Mr. Alexander Adams
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Subject: Dow Chemical Company- Response to the Request for Additional Information regarding Environmental Assessment for the renewed license of the TRIGA research reactor. License No. R-108; Docket No. 50-264

Dow is providing this information in support of its Application for License Renewal for the Dow TRIGA research reactor. Portions of this document constitute security and emergency response that should be held in confidence by the NRC under 10 C.F.R. 2.790(a)(1) and 10 C.F.R. 9.17(a)(1), because:

i. This information is and has been held in confidence by Dow.
ii. This information is of a type that is held in confidence by Dow and there is a rational basis for doing so because it is sensitive information concerning Dow's physical security and emergency response
iii. This information is not available in public sources and could not be gathered readily from other publicly available information.
iv.

Accordingly, Dow requests that localized site map and photographs be withheld from public disclosure under 10 C.F.R. 2.790 (a)(1) and 10 C.F.R. 9.17(a)(1). The map and photographs are not publically available in keeping with Dow's internal policies and

- Chemical Facilities Anti-terrorism standards CFATS originally authorized in Section 550 of the Department of Homeland Security Appropriations Act of 2007 Pub. L. 109–295.
- U.S. Nuclear Regulatory Commission, Division of Nuclear Security Stapleton, "Designation Guide for Safeguards Information, Criteria and Guidance" DG-SGI-1 321.3 M

The response to the RAI is organized by individual question. Each question will be reiterated and our response will follow.

1. In order to comply with the National environmental Policy Act (NEPA) requirements of title 10 Code of Federal Regulations (CFR) part 51, the NRC must prepare an Environmental Assessment (EA) as part of the license renewal process. Please provide the following additional information.

a. The amount of water usage by the primary coolant system and secondary coolant system and the amount of water discharged to the sewer. Additionally for comparison, list the amount of water used and discharged to the sewer by the DTRR and Dow complex.

    a. DTRR response-

    The coolant loop is a closed-loop system- water is circulated from the pool at 70 gallons per minute. The heat is transferred to the refrigeration unit through glycol in the second loop and R134a in the chiller located outside of building 1602. No water is released to the sewer from this coolant system. The back-up system (Huron heat exchanger) is not in use. The Huron heat exchanger is a shell and tube type. The secondary water passes through the heat exchanger at less than 100 gallons per minute and is directly released to Dow waste water treatment system. The pressure in the secondary loop is kept high than the pool water loop

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to ensure pool water does not leak into the Dow waste water treatment system As stated this Huron system is not used and is flushed monthly with ~ 4500 gallons. The water is released to the Dow waste water treatment system. For comparison the Dow waste water system treats 17.6 million gallons per day.

The following table summarizes water discharged to the Dow Waste Water treatment system.

	Average monthly discharge	Yearly discharge
DTRR	4500 gallons	54,000 gallons
BLDG 1602 (houses DTRR)	50,000 gallons	600,000 gallons
Dow Complex	528 million gallons	6336 million gallons

b. Please provide the environmental dosimeter readings for the previous 6 years and describe the abnormal readings and the reason for the abnormal readings
   a. DTRR Response-

Environmental dosimeters are placed on the exterior walls of the reactor room and record radiation levels in the four cardinal directions around the reactor. Dosimeters are kept in place 24 hours per day, 365 days per year, and are processed quarterly. These readings are intended to represent potential exposure to Dow employees and contractors who are performing work on the Michigan Operations site, and not a fenceline dose, which would be substantially lower.

The following table summarizes the readings on the environmental dosimeters (Deep Dose Equivalent readings) for the previous 6 years. No significant abnormal readings have occurred during this time.

Location	2005 (mrem/ yr)	2006 (mrem/ yr)	2007 (mrem/ yr)	2008 (mrem/ yr)	2009 (mrem/ yr)	2010* (mrem/ yr)
Reactor Console Room (North of Reactor)	11	11	14	14	7	5
East Wall of Reactor Room	39	27	51	42	29	9
South Wall of Reactor Room	N/A**	32***	8	7	4	4
West Wall of Reactor Room	N/A**	33****	83	101	71	30
*1st Quarter results ** Dosimeter not in use at this time ***Represents two quarters on interior of wall. Dosimeter moved to exterior of wall in 2007 ****Represents two quarters of use						

c. Quantify the amount of solid and liquid waste by material form, volume, weight and activity of waste materials transferred each year for the last 6 years
   a. DTRR Response-

The majority of radioactive waste generated at the TRIGA reactor is short-lived radioactive waste. This material is collected and held in decay-in-storage before being disposed as non-radioactive waste.

The TRIGA reactor generates only small quantities of long-lived radioactive waste. The following table summarizes the waste materials transferred over the last 6 years.

	2005	2006	2007	2008	2009	2010
Form	N/A	N/A	N/A	N/A	N/A	Solid
Volume (m3)	0	0	0	0	0	0.15
Weight (kg)	0	0	0	0	0	70
Activity (mCi)	0	0	0	0	0	0.03

d. Please provide the last 6 years annual person-rem per group, number of people in a group, average annual individual dose per group and the maximum individual dose per group. Your groups should account for all persons assigned dosimeters for the purpose of the research reactor
   a. DTRR Response-

   Dosimetry results for persons assigned dosimeters for the research reactor are tracked within two groups – TRIGA reactor employees, who operate and run the reactor, and support personnel in the radiation safety group, such as the Radiation Safety Officer and the Health Physics Technician. Support personnel in the radiation safety group also support other radiation activities at the Michigan Operations site, and their recorded dose would include exposures from sources other than the TRIGA reactor.

   The following table shows a summary of the annual doses received by members of each of these groups (all doses represent deep dose equivalent, as measured on their whole-body dosimeter):

TRIGA reactor employees	**2005**	**2006**	**2007**	**2008**	**2009**	**2010***
Person-rem	0.019	0.020	0.027	0.022	0.025	0.031
# of employees	4	4	4	5	3	3
Average dose (mrem)	5	5	7	4	8	10
Maximum dose (mrem)	7	11	13	10	10	24
Support Personnel	**2005**	**2006**	**2007**	**2008**	**2009**	**2010***
Person-rem	0.004	0.003	0.016	0.004	0.011	0.000
# of employees	3	3	3	3	3	3
Average dose (mrem)	1	1	5	1	4	0
Maximum dose (mrem)	2	2	11	4	11	0
*1st Quarter only						

e. Please provide the estimated dose to the public for the previous 6 years, along with the basis for estimation.
   a. DTRR Response-

   Doses to members of the public are determined by calculation based on radioactive emissions from the reactor and direct exposure from reactor operations. Operations and emissions from the reactor are consistent from year-to-year, and calculations to estimate potential exposures are conservative. The annual exposure to a member of the public present at the fenceline nearest to the reactor facility is less than 3 mrem/yr. The calculations used to estimate radiation exposure to members of the public are summarized in Attachment 1.

2. As part of the environmental assessment review, the NRC must apply to the Michigan State Historic Preservation Office for a section 106 of the National Historic Preservation Act review of the DTRR building. This application requires photographs of the DTRR building and a localized map. Photographs must provide clear views (i.e. subject of the photograph should not be obscured by shadows, trees, cars, or any other type of obstruction) of the DTRR building. A localized map highlighting the location of the DTRR building and including road names that are legible. The items submitted for the Section 106 reviews have typically been made publically available. If it is the desire of the DTRR to have this information withheld from public disclosure, please provide the justification as specified in 10 CRF 2.390 for NRC consideration.

DTRR response

As requested the localized map highlighting the location of the DTRR building and an unobstructed photograph of the DTTR are included in Attachment 2. The map and photographs are not publically available in keeping with Dow's internal policies and

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- U.S. Nuclear Regulatory Commission, Division of Nuclear Security Stapleton, "Designation Guide for Safeguards Information, Criteria and Guidance" DG-SGI-1 321.3 M

Should you have questions or need additional information, please contact the undersigned at 989-638-6932.

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

Executed on Aug 11, 2010

Melinda Krahenbuhl, P.h. D.
Director
Dow TRIGA Research Reactor

Amy L Wagner
Notary Public
Commission Expires: 6/9/2013

Attachment 1
Calculation of Dose to Members of the Public from the Reactor Operations

TRIGA Reactor Exposures to Members of the Public

1. Purpose

The purpose of this calculation is to demonstrate that exposures to members of the public from TRIGA reactor operations will not exposure members of the public to more than the NRC annual allowable limit of 100 mrem/yr and that material released from the lab hoods in the reactor building would not exceed allowable concentrations of airborne releases in Appendix B of 10 CFR Part 20 or the 10 mrem/yr ALARA dose limit for airborne releases in 10 CFR 20.1101(d).

2. Method

Of the radionuclides that are commonly produced during activation of samples in the TRIGA reactor, identify the gaseous/volatile and solid radionuclides with the smallest acceptable release concentration in air in 10 CFR Part 20, Appendix B. Due to differences in release fractions and efficiency of the filtration system, these radionuclides must be identified separately. Using these two radionuclides and bounding throughput assumptions (100 μCi/sample, 1000 samples per year), show that actual releases are well below 10 CFR Part 20, Appendix B concentrations. Note that according to emails from the Reactor Director, only Br-82 is likely to have that much throughput in a year (99.2 mCi handled in 2007). These concentrations are applicable to an annual average concentration, and the hood vents operate 24 hours a day. It will be assumed that the hood vents are down 5% of the time for repairs.

3. Identification of Limiting Radionuclide

The physical states and 10 CFR 20, Appendix B Airborne Release Concentrations for the radionuclides most commonly produced in the reactor with relatively long half-lives (~a day or more) are listed in Table 1.

Table 1. Common Radionuclides Produced in the Reactor

Radionuclide	**Physical State**	**10 CFR 20, Appendix B Concentration**
Hg-203	Volatile	1×10^{-9}
As-76	Volatile	2×10^{-9}
Br-82	Gaseous	5×10^{-9}
I-128	Volatile	2×10^{-7}
Ar-41	Volatile	1×10^{-8}
Cu-64	Solid	3×10^{-8}
Na-24	Solid	7×10^{-9}
Sb-124	Solid	3×10^{-10}
Eu-152	Solid	3×10^{-11}

Based on this information, the most limiting solid radionuclide is Eu-152 at 3×10^{-11} μCi/ml and the most limiting gaseous/volatile radionuclide is Hg-203 at 1×10^{-9} μCi/ml.

4. Calculation of Release Concentration

The concentration of radionuclides in the air released from the laboratory hoods can be calculated by the following formula:

$$C = \frac{Cs \times N \times RF \times FF}{VF \times 24hr/day \times 365.25day/yr \times U}$$

where,

C	=	Concentration of released radionuclide (µCi/mL)
Cs	=	Concentration of radionuclide in sample (µCi/sample) – assumed to be 100 µCi
N	=	Number of samples handled each year – assumed to be 1000
RF	=	Fraction of radionuclide that is released during handling
FF	=	Fraction of radionuclide that passes through the filter
VF	=	Vent flow rate (mL/hr) = 1.7×10^9 mL/hr
U	=	Usage fraction of hood – assumed to be 0.95 (5% of year, hood is not running due to maintenance or repairs)

For the solid radionuclide (Eu-152), RF = 2×10^{-4} (value for drop of loose powder from up to a 3 meter height (NUREG/CR-6642)) and FF = 0.03 (lowest acceptable filtration efficiency of filter is 97%). Using these values in the above formula results in a concentration of Eu-152 in the effluents of 4.2×10^{-14} µCi/mL. The 10 CFR Part 20, Appendix B concentration of releases to the air is 3×10^{-11} µCi/mL, so this concentration is significantly below the 10 CFR Part 20, Appendix B concentration. If a person was to continuously breathe air with the 10 CFR Part 20, Appendix B concentration of a radionuclide in it, their annual dose would be 50 mrem. Therefore, maintaining the concentration at the release point below this value demonstrates that the dose to the maximally exposed member of the public will be less than the NRC public dose limit of 100 mrem/yr. Since the release concentration is more than a factor of 5 below the 10 CFR Part 20, Appendix B concentration, the ALARA dose constraint for airborne releases in 10 CFR 20.1101(d) will not be exceeded. The estimate of dose to a maximally exposed member of the public based on the 10 CFR Part 20, Appendix B concentrations being equivalent to 50 mrem/yr is 0.07 mrem/yr.

For the volatile radionuclide (Hg-203), RF is conservatively assumed to be equal to 1 (100% released) and FF is also assumed to be 1 (100% passes through filter). Using these values in the above formula results in a concentration of Hg-203 of 7.1×10^{-9} µCi/mL. This is higher than the 10 CFR Part 20, Appendix B concentration, so transport calculations will have to be conducted to demonstrate compliance with the NRC public dose limit.

4.1. Calculation of Release of Ar-41

Ar-41 is produced when Ar-40 in air gaps in the reactor (empty sample tubes and rabbit terminus) absorbs a neutron and is activated to Ar-41. The rabbit system discharges into a hood exhaust duct which flows at 1100 CFM (34,000 L/min). The volume of Rabbit Terminus is 301.5 cc (1.25" dia. x 15" long). The weight of Ar in this volume at 1 atm. is 3.64 mg (0.001293 g/cm3 x 0.934% x 301.5 cc). Per Erdtmann NAA tables, production of 41-Ar at 250Kw is 250 dpm/microgram per irradiation minute.

3.64 mg x 252 dpm/ug / 37,000 dps per microcurie = 0.4 uCi/min

The reactor is operated approximately 15 minutes per day with the rabbit system in operation, so total generation rate of Ar-41 is approximately 6 uCi/day.

The rabbit system discharges into a hood exhaust duct which flows at 1100 CFM (34,000 L/min). Again, assuming a 95% usage fraction of the hood, the daily exhaust rate of the hood is:

34,000 L/min x 60 min/hr x 24 hr/day x 0.95 = 4.65×10^7 L/day

Using these numbers, the daily average concentration of Ar-41 being exhausted from the 1602 Building is 1.29×10^{-10} uCi/mL. The 10 CFR 20, Appendix B allowable effluent release concentration of Ar-41 through the air pathway is 1×10^{-8} uCi/ml. The Ar-41 releases from the reactor are 1.3% of the allowable release concentration, which corresponds to a dose of approximately 0.65 mrem/yr, assuming somebody was continuously present at the location of release. This is well below regulatory limits for releases of radioactive material in 10 CFR Part 20 and the ALARA goal of 10 mrem/yr. Therefore, transport calculations will not be performed for the Ar-41 release.

5. Transport Calculations

The concentration of a radionuclide at a given distance from a point source is calculated by (Regulatory Guide 1.111, Nuclear Regulatory Commission, 1977):

$$\chi_{\sec avg} = \sum_j \frac{n_{ij}}{N} \frac{2.032Q}{\overline{u}_j \Sigma_z x} e^{-\frac{h^2}{2\sigma_z^2}}$$

where,

$\chi_{\sec avg}$	=	Concentration of radionuclide in the air, averaged over a 22.5 degree sector (Ci/m^3)
n_{ij}	=	Number of hours that wind is blowing in direction i (towards receptor) in wind speed group j [From wind speed and direction data]
N	=	Total number of hours of wind speed and direction data
Q	=	Release rate of radionuclide (Ci/sec) [calculated to be 3.17e-9 Ci/sec]
u_j	=	Average wind speed in wind speed group j (m/sec) [From wind speed and direction data]
σ_z	=	Vertical diffusion coefficient (m)[From Regulatory Guide 1.111 (U.S. Nuclear Regulatory Commission, 1977), dependent on stability class]
S_z	=	Vertical plume spread with a volumetric correction for a release within the building wake cavity [equivalent to σ_z to conservatively not take credit for building wake effects]
x	=	Distance from release point to receptor (m) [23 m]
h	=	Release height (m) [Assumed to be a ground release (h=0) because vent height is less than two times the building height and the exit velocity is less than 5 times the horizontal wind velocity]

Using this formula and wind speed and direction data from Flint, MI as supplied with the CAP88-PC version 2.1 code (U.S. Environmental Protection Agency, 2000), the concentration of Hg-203 is calculated to be 3.08e-12 Ci/m^3 at the nearest fence, which is 75 feet to the west of the reactor building . If a member of the public was continuously present at this location for 24 hours a day, 365 days per year, the dose to this individual would be calculated by:

$$D = \chi_{\sec avg} \times BR \times DCF \times 365$$

D	=	Dose from radionuclide (mrem/yr)
$\chi_{\sec avg}$	=	Concentration of radionuclide in the air, averaged over a 22.5 degree sector (Ci/m^3)
BR	=	Breathing Rate (m^3/day) [22.8 m3/day – value for reference man from Radiological Health Handbook (1970)]
DCF	=	Dose Conversion Factor for Inhalation (mrem/Ci) [7.3e6 mrem/Ci for Hg-203 from FGR 11 (U.S. Environmental Protection Agency, 1988)]
365	=	Days per year

The dose from Hg-203 to this individual would be 0.19 mrem/yr.

6. **Direct Exposure:** The nearest fenceline to the TRIGA reactor is located 75 feet to the west of the reactor facility. The west wall of the TRIGA reactor is approximately 10 feet from the reactor core. The environmental dosimeter located on the west wall of the TRIGA reactor building has shown an average exposure level of 85 mrem/yr over the last three years, with a maximum of 101 mrem. Treating the reactor core as a point source of radiation exposure, the inverse-square law predicts a maximum fenceline exposure rate of:

Fenceline Dose = 101 mrem/yr * (10 ft)^2/(75 ft)^2

Fenceline Dose = 1.8 mrem/yr

This calculation ignores the potential attenuation of additional shielding materials between the west wall of the reactor and the fenceline and attenuation in the air, and therefore, should be conservative.

7. **Conclusion:** The calculated dose to the maximally exposed offsite individual assuming that all of the material produced in the reactor was the limiting solid or the limiting gaseous/volatile radionuclide is very small (0.07 mrem/yr for Eu-152, the limiting solid radionuclide; 0.19 mrem/yr for Hg-203, the limiting volatile/gaseous radionuclide, and 0.65 mrem/yr from the generation and release of Ar-41 from the reactor). The actual dose that would be received by a member of the public would be smaller than any of these values because much of the activity produced in the reactor would not be these limiting radionuclides and no credit was taken for the reduction of dose due to transportation of the Ar-41 from the release point to a location where members of the public may be exposed. The summation of these exposures plus the 1.8 mrem/yr calculated for direct exposure from reactor operations results in a total estimated exposure level from reactor operations of 2.7 mrem/yr.

Therefore, it can be concluded that direct radiation exposure and the releases from the lab hoods and the reactor will not expose members of the public to doses greater than the NRC limits in 10 CFR 20.1301 or the ALARA limit for airborne releases of 10 mrem/yr in 10 CFR 20.1101(d).