10CFR50.90

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
8901 WISCONSIN AVENUE
BETHESDA, MARYLAND 20889-5603

June 20, 2011

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)

Sir:

By letter dated July 19, 2010, the Nuclear Regulatory Commission requested additional information necessary to allow processing of our research reactor license renewal application (License R-84, Docket 50-170).

In subsequent conversations with Mr. Walter Meyer, we were granted an extension until June 30, 2011 to provide an answer for question 4. Our response to that question is enclosed. Remaining to be answered are questions 3, 5, 6, and 12 for which we have requested an additional extension.

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 June 20, 2011.

MARK A. MELANSON
COL, MS, USA
Director

Enclosure:
as

A020
LRR

4. NUREG-1537, Part 1, Section 13.1.1 provides guidance identifying as an acceptable Maximum Hypothetical Accident (MHA) for TRIGA reactors, the failure of one fuel element in air. The potential consequences of the postulated MHA scenario excess and bound all credible accidents. Section 13.2.3 of the SAR discusses the Design Basis Accidents (DBA) providing dose consequences to members of the public. The SAR does not specifically present an MHA analysis. Instead two DBA analyses are presented, which incorporate assumptions that result in doses that appear to not be bounding. Identify an MHA scenario whose potential consequences bound all credible accidents.

The Maximum Hypothetical Accident (MHA) for the AFRRI TRIGA Mark F reactor is cladding failure, in air, of one highly-irradiated fuel element without concomitant radioactive decay, resulting in the instantaneous release of the noble gas and halogen fission products. This instantaneous release bounds all credible accident scenarios, and represents the maximum radiological hazard to personnel under the postulated MHA. The probability of such an accident is considered exceedingly remote; the probability is further reduced under the postulated accident conditions with the complete replacement of all aluminum-clad fuel elements with stainless steel-clad elements.

The assumption underlying this MHA analysis was continuous reactor operation at full licensed power (1.1 MW) for an entire year (365 days) prior to reactor shutdown. Although the assumed continuous rate of operation is within the realm of possibility for the AFRRI TRIGA Mark F reactor, this condition exceeds the known 50-year operating history of the AFRRI reactor. The length of time for continuous, full power operation of the AFRRI reactor has never exceeded more than 12 hours. Thus, enormous conservatism has been incorporated into the radiological consequences for this MHA analysis. Additionally, fuel elements are only moved when the reactor is in a shutdown condition.

In this MHA accident scenario, all halogens and noble gases (except Kr-85) are at their saturation activity. In order to determine the highest power density element within the AFRRI core, it is assumed that the core is loaded in the 85-3 configuration. This is a conservative estimate, given that the 87-3 core configuration would yield a lower value for the highest power density element. As described in Chapter 4 of the SAR, the hot element power factor for the AFRRI reactor operating in steady state at 1.1 MW is 1.56. Therefore, the highest power density element has a power density of 20.2 kW. The fission product inventory of halogen and noble gases for this highest power density element are given in Table 1, and are based upon the fission yield for each isotope. Low-enriched uranium (LEU) TRIGA fuel contains a significant amount of U-238, which leads to the presence of Pu-239 as the fuel approaches end-of-life. To quantify this, depletion calculations for LEU cores have shown that the percentage of fission events originating from Pu-239 at the end-of-life will be 13.3%. Therefore, the difference in fission yield curves for Pu-239 and U-235 are considered in the source term inventory.

TABLE 1. Isotope activities for the highest power density fuel element.

Isotope	Half Life	Activity in Element (mCi)	Activity Released to Environment (mCi)
Br-82	1.47d	[redacted]	[redacted]
Br-83	2.4 h	[redacted]	[redacted]
Br-84m	6.0 m	[redacted]	[redacted]
Br-84	31.8 m	[redacted]	[redacted]
Br-85	2.87 m	[redacted]	[redacted]
Br-86	55.5 s	[redacted]	[redacted]
Br-87	55.9 s	[redacted]	[redacted]
I-131	8.02 d	[redacted]	[redacted]
I-132	2.28 h	[redacted]	[redacted]
I-133	20.8 h	[redacted]	[redacted]
I-134	52.6 m	[redacted]	[redacted]
I-135	6.57 h	[redacted]	[redacted]
I-136	1.39 m	[redacted]	[redacted]
Kr-83m	1.86 h	[redacted]	[redacted]
Kr-85m	4.48 h	[redacted]	[redacted]
Kr-85	10.76 y	[redacted]	[redacted]
Kr-87	1.27 h	[redacted]	[redacted]
Kr-88	2.84 h	[redacted]	[redacted]
Kr-89	3.15 m	[redacted]	[redacted]
Xe-131m	11.9 d	[redacted]	[redacted]
Xe-133m	2.19 d	[redacted]	[redacted]
Xe-133	5.24 d	[redacted]	[redacted]
Xe-135m	15.3 m	[redacted]	[redacted]
Xe-135	9.1 h	[redacted]	[redacted]
Xe-137	3.82 m	[redacted]	[redacted]
Xe-138	14.1 m	[redacted]	[redacted]

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 in Ref. 1-9. Using this data, GA developed a conservative correlation factor for fission product release to be:

$$e = 1.5x10^{-5} + 3.6x10^{3} exp\left\{\frac{-1.34x10^{4}}{T + 273}\right\}$$

At an average fuel temperature of 500°C, the release fraction is $1.22x10^{-4}$. This fuel temperature is 60°C higher than the reported peak fuel temperature of 440°C for the hot element described in Chapter 4 of the SAR. Using a fuel temperature of 440°C would decrease the release fraction by a factor of approximately 3, thus the higher average fuel temperature of 500°C was used in this MHA scenario.

Once the fission products are released to the gap, this activity would then be released to the reactor room air in the MHA scenario. After release into the reactor room air, a further reduction of the halogen activity is expected to occur due to plateout in the building. Thus, the fraction (*w*) of the fission product inventory released from a single fuel element which reaches the reactor room air and, subsequently, the atmosphere in the unrestricted environment is given by:

$$w = e * f * g$$

e = fraction released from the fuel to the fuel-cladding gap

f = fraction released from the fuel-cladding gap to the reactor room air

g = fraction released from the reactor room air to the outside unrestricted environment

It is very conservatively assumed that 25% of the halogens released to the cladding gap are eventually available for release from the reactor room to the outside environment. This value is based on historical usage and recommendations (Ref. 1-9), where Ref. 1 recommends a 50% release value for the halogens from the gap to the air. 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 gap with the 50% plateout. However, this 25% value appears to be quite conservative, since Ref. 6 and Ref. 7 quote a 1.7% release from the gap rather than a 50% release fraction from the gap. The experience at TMI-2, along with recent experiments, indicates that the 50% halogen release fraction is much too large. 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 assumed that 100% of the noble gases are available for release to the unrestricted environment.

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 will 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 MHA analysis that only ground level effluent releases occur, and no credit is taken for either release heights or building wake effects. Furthermore, atmospheric modeling indicates that the more stable the atmospheric class and the lower the wind speed, the higher the effluent concentration. Therefore, this MHA analysis assumed both the most stable atmospheric class (Pasquill F) and a low wind speed (1 m/s) were 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 was used (Ref. 10) in the MHA accident analysis. For distances greater than 100 m, the values for horizontal and vertical dispersion coefficients were 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 was 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 was assumed that all of the fission products were 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 were:

- 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:

$$(DDE_{thyroid}\ or\ DDE_{WB})_D = \sum_i \left[\frac{\left(\frac{x}{Q}\right)_D DCF_{ext,i}\ A_i \lambda_v \left(e^{-\lambda_i t_{11}} - e^{-\lambda_i t_2}\right)}{\lambda_i} \right]$$

$DCF_{ext,i}$ = 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 was 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:

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

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

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:

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

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

There were two different scenarios analyzed in this MHA. 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)	$CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	TEDE (mrem)
10	488	22
50	224	10
100	76	4
150	33	1
200	19	< 1
250	13	< 1

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

Reactor Room Occupancy (min)	$CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	TEDE (mrem)
2	971	37
5	2077	93

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

Reactor Room Occupancy (min)	$CDE_{Thyroid}$ + $DDE_{Thyroid}$ (mrem)	TEDE (mrem)
2	1082	50
5	2698	122

Direct external exposures to individuals outside of the reactor room originating from airborne radioactive material inside the reactor room were calculated assuming the source term to be the entire reactor room volume. These exposure rates encompassed three distinct locations, and were calculated using MicroShield ™ V8.02. Receptor A was located 3 ft. from any reactor wall, but not within the reactor room. Receptor B was located 20 ft. from any reactor wall, with an additional concrete block wall between receptor B and the reactor wall. Receptor C was 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 was 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	7.8	5	0.65
B	1.3	20	0.43
C	0.13	120	0.26

It is important to note that these dose rates are at the time of the failure of the fuel element 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 negligible.

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 PowerNuclear Engineering, Theory, and Technology of Commercial Nuclear 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.