United States Department of the Interior
GEOLOGICAL SURVEY
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IN REPLY REFER TO

August 17, 1995

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: USGS TRIGA Reactor Facility
Docket No. 50-274, License R-113

Gentlemen:

This letter is in response to your request for additional information dated July 13, 1995, concerning the use of TRIGA 12 w% fuel at the USGS TRIGA facility.

The information in the following pages is referenced to the numbered questions in your July 13 letter.

If you have any questions concerning this information, please contact Tim DeBey, the Reactor Supervisor, at (303) 236-4726.

Sincerely,

T D Fouch

Thomas Fouch
Reactor Administrator

Subscribed and affirmed before me this 17th day of August, 1995.

Carol Case

Carol Case, Notary

My Commission expires July 26, 1999.

Copies to:
Ross Scarano, USNRC, Region IV
Tim DeBey, MS 974

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1.a. The analytical method used to calculate the thermal power in the hottest fuel element was the Monte Carlo N-particle (MCNP) Transport Code developed at the Los Alamos National Laboratory. The MCNP code is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron-photon-electron Monte Carlo transport code system. The energy ranges covered are 0-20 MeV for neutrons, and 1 keV-1GeV for photons and electrons. Version 4 of the MCNP code has been compiled to run on personal computers with DOS version 4 or higher.

The MCNP code is supplemented with a pre-processor and post-processor code called Research Reactor Analysis Program (RRAP). The RRAP code was developed by Atom Analysis, Inc. RRAP provides a user-friendly interface to simplify the production of the input data for MCNP and also retrieves the MCNP output data and displays the information in a format that is easier to evaluate.

The fuel cell heating is determined by retrieving the cell power peaking factors from MCNP and multiplying them by the average power produced per fuel cell. For example, if the maximum operating power is 1000 kW and there are 100 fuel elements in the core, then the average cell power is 10 kW. If MCNP reports a maximum cell power peak of 1.51, then the maximum power produced in that fuel element is 1.51 x 10 = 15.1 kW. If MCNP reports an error of 3.5%, then the maximum power produced in that fuel element would be (1.51 x 1.035) x 10 = 15.6 kW. The MCNP code reports errors that represent one standard deviation and it is recommended that calculations be performed with sufficient simulations to keep the standard deviations below 5%. This recommendation was followed in performing the GSTR calculations.

The fuel cell power peaking factors were determined by integrating neutron interaction energy deposition (including fissions) over the volume of the cell. Although the calculations did not include gamma energy deposition, the local gamma field will be proportional to the neutron interactions and the power peaking factors will be valid.

1.b. The MCNP model used for the GSTR core is a full-scale, 3-dimensional model. Actual core size and configuration are used, including the graphite reflectors. The uranium loading of each element is used, with burnup calculations being applied to elements that are not new. Actual control rod positions are modeled, with the control rods accurately modeled as fueled followers or void follower, whichever applies. All in-core experimental facilities have been modeled, including air-filled voids, water-filled voids, and cadmium-lined terminii. The MCNP code includes the effects of all materials in the model.

1.c. The GSTR application of the MCNP-RRAP calculations have been benchmarked against known reactivity changes and experimental flux measurements. The largest reactivity change benchmarked was the removal of a fuel element from the G-ring and placement of a cadmium-lined terminal in the vacated grid. This change has been made a number of times at the facility for experimental reasons with a reactivity change of about 45 cents, varying from about 40 cents to 50 cents. The MCNP-RRAP calculation of this same change predicted a reactivity change of 48 cents, $\pm$ 16 cents. The calculation is slightly high, but represents good agreement. (Note: all MCNP/RRAP errors represent one standard deviation.)

The flux benchmarks result from measurements performed in the central thimble of the GSTR. The flux data are given below:

	Experimental	MCNP/RRAP
thermal flux	2.9 $\pm$0.15 e13	3.1 $\pm$0.04
1 MeV flux	6.9 $\pm$0.35 e12	7.5 $\pm$0.15

These results again show the calculated values to be slightly high, with overlap of the ranges occurring at the two standard deviation level. It may be that self-absorption affects cause the experimental data to be biased low.

1.d. The MCNP analyses showed that if the core has at least 100 fuel elements installed and the worst case fuel loading with new 12 w% elements clustered in the core center around a water hole, the peak power produced in any fuel element will be less than 22 kW. We propose that recalculation of the power per element data will not be required unless the reactor is operated above 100 kW with less than 100 fuel elements in the core. This will allow experiments such as fuel loading and approach to criticality with less than 100 fuel elements in the core to be performed without the lengthy MCNP calculations. Any operations above 100 kW with less than 100 fuel elements in the core would require power per element calculations at 6 ± 1 month intervals or whenever the fuel loading was changed. The proposed surveillance requirement is given below:

> Page 5a, New Item D.7.
>
> The power produced by each fuel element while operating at the rated full power shall be calculated before the reactor is operated at greater than 100 kW with less than 100 fuel elements in the core and recalculations shall be performed:
>
> a) at 6±1 month intervals, or
>
> b) whenever a core loading change occurs.
>
> Power per element calculations are not required at any time if core contains at least 100 fuel elements or if reactor power is limited to 100 kW .
>
> If the calculations show than any fuel element would produce more than 22 kW, the reactor shall not be operated in that configuration.

1.e. There are no methods proposed to be used to determine the relationship between the power in the hottest fuel element and the temperature measurements in the instrumented elements. This relationship is very dynamic, changing as a function of fuel burnup, pulsing history, experiment configuration, control rod positions and fuel loading. This changing relationship is the reason we have proposed using calculated power per element values as a limit instead of measured fuel temperatures. We will still measure fuel temperatures as specified in our Technical Specifications; however, that alone will not ensure that the power produced in each fuel element is limited to an appropriate value.

2.a-d. Numerous fuel temperature "limits" are discussed in various documents concerning TRIGA fuel. It is important to differentiate among measured temperature, calculated temperature, steady state temperature and pulsing temperature. I will attempt to clearly discuss temperature limitations of the fuel.

The GA document "The U-ZrH Alloy: Its Properties and Use In TRIGA Fuel", GA E-117-833, Feb. 1980 specifies limiting design basis parameters and values for TRIGA fuel elements. (See Section 4.) It states that a safety limit of 1150 C for pulsing operations is used to preclude a loss of clad integrity when the clad temperature is below 500 C. It also states that a maximum temperature of 750 C is used as the operational (steady state) design basis because the resulting average fuel temperature will result in insignificant calculated fuel growth from temperature-dependent irradiation effects. These limits (1150 C and 750 C) apply to the peak fuel temperatures, not the measured temperatures. Since the peak temperatures cannot be measured in operational reactors, these values must be calculated.

The original USGS Hazards Summary Report, page 8-15, predicts peak fuel temperatures from pulsing to be limited to less than 500 C. This information applies only to the original 8.5 wt% fuel elements that have been used in the reactor to date. Clearly, use of the 12 wt% fuel will result in higher fuel temperatures.

We have predicted a maximum pulsing fuel temperature in the 12 wt% fuel of 770 C. This is higher than the 500 C predicted for the 8.5 wt% fuel but much less than the 1150 C safety limit determined by GA for the TRIGA fuel.

Our Technical Specification D.3 that restricts the measured fuel temperature to 800 C provides a 350 C margin from the 1150 C safety limit for pulsing. This allows for differences between the measured temperature and the calculated peak temperature and for uncertainty in the fuel temperature measurements. Clearly this specification of 800 C is not protecting from overpower, steady state operation, but rather from overpower transient operation. Our proposed limit of 22 kW per element will prevent overpower, steady state operation of any fuel element.

Data of measured fuel temperatures versus power are obtained from experimental results of tests performed by GA. Figure 3-25 of the Torrey Pines TRIGA Mk III SAR shows estimated maximum and average core temperatures versus power for a 100 element core. This figure shows that as power increases from 0.5 MW to 1.5 MW, the maximum fuel temperature would increase from approximately 280 C to 450 C. If a bulk water temperature of 30 C existed, this factor of 3 increase in power resulted in a differential temperature increase of a factor of 1.68 ((450-30)/(280/30)). Thus, assuming a fuel temperature change that is directly proportional to power density change is very conservative in this power range since it overestimates the fuel temperature change by about 78%. This power change from 0.5 MW to 1.5 MW represents a peak element power change from 10.45 kW to 31.35 kW. (A 100 element core with a peak-to-average power factor of 2.09) The figure (3-25) shows the power-to-temperature relationship is approximately linear over this range of 0.5 MW to 1.5 MW and this range clearly covers the limiting power values being proposed for the USGS TRIGA reactor.

The USGS TRIGA is limited to a $3.00 maximum pulse size per Technical Specification E.4. In addition, pulses that result in a peak power greater than 2500 MW require reactor shutdown and investigation of the excessive magnitude, per Technical Specification D.4. Our reply of February 23, 1995 indicated that the 12 wt% fuel has a prompt negative temperature coefficient that is 85% of the 8.5 wt% fuel. Thus, an additional 17.6% temperature increase would be achieved in a core made entirely of 12 wt% fuel compared to a core of 8.5 wt% fuel, for pulses of the same reactivity.

The pulsing temperature predictions for the 12 wt% fuel were previously discussed to be 770 C. The pulsing power scales as the square of the prompt reactivity; therefore a pulse power increase of 38.3% ($(1.176)^2$) would be expected from a core made entirely of 12 wt% fuel. Empirical pulsing data from the GSTR can be used to predict typical power peaks with a 12 wt% core. The pulse rod at the GSTR, in operational cores, has always had a total worth of less than $3.00. Data for $2.80 pulses resulted in the following predictions for a 12 wt% core:

	Existing data (8.5 wt% core)	Predicted data (12 wt% core)
$2.80 pulse peak power (MW)	1075	1490
$3.00 pulse peak power (MW)	1330	1840

These data indicate that pulsing parameters should be within the existing Technical Specifications; therefore, we do not believe any adjustments are required in pulsing-related specifications in the license or Technical Specifications. Operationally, adjustments may have to be made in pulsing insertions but those will be dependent on the amount of 12 wt% fuel in the core, the core size, core configuration, etc. The operational adjustments will be made on a case-by-case basis, within the constraints of the license and Technical Specifications.

3. We have re-evaluated the fission product release accident scenario for a fuel element clad failure in air. The data given in previous analyses assumed infinite volumes for immersion doses, in both the reactor room and unrestricted areas. The following table assumes failure of a 12 wt% element operating at 22 kW for an infinite time, a release fraction of 3.146×10^{-4} and a reactor room volume of 3.48×10^{8} ml.

GASEOUS FISSION PRODUCT RELEASE FROM A 12 wt% FUEL
ELEMENT AFTER OPERATION AT 22 KW.

Isotope	Fission Yield %	Half-life	Activity (Ci) in Core	Activity (Ci) in 12 w% fuel	Reactor room conc
Kr-83m	0.544	1.86 h	4560	100	9.0 e-5
Kr-85m	1.01	4.48 h	8460	186	1.7 e-4
Kr-85	0.054	10.5 y	460	10	9.0 e-6
Kr-87	2.76	76 min	23120	509	4.6 e-4
Kr-88	4.38	2.84 h	36700	807	7.3 e-4
Kr-89	5.47	3.15 min	45830	1010	9.1 e-4
Xe-131m	0.03	12.0 d	250	5.5	5.0 e-6
Xe-133m	0.16	2.19 d	1340	29	2.6 e-5
Xe-133	6.62	5.25 d	55400	1220	1.1 e-3
Xe-135m	1.83	15.3 min	15330	337	3.0 e-4
Xe-135	6.3	9.09 h	52780	1160	1.0 e-3
Xe-137	6.17	3.86 min	51700	1140	1.0 e-3
Xe-138	5.49	14.2 min	46000	1010	9.1 e-4
Br-82	0.125	35.3 h	1044	23	2.1 e-5
Br-83	0.51	2.4 h	4270	94	8.5 e-5
Br-84	0.90	31.8 min	7540	166	1.5 e-4
Br-85	1.1	2.87 min	9220	203	1.8 e-4
I-129	0.6	1.7e7 y	5045	111	1.0 e-4
I-130m	0.51	9.2 min	4280	94	8.5 e-5
I-131	3.1	8.04 d	25970	571	5.2 e-4
I-132	4.38	2.29 h	36700	807	7.3 e-4
I-133	6.9	20.8 h	57800	1270	1.1 e-3
I-134	7.8	52.6 min	65350	1440	1.3 e-3
I-135	6.1	6.585 h	51100	1120	1.0 e-3

At one hour after the release, most of the isotopic airborne concentrations in the reactor room are below the occupational limits for airborne activity, and after six hours the reactor room would no longer be an airborne radioactivity area.

Data were used from Reg. Guide 1.109 and Appendix B of the revised 10 CFR 20 to calculate radiation doses from both submersion and inhalation of the released fission products. The krypton and xenon isotopes result in only submersion exposures. Data from 10 CFR 20 are used preferentially to data from RG 1.109. The isotopes for which RG 1.109 data were used are Kr-89, Xe-137 and Br-85. Dose conversions from the 10 CFR 20 data (DAC and ALI) were performed using the following:

1 DAC-hr = 2.5 mRem, and 2000 DAC-hr = 1 ALI.

1 ALI = CEDE of 5 Rems or 50 Rems to critical organ/tissue.

Table 2, Col. 1 air concentrations give 50 mRem in 8760 hrs.

PERSONNEL EXPOSURES FROM 12 wt% FUEL CLAD FAILURE IN AIR

Isotope	Occupational doses (mrem) (stay time in reactor bay)				Public doses (mrem) (stay time near building)			
	1 min	5 min	1 hr	6 hr	1 min	5 min	1 hr	6 hr
Kr-83m	0	0	0	0	0	0	0	0
Kr-85m	<1	2	11	14	0	0	<1	<1
Kr-85	0	0	<1	<1	0	0	0	0
Kr-87	4	18	104	123	<1	<1	<1	1
Kr-88	15	71	461	583	<1	<1	2	2
Kr-89								
Xe-131m	0	0	<1	<1	0	0	0	0
Xe-133m	<1	<1	<1	<1	0	0	0	0
Xe-133	<1	2	15	21	0	<1	<1	<1
Xe-135m	1	6	20	21	0	<1	<1	<1
Xe-135	4	20	135	179	<1	<1	<1	<1
Xe-137	<1	1	1	1	<1	<1	<1	<1
Xe-138	9	40	132	134	<1	<1	<1	<1
Br-32	<1	3	17	24	0	0	<1	<1
Br-83	<1	1	4	5	0	0	<1	<1
Br-84	<1	1	7	7	0	<1	<1	<1
Br-85	<1	<1	<1	<1	0	0	0	0
I-129	330	1578	11104	15146	2	11	79	107
-thyroid	10990	52610	370130	504870	78	373	2625	3580
I-130	4	20	139	185	0	<1	1	1
-thyroid	120	574	3960	5290	1	4	28	38
I-131	257	1230	8650	11780	2	9	61	84
-thyroid	10300	49200	346000	471000	73	349	2450	3340
I-132	7	34	217	270	<1	<1	2	2
-thyroid	90	427	2710	3370	1	3	19	24
I-133	121	578	4020	5420	1	4	29	38
-thyroid	3630	17300	121000	163000	26	123	855	1150
I-134	3	12	68	77	<1	<1	1	1
-thyroid	-	-	-	-	-	-	-	-
I-135	25	118	800	1050	<1	1	6	113
-thyroid	494	2360	16000	21000	4	17	113	149

Exposures from 10 CFR 20 data were calculated in DAC-hrs (for external exposure) and ALI's (for internal exposure) by continuous integration of the exposure rate, using a reactor room ventilation decay constant of 0.022 per minute and isotopic decay constant based on the half lives given in the first table of isotopes. Exposures from Reg. Guide 1.109 data were calculated using Table B-1 or E-7 and the average concentration over the time period stated in the above table.

The reactor room submersion doses (from Krypton and Xenon isotopes) were adjusted for the finite dimensions of the room. A hemisphere of 550 cm radius was used to simulate the reactor room. The equation used to evaluate the dose reduction factor due to finite dimensions is $(1-e^{-ur})^{-1}$. The air density was taken to be 0.001116 at 5000 ft. altitude. This yields the following data:

Energy (MeV)	Dose reduction factor
0.01	1.0
0.05	8.3
0.1	11.9
0.2	15.7
0.4	17.6
0.6	20.7
0.8	23.6
1.0	26.1
2.0	37.1

The exposures in the table of the previous page were totalled to give the following data:

FISSION PRODUCT RELEASE EXPOSURE DATA					
	Occupational dose (mrem)				
	1 minute stay	5 minute stay	1 hour stay	6 hour stay	Annual CFR Limit
TEDE	780	3735	25905	35040	5000
thyroid	25624	122471	859800	1168530	50000
	Maximum public dose (mrem)				
	1 minute stay	5 minute stay	1 hour stay	6 hour stay	Annual CFR Limit
TEDE	9	30	186	355	100
thyroid	183	869	6090	8281	~3000

The conditions required to achieve the doses in the previous table are: (1) worst case loading of a 12 w% fuel element to achieve 22 kW per element, (2) a 12 w% element with essentially no fission product poisons (no burnup), (3) reactor operated continuously at full power for at least 40 days, 24 hours a day, (4) the 12 w% element removed from the reactor tank instantly after shutdown and (5) a large cladding failure occurs in the element immediately after being removed from the water.

Our experience from many high radiation alarm evacuation tests shows that the reactor facility would be evacuated within one minute and the building would be evacuated within 5 minutes. The security personnel would arrive within 5 minutes to evacuate as much of the surrounding area as is necessary. Under these conditions, the 10 CFR limits for occupational and public exposure would not be exceeded. The simultaneous occurrence of the postulated conditions is not considered credible, but represents a worst case scenario.

In summary, historical experience at other research reactor facilities has shown that cores fueled with both 8.5 w% and 12 w% TRIGA fuel can be operated safely in both steady state and pulsing reactors. Past analyses show that the TRIGA fuel behaves essentially the same with the two different uranium loadings. Facility-specific analyses for the GSTR show that it is not credible for a cladding rupture to cause personnel to receive radiation doses above the dose limitations of 10CFR20.

THE FOLLOWING PAGES ARE PROPOSED REPLACEMENT PAGES FOR THE USGS TECHNICAL SPECIFICATIONS.

2. The pool water shall be sampled for conductivity at least weekly. Conductivity averaged over a month shall not exceed 5 micromhos per cm^2. This item is not applicable if the reactor is completely defueled and the pool level is below the water treatment system intake.

D. Reactor Core

1. The core shall be an assembly of TRIGA stainless steel clad fuel-moderator elements, nominally 8.5 to 12 wt% uranium, arranged in a close-packed array except for (1) replacement of single individual elements with incore irradiation facilities or control rods; (2) two separated experiment positions in the D through E rings, each occupying a maximum of three fuel element positions. The reflector (excluding experiments and experimental facilities) shall be water or a combination of graphite and water. The reactor shall not be operated in any manner that would cause any fuel element to produce a calculated steady state power level in excess of 22 kW.

2. The excess reactivity above cold critical, without xenon, shall not exceed 4.9% delta k/k with experiments in place.

3. Fuel temperatures near the core midplane in either the B or C ring of elements shall be continuously recorded during the pulse mode of operation using a standard thermocouple fuel element. The thermocouple element shall be of 12 wt% uranium loading if any 12 wt% loaded elements exist in the core. The reactor shall not be operated in a manner which would cause the measured fuel temperature to exceed 800°C.

4 Power levels during pulse mode operation that exceed 2500 megawatts shall be cause for the reactor to the shut down pending an investigation by the reactor supervisor to determine the reason

Any element which exhibits a clad break as indicated by a measurable release of fission products shall be located and removed from service before continuation of routine operation.

7. The power produced by each fuel element while operating at the rated full power shall be calculated if the reactor is to be operated at greater than 100 kW with less than 100 fuel elements in the core. Recalculations shall be performed:

   a) at 6 $\pm$ 1 month intervals, or

   b) whenever a core loading change occurs.

   Power per element calculations are not required at any time that the core contains at least 100 fuel elements or if reactor power is limited to 100 kW. If the calculations show that any fuel element would produce more than 22 kW, the reactor shall not be operated with that core configuration.

E. Control and Safety Systems

1. The standard control rods shall have scram capability and the poison section shall contain borated graphite, or boron and its compounds in solid form as a poison in an aluminum or stainless steel clad.