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Denver, Colorado 80225

Thursday, October 22, 2020

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: U.S. Geological Survey TRIGA Reactor, Docket Number 50-274, License Number R-113, License Amendment Request LAR 14, Supplemental Information Letter 02 Provided in Response to Audit Visit on 2020-10-15

Purpose

By way of this letter, the U.S. Geological Survey (USGS) is providing supplemental information requested by NRC via during an audit visit on 15 October 2020, performed as indicated by ADAMS accession number ML20279A591, regarding license amendment request #14 by the USGS TRIGA Reactor facility.

Discussion

At the request of the NRC Project Manager during the audit visit, additional information is being supplied via this letter in support of license amendment request #14, ADAMS accession number ML20275A267, are being submitted by way of this letter. Six items are to be addressed as follows:

1. We request to modify our requested added license condition 2.B.2.f to read:

   f. to receive, possess, but not use or separate, in connection with the operation of the facility, up to 4 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA-type reactor fuel received from the VTT Technical Research Centre of Finland.

   This is being requested to clarify and separate the fuel in question from other such fuel that is currently in the USGS inventory.

2. We request to add two additional license conditions to specify the special nuclear material and byproduct materials acquired in tandem with the fuel, that would read as conditions 2.B.2.g and 2.B.3.h in order:

   g. to receive, possess, but not use or separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of other facilities in the form of TRIGA-type reactor fuel received from the VTT Technical Research Centre of Finland.

h. to receive, possess, but not use or separate, in connection with operation of the facility, such byproduct materials as may be produced by operation of other facilities in the form of TRIGA-type reactor fuel received from the VTT Technical Research Centre of Finland.

These conditions are being requested to clarify and separate the materials in question from other such materials currently in the USGS inventory.

3. Concerning the non-fuel research reactor parts or components stated in the original license amendment request, the items in question are the shipping cask internal fuel baskets, which are expected to have a small degree of removable contamination present when possession is transferred to the USGS. All expected reactor parts or components (i.e., baskets) received with the proposed fuel shipment will be within the requirements of the current license condition 2.B.3.e., and thus, no change to license condition is needed. Also, with regard to the applicability of 10 CFR Part 37, "Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material," all isotopes and associated activities of the reactor parts and components to be received with the fuel shipment are expected to be well below the limits stated in Appendix A to that regulation, and will be confirmed upon receipt through removable contamination testing and portable radiation detector measurements.

4. Concerning the applicability of 10 CFR 70.24, "Criticality Accident Requirements," the amount of material to be stored dry and not under water (i.e. the damaged TRIGA fuel elements in the storage pits) is below the limits listed in the regulation, and therefore does not impose any of the listed requirements. If additional material is to be stored dry and not underwater such that the total amount is above the limits in 10 CFR 70.24, the requirements of the regulation will be followed.

5. In order to provide clarity on the submission of attachment 6 to the original license amendment request, an attachment describing the MicroShield case analyses is provided to this letter. The attachment contains additional details to the geometry, isotope inventories, and more adequately summarizes the work done on these simulations.

6. Concerning demonstration that 20 centimeters of space between hexagonal fuel storage racks in the pool being adequate to ensure that each rack is unaffected by neutrons from any adjacent racks, multiple sources of information are cited:

    a. Using simplified calculations, a result of approximately 1.05 cm is obtained for the mean free path (MFP) of thermal neutrons in room temperature water, which would be the case for all populated racks stored in the pool. With a 20-centimeter spacing between adjacent racks, this provides for many MFPs, effectively decoupling the racks with regards to neutron transmission.

    b. The publication cited in the license amendment request, "Benchmarking criticality analysis of TRIGA fuel storage racks" by Robinson, DeBey, and Higginbotham, available at http://dx.doi.org/10.1016/j.apradiso.2016.08.019 with proper subscription, demonstrates through empirical measurements with validated neutronic modeling that the racks are effectively decoupled in several configurations. Specifically, table 6 is reproduced here to show the results of stacking multiple racks vertically on top of one another on a wall of the reactor tank, which provides a minimum of 13 inches or 33 centimeters of distance between fueled sections of the elements:

Table 6
Calculated results showing effect of stacking racks with mixed fuel.

Configuration	k_{eff}
Single rack on wall	0.7185 ± 0.0003
Two racks stacked vertically on wall	0.7192 ± 0.0003
Three racks stacked vertically on wall	0.7187 ± 0.0003

Also included from the publication, and more relevant to the 20 centimeter parameter established, figure 9 shows the arrangement of racks in the reactor tank along the interior wall analyzed at 20 centimeter spacing with table 7 showing the results of the analysis:

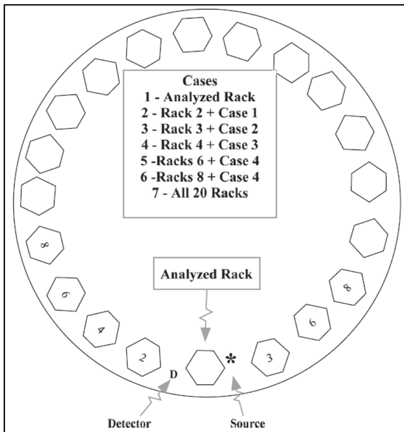

Fig. 9. Locations of hexagonal storage racks along the outer tank wall for the 7 cases of calculated keff as a function of adding additional rack storage.

Table 7
Calculated results of having fuel storage racks 20 cm apart around reactor tank.

Number of Racks (20 cm spacing between any two racks)	k_{eff}
1	0.7056 ± 0.0001
2	0.7075 ± 0.0001
3	0.7082 ± 0.0001
4	0.7085 ± 0.0001
6	0.7085 ± 0.0001
8	0.7085 ± 0.0001
20	0.7087 ± 0.0001

The publication presents additional data regarding the experimental setup and validation steps, but overall presents a demonstrable lack of neutron coupling between racks at the specified spacing.

c. Additional simulations were performed and submitted in response to a request for additional information during license renewal, ADAMS accession number ML16277A216, question 2 response and attachment 3, demonstrating the effective multiplication factor of two racks spaced at 20 centimeters apart fully populated with fresh 12 wt% fuel elements in response. The results, in conjunction with those presented by the publication above, demonstrates that with the most reactive fuel potentially held by the facility the neutronic coupling between hexagonal fuel racks is negligible in the proposed configuration.

d. The safety analysis for the hexagonal fuel racks as performed by General Atomics was submitted in response to a request for additional information for relicensing of the GSTR, ADAMS accession number ML16110A008, technical review question 3 response and attachment 1, which analyzed the effective multiplication factor of the racks in a stacking configuration.

Contact
If you have any questions regarding this matter, please contact me at (303) 236-4727.

Affirmation
I declare under penalty of perjury that the foregoing is true and correct.

Digitally signed by
JONATHAN WALLICK
Date: 2020.10.22
12:20:51 -06'00'

Sincerely,
Jonathan Wallick, Reactor Director

Copied to:
Dr. Robert Horton, Reactor Administrator, USGS
Geoffrey Wertz, Project Manager, US NRC
Craig Bassett, Inspector, US NRC

Attachments:
(1) MicroShield Cases – Assumptions and Detailed Summaries.

Attachment 1

Microshield Cases – Assumptions and Detailed Summaries

Dose Rate Analyses Assumptions

- No fuel self-shielding of the fuel or other materials.
- Activities used were produced from SERPENT code output with t=2015, using radionuclide inventory fuel element as the model for all elements.
- Nuclides with half-life less than 100 days ignored.
- Standard fuel storage rack modeled as cylinder with 5 inches radius and 28 inches in height, with source material as air (no self-shielding).
- SDFC fuel storage rack modeled as a cylinder with 5 inches radius and 42.7 inches in height, with source material as air (no self-shielding).
- Activity is evenly distributed throughout the source cylinder.
- Nuclides with full rack activity (worst-case element activity x 19) less than 1 Ci neglected.
- Cases with lead shielding thickness ranging from 0 to 9 inches of lead analyzed.
- Ba-137m activity set to Cs-137 activity due to equilibrium.
- Y-90 activity set to Sr-90 activity due to equilibrium.

Description of MicroShield Cases

Canned Elements Worst 6 Case

- The “Canned Elements Worst 6 Case” involves simulating a full rack of canned elements, which is comprised of 6 total fuel elements paired up into 3 total Sealed Damaged Fuel Canisters. The source term is calculated by finding the highest-activity SDFC element from the information spreadsheets provided by DOE and VTT, and then multiplying it by 6, resulting in a conservative analysis that is agnostic to any possible rearrangement.
- The source is modeled in the shape of a cylinder, with radius 5 inches and length 42.7 inches. This represents the full radius of the storage pit, and the height of the entire SDFC can; however, the fuel inside the can is much shorter, around 28 inches tall, so the shape of the source is conservative, as the top of the actual source will be further away from the top of the storage pit.
- The material of the source is modeled as air, given that the pit is not intended to be flooded in normal storage. However, this ignores any self-shielding that the fuel and canister will provide, resulting in a very conservative estimate.
- The source is also assumed to be evenly distributed throughout this hypothetical cylinder.
- The dose calculation point is about 12 ft 2 inches above the bottom of the pit, which is a few inches above the deck.
- Multiple cases are run for different thicknesses of lead shielding; this was done to inform the design of the shield plug and to demonstrate the dose rate’s sensitivity to any change in thickness of the shield; the final design thickness is 8 inches.
- The dose rates reported for the final results will be the “Dose rate mRad/hr with buildup column”, as the results will be in the correct units and account for buildup in the shield.

RESULTS (Taken from Microshield output):

Lead Thickness (in)	Dose Rate (mRad/hr with Buildup)
0	1.316E+3
1.125	2.798E+1
2.25	1.489E+0
3.375	7.626E-2
4.5	1.151E-2
5.625	2.750E-3
6.75	7.018E-4
7.875	**1.793E-4**
9.00	4.544E-5

The bolded row indicates the dose rate that will most closely match the designed shield plug; it is a very low dose rate, essentially background compared to the radiation readings normally present above the deck plates.

1-4 Racks Worst Case

- Each of these cases simulates a different number of TRIGA fuel storage racks being dry-stored in a storage pit. This situation is not anticipated to be used at GSTR upon receipt of the VTT fuel, but it is analyzed to show that in the case of a full core and tank unload, even with the extra fuel received from VTT, dry storage in the pits results in acceptable dose rates once the shield plug is placed.
- Each rack stacks on top of the previous one, with each being 28 inches in height. Thus, the source is modeled as a cylinder with radius of 5 inches and height of 28 x *number of racks* inches.
- The source inventory was determined by finding the highest-activity fuel element out of the VTT shipment and multiplying its inventory by 19 (the number of fuel elements in each rack) and the number of racks stacked in each case.
- The material of the source is air, given that the elements will be dry-stored and that this will result in a conservative estimate due to the neglect of self-shielding in any fuel or structural material.
- The source is also assumed to be evenly distributed throughout this hypothetical cylinder.
- The dose calculation point is about 12 ft 2 inches above the bottom of the pit, which is a few inches above the deck.
- Multiple cases are run for different thicknesses of lead shielding; this was done to inform the design of the shield plug and to demonstrate the dose rate's sensitivity to any change in thickness of the shield; the final design thickness is 8 inches.
- The dose rates reported for the final results will be the "Dose rate mRad/hr with buildup column", as the results will be in the correct units and account for buildup in the shield.

Results (from Microshield Outputs 1-4 Racks Worst Case files)

Lead Thickness (in)	Dose Rate 1 Rack (mRad/hr with Buildup)	Dose Rate 2 Racks (mRad/hr with Buildup)	Dose Rate 3 Racks (mRad/hr with Buildup)	Dose Rate 4 Racks (mRad/hr with Buildup)
0	7.404E3	1.953E4	4.35E4	1.045E5
1.125	6.172E3	1.626E3	3.616E3	8.640E3
2.25	4.844E1	1.275E2	2.833E2	6.746E2
3.375	6.406E0	1.685E1	3.740E1	8.891E1
4.5	1.548E0	4.069E0	9.026E0	2.144E1
5.625	4.675E-1	1.229E0	2.725E0	6.465E0
6.75	1.461E-1	3.839E-1	8.51E-1	2.015E0
7.875	**4.562E-2**	**1.199E-1**	**2.656E-1**	**6.280E-1**
9.00	1.415E-2	3.719E-2	8.236E-2	1.944E-1

Final Summary Statement:

Even with 4 racks full of fuel of the highest isotope inventory, the dose rate above the deck plates will be less than 1 mRem/hr.