DOW CHEMICAL TRIGA
RESEARCH REACTOR
LICENSE NO. R-108
DOCKET NO. 50-264

TECHNICAL RAI RESPONSES (10/12/2011)

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

DOW

The Dow Chemical Company
Midland, Michigan 48667

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

Subject: The Dow Chemical Company- License No. R-108; Docket No. 50-264

Enclosed are the DTRR Revised responses to RAI questions 11, 14, 15-1, 16, 17-1, 17-3, 18, 19, 41, 52, 53, 54, 56 in support of the license renewal.

Should you have any questions or need additional information, please contact the Facility Director, Paul O'Connor, at 989-638-6185.

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

Executed on October 12, 2011

Paul O'Connor

Paul O'Connor, Ph.D.
Director
Dow TRIGA Research Reactor

Subscribed and sworn to before me this 12 day of October, 2011

Stacy L. McKeon
Notary Public
________County, Michigan
My Commission Expires:

____

STACY L. McKEON
NOTARY PUBLIC
COUNTY OF BAY
My Commission Expires
June 28, 2017
Acting in the County of
Midland
STATE OF MICHIGAN

cc: Wayde Konze, R&D Director - Analytical Sciences
Paul O'Connor, Director
Siaka Yusuf, Reactor Supervisor

A020
NRR

DTRR Revised response to questions 11, 14, 15-1, 16, 17-1, 17-3, 18, 19, 41, 52, 53, 54, 56 in support of the license renewal.

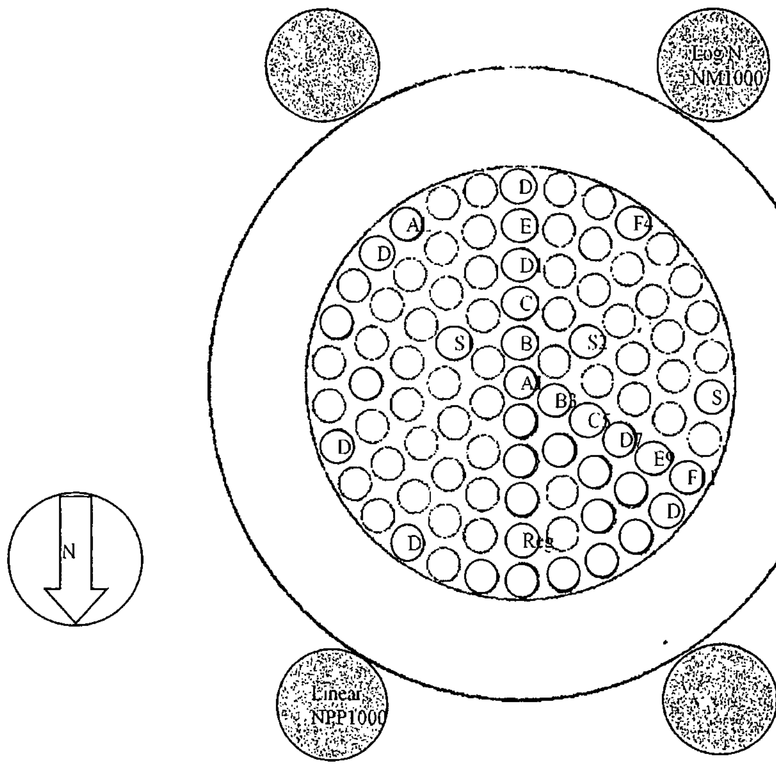
October 2011

11. NUREG–1537, Part 1, Section 4.2.5, "Core Support Structure" requests the applicant to provide design information pertaining to the core support structure. DTRR SAR, Chapter D, does not provide sufficient information. Please provide figures depicting the upper and lower core plates and provide the dimensions and locations of all penetrations that allow coolant to flow through them.

DTRR response:
Two aluminum grid plates fix the position of fuel elements, dummy elements and neutron source. Figure 4 is the schematic drawing of the upper grid plate with the position of the control rods, pneumatic transfer location and source indicated. The upper plate is ¾ inch aluminum and 1.5 inch diameter holes to position the fuel, dummy elements, control rods, etc. The bottom plate is ¾ inch aluminum and has holes to receive the end pins of the fuel and dummy elements. Thirty-six holes for the natural convection cooling are found on the lower grid plate. The water passes through the upper grid plate by means of the gap between the triflute section of the fuel and the upper grid plate. The penetrations on the lower grid plate are in concentric rings with 7, 12, and 17 holes, respectively. Figure 5 is a photograph of the lower grid plate. This photograph substantiates the as built drawings given by General Atomics, and previous Safety Analysis Reports submitted by The Dow Chemical Company.

In addition to these holes, the fuels are raised above the bottom grid plate by the fuel pin (5/8" diameter by 2" long). The gap between the fuel and the lower grid plate therefore allows the bottom coolant to flow, in an "open" configuration through the core from the bottom.

Visual inspections of the core support structure indicate there are no corrosion, no deformation, and no cracking. We believe that the structures can support the core for the next 20 and more years.



- F4-Rabbit Terminus
- S-AmBe Source
- D-Dummy element
- AL- Aluminum Fuel element
- S1- Shim1 Control Rod
- S2- Shim2 Control Rod
- Reg-Regulating Control Rod

Figure 4. Upper grid plate

Figure 5. Lower grid plate.

14. NUREG–1537, Part 1, Section 4.5, "Nuclear Design" requests the applicant to provide a detailed description of analytical methods used in the nuclear design, including computer codes used to characterize technical parameters pertaining to its reactor. DTRR SAR, Chapter D, does not provide sufficient information. Please provide descriptions of the DTRR nuclear design analyses, including the methods and the computer codes used for the analyses.

DTRR Response
The response to this question is awaiting the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

15. NUREG–1537, Part 1, Section 4.5.1, "Normal Operating Conditions" requests the applicant to provide a description of the limiting core configuration (LCC), the core configuration that would yield the highest power density using the fuel specified for the reactor. All other core configurations utilized by the applicant should be encompassed by the safety analysis of this configuration. The description should indicate the number, types, and locations of all core components on the grid plate including fuel, control rods, neutron reflectors, and moderators.

15.1 DTRR SAR, Section D.5.5, provides a list of reactivity worths but control rod worths are not included. Please provide control rod worths specific to the LCC at the requested power level.

15.2 DTRR SAR, Section A.3, describes the original fuel configuration as having 75 stainless steel (SS)-clad elements and one Aluminum (Al)-clad element. The DTRR SAR does not provide information relating to the DTRR fuel element and control rod layout for the requested power level. Please provide a complete description of the LCC for the requested power level and provide a core diagram showing all components.

15.3 The limit on excess reactivity is established in DTRR SAR Table 4. However, the actual excess reactivity of the DTRR LCC is not identified in the DTRR SAR. Please provide the calculated excess reactivity for the LCC at the requested power level.

DTRR response:
15.1 At 300 kW the excess reactivity is limited to \$3.00, and shutdown margin is \$0.50 based on a cold xenon negligible condition (<\$0.30). There are three control rods, namely, Shim1, Shim2 and the Regulating rod. The control rod worth for the Shim1 and Shim2 is approximately \$3.00 each. The Regulating rod is worth, approximately, \$1.00.
The reactivity worth of the control rods are Shim1 \$2.68, Shim2 \$2.73, Regulating rod \$1.01 as measured on January 11th, 2011.

15.2 The core configuration is found in Figure 1. The core is loaded with 79 Stainless Steel clad fuel elements, 1 Aluminum clad fuel element and 5 graphite dummies.

15.3. The excess reactivity of the DTRR as currently configured is \$2.28 as measured at 5 Watts on January 11th, 2011.

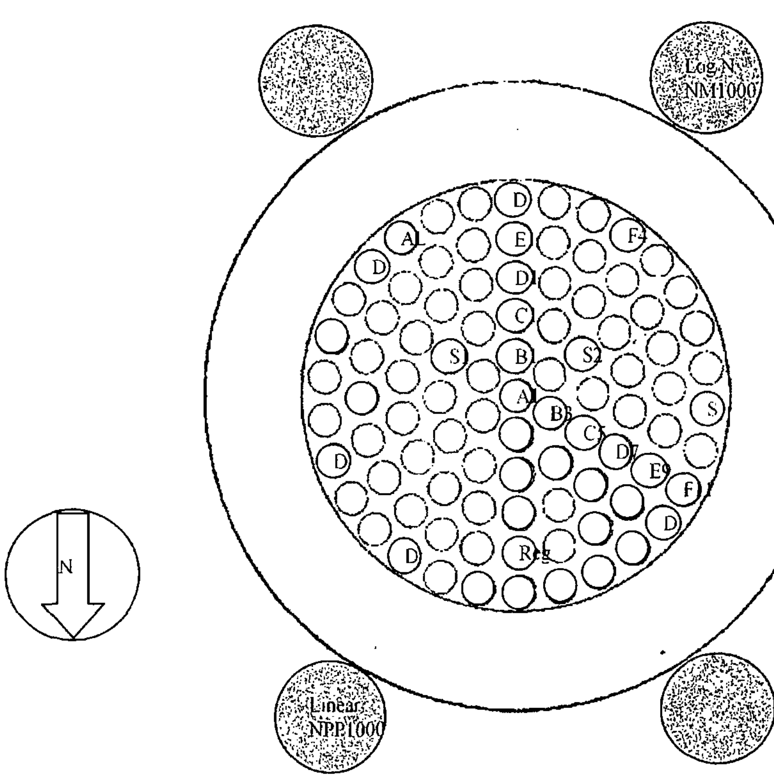


Figure 1. Upper grid plate

16. NUREG–1537, Part 1, Section 4.5.2, "Reactor Core Physics Parameters" requests the applicant to provide a description of the full set of core physics parameters for the LCC that are used in their safety analyses and the methods used to determine them. DTRR SAR, Table 4, provides some of the values cited (i.e., β_{eff}, prompt-neutron-lifetime, fuel temperature and the void coefficient). However, it is unclear if these are generic values or if they are applicable to the LCC of the DTRR and to the safety analyses in Chapter M. Please provide a description of the full set of core physics parameters for the LCC that are used in the DTRR safety analyses and the methods used to determine them.

DTRR Response
The response to this question is awaiting the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

17. NUREG–1537, Part 1, Section 4.5.3, "Operating Limits" requests the applicant to provide information regarding the operating limits applicable to the LCC of its reactor. DTRR SAR, Section D does not provide sufficient information.

17.1 Please describe any limits or conditions on the evaluation of excess reactivity contributors, such as those due to temperature variations and poisons (e.g., xenon and samarium). Please describe algebraically how DTRR determines excess reactivity showing all components.

17.2 Please describe any limits or conditions on the evaluation of shutdown margin, including a discussion of uncertainties.

17.3 Safety Limit (SL) is based on fuel temperature, and the Limiting Safety System Setting (LSSS) is based on core power (DTRR TS 2.1 and DTRR TS 2.2). Please describe the relationship between these parameters and how the DTRR operation using the LCC at the new requested power level will result in fuel temperatures that are bounded by the SL.

DTRR response:
17.1 The response to this question is awaiting the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

17.2 The response to this question is awaiting the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

17.3 The DTRR has since withdrawn the request for a new power level other than 300kW. However, the relationship between the SL, Temperature, LSSS, 300kW reactor power and the TS will be described after the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

18. NUREG–1537, Part 1, Section 4.6, "Thermal-Hydraulic Design" requests the applicant to provide information and analyses of thermal-hydraulic conditions in its reactor demonstrating that sufficient cooling capacity exists for steady-state operations at the maximum licensed power level. DTRR SAR, Chapter D, does not provide sufficient information. Please provide information pertaining to the minimum DNBR for the DTRR using the LCC at the new requested power level. Please describe the analytical methods used to determine the DNBR, including the core inlet and exit conditions assumed and other assumptions and correlations employed.

DTRR Response
The response to this question is awaiting the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

19. NUREG–1537, Part 1, Section 5.2, "Primary Coolant System" requests the applicant to provide a description of the primary coolant system, including information to substantiate the removal of heat from the fuel during maximum licensed power operation and decay heat when the reactor is shutdown. DTRR SAR, Sections E.1 and E.3, do not provide information demonstrating the adequacy of the primary system to perform this task. Please provide information showing the adequacy of the primary system to cool the reactor under all anticipated conditions of operation at the new requested power level.

DTRR Response
The response to this question is awaiting the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

41. NUREG–1537, Part 1, Section 10.2, "Experimental Facilities" requests the applicant to provide a description of the radiological considerations associated with the design and the use of the experimental facilities, generation of radioactive gases, release of fission products or other radioactive contaminants, and exposure of personnel to neutron and gamma beams. DTRR SAR, Section J.2, does not provide this information. Please provide this information for operation at the new requested power level.

DTRR response

The DTRR is The DTRR is currently operated at varying power levels up to the licensed power of 300 kW. At this time the DTRR is withdrawing its request to up rate in power. The DTRR will continue to operate at 300 kW. All routine experiments are reviewed prior to irradiation. Dose rate on removal is estimated using power, irradiation time, decay time and composition of the samples. These estimates are noted on the TRIGA activation request form. Dose rate to the experimenter is controlled using ALARA. Samples are unloaded from the rotary specimen rack using a "fishing pole". Samples may be returned to the rotary specimen rack or a lead cave in the reactor room if the dose rate exceeds expectation. Samples are unloaded from the secondary capsules using long handled tongs in side of the fume hood which is vented HEPA filter. Operation of the pneumatic system is not allowed when individuals are on the roof. Personnel are not typically in the reactor room during operation and therefore not as risk for direct exposure to neutron or gamma beams. Bends are located in both the pneumatic system and rotary specimen rack system to minimize dose rates in the reactor room. The ARM, CAM and Geiger tube continuously provide an indication of the condition in the reactor room. The schematics of the rabbit system and the directional flow of air during operation are shown in Figures 41-1a and 41-1b.

[redacted]

Inserting a Rabbit

Figure 41-1a: The Pneumatic system showing air flow direction during insertion of a rabbit capsule

[redacted]

Withdrawing a Rabbit

Figure 41-1b: The Pneumatic system showing air flow direction during withdrawal of a rabbit capsule

Ar-41 is produced when Ar-40 in the air (P-tube, sample capsules 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/cm^3 x 0.934% x 301.5 cm^3). Per Erdtmann NAA tables, production of 41-Ar at 300 KW is 300 dpm/microgram per irradiation minute.

$$A = \frac{3.64\ mg \times 300 \frac{dpm}{\mu g - min} \times 1000 \frac{\mu g}{mg}}{2.2\ x\ 10^{6} \frac{dpm}{\mu Ci}} = 0.49 \frac{\mu Ci}{min}$$

The rabbit system is typically only operated approximately 15 minutes per day. In a worst case scenario, if the reactor was operated continuously during working hours (8 hours per day) with the rabbit system in operation, the total generation rate of Ar-41 would be approximately 235.2 µCi/day.

The rabbit system discharges into a hood exhaust duct which flows at 1100 CFM (31,000 L/min). Assuming a 95% usage fraction of the hood, the daily exhaust rate of the hood is 4.26 x 10^7 L/day

Using these numbers, the daily average concentration of Ar-41 being exhausted from the 1602 Building is:

$$C = \frac{235.2 \frac{\mu Ci}{day}}{4.26\times 10^7 \frac{L}{day} \times 1000 \frac{mL}{L}} = 5.52 \times 10^{-9}\ \mu Ci/mL. \quad (2)$$

The 10 CFR 20, Appendix B allowable effluent release concentration of Ar-41 through the air pathway is 1 x 10^{-8} μCi/ml. The Ar-41 releases from the reactor are less than the allowable release concentration, which corresponds to a Total Effective Dose Equivalent of approximately 28 mrem/yr, assuming somebody was continuously present at the location of release. This is below regulatory limits for releases of radioactive material in 10 CFR Part 20. Note that actual operation of the rabbit system occurs for only about 15 minutes per day, which keeps routine releases well below the 10 CFR 20 ALARA goal of 10 mrem/yr.

If this material was released directly into the reactor room, it would be dispersed throughout the reactor room and be ventilated out of the area via the room ventilation.
Ar-41 concentrations in the reactor room can be calculated using a steady-state well-mixed box model (AIHA, 2000) by:

$C = G/Q$

Where,

C = room concentration $(\mu Ci/m^3)$
G = generation rate $(\mu Ci/min)$ (0.49 $\mu Ci/min$)
Q = room ventilation rate (m^3/min) (50 m^3/min)

An Ar-41 concentration of 9.8 x 10^{-9} $\mu Ci/mL$ could be generated. If a worker was continuously present in an environment of this concentration, during reactor operations, their Total Effective Dose Equivalent would only be 17 mrem/yr.

52. NUREG–1537, Part 1, Section 13.1.1, "Maximum Hypothetical Accident" requests the applicant to provide a maximum hypothetical accident (MHA) and demonstrate that it bounds all potential credible accidents at the facility. The MHA for TRIGA reactors is typically the failure of one fuel element in the air with the release of gaseous fission products. DTRR SAR M.1.3, analyzes a fuel failure in the pool, but it does not meet the expectation of being a bounding accident analysis. Please provide an analysis of the MHA for the DTRR that bounds all other accident analysis. Please include all assumptions, sequence of events and the potential radiological consequences.

DTRR response

The rupture of a fuel element outside of the pool water is the DTRR "Maximum Hypothetical Accident" because it has the potential to result in the release of fission products and cause maximum exposure to personnel. Fuel elements are rarely, if ever, removed from the pool water, but, in order to bound the consequence analysis, it is assumed that the damaged fuel element is outside the reactor pool at the time of the accident. This type of event has been analyzed by F. C. Foushee and R. H. Peters, "Summary of TRIGA Fuel Fission Product Release Experiments", Gulf Energy and Environmental Systems Report A-10801, 1971. Similar conclusions are reported by S. C. Hawley and R. L. Kathren, "Credible Accident Analyses for TRIGA and TRIGA-fueled Reactors", NUREG/CR-2387, PNL-4028 (1982). The Dow TRIGA Reactor contains fuel with similar characteristics as the Oregon State LEU fuel (Oregon State, 2007). The Oregon State SAR calculates fission product inventory in a fuel element after being run for one year at a power level of 1.1 MW at a peak power density of 18.52 kW per fuel element with zero decay time. Correcting this initial fission product inventory for the Dow TRIGA reactor's lower peak power density of 6.08 kW per fuel element (from the analysis of a limiting core configuration, to be reported), the fission product inventory predicted for the worst-case single fuel element is presented in Tables 52-1, 52-2, 52-3, 52-4, 52-5 and 52-6. The use of the Dow TRIGA research reactor over the past 40 years indicates that the reactor has not been continuously operated for a sufficient period of time to achieve saturation of the fission product inventory and is not likely to be so operated, therefore the inventory is conservatively estimated and the doses estimated are for emergency planning only.

The fraction of gaseous fission products that will be released during a failure of a TRIGA reactor fuel element at a temperature of less than 350° C is 1.5×10^{-5} (Foushee and Peters, 1971; Stahl, 1982). The volume of the reactor room is about 130 cubic meters, with a turnover rate of about 50 cubic meters per minute. However, no credit was taken in the consequence calculations for reductions in air concentrations due to the ventilation of the room. The effective dose equivalent for worker exposure was calculated assuming perfect mixing in the room and instantaneous release to the reactor room.

Restricted area

The effective dose equivalent to an individual in the restricted area enveloped in the radioactive cloud of released halogens and noble gases was calculated as the sum of the inhalation dose of the halogens and the submersion dose of the noble gases. Dose conversion factors for inhalation were taken from 10 CFR 20, Appendix B, column 2 for the calculation of effective dose equivalent to the whole body and for the dose calculations to the thyroid for the iodine compounds. For the thyroid dose from the bromine compounds and the submersion dose, dose conversion factors from FGR No. 11 (Environmental Protection Agency, 1988) were used.

Dose rates from inhalation and submersion doses in the reactor room are calculated to be 1.76 rem /hour. The committed dose equivalent to the thyroid is calculated as 52 rem/hour. Calculations of the dose rate by isotope are summarized in Tables 52-1 and 52-2. Assuming that it takes no more than 30 minutes for the operators to bring the reactor to a safe condition and evacuate the area, if necessary, the total effective dose equivalent to reactor operators from this incident would be 0.88 rem, and the committed dose equivalent to the thyroid would be 26 rem.

Table 52-1. Dose Inside the Reactor Room from Halogens from Fuel Element Rupture Accident Scenario

Halogens	Activity in Fuel Rod (Ci)	Concentration in Reactor Room (Ci/m3)	Inhalation Dose Conversion Factor (Sv/Bq)	Dose Contribution (rem/hr)
Br-82	[redacted]	[redacted]	3.51E-10	2.30E-06
Br-83	[redacted]	[redacted]	2.41E-11	3.26E-04
Br-84	[redacted]	[redacted]	2.61E-11	6.45E-04
Br-84M	[redacted]	[redacted]	2.61E-11	2.41E-05
Br-85	[redacted]	[redacted]	2.61E-11	8.16E-04
Br-86	[redacted]	[redacted]	2.61E-11	1.14E-03
Br-87	[redacted]	[redacted]	2.61E-11	1.28E-03
I-131	[redacted]	[redacted]	8.89E-09	6.91E-01
I-132	[redacted]	[redacted]	1.03E-10	1.24E-02
I-133	[redacted]	[redacted]	1.58E-09	2.87E-01
I-134	[redacted]	[redacted]	3.55E-11	7.34E-03
I-135	[redacted]	[redacted]	3.32E-10	5.64E-02
I-136	[redacted]	[redacted]	8.89E-09	5.67E-01
Total Halogens				[redacted]

Table 52-2. Dose Inside the Reactor Room from Noble Gases from Fuel Element Rupture Accident Scenario

Noble Gases	Activity (Ci)	Concentration in Reactor Room (Ci/m3)	Submersion Dose Conversion Factor (Sv/Bq)	Dose Contribution (rem/hr)
Kr-83M	[redacted]	[redacted]	2.98E-11	3.36E-04
Kr-85	[redacted]	[redacted]	4.70E-13	2.65E-06
Kr-85M	[redacted]	[redacted]	2.98E-11	7.73E-04
Kr-87	[redacted]	[redacted]	1.42E-10	7.28E-03
Kr-88	[redacted]	[redacted]	3.60E-10	2.60E-02
Kr-89	[redacted]	[redacted]	3.60E-10	3.35E-02
Xe-131M	[redacted]	[redacted]	1.48E-12	1.17E-06
Xe-133M	[redacted]	[redacted]	5.38E-12	2.43E-05
Xe-133	[redacted]	[redacted]	6.07E-12	9.19E-04
Xe-135M	[redacted]	[redacted]	7.53E-11	2.17E-03
Xe-135	[redacted]	[redacted]	4.68E-11	7.02E-03
Xe-137	[redacted]	[redacted]	1.92E-10	2.63E-02
Xe-138	[redacted]	[redacted]	1.92E-10	2.67E-02
Kr-83M	[redacted]	[redacted]	2.98E-11	3.36E-04
Kr-85	[redacted]	[redacted]	4.70E-13	2.65E-06
Kr-85M	[redacted]	[redacted]	2.98E-11	7.73E-04
Kr-87	[redacted]	[redacted]	1.42E-10	7.28E-03
Total Noble Gases:				[redacted]

Unrestricted area

The only occupied space that the reactor room is connected to in the building is the reactor control room. If the reactor room ventilation system is operational, the reactor room is at negative pressure as compared to surrounding facilities, and there will be no potential for release of the radionuclides to other areas in the building. If the reactor room ventilation system was not functional, air within the reactor control room could leak around the door to the reactor room into the control room. However, in this circumstance, the building would be evacuated of non-radiation workers to prevent additional personnel exposure. Exposures in other locations within the building would be bounded by dose estimates from air leakage into the control room.

Air leakage through a narrow opening can be predicted by:

$$Q = C_d A[\frac{2\Delta p}{\rho}]^{0.5}$$

Where,

Q = Flow Rate (m^3/sec)
C_d = Discharge coefficient (0.61 for a flat plate orifice)
A = Area of opening (m^2) (0.0095 m^2)
Δp = pressure differential between rooms (Pa)
ρ = density of air (kg/m^3) (1.225 kg/m^3)

Using the characteristics of the reactor room door and a conservative pressure differential between the rooms of 20 Pa, a flow rate of 0.023 m^3/sec is calculated. This air flow rate is assumed to flow into the control room for 15 minutes before the building is evacuated. It is conservatively assumed that no air is released from the control room. Average concentrations of radionuclides generated in the control room from this scenario and calculations of the dose rate for members of the public located inside the building are summarized in Tables 52-3 and 52-4.

The total effective dose equivalent rate from inhalation and submersion in the control room are calculated to be 186 mrem /hour. The committed dose equivalent to the thyroid is calculated as 5.21 rem/hour. Assuming that it takes no more than 15 minutes for the building to be evacuated, the total effective dose equivalent to members of the public in the reactor building from this incident would be less than 47 mrem, and the committed dose equivalent to the thyroid would be less than 1.3 rem.

Exposures to operators in the control room would be bounded by the exposure estimates inside the reactor room, above.

Table 52-3. Dose Inside the Control Room from Halogens from Fuel Element Rupture Accident Scenario

Halogens	Release Rate into Control Room (Ci/sec)	15-Minute Average Concentration in Control Room (Ci/m3)	Inhalation Dose Conversion Factor (Sv/Bq)	Dose Contribution (rem/hr)
Br-82	[redacted]	[redacted]	3.51E-10	2.47E-08
Br-83	[redacted]	[redacted]	2.41E-11	5.09E-05
Br-84	[redacted]	[redacted]	2.61E-11	1.01E-04
Br-84M	[redacted]	[redacted]	2.61E-11	3.77E-06
Br-85	[redacted]	[redacted]	2.61E-11	1.28E-04
Br-86	[redacted]	[redacted]	2.61E-11	1.78E-04
Br-87	[redacted]	[redacted]	2.61E-11	2.00E-04
I-131	[redacted]	[redacted]	8.89E-09	1.08E-01
I-132	[redacted]	[redacted]	1.03E-10	1.93E-03
I-133	[redacted]	[redacted]	1.58E-09	4.48E-02
I-134	[redacted]	[redacted]	3.55E-11	1.15E-03
I-135	[redacted]	[redacted]	3.32E-10	8.81E-03
I-136	[redacted]	[redacted]	8.89E-09	3.31E-03
Total Halogens				**0.165**

Table 52-4. Dose Inside the Control Room from Noble Gases from Fuel Element Rupture Accident Scenario

Noble Gases	Release Rate into Control Room (Ci/sec)	15-Minute Average Concentration in Control Room (Ci/m3)	Submersion Dose Conversion Factor (Sv/Bq)	Dose Contribution (rem/hr)
Kr-83M	[redacted]	[redacted]	2.98E-11	5.25E-05
Kr-85	[redacted]	[redacted]	4.70E-13	4.15E-07
Kr-85M	[redacted]	[redacted]	2.98E-11	1.21E-04
Kr-87	[redacted]	[redacted]	1.42E-10	1.14E-03
Kr-88	[redacted]	[redacted]	3.60E-10	4.07E-03
Kr-89	[redacted]	[redacted]	3.60E-10	5.23E-03
Xe-131M	[redacted]	[redacted]	1.48E-12	1.82E-07
Xe-133M	[redacted]	[redacted]	5.38E-12	3.79E-06
Xe-133	[redacted]	[redacted]	6.07E-12	1.44E-04
Xe-135M	[redacted]	[redacted]	7.53E-11	3.40E-04
Xe-135	[redacted]	[redacted]	4.68E-11	1.10E-03
Xe-137	[redacted]	[redacted]	1.92E-10	4.11E-03
Xe-138	[redacted]	[redacted]	1.92E-10	4.17E-03
Kr-83M	[redacted]	[redacted]	2.98E-11	5.25E-05
Kr-85	[redacted]	[redacted]	4.70E-13	4.15E-07
Kr-85M	[redacted]	[redacted]	2.98E-11	1.21E-04
Kr-87	[redacted]	[redacted]	1.42E-10	1.14E-03
Total Noble Gases:				**0.0205**

Unrestricted area - offsite

Doses to unrestricted areas are calculated using a normal ventilation rate for the reactor room of 50 m^3/min. Based on this ventilation rate and the reactor room volume of 130 m^3, it will take 2.6 minutes for the released radionuclides to be released to the environment when the ventilation system is running. Using the Pasquill categories and a Gaussian approach to dispersion, the maximum concentration of fission products to a member of the public is at the fence-line, 23 m from the reactor building. The short range is due to the low height of the ventilation exit, 2.73 meters (8 ft) above the building. The wind is assumed to be blowing in the direction of the fence at the time of the accident, and conservative atmospheric stability conditions (category F) are assumed. Transport calculations are performed following the guidance in Regulatory Guide 1.145 (U.S. Nuclear Regulatory Commission, 1982). To be conservative, the downwind conditions are calculated assuming a fumigation condition and equation (2) in Section 1.3.1 of the Regulatory Guide is used, as it is higher than the result from equation (1) for this scenario. Radioactive decay is conservatively not considered during transport calculations. The total effective dose equivalent to a member of the public at the unrestricted location is 20 mrem and the committed dose equivalent to the thyroid at this location is 382 mrem. Calculations of the dose rate by isotope for offsite exposures are summarized in Tables 52-5 and 52-6.

Table 52-5. Downwind Dose from Halogens from Fuel Element Rupture Accident Scenario

Halogens	Release Rate (Ci/sec)	Inh DCF (Sv/Bq)	Downwind Concentration (Ci/m3)	Dose Rate (rem/hr)
Br-82	[redacted]	3.51E-10	3.92E-10	6.11E-07
Br-83	[redacted]	2.41E-11	8.08E-07	8.65E-05
Br-84	[redacted]	2.61E-11	1.48E-06	1.71E-04
Br-84M	[redacted]	2.61E-11	5.53E-08	6.41E-06
Br-85	[redacted]	2.61E-11	1.87E-06	2.17E-04
Br-86	[redacted]	2.61E-11	2.60E-06	3.02E-04
Br-87	[redacted]	2.61E-11	2.54E-06	2.94E-04
I-131	[redacted]	8.89E-09	4.64E-06	1.83E-01
I-132	[redacted]	1.03E-10	7.19E-06	3.29E-03
I-133	[redacted]	1.58E-09	1.09E-05	7.62E-02
I-134	[redacted]	3.55E-11	1.24E-05	1.95E-03
I-135	[redacted]	3.32E-10	1.02E-05	1.50E-02
I-136	[redacted]	8.89E-09	3.46E-06	1.37E-01
Total:				**0.417**

Table 52-6. Downwind Dose from Noble Gases from Fuel Element Rupture Accident Scenario

Noble Gases	Release Rate (Ci/sec)	Submersion Dose Conversion Factor (Sv/Bq)	Downwind Concentration (Ci/m3)	Dose Rate (rem/hr)
Kr-83M	[redacted]	3.36E-04	8.08E-07	8.91E-05
Kr-85	[redacted]	2.65E-06	4.05E-07	7.05E-07
Kr-85M	[redacted]	7.73E-04	1.86E-06	2.05E-04
Kr-87	[redacted]	7.28E-03	3.68E-06	1.93E-03
Kr-88	[redacted]	2.60E-02	5.19E-06	6.91E-03
Kr-89	[redacted]	3.35E-02	6.39E-06	8.51E-03
Xe-131M	[redacted]	1.17E-06	5.65E-08	3.09E-07
Xe-133M	[redacted]	2.43E-05	3.24E-07	6.44E-06
Xe-133	[redacted]	9.19E-04	1.09E-05	2.44E-04
Xe-135M	[redacted]	2.17E-03	2.07E-06	5.77E-04
Xe-135	[redacted]	7.02E-03	1.08E-05	1.86E-03
Xe-137	[redacted]	2.63E-02	9.48E-06	6.74E-03
Xe-138	[redacted]	2.67E-02	9.97E-06	7.08E-03
Kr-83M	[redacted]	3.36E-04	8.08E-07	8.91E-05
Kr-85	[redacted]	2.65E-06	4.05E-07	7.05E-07
Totals:				**3.42E-02**

References

F.C. Foushee and R. H. Peters, Summary of TRIGA Fuel Fission Product Release Experiments, Vol. 11, General Atomic Company Report Gulf EES-A108011; and S. Langer and N. L. Baldwin, Fission Product Release Experiments on Uranium-Zirconium Hydride Fuels, Vol. I, General Atomic Company Report Gulf GA-A10781 (1971).

D. Stahol. Fuels for Research and Test Reactors, Status Review: July 1982. Argonne National Laboratory report ANL-83-5 (1982).

53. NUREG–1537, Part 1, Section 13.1.2, "Insertion of Excess Reactivity" requests the applicant to provide an analysis of reactivity insertion events. Similarly, NUREG–1537, Part 1, Section 4.5.3,"Operating Limits," requests that the applicant provide an analysis of the uncontrolled withdrawal of the highest reactivity control rod. DTRR SAR, Section M.1.2, does not provide sufficient information regarding reactivity insertion events.

53.1 Please provide an analysis of possible reactivity insertion events for the DTRR.

53.2 Please provide an analysis of the uncontrolled rod withdrawal event for DTRR using the highest reactivity control rod.

DTRR response:
A revision to the response to this question is awaiting the results of the Neutronics and Thermal-hydraulic models using MNCP and RELAP codes, which are being carried out specifically for the DTRR. A time extension of 30 days is hereby requested for this RAI.

54. NUREG–1537, Part 1, Section 13.1.3, "Loss of Coolant" requests the applicant to provide analysis that assures that doses to the public that could result from a loss of coolant accident do not exceed 10 CFR Part 20 limits. DTRR SAR, Section M.1.1, Table 7 presents exposures resulting from a loss of coolant accident. There is no statement regarding occupational or public dose limits and whether they are met. Please explain this accident analysis in further detail and in terms of meeting the regulatory limits.

DTRR response:

The water level in the surrounding area is above the core height and therefore tank breach will not result in a total loss of coolant. However, a site-specific analysis was completed for an uncovered core after several hours of operation at 300 kW and reported in the SAR for DTRR, U.S. Nuclear Regulatory Commission, 1989). The results are in the following table:

Time after complete loss of coolant	Direct radiation – 18 ft directly above core (R/hr)	Indirect Radiation shield top edge of the tank (R/hr) (Position 1 or 4 in Figure 54-1)
10 seconds	3000	0.78
1 day	360	0.090
1 week	130	0.042
1 month	35	0.012

Exposures inside the Reactor Room

The elevated radiation fields generated from this hypotheticalaccident will be highly collimated above the reactor pool. The core sits inside a 17" diameter opening in the reflector. The top of the reflector is 16' below the top of the 76" diameter reactor pool. The reactor room roof is 12' above the top of the reactor pool. Based on the scattering angle that direct radiation may be emitted from the reactor, the direct radiation beam from the core will only have a diameter of 12.3 feet at the roof of the reactor room, which is still much smaller than the entire room. Therefore, workers and members of the public located outside of the reactor room will not be exposed to the direct radiation from the reactor core. Responders to the incident will also avoid the area of the room directly above the core in order to avoid exposure to the direct radiation from the reactor core.

<u>Exposures outside the Reactor Building</u>

To estimate potential radiation exposure levels from scattered radiation outside of the reactor room (indicated as position 2 in Figure 54-1, which is outside of the reactor building) measured radiation scatter data from dose rates during the operation of the neutron beam tube will be used to estimate potential dose rates generated around the reactor building. Specifically, during an operation of the central beam tube in 1991, measurement survey were made and reported to the Radiation Safety Committee, The Dow Chemical Company. This report documented measured gamma and neutron doses during the operation of a neutron beam tube that consisted of a 1.5" streaming pathway that allowed collimated neutrons to travel through a helium filled aluminum pipe without being shielded. Figure 54-1 shows the layout of the reactor pool and the areas surrounding the reactor room.

The highest total gamma and neutron dose rate measured during this survey was 4.2 mrem/hr and was located on the east side of the building (marked as location #2 in the drawing). During this survey, a measurement of 5.9 mrem/hr was made also on the east side of the reactor but outside of the intense direct radiation field (marked as location #1 in the drawing). This is proportional to the 1991 predicted radiation field of 780 mrem/hr (a factor of 132), immediately after an incident. Therefore, using this factor, the 4.2 mrem/hr measured outside of the reactor building will be equivalent to 554 mrem/hr. This number represents the highest expected dose rate, outside the reactor room, from a totally exposed core. Note that this takes no credit for the decay of the fission products during the time that it would take to drain the reactor pool, which would lower the calculated dose rates significantly.

Reactor room radiation alarms are monitored by Dow Security. In the event of an alarm, Dow Security would immediately respond and clear the area around the reactor of personnel. This response would occur within 30 minutes. Therefore, the highest potential dose to a member of the public from this incident would be a Total Effective Dose Equivalent of 227 mrem, assuming an individual was located immediately outside the emergency door on the east side of the reactor for the entire duration of the incident until they were cleared from the area by security.

D (mrem) = 554 mrem/hr * 0.5 hr =227 mrem

This is the maximum conservative estimate of the public dose during a hypothetical loss of coolant accident.

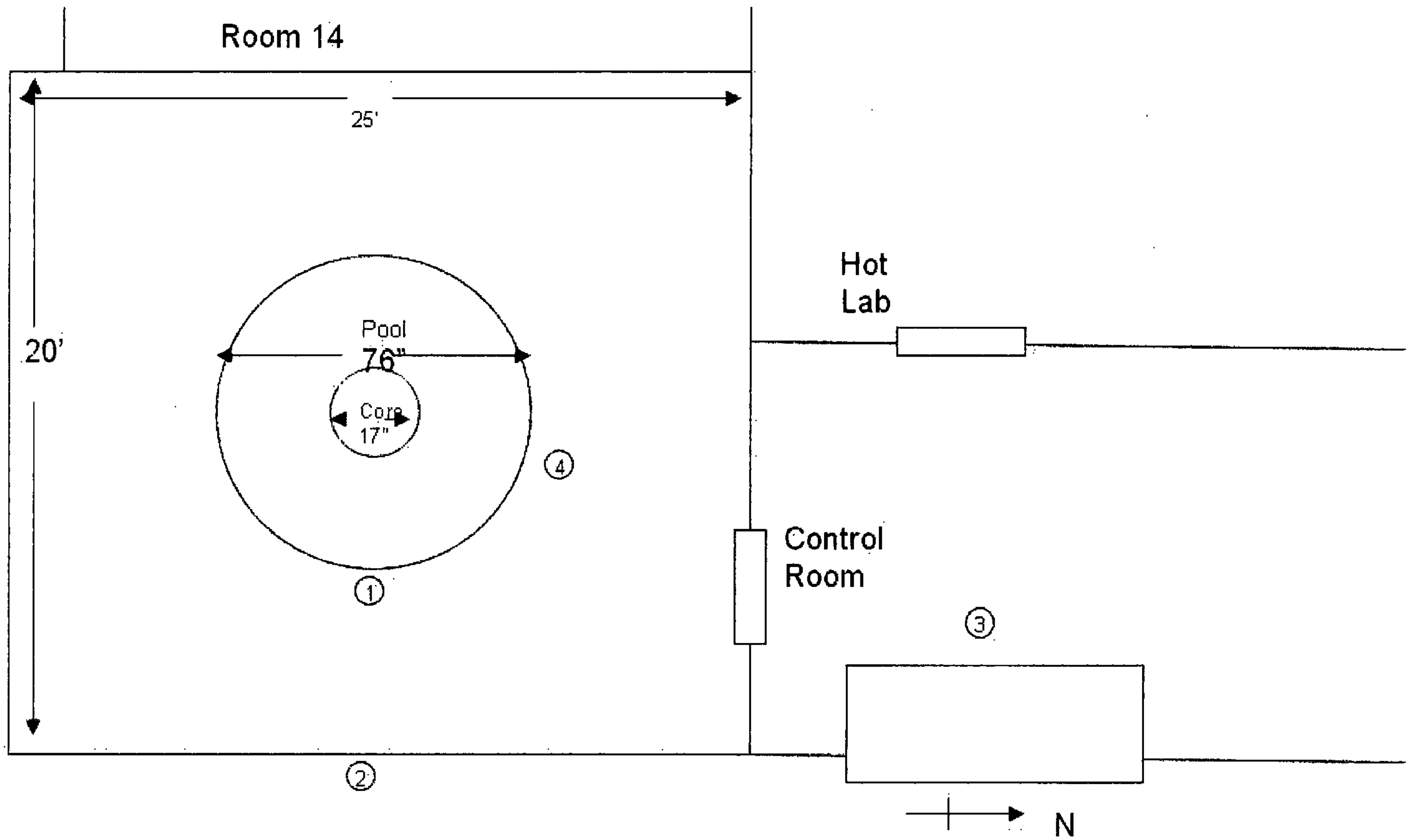
Exposures in the Control Room
To estimate potential radiation exposure levels from scattered radiation inside the reactor control room (position 3 in Figure 54-1), a similar method will be used to predict scattered radiation dose rates. The dose rate measured during the beam tube experiment in the control room reactor console was 0.55 mrem/hr and the dose rate on the north side of the reactor as 8.5 mrem/hr (marked as location #4 in Figure 54-1). Scaling the dose rates using the factor 91 (i.e. 780/8.5), as before, a control room dose rate of 50 mrem/hr from scattered radiation is the calculated expected dose rate immediately following such an incident.

The Dow Chemical Company operates an on-site fire department, which would respond to this incident and be able to refill the reactor pool within 4 hours of the incident occurring. Assuming that this response requires an individual to be located in the reactor room for 30 minutes and in the control room for the remaining 3.5 hours, a total employee Total Effective Dose Equivalent of less than 565 mrem would be received by an employee due to this incident.

D (mrem) = (780 mrem/hr * 0.5 hr) + (50 mrem/hr * 3.5 hr) = 565 mrem

This is the maximum conservative estimate of the occupational dose during a hypothetical loss of coolant accident.

It is important to note that location 3 shown in figure 54-1 and whose dose rate is addressed above, is the only occupied location just outside of the reactor room.



Note: Drawing not to scale

Figure 54-1. Diagram of Reactor Room and Adjacent Areas

56. NUREG–1537, Part 1, Section 13.1.6, "Experiment Malfunction" requests the applicant to provide analysis of an experiment malfunction event. DTRR SAR, Section M.1.4, does not include analysis of an experiment failure with release of radioactivity. Please provide an analysis and consequences of an experiment malfunction for the experiment with the highest potential release of radioactivity.

DTRR response:

All experiments are reviewed before insertion and all experiments are separated from the fuel cladding by at least one barrier for example, the pneumatic tube and central thimble. All experiments that could damage components of the reactor are required by technical specification to be double encapsulated. Samples are typically under 8 grams, with a majority of the samples irradiated consisting of carbon, hydrogen and oxygen (plastic and organics). The dose consequences of the release of 10 microCi of I-131-I-135 are calculated for workers assuming that 100% of the material is released into the reactor room, and the ventilation system is shut off, causing the material to be trapped within the rector room and the worker spends 60 minutes within the reactor room to resolve the incident. The release of the iodine would generate a concentration of 7.69×10^{-8} Ci/m^3. Worker doses are calculated using Dose Conversion Factors for effective dose and dose to the thyroid for I-131 (conservative for iodine radionuclides) from Federal Guidance Report #11 (Eckerman, et al. 1988). This calculation will bound the dose to any member of the public who is located within the building or fence-line and any exposure estimates to workers located within laboratories adjacent to the reactor room.

The total effective dose to the worker is calculated to be 3.04 mrem, and the thyroid dose is calculated to be 99.7 mrem.

Note that from the exposure scenarios for a ruptured fuel element, in response to RAI 52, shows that I-131 contributed 42% of the total dose for the scenario. Since this accident scenario would have a similar mix of radionuclides, it is not anticipated that contributions from additional radionuclides would increase this dose estimate by more than a factor of 3, which would keep exposure estimates from this scenario well below the dose estimates for the fuel rupture accident scenario and 10 CFR Part 20 exposure limits.

For exposures to members of the public, it is assumed that the ventilation system is operational and vents the released iodine outside the reactor building. Based on the ventilation rate and volume of the reactor room, it would take 2.6 minutes to have one full air change of the reactor room and release all of the iodine. Downwind air concentrations at the plant fence-line located 23 m to the west of the reactor building are determined following guidance in Regulatory Guide 1.145 (U.S. Nuclear Regulatory Commission, 1982) to be 3.40 x 10^{-10} Ci/m^3. Doses to members of the public are calculated using Dose Conversion Factors for effective dose and dose to the thyroid for I-131 (conservative for iodine radionuclides) from Federal Guidance Report #11 (Eckerman, et al. 1988).

The total effective dose equivalent to the maximally exposed offsite member of the public is calculated to be 0.01 mrem and the thyroid dose is calculated to be 0.44 mrem.

References

U.S. Environmental Protection Agency. 1988. Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion. Federal Guidance Report No. 11. Washington, D.C.: U.S. Environmental Protection Agency.

The Dow Chemical Company. 1991. Radiation Dose and Exposure Rate Evaluations During Neutron Radiographic Operation of the Dow TRIGA* Research Reactor at 100 and 240 Kilowatts, Special Analysis, Michigan Division Analytical Laboratory, 1602 Building, November 19, 1991. HEH RAD14(8).

U.S. Nuclear Regulatory Commission. 1983. Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. Regulatory Guide 1.145. Washington, DC: U. S. Nuclear Regulatory Commission.

U.S. Nuclear Regulatory Commission. 1989. Safety Evaluation Report related to the renewal of the facility license for the research reactor at the Dow Chemical Company. NUREG-1312. Washington, DC: U.S. Nuclear Regulatory Commission.

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