	1.66 E -5	1.66 E-5	1.66 E-5
Halogens	8.30 E -6	4.15 E -7	4.15 E-6	2.08 E-7

For the GSTR, the prevailing wind is from the west, blowing to the east. The minimum distance to the unrestricted environment (350 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 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 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]	0.19	0.10	2.00E-06	0.31	0.01	0.01	0.00	0.01	0.00
Br-83	[redacted]	0.78	0.39	3.00E-05	0.08	0.00	0.04	0.02	0.00	0.00
Br-84	[redacted]	1.38	0.69	2.00E-05	0.22	0.01	0.07	0.03	0.01	0.00
Br-85	[redacted]	1.68	0.84	1.00E-07	54.35	1.81	0.08	0.04	1.36	0.05
Total Bromine	[redacted]	4.03	2.02	-	54.97	1.83	0.20	0.10	1.37	0.05
I-129	[redacted]	0.92	0.46	4.00E-09	742.98	24.77	0.05	0.02	18.57	0.62
I-131	[redacted]	4.74	2.37	2.00E-08	764.40	25.48	0.24	0.12	19.11	0.64
I-132	[redacted]	6.70	3.35	3.00E-06	7.20	0.24	0.33	0.17	0.18	0.01
I-133	[redacted]	10.54	5.27	1.00E-07	340.03	11.33	0.53	0.26	8.50	0.28
I-134	[redacted]	11.95	5.98	2.00E-05	1.93	0.06	0.60	0.30	0.05	0.00
I-135	[redacted]	9.30	4.65	7.00E-07	42.84	1.43	0.46	0.23	1.07	0.04
Total Iodine	[redacted]	**44.15**	**22.07**	-	1899.39	63.31	2.21	**1.10**	47.48	1.58
Kr-83m	[redacted]	1.66	1.66	1.00E-02	0.00	0.00	1.66	1.66	0.00	0.00
Kr-85m	[redacted]	3.09	3.09	2.00E-05	0.50	0.02	3.09	3.09	0.50	0.02
Kr-85	[redacted]	0.17	0.17	1.00E-04	0.01	0.00	0.17	0.17	0.01	0.00
Kr-87	[redacted]	8.45	8.45	5.00E-06	5.45	0.18	8.45	8.45	5.45	0.18
Kr-88	[redacted]	13.40	13.40	2.00E-06	21.61	0.72	13.40	13.40	21.61	0.72
Kr-89	[redacted]	16.77	16.77	1.00E-07	540.84	18.03	16.77	16.77	540.84	18.03
Total Krypton	[redacted]	43.53	43.53	-	568.40	18.95	43.53	43.53	568.40	18.95
Xe-131m	[redacted]	0.09	0.09	4.00E-04	0.00	0.00	0.09	0.09	0.00	0.00
Xe-133m	[redacted]	0.48	0.48	1.00E-04	0.02	0.00	0.48	0.48	0.02	0.00
Xe-133	[redacted]	20.25	20.25	1.00E-04	0.65	0.02	20.25	20.25	0.65	0.02
Xe-135m	[redacted]	5.59	5.59	9.00E-06	2.01	0.07	5.59	5.59	2.01	0.07
Xe-135	[redacted]	19.26	19.26	1.00E-05	6.21	0.21	19.26	19.26	6.21	0.21
Xe-137	[redacted]	18.91	18.92	1.00E-07	610.45	20.35	18.91	18.92	610.45	20.35
Xe-138	[redacted]	16.77	16.77	4.00E-06	13.52	0.45	16.77	16.77	13.52	0.45
Total Xenon	[redacted]	**81.36**	**81.36**	-	**632.86**	**21.10**	**81.36**	**81.36**	**632.86**	**21.10**
Total Halogens	[redacted]	**48.18**	**24.09**	-	**1954.35**	**65.15**	**2.41**	**1.20**	**48.86**	**1.63**
Total Noble Gases	[redacted]	**124.89**	**124.89**	-	**1201.26**	**40.04**	**124.89**	**124.89**	**1201.26**	**40.04**
Total Exposure for 2-minute stay time	[redacted]	-	-	-	-	**105.19**	-	-	-	**41.67**

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

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

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

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

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 13.5
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-129	0.92	9	0.12	1.32E-02	659.77
I-131	4.74	50	0.61	1.22E-02	610.91
I-132	6.70	8.00E+03	0.86	1.08E-04	5.40
I-133	10.54	3.00E+02	1.36	4.53E-03	226.46
I-134	11.95	5.00E+04	1.54	3.08E-05	1.54
I-135	9.30	2.00E+03	1.20	5.99E-04	29.96
Total Iodine	**44.15**	-	-	-	**1534.04**

The released amounts of iodine radionuclides in the reactor bay a 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	1534	263	77	104
5	3835	657	192	260

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: general plume model, mixture of isotopes from Table 13.4, release height of 6.4 m, plume rise calculation with vertical exit velocity of 10.1 m/s, ambient air temperature of 20°C, effluent temperature of 20°C, stack effective diameter of 0.2 m, 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, standard terrain, wind reference height is 10 m, sample time is 10 min, source geometry is simple, DCF library is FCR11 corresponding to ICRP 30 series, and the breathing rate is 3.33e-4 m^3/s.

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

Distance (m)	$CDE_{Thyroid}$ (no water) (mrem)	TEDE (no water) (mrem)	$CDE_{Thyroid}$ (water) (mrem)	TEDE (water) (mrem)
10	0.0	0.0	0.0	0.0
50	2.9e-7	1.2e-8	1.6e-8	2.9e-9
100	0.13	5.3e-3	7.0e-3	1.2e-3
150	.87	0.036	.048	8.3e-3
200	1.4	.059	.078	0.013
250	1.6	.066	.088	0.015
300	1.6	.066	.087	.014
350 (fence)	1.5	.061	.080	.013
640 (residence)	0.77	.031	.041	6.4e-3
720 (school)	0.65	.026	.035	5.3e-3

The neutronic and thermo-hydraulic analyses are continuing at the Colorado School of Mines with decisions being made about the versions of ENDF libraries and MCNP codes to use. Testing the recent ENDF zirconium cross-section libraries against established benchmarks has shown that the ENDF/B-VII.0 libraries result in over-estimates of the k_{eff} values inside a reactor containing significant amounts of zirconium hydride. This corresponds with the current results from the MCNP model of the GSTR. Additionally, the ENDF/B-VI.6 libraries do not include several of the isotopes needed for the burnup analysis, requiring an ENDF/B-VI based model to be a hybrid of both the VI.6 and VII.0 libraries. Finally, the ENDF/B-VI.6 libraries for MCNP5 are only provided at room temperature, while the ENDF/VII.0 libraries are available at a wide range of temperatures suitable for producing accurate cross sections at specific temperatures, which is necessary to accurately calculate temperature reactivity coefficients. Therefore, the final MCNP model of the GSTR will use the ENDF/B-VII.0 libraries while acknowledging the resulting positive bias. In addition to this change, the MCNP model of the GSTR has been tested using the latest version of MCNP5 (Version 1.60). This version includes several capabilities which will prove useful for executing the model at different temperatures. To ensure that the version change did not compromise the results of the MCNP model, the same critical core model (using the .70c libraries) was run using one-hundred thousand neutrons per cycle for one-thousand cycles. The results show that the two models produce results that are statistically the same.

Sincerely,

Tim DeBey

Tim DeBey

USGS Reactor Supervisor

I declare under penalty of perjury that the foregoing is true and correct.
Executed on 1/3/12

Copy to:

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