10 CFR 50.90

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January 17, 2012

Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE APPLICATION FOR LICENSE RENEWAL (TAC NO. ME1587)
(License R-84, Docket 50-170)

Sir:

After conversation with Mr. Walter Meyer, we are submitting a complete revision of our response to Question 12 of your initial RAI. Our November 28, 2011 response to Question 12 should be withdrawn and replaced by this version. The only changes from the November 28, 2011 version are indicated by a vertical line in the left margin on page six. Because this item contains security-related information, we request that all eight pages be withheld from public disclosure under 10 CFR 2.390.

If you need further information, please contact Mr. Steve Miller at 301-295-9245 or millers@afrri.usuhs.mil.

I declare under penalty of perjury that the foregoing and all enclosed information is true and correct to the best of my knowledge. Executed on January 17, 2012.

Enclosure:
as

MARK A. MELANSON
COL, MS, USA
Director

AD20
NRR

12. NUREG-1537, Part 1, Section 13.1.6 provides guidance for the licensee to discuss events that could result from experiment malfunction. The licensee is requested to justify its assumption that the release of irradiation Argon accident scenario is the worst conceivable case for radiological consequences from an experiment. The licensee should present a range of experimental malfunction accidents considered. The Argon activation assumptions and calculations should be presented in more detail.

All experiments performed as part of the TRIGA reactor operations are reviewed by the Reactor and Radiation Facilities Safety Subcommittee and supervised by trained, licensed, supervisory personnel. The Technical Specifications contain requirements that must be met before performing experiments using the AFRRI-TRIGA reactor, including the requirement that the failure of one experiment cannot contribute to the failure of any other experiment. Although improbable, an experiment could fail and therefore the consequences of a variety of experimental accidents have been considered.

The most common experiments performed at AFRRI involve the irradiation of biological samples. The radiological consequences from the failure of an experiment of this type are very minimal, as failure would not pose any risk to the reactor structure itself or result in a release of significant quantities of radioactive material to the staff or public.

The consequences of experiment malfunction of non-biological samples are described below:

According to the Technical Specifications, the irradiation of explosive materials in quantities greater than 25 mg is prohibited. Smaller quantities may be irradiated assuming they are housed in a container capable of withstanding a pressure burst greater than twice the pressure resulting from detonation of the sample. The calculations demonstrating the ability of the container to withstand the pressure burst are to be reviewed by the Reactor and Radiation Facilities Safety Subcommittee and approved by the Reactor Facility Director. Failure of an explosive experiment therefore does not classify as a worst conceivable event.

Samples containing corrosive material must be doubly encapsulated according to the Technical Specifications and therefore the consequences of a failure are limited. Failure of a corrosive experiment in the reactor pool would be diluted by the primary coolant and, while resulting in the need for a cleanup and inspection of reactor fuel and instrumentation, would not present a worst case radiological event.

It is also notable that the sum of all experiments will not exceed $3.00. The AFRRI-TRIGA reactor is capable of pulsing up to $3.00 and has proven that changes in reactivity of this magnitude are not damaging to the reactor. Therefore, simultaneous failure of all experiments would result in reactor conditions that are more conservative than a standard pulse operation.

Another experiment identified in the SAR as the irradiation of a 20 liter container of argon gas in ER1 for one hour at 1 MW. This results in a total argon-41 inventory of 5.6 Ci. Release of this quantity from the reactor stack yields an effective dose to the closest member of the public of 0.2 mrem and is not considered the worst case radiological event resulting from an experiment malfunction.

The irradiation of fueled experiments at AFRRI are limited so that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.0 Ci and the maximum strontium-90 inventory is not greater than 5 mCi. Given the fission product yields reported in NUREG/CR-2387, the 1.0 Ci limit of radioiodines 131 through 135 is reached before 5 mCi of strontium-90 in fueled experiments. The release of these quantities directly from the AFRRI stack has been analyzed and determined to represent the worst case scenario of a release of radioactive material from experimental failure. In this accident scenario, a fueled experiment is irradiated until 1.01 Ci of radioiodines 131-135 are present.

For this experiment, it is assumed that 1 g of 19.75% enriched LEU is irradiated in the AFRRI core for 42 minutes at 1 MW. The assumed thermal neutron flux at this power level and sample location is $1x10^{13}$ n/cm^2-s. The source term for this experiment was generated using ORIGEN, with radioisotope activities of interest shown in Table 1.

TABLE 1. ORIGEN source term for fueled experiment.

Isotope	Half Life	Activity in Experiment (mCi)	Activity Released to Environment (mCi)
Br-82	35.3 h	4.00E-04	1.00E-04
Br-83	2.4 h	4.15E+01	1.04E+01
Br-84m	6.0 m	9.55E+00	2.39E+00
Br-84	31.8 m	3.38E+02	8.46E+01
Br-85	2.87 m	6.95E+02	1.74E+02
Br-86	55.5 s	9.86E+02	2.46E+02
Br-87	55.9 s	1.18E+03	2.95E+02
I-131	8.02 d	5.35E-01	1.34E-01
I-132	2.28 h	2.46E+00	6.16E-01
I-133	20.8 h	3.48E+01	8.70E+00
I-134	52.6 m	7.17E+02	1.79E+02
I-135	6.57 h	2.56E+02	6.40E+01
I-136	83.4 s	1.50E+03	3.75E+02
Kr-83m	1.86 h	4.69E+00	4.69E+00
Kr-85m	4.48 h	6.72E+01	6.72E+01
Kr-85	10.76 y	1.16E-04	1.16E-04
Kr-87	76.2 m	4.61E+02	4.61E+02
Kr-88	2.84 h	3.17E+02	3.17E+02
Kr-89	3.15 m	2.59E+03	2.59E+03
Xe-131m	11.9 d	3.20E-06	3.20E-06
Xe-133m	2.19 d	1.34E-02	1.34E-02
Xe-133	5.24 d	4.96E-02	4.96E-02
Xe-135m	15.3 m	1.10E+02	1.10E+02
Xe-135	9.1 h	1.12E+01	1.12E+01
Xe-137	3.82 m	3.47E+03	3.47E+03
Xe-138	14.1 m	3.16E+03	3.16E+03

It is very conservatively assumed that 25% of the halogens released from the sample into the fueled experiment sample holder are eventually available for inhalation by a radiation worker in the reactor room or a member of the public in the unrestricted environment. This value is based on historical usage and recommendations (Ref. 1-9), where Ref. 1 recommends a 50% release fraction for the halogens from the gap of a fuel element to the air. For the purpose of evaluating the consequences of a failed fueled experiment, the release fraction due to plateout from the gap of a fuel element is assumed to be equal to the release fraction due to plateout from the sample holder. Ref. 2 and Ref. 3 apply a natural reduction factor of 50% due to plateout in the reactor building. The 25% total halogens released results from combining the 50% release from the sample holder with the 50% plateout. However, this 25% value appears to be quite conservative, as Ref. 6 and Ref. 7 quote a 1.7% release fraction from the gap of a fuel element rather than a 50% release fraction from the gap. The experience at TMI-2, along with recent experiments, also indicates that the 50% halogen release fraction from the gap is much too large and reports that possibly as little as 0.06% of the iodine reaching the cladding gap may be released into the reactor room, 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 (Ref. 7). It is reasonable to assume that this same reaction occurs in the fueled experiment sample holder, therefore reducing the amount of radioiodine released from the fueled experiment sample holder to the reactor room. It is assumed that 100% of the noble gases are available for release to the unrestricted environment.

This accident analysis assumes that 100% of the fission products present in the sample are released to the sample holder, with no restrictions on release from the sample matrix itself (unlike the fuel element gap release where an additional fission product reduction is attributed to the design of the TRIGA fuel matrix and its ability to restrict the release of fission products). Because of this conservative assumption, the physical state of the fueled experiment does not need to be specified. For example, it is understood that a larger fraction of fission products release from a liquid sample than a solid sample. By assuming 100% release, the accident analysis provides a true worst case scenario regardless of the physical properties of the fueled experiment sample. In reality, both liquid and solid fueled experiments would restrict the release of fission fragments to the sample holder to some extent, thus reducing the doses to radiation workers and members of the public.

The minimum distance to the unrestricted environment, as well as the minimum distance to the nearest occupied building, are assumed to be in the same direction as the prevailing wind. These assumptions result in the highest possible radiation doses to members of the public.

For any atmospheric stability (Pasquill) class, a ground-level release always leads to a higher effluent concentration at any given distance than an elevated release. Accordingly, it is assumed for this accident analysis that only ground level effluent releases occur, and no credit is taken for either release heights or building wake effects. In reality, the release of radioactive material would emit from the AFRRI stack at a height of 13 meters; resulting in lower doses to the public than those reported in Table 3. A COMPLY calculation estimates the dose at 10 meters from the AFRRI stack to be reduced by a factor of 10 when accounting for the elevated release. Furthermore, atmospheric modeling indicates that the more stable the atmospheric class and the lower the wind speed, the higher the effluent

concentration. Therefore, this analysis assumes both the most stable atmospheric class (Pasquill F) and a low wind speed (1 m/s) are present. The time that a receptor is exposed to the plume is determined by calculating the time required to exhaust the reactor room at the standard ventilation exhaust rate. For this analysis, the time is 9.1 min.

The methodology for atmospheric diffusion models presented in NRC Regulatory Guide 1.145 are used (Ref. 10) in the accident analysis. For distances greater than 100 m, the values for horizontal and vertical dispersion coefficients are also taken from Regulatory Guide 1.145. For distances from 10 m to 100 m, not addressed in Regulatory Guide 1.145, data from the OSTR SAR are used (Ref. 11). The values for the dispersion coefficients and x/Q are given in Table 2.

TABLE 2. Atmospheric Dispersion Coefficients and x/Q Values for Pasquill F and Mean Wind Speed of 1 m/s.

Distance (m)	σ y (m)	σ z (m)	x/Q (s/m^3)
10	1.29	1.04	5.93E-02
50	2.45	1.2	2.71E-02
100	3.9	2.2	9.27E-03
150	6.18	3.22	4.00E-03
200	8.21	4.13	2.35E-03
250	10.21	4.98	1.57E-03

Furthermore, it is assumed that all of the fission products are released to the unrestricted area by a single reactor room air change, which would maximize the dose rate to persons exposed to the plume during the accident.

Additional parameters used in this accident are:

- **Reactor room ventilation exhaust rate: 1.68 m^3/s**
- **Reactor room volume: 917 m^3**
- **Receptor breathing rate: 3.3x10^{-4} m^3/s (NRC "light work" rate)**
- **Dose conversion factors:**

  Internal based on DOE/EH-0071 (Ref. 12)

  External based on DOE/EH-0070 (Ref. 13)

The committed dose equivalent (CDE) to the thyroid and the committed effective dose equivalent (CEDE) for members of the general public at a given distance downwind from the facility for all isotopes of concern are calculated by:

$$(CDE\ or\ CEDE)_D = \sum_i \left[\frac{\left(\frac{x}{Q}\right)_D BR\ DCF_{int,i}\ A_i \lambda_v \left(e^{-\lambda_i t_1} - e^{-\lambda_i t_2}\right)}{\lambda_i} \right]$$

$(x/Q)_D$ = atmospheric dispersion factor at a given distance D (s/m^3)

BR = breathing rate (m^3/s)

$DCF_{int,i}$ = internal dose conversion factor for isotope i (mrem/μCi)

A_i = initial activity of isotope I (μCi)

R_v = ventilation of air from the reactor room (m^3/s)

V = reactor room volume (m^3)

λ_v = ventilation constant = R_v/V (s^{-1})

λ_i = decay constant for isotope i (s^{-1})

t_1 = time when the plume first arrives at the receptor point (s)

t_2 = time when plume has passed the receptor point (s)

The deep dose equivalent (DDE) to members of the general public at a given distance downwind from the facility for both the thyroid and whole body are each calculated by:

$$(DDE_{thyroid}\ or\ DDE_{WB})_D = \sum_i \left[\frac{\left(\frac{x}{Q}\right)_D DCF_{ext,i}\ A_i \lambda_v \left(e^{-\lambda_i t_{i1}} - e^{-\lambda_i t_2}\right)}{\lambda_i} \right]$$

$DCF_{ext,i}$ = external dose rate conversion factor for isotope i (mrem m^3/μCi s)

For calculating the dose to occupational workers in the reactor room, a stay time of 5 minutes is used. Experience indicates that the reactor room can easily be evacuated in less than 2 minutes however; the value of 5 minutes is used to account for any time the worker may be delayed performing a task. The CDE and CEDE for personnel in the reactor room for a given stay-time may each be calculated by:

$$(CDE\ or\ CEDE)_{ST} = \sum_i \left[\frac{DCF_{int,i}\ A_i\ BR\ \left(1 - e^{-\lambda_{eff} t_{ST}}\right)}{\lambda_{eff}\ V} \right]$$

$\lambda_{eff} = \lambda_i + \lambda_v$

t_{ST} = stay time of personnel

The DDE to personnel in the reactor room for a given stay time for both the thyroid and the whole body are calculated by:

$$(DDE_{thyroid} \text{ or } DDE_{WB})_{ST} = \sum_i \left[\frac{DCF_{ext,i}\, A_i\left(1 - e^{-\lambda_{eff} t_{ST}}\right)}{\lambda_{eff}\, V}\right]$$

The results of these calculations are shown in Tables 3-5. In all cases, doses for the general public and occupational workers are below the annual dose limits specified by 10 CFR 20.

There are two different scenarios analyzed in this accident. In scenario #1, the isolation dampers fail following the release of radioactive material into the reactor room. As a result, the radioactive material is vented from the AFRRI stack to the unrestricted public. In scenario #2, the isolation dampers operate as designed and limit the radioactive material release from the reactor room. This latter scenario results in a higher exposure to the reactor staff member in the reactor room. As the radioactive materials disperse in the reactor room, the room becomes a source term for external exposure to staff members within the building, as well as to members of the public outside in the vicinity of the AFRRI facility. Although the reactor room does not completely seal when the dampers are closed, the slow leakage of radioactive material results in a lower dose to the public than the instantaneous release analyzed in scenario #1. Therefore, the release through room leakage as an internal exposure is not detailed in this analysis.

TABLE 3. Radiation Doses to Members of the Public for Scenario #1.

Distance (m)	TEDE (mrem)
10	76
50	33
100	11
150	5
200	3
250	2

TABLE 4. Occupational Radiation Doses in the Reactor Room for Scenario #1.

Reactor Room Occupancy (min)	TEDE (mrem)
5	401

TABLE 5. Occupational Radiation Doses in the Reactor Room for Scenario #2.

Reactor Room Occupancy (min)	TEDE (mrem)
5	508

Direct external exposures to individuals outside of the reactor room originating from airborne radioactive material inside the reactor room are calculated assuming the source term to be the entire reactor room volume. These exposure rates encompass three distinct locations, and are calculated

using MicroShield ™ V8.02. Receptor A is located 3 ft. from any reactor wall, but not within the reactor room. Receptor B is located 20 ft. from any reactor wall, with an additional concrete block wall between receptor B and the reactor wall. Receptor C is located 100 ft. from any reactor wall, with an additional concrete block wall between receptor C and the reactor wall.

Receptor A represents the staff member in closest proximity to the reactor, typically able to evacuate the area in less than 2 minutes. To incorporate further conservatism, the evacuation time for Receptor A is set at 5 minutes.

Receptor B represents the closest proximity to the reactor's Controlled Access Area within the AFRRI complex. Receptor B's location represents the highest exposure rate to a staff member who is outside of the Controlled Access Area. All other staff locations throughout AFRRI are a greater distance from the reactor room, and have significantly more shielding. From past emergency drill experiences, it is estimated that the entire AFRRI complex can be evacuated in less than 20 minutes.

Receptor C represents the closest location of an emergency evacuation assemblage point. For the purposes of this calculation, it was assumed that a member of the public could stay at this assemblage point for 2 hours following the accident. In reality, personnel would be evacuated to a more distant location in this type of accident. The exposures for each receptor are presented in Table 6.

TABLE 6. Radiation Exposures Outside of the Reactor Room in Scenario #2.

Receptor	Exposure Rate (mR/hr)	Evacuation Time (min)	Exposure (mR)
A	118	5	9.9
B	21	20	7
C	2	120	4

It is important to note that these dose rates are at the time of the failure of the fueled experiment and do not include decay corrections for the duration of any of the evacuation times. This adds a significant conservatism into the estimated exposures. The results presented indicate the contribution of exposure from the source term inside the reactor room to anyone outside the reactor room is well within the 10 CFR 20 limits.

REFERENCES

1. "The Calculations of Distance Factors for Power and Test Reactor Sites" DiMunno, JJ. et al., TID-14844, U.S. Atomic Energy Commission, March 1962.
2. Regulatory Guide 3.33 "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant" U.S. Nuclear Regulatory Commission, April 1977.
3. Regulatory Guide 3.34 "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Uranium Fuel Fabrication Plant" U.S. Nuclear Regulatory Commission, July 1979.

4. Regulatory Guide 1.5 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" U.S. Nuclear Regulatory Commission, June 1974.
5. "A Guide to Radiological Accident Considerations for Siting and Design of DOE Nonreactor Nuclear Facilities" Elder, JC. et al., LA-10294-MS, Los Alamos National Laboratory, January 1986.
6. Nuclear Power Reactor Safety Lewis, EE., John Wiley and Sons, 1977, p.521.
7. Nuclear Engineering, Theory, and Technology of Commercial Nuclear Power Knief, RA., Hemisphere Publishing, 1992, pp.353, 431.
8. "Fuel Elements for Pulsed TRIGA Research Reactors" Simnad, MT. et al., Nuc. Tech. 28, January 1976.
9. "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel" Simnad, MT. General Atomic Report E-117-883, February 1980.
10. Regulatory Guide 1.145 "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" U.S. Nuclear Regulatory Commission, August 1979.
11. "Calculated Atmospheric Radioactivity from the OSU TRIGA Research Reactor Using the Gaussian Plume Diffusion Model" Bright, MK. et al., Oregon State University Department of Nuclear Engineering Report 7903, August 1979.
12. "Internal Dose Conversion Factors for Calculation of Dose to the Public" DOE/EH-0071, U.S. Department of Energy, Washington DC, 1988.
13. "External Dose Conversion Factors for Calculation of Dose to the Public" DOE/EH-0070, U. S. Department of Energy, Washington DC, 1988.