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 is the DTRR Revised responses to RAI questions 6, 11, 15-1, 17-1, 17-3, 34, 35, 36-2, 41,52, 54, 56, 57, 61-3, 62-2, 64-1, 65-1, 65-2 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 August 31, 2011

Paul O'Connor

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

Subscribed and sworn to before me this 31st day of August, 2011

Kimberly A. Hartman
Notary Public
Gladwin County, Michigan
My Commission Expires:
December 15, 2016

KIMBERLY ANN HARTMAN
NOTARY PUBLIC - STATE OF MICHIGAN
COUNTY OF GLADWIN
My Commission Expires December 15, 2016
Acting in the County of Midland

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

Gladwin

A020

DTRR Revised response to questions 6, 11, 15-1, 17-1, 17-3, 34, 35, 36-2, 41, 52, 54, 56, 57, 61-3, 62-2, 64-1, 65-1, 65-2

August 2011

6. NUREG–1537, Part 1, Section 2.3, "Meteorology" requests the applicant to indicate how the local (site) meteorology supports the dispersion calculations of airborne releases under normal and accident conditions. Please provide a description of the dispersion model based on this meteorological data.

DTRR response:
The public dose calculations are completed using wind speed, number of hours and wind direction determined from the meteorological data and using Regulatory Guide 1.111 (Nuclear Regulatory Commission, 1977) and CAP88-PC version 2.1 code (U.S. Environmental Protection Agency, 2000).

Under normal conditions, the concentration of a radionuclide at a given distance from a point source is calculated by

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

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)

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]

Σ_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]

Under accident conditions, 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.

$$\frac{\chi}{Q} = \frac{1}{\overline{U}_{10}(3\pi\sigma_Y\sigma_z)}$$

χ	=	Concentration of radionuclide in the air, (Ci/m^3)
Q	=	Release rate of radionuclide (Ci/sec)
U_{10}	=	Average wind speed at 10 m (m/sec) [conservatively use 1 m/sec]
σ_z	=	Vertical diffusion coefficient (m) [From Regulatory Guide 1.111 (U.S. Nuclear Regulatory Commission, 1977), dependent on stability class]
σ_y	=	Lateral plume spread, [From Regulatory Guide 1.111 (U.S. Nuclear Regulatory Commission, 1977), dependent on stability class]

Reference

U.S. Nuclear Regulatory Commission. 1982. Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. Regulatory Guide 1.145. Washington, DC: U.S. Nuclear Regulatory Commission.

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.

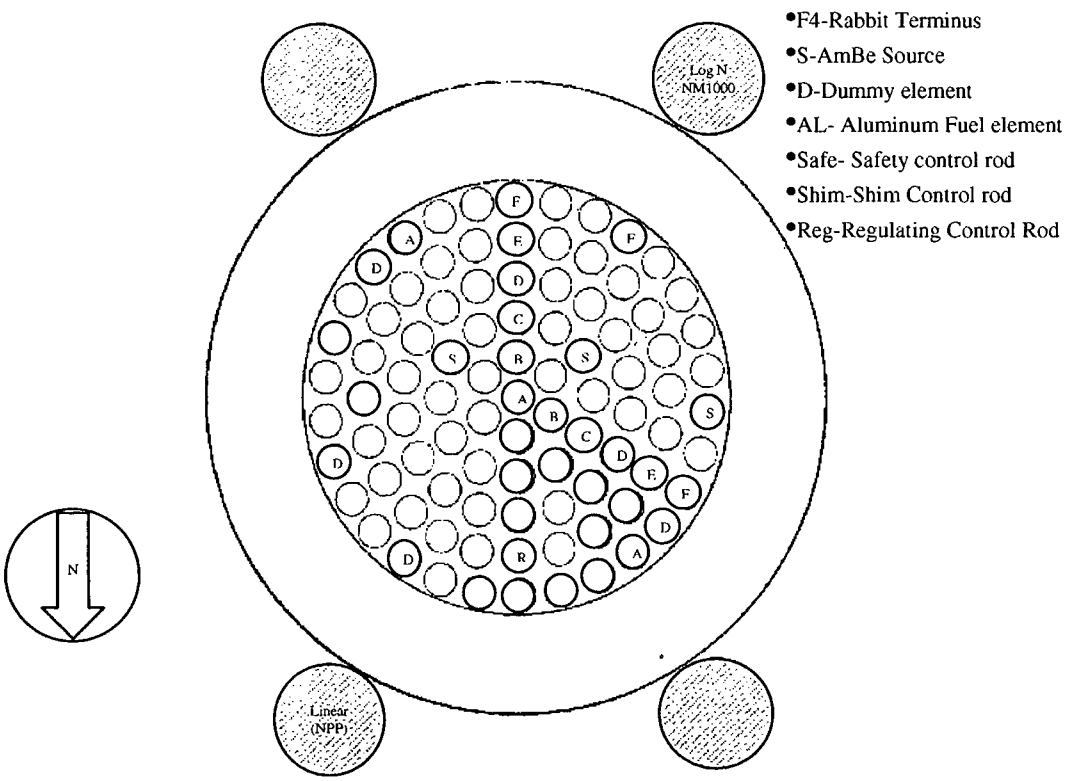

Figure 4. Upper grid

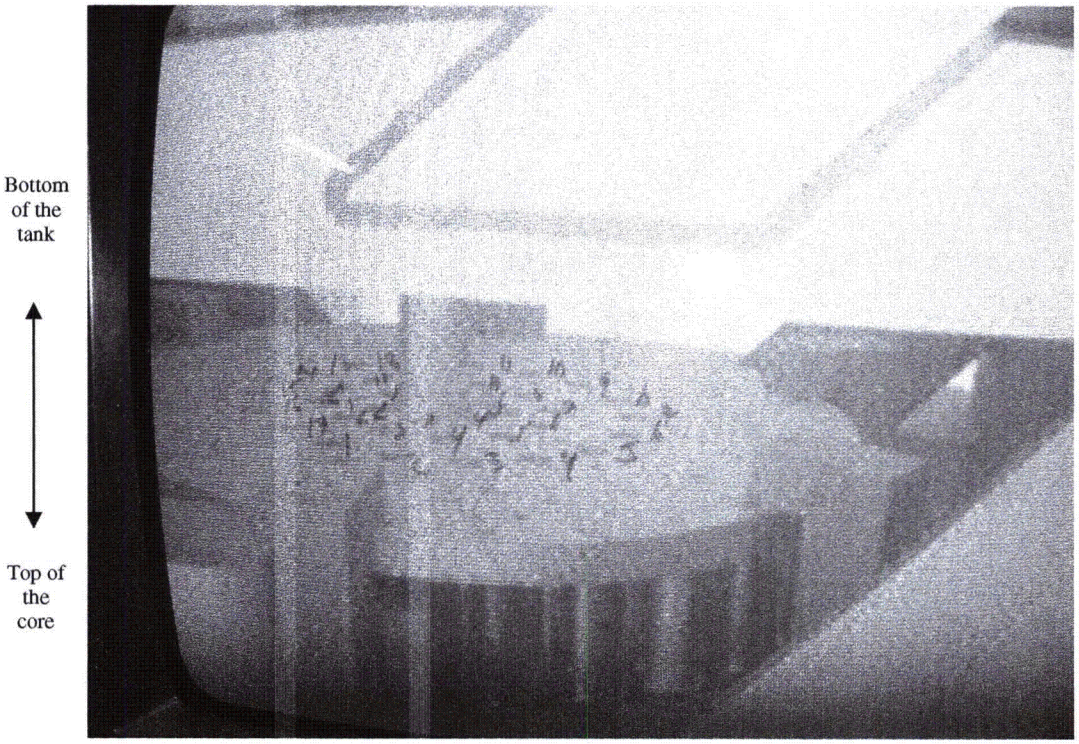

Figure 5. Lower grid plate.

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.

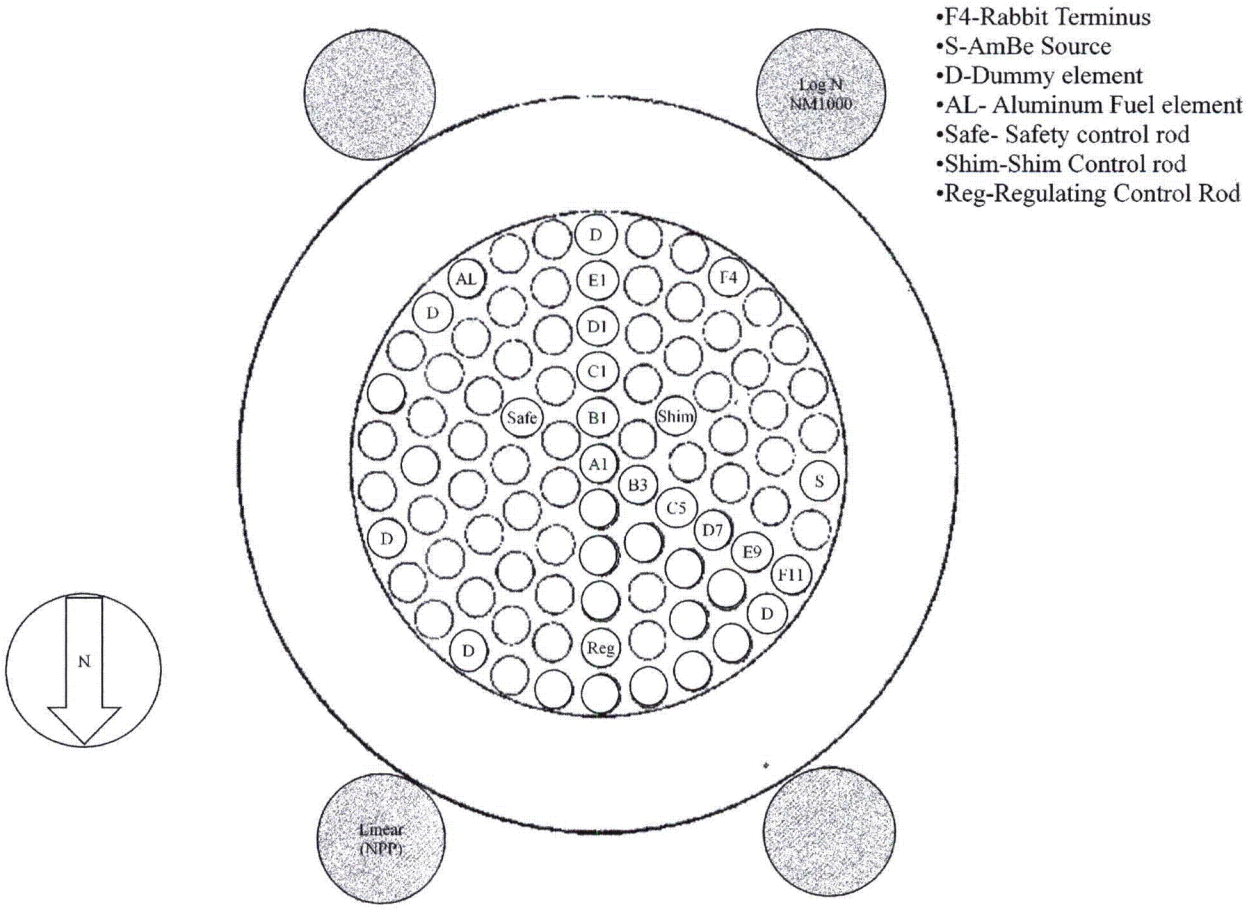

Figure 1. Upper grid plate

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 60 days is hereby requested for this RAI.

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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 60 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 60 days is hereby requested for this RAI.

26. NUREG–1537, Part 1, Section 5.6, "Nitrogen-16 [N^{16}] Control Systems" requests the applicant to provide a description of the nitrogen control system employed and how personnel exposure to N^{16} is consistent with the facility's as low as reasonably achievable (ALARA) Program. DTRR SAR, Section E.6, does not provide a description of the system or information for estimating the dose.

DTRR response:
Personnel dosimetry records show that radiation exposure has historically been very low at this facility. The reactor room is typically unoccupied during reactor operation. This practice will not be altered and is consistent with the as low as reasonably achievable. At 300 kW, the dose rate at the top of the pool is ~ 36 mrem/hr. With the cooling system in operation the transport time increases due to the interruption of the vertical convective column. Dose rate measured at the top of the pool is reduced to 5 mrem/hr.

34. NUREG–1537, Part 1, Section 9.1, "Heating, Ventilation and Air Conditioning System" requests the applicant to provide a description of the heating, ventilation and air conditioning system (HVAC) and any manual or automatic functions. There is a brief description in DTRR SAR I.1 regarding this function with no schematics or illustrations. Please provide a more detailed description of the HVAC system and system functions and provide schematics of the HVAC system.

DTRR response:
The HVAC for the reactor room of the DTRR is shown in Figure 34. The inlet to the reactor room is located to the north side of the reactor room. The outlet is located on the east wall of the reactor room. Both the inlet and the outlet fans can be shut off using a switch in the control room. There are two modes of operation of the HVAC system, 1) Normal operating mode and 2) Isolation mode. In the normal operating mode, both fans are operating and air is being sucked out of the reactor room and "conditioned-air" is being pumped in at the specified turnover rate. There are indicators, available to the operator, of the system status. These indicators are checked daily or whenever the reactor is operated. In the isolation mode, both fans are turned off and the doors are closed. There is a manually operated shutter outside the building which covers the exhaust. This shutter, which is normally opened, will be closed during isolation mode. There are no filters in the exhaust.

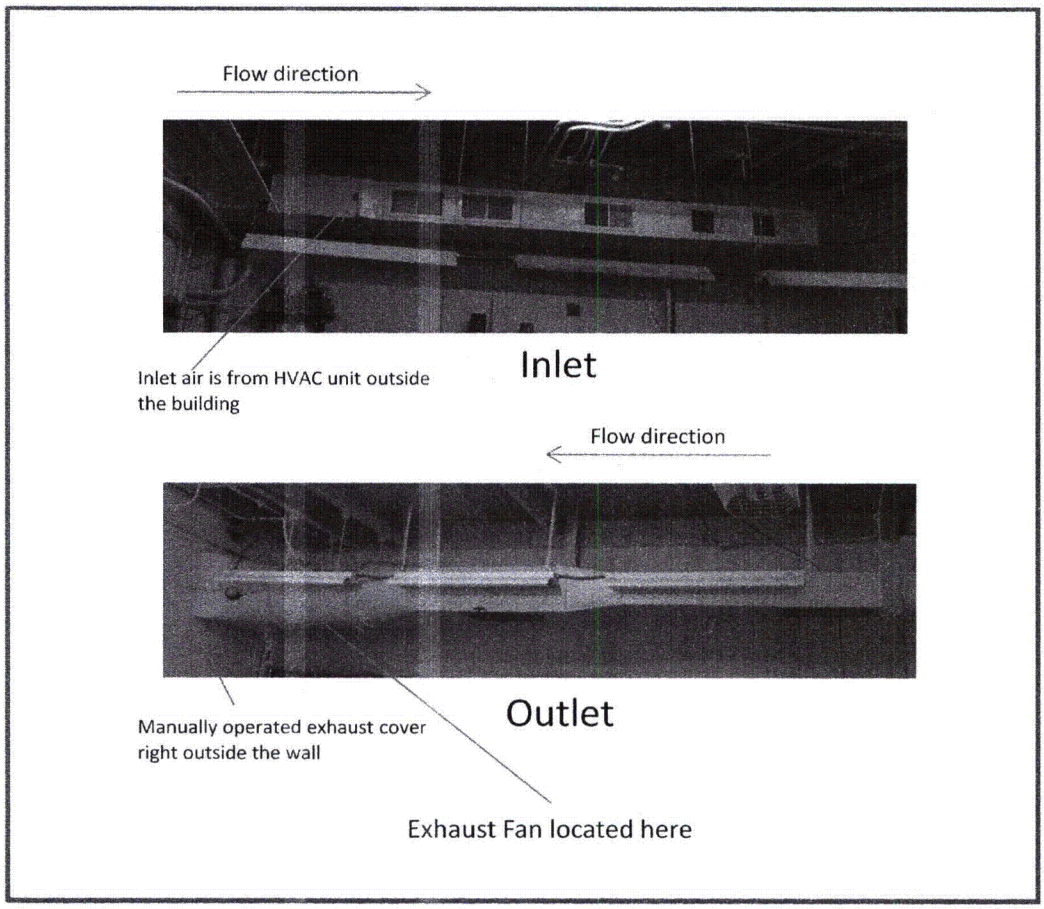

Figure 34: Reactor room HVAC system showing Inlet and outlet units

35. NUREG–1537, Part 1, Section 9.2, "Handling and Storage of Reactor Fuel" requests the applicant to provide an analysis or a reference to an analysis that shows that subcriticality is assured under all conditions of fuel handling and storage. The DTRR SAR does not provide this information. Please provide an analysis that shows that K_{eff} is maintained below 0.90 for all storage configurations and that dose limits are met.

DTRR response:
The DTRR does not currently store fuel outside of the core. Storage of fuel is limited to in-pool only, and this is typically a temporary storage. For example, during the annual fuel inspection or similar special experiments or control rod inspections, the core is made subcritical by removing two fuel elements from the B or C rings and placing them in the in-pool storage. Fuel inspection is carried out in the pool and taking one fuel at a time and using standard fuel handling and inspection tools according to DTRR procedure 4.3. By handling fuel in the pool, dose limits are met by taking advantage of the shielding provided using 15 ft of water.

36. NUREG–1537, Part 1, Section 9.2 requests the applicant to provide a discussion of the handling and storage of new, spent, and failed fuel elements.

36.1 DTRR SAR, Section I.2, does not discuss the tools used to insert or remove fuel from the core, as well as the physical and administrative methods specified to control their use. Please provide this information.

36.2 DTRR SAR, Section I.2, does not discuss whether Technical Specifications are required for or are applicable to the handling and storage of spent or damaged fuel. Please provide this information.

DTRR response:
36.1. Under the Technical specifications, all movement of fissile material is performed as a special experiment and reviewed by the Reactor Operations committee. DTRR procedure 4.3 describes and defines fuel movement, including requirements for sub criticality and number of operators present. Fuel is moved with the use of a General Atomics fuel handling tool. The fuel handling tool is secured when not in use.

36.2. Damaged fuel will be removed from the core and stored in-pool. The DTRR uses a General Atomics fuel handling tool if possible. Other tools can be fabricated if necessary for the removal of individual damaged fuel if the standard tool is not applicable. A definition for damaged fuel has been added to the Technical Specifications, TS definition 1.9. Each situation will be evaluated appropriately and arrangement to package the fuel and transport it to the appropriate agency will be made.

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.

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 300 KW is 300 dpm/microgram per irradiation minute.

$$A = \frac{3.64\ mg \times 300\ \frac{dpm}{ug-min}}{37000\ \frac{dps}{\mu Ci}} = 0.49\ \frac{mCi}{min} \qquad (1)$$

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 uCi/day.

The rabbit system discharges into a hood exhaust duct which flows at 1100 CFM (34,000 L/min). Assuming a 95% usage fraction of the hood, the daily exhaust rate of the hood is 4.65 x 10^7 L/day

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

$$C = \frac{235.2\ \frac{mCi}{day}}{4.65 \times \frac{10^7 L}{day} \times 1000 \frac{mL}{L}} = 5.06 \times 10^{-9}\ \text{uCi/mL.} \quad (2)$$

The 10 CFR 20, Appendix B allowable effluent release concentration of Ar-41 through the air pathway is 1 x 10^{-8} uCi/ml. The Ar-41 releases from the reactor are less the allowable release concentration, which corresponds to a dose of approximately 25 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 (uCi/m3)
G = generation rate (uCi/min) (0.49 uCi/min)
Q = room ventilation rate (m3/min) (50 m3/min)

An Ar-41 concentration of 9.8 x 10^{-9} uCi/mL could be generated. If a worker was continuously present in an environment of this concentration, during reactor operations, their exposure 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
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 60 days is hereby requested for this RAI.

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

To estimate potential radiation exposure levels from scattered radiation outside of the reactor room, measured radiation scatter data from dose rates outside of the reactor room during the operation of the neutron radiographic beam tube will be used to estimate potential dose rates generated around the reactor room during this incident (The Dow Chemical Company, 1991). This report documents measured gamma and neutron doses during the operation of a neutron radiographic beam tube that consisted of a 1.5" streaming pathway that allowed neutrons to travel through the reactor pool without being shielded. Figure 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 of 4.2 mrem/hr was located on the east side of the building, outside of the east wall emergency door (marked as location #2 in the drawing). During this survey, a measurement of 5.9 mrem/hr was made on the east side of the reactor outside of the intense direct radiation field (marked as location #1 in the drawing), compared to a predicted radiation field of 780 mrem/hr at the top edge of the tank immediately after the incident occurs, or 132 times higher. Therefore, the highest predicted radiation dose outside the reactor room would be expected to be no higher than 554 mrem/hr. 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 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 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

The survey report of the dose rates around the reactor room during beam tube operation also reported the dose in the control room reactor console as 0.55 mrem/hr (marked as location #3) and the dose rate on the north side of the reactor as 8.5 mrem/hr (marked as location #4). Scaling the dose rates as before, a control room dose rate of 50 mrem/hr from scattered radiation is calculated immediately following the 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 dose 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

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 60 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, the analysis was completed for an uncovered core after an extended period of operation at 300 kW. 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)
10 seconds	3000	0.78
1 day	360	0.090
1 week	130	0.042
1 month	35	0.012

The elevated radiation fields generated from this accident 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. Based on the largest potential 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.

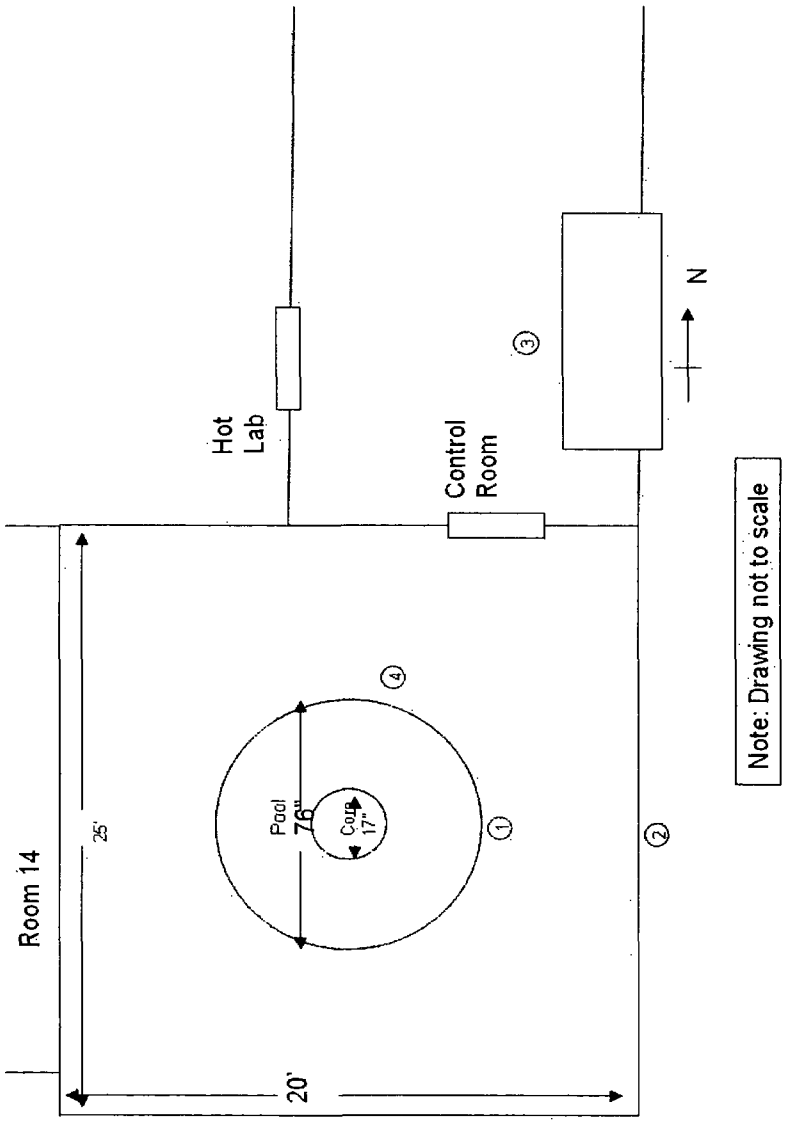

Figure 54. Diagram of Reactor Room and Adjacent Areas

55. NUREG–1537, Part 1, Section 13.1.5, "Mishandling or Malfunction of Fuel" requests the applicant to provide analyses regarding the mishandling or malfunction of fuel. DTRR SAR, Section M.1.3, provides an analysis that assumes that a damaged fuel element is submerged in the reactor pool at the time of the event and only halogens and noble gases are released. The DTRR SAR does not discuss how the accident terminates and does not provide exposures to the staff or the public. Please provide additional information regarding this analysis indicating how the accident terminates and the dose consequences of this accident analysis.

DTRR response:
The rupture of a fuel element would result in the release of fission products into the water. Fuel elements are rarely, if ever, removed from the pool water, so it is assumed that the damaged fuel element is submerged in the reactor pool at the time of the accidents, and only halogens and noble gases may be released to the atmosphere. This type of event has been analysis 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 dose consequences for this accident are bounded by the MHA discussed in question 52. The accident will be terminated by the Facility director once the releasable gaseous fission products have been released and cleared from the reactor room, area radiation monitors return to normal levels, and the fuel rod has been recovered and stored in a secure location (core or storage rack). The Facility Director with advice from the First Responders including the Radiation Safety officer terminates the accident. The event will be transitioned in to clean-up based on dose rate.

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 on barrier such as the pneumatic transfer and irradiation 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/m3. 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 fenceline 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.

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 3.40×10^{-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. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, Washington, DC.

57. NUREG–1537, Part 1, Section 13.1.9, "Mishandling or Malfunction of Equipment" requests the applicant to provide analysis regarding equipment mishandling or malfunction. DTRR SAR, Section M.1.7, describes an accident that involves dropping a lead transfer cask in the pool but does not discuss the accident results and dose consequences. Please provide information pertaining to the results and consequences of the accident scenario with the dropped lead cask.

DTRR response:
Under administrative prohibition, the DTRR do not handle casks over the reactor core. If and when this becomes necessary, a 50-59 review and amendment will be made, especially during

decommissioning. Therefore, we will withdraw the statement about "dropping transfer cask in the pool" in the DTRR SAR section M.1.7 in its entirety.

61. ANSI/ANS-15.1-2007, Section 3, "Limiting Conditions for Operations" and NUREG-1537, Part 1, Appendix 14.1, Section 3.8.2, "Materials," provide recommended LCOs. Some differences were noted with the DTRR TS LCOs. Please explain and justify why the DTRR TS LCOs differ from these guidance documents or consider revising the LCOs.

61.1 ANSI/ANS-15.1-2007, Section 3.1, "Reactor Core Parameters" recommends an LCO for fuel inspection not found in the DTRR TS LCOs.

61.2 ANSI/ANS-15.1-2007, Section 3.2, "Reactor Control and Safety Systems" recommends a specification for permitted bypassing of channels for checks, calibrations, maintenance, or measurements. DTRR TS 3.3, "Reactor Control and Safety Systems" does not specify when it is permitted to bypass channels for checks, calibrations, maintenance or measurements.

61.3 ANSI/ANS-15.1-2007, Section 3.3, "Coolant Systems" recommends an LCO for monitoring pool leaks, loss-of-coolant, and isolation valve positions which were not found in the DTRR TS 3.4, Coolant System.

61.4 ANSI/ANS-15.1-2007, Section 3.7, "Radiation Monitoring Systems and Effluents" recommends an LCO for monitoring environmental conditions which is not found in DTRR TS 3.6, Radiation Monitoring Systems.

61.5 ANSI/ANS-15.1-2007, Section 3.8.1 recommends a limit for the sum of the absolute values of the reactivity worth of all experiments. DTRR TS 3.7, "Experiments" provides a limit for the total absolute reactivity worth of in-core experiments that is inconsistent with this recommendation.

61.6 ANSI/ANS-15.1-2007, Section 3.8.1 recommends a specification for the absolute reactivity worth of individual experiments. DTRR TS 3.7, "Experiments" does not provide reactivity worth limits for individual unsecured, secured or movable experiments.

61.7 NUREG-1537, Part 1, Appendix 14.1, Section 3.8.2, "Materials," recommends that containers for experiments containing known explosive materials be designed such that the design pressure of the container is twice the pressure the experiment can potentially produce. DTRR TS 3.7, "Experiments" Specification 5 does not include this guidance for known explosive material containers.

DTRR response:

61.1. An LCO for fuel burn-up has been added to the TS 3.2.4 under the caption, "damaged fuel". The definition of "damaged fuel", TS definition 1.9, now includes fuel burn-up. The corresponding surveillance is TS 4.2d.

61.2. A specification has been added, TS 3.3.7., which states that: "Bypassing of channels and interlocks in table 3.3A is not permitted.

61.3. The pool water level is kept at a level much higher than the TS 3.4.4 level. The pool water is equipped with a surface "skimmer" which, among other steps, helps to keep the pool water clean and to maintain the TS pool water conductivity requirement. For the surface "skimmer" to work effectively, the pool water level is controlled within a narrow range for the water levels. Therefore, the pool water level is monitored daily and whenever the reactor is operated. The DTRR has a pool water level monitoring and alarm system to assure that the pool water level never goes below or above the given set points. It is a combination of mechanical and electrical system. The mechanical part is composed of dual air gaps. When the water level is lower than a set level, both air gaps have no water and the lower level alarm will sound, which is loud and audible in the control room. When the water level is higher than a set level, both air gaps have water in them and the high level pool water alarm will sound. The distance between these air gaps, a narrow range, is the normal operating range within which water level is maintained.

A specification has been added requiring that this pool water level monitoring and alarm system be tested annually, TS 3.4.6.

61.4. An LCO for an environmental monitor has been added to TS 3.6.3

61.5. An LCO was added regarding the sum of the absolute reactivity worths has been added, TS 3.7.2.

61.6. TS 3.7.3 is the specification for movable experiments.

61.7. TS3.7.5 has been revised to include that the pressure produced by an experiment is less than half the design pressure of the container.

62. ANSI/ANS-15.1-2007, Section 4, "Surveillance Requirements," identifies recommended Surveillance Requirements (SRs). The following differences were noted in comparison to DTRR TS surveillance requirements. Please provide justification regarding the differences noted or consider revising the surveillance requirements noted below.

62.1 DTRR TS 4.0, "Surveillance Requirements" does not specify which surveillances, if any, are required for safety while the reactor is shutdown and thus should not be deferred during a period when the reactor is shutdown.

62.2 DTRR TS 4.0, "Surveillance Requirements" does not specify which surveillances are required prior to, or following maintenance, inspection, and fuel movement activities.

62.3 DTRR TS 4.2, "Reactor Control and Safety Systems" Specification 2 accomplishes calibration of NM1000, however, no equivalent specification applies to NPP1000.

62.4 ANSI/ANS-15.1-2007, Section 4.3(4) recommends that reactor coolant be analyzed for radioactivity annually. This specification was not found in DTRR TS 4.3, "Coolant Systems."

62.5 DTRR TS 3.5, "Confinement" has no corresponding SR.

62.6 DTRR TS 3.8, "Experiments" has no corresponding SR.

62.7 DTRR TS 4.4, "Radiation Monitoring Systems" does not specify the frequency of evaluation of environmental monitors.

DTRR response:

62.1. TS 4.0 has been changed to include the following. "Required surveillances of the CAM and ARM shall not be deferred for extended reactor shutdown."

62.2. TS 4.1 requires that the reactivity worth of each control rod, reactor core excess and reactor shutdown margin be measured at least annually and after each time the core fuel is moved. The testing or surveillance required prior to or following these maintenance, inspection, and fuel movement activities are a part of the procedure for such surveillance or maintenance. These are documented in the DTRR operating procedures. In the case of unspecified maintenance item, the maintenance log contains sufficient requirement that requires the operator to determine and indicate the necessary tests prior to or following maintenance and record them down in the maintenance log before carrying out the maintenance or before putting the item back into operation. Whether there is a change in facility or not, the maintenance log also required that the operator determines, if a maintenance or a change requires a 50-59 review.

62.3. The NPP1000 channel calibration has been added to TS 4.3.2.

62.4. TS 4.4.1 requires monthly testing of the pool water. This specification exceeds the ANSI/ANS-15.1-2007, Section 4.3(4).

62.5. TS 3.5 requires that the ventilation systems shall be operational when the reactor is operated fuel is manipulated, or radioactive materials with the potential of airborne releases are handled in the reactor room. The corresponding surveillance requirement is TS 4.5

62.6. TS 3.7, "Experiments" has a corresponding surveillance requirement, which is TS 4.7. Note that there is no more TS 3.8.

62.7. The statement "The environmental monitors shall be changed at least semi-annually", has been added as TS 4.6.3 to be the corresponding environmental monitoring surveillance requirement for TS 3.6.3

64. ANSI/ANS-15.1-2007, Section 5.0, "Design Features" provides information, identified below, regarding content and format that was not found in the DTRR TS. Please provide additional information for each of the following:

64.1 ANSI/ANS-15.1-2007, Section 5.1, "Site and Facility Description" recommends

a description of the site and the facility expressly identifying the extent of the reactor license coverage.

64.2 ANSI/ANS-15.1-2007, Section 5.2, "Reactor Coolant System" requests a description of the reactor coolant system including materials and applicable temperatures.

64.3 ANSI/ANS-15.1-2007, Section 5.3, "Reactor Core and Fuel" recommends providing a description of: 1) core parameters including fuel enrichment; 2) conditions for operation of the reactor with damaged or leaking fuel elements; and 3) fuel burn-up limits.

DTRR response:

64.1. A definition of licensed area has been added to the TS definitions in TS 1.13

64.2. The reactor coolant is de-ionized water. A sentence has been added to TS 3.4 for clarity. Additionally the temperature, pH and conductivity limits are defined in TS 3.4.

64.3. Fuel enrichment is defined in the TS definition 1.39. Conditions for operation with damaged fuel are described in TS 3.2.4. Fuel burn-up has been added to the definition of damaged fuel, TS definition 1.9.

65. ANSI/ANS-15.1-2007, Section 6, "Administrative Controls" provides recommendations regarding content and format. DTRR TSs differences from these recommendations were noted. Please provide additional information for the following:

65.1 ANSI/ANS-15.1-2007, Section 6.1, "Organization" recommends organizational structures including levels and reporting authority. DTRR TS Figure 6.1, the DTRR organization structure, includes no level 3 or 4 staff and differs from Figure 8.0 of the DTRR SAR.

65.2 ANSI/ANS-15.1-2007, Section 6.1.2, "Responsibility" describes responsibilities for the operation and safeguarding of the public which was not fully described in DTRR TS 6.1.2. Please describe the Facility Director's responsibilities and clarify what is meant by "management sense."

65.3 ANSI/ANS-15.1-2007, Section 6.1.3(3), "Staffing" lists those events requiring the senior reactor operator to be present at the facility. DTRR TS 6.1.3 does not include initial startup and approach to power recommended by ANSI/ANS-15.1, Section 6.1.3(3)(a) and required by 10 CFR Section 50.54(m)(1).

65.4 ANSI/ANS -15.1-2007, Section 6.1.4, "Selection and Training of Personnel" recommends meeting or exceeding the criteria in ANSI/ANS-15.4-1988 (R1999). DTRR TS 6.1.4 states that the implementation shall be consistent with all current regulations but does not indicate what regulations or guidance is being met.

65.5 ANSI/ANS-15.1-2007, Section 6.2.2, "Charter and Rules" provides recommendations that are incorporated into DTRR TS 6.2.1 except for the provision for "quick action" in DTRR TS 6.2.1. Please explain what this

means and when this would be necessary.

65.6 ANSI/ANS-15.1-2007, Section 6.2.4, "Audit Function" recommends an audit of the facility emergency plan and implementing procedures and the operator requalification program that is performed by an individual not immediately responsible for the audited area. DTRR TS 6.2.3.b states that these audits may be satisfied by the annual review of these plans for the requalification program.

Please explain how this audit process meets the recommendation that the audit be performed by an individual not immediately responsible for the area audited.

65.7 ANSI/ANS-15.1-2007, Section 6.4, "Procedures," recommends procedures for several categories of activities. DTRR TS 6.3 does not apply this recommendation to emergency and security plans, surveillances, and experiments.

65.8 ANSI/ANS-15.1-2007, Section 6.5, "Experiment Review and Approval" recommends that experiments be carried out in accordance with approved procedures. DTRR TS 6.4 does not describe the process of experiment review and approval for new experiments.

65.9 ANSI/ANS-15.1-2007, Section 6.7, "Reports" provides recommendations for reporting activities. The DTRR TS Section 6.6.1 (c) uses the outdated terminology "un-reviewed safety question."

65.10 ANSI/ANS-15.1-2007, Section 6.7.2(1), "Special Reports" specifies facsimile or similar conveyance of the special report. DTRR TS 6.6.2.a specifies telegraph of similar conveyance.

65.11 ANSI/ANS-15.1-2007, Section 6.8, "Records" provides recommendations for record retention. ANSI/ANS-15.1-2007, Section 6.8.2 recommends an administrative control that retraining and requalification records for operators be retained for at least one certification cycle (per 10 CFR 55.55(a) this period is 6 years) and be maintained at all times the individual is employed or until the certification is renewed. DTRR TS 6.7.2 is not consistent with this ANSI/ANS Section 6.8.2 guidance.

65.12 ANSI/ANS-15.1-2007, Section 6.8.3, "Records to be retained for the lifetime of the reactor facility" recommends certain records be maintained for the lifetime of the facility. DTRR TS 6.7.3 does not implement the full extent of those recommendations (e.g. it does not include records of violations of safety limits, LSSS, LCOs; environmental monitoring; or approved changes in operating procedures).

DTRR response:

65.1. Figure 6.1 of the TS has been changed to comply with ANSI/ANS-15.1, by clearly stating the designations of Level-1, Level-2, Level-3 and Level-4. Also Figure 6.1 now includes the chain of command for the Radiation Safety Officer within the organization. This figure will replace Figure 8 in the revised SAR for the DTRR.

The Radiation Safety Committee handles matters concerning radiation safety for the larger Dow Chemical Company, Midland Operations. It holds a broad scope license for

Dow Midland operations. It administers the ALARA program for several radiation generating machines, loose isotope uses and sealed sources for the Midland operations. In this role the RSC, through the radiation officer and a health physics technician, oversees and administers the ALARA program for the DTRR. This includes but not limited to, training of reactor operational staff on matters of radiation safety and uses, issuance of permission to the reactor operational staff to handle radioactive material and personal dose meter assignment and evaluation for the DTRR personnel. To facilitate this, the reactor supervisor is a member of the RSC and reports to it at least twice per year. Radiation safety officer conducts internal audits of the DTRR facility for the RSC as required by the RSC. Also, the Radiation Safety Officer is a member of the ROC and communicates to it regularly.

65.2. The facility director's role has been defined in TS 6.1.2. The term "management sense" has been removed from TS 6.1.2.

65.3. Initial start-up and approach to power has been added to the TS 6.1.3 to comply with ANSI/ANS-15.1.

65.4. ANSI/ANS -15.4, Sections 4 through 7 have been added to TS 6.1.4 as the reference for selection and training of personnel.

65.5. Reference to a "quick action" has been removed from the TS. TS 6.2.1e is required to allow the Facility Director the ability to poll the ROC via phone or email.

65.6. TS 6.2.3b has been changed to state that the ROC will direct a biennial audit of the emergency plan.

65.7. Emergency Plan procedures are required in TS 6.3b. Surveillances procedures are addressed in TS 6.3e. A requirement for procedures for operation of each experimental facility has been added as TS 6.3g.

65.8. The following restriction has been added to TS 6.4. No experiment shall be performed without review and approval by Reactor Operations Committee. Experiments are approved and classified by the ROC as Routine, Modified Routine or Special. Experiments shall be reviewed with respect to 10 CFR part 20, and TS 3.7.

65.9. The phrase "un-reviewed safety question" has been replaced with "they are allowed without prior authorization by the Nuclear Regulatory Commission"

65.10. Reference to use of a telegraph has been eliminated from TS 6.6.2a.

65.11. TS 6.7.2 has been revised to include the retention time specified in ANSI/ANS section 6.8.2.

65.12 TS 6.7.3 has been revised to include the retention of safety violations, LSSS, LCO; environmental monitoring, and approved changes in the operating procedures.