The Dow Chemical Company
Midland, Michigan 48667

April 9, 2014

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 responses to RAI letter, dated March 6, 2014, and the DTRR revised Technical Specifications. These documents are submitted as attachment I and attachment II respectively.

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 April 9, 2014

Paul O'Connor

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

Subscribed and sworn to before me this 9th day of April, 2014

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, Facility Director
Siaka Yusuf, Reactor Supervisor

A020
NRR

Attachment I

DTRR response to RAI letter dated March 6, 2014: Questions 1 through 24.

Today: 4/9/2014

RAI-1. The proposed DTRR TS 3.7.2, "Materials," Specification b, does not appear to provide an upper limit for the quantity of explosives (e.g., greater than 25 milligrams TNT equivalent) that may be irradiated. NUREG-1537 provides guidance that licensees should establish an upper limit for the quantity of explosive material that may be irradiated. Provide justification for the proposed TS 3.7.2, or revise to follow the guidance provided in NUREG-1537 and establish an upper limit for the quantity of explosive material that may be irradiated.

DTRR response:
TS 3.7.2 specification b has been revised by providing an upper limit using the phrase "up to 25 milligrams of TNT equivalent" in the DTRR TS (April 2014).

RAI-2. The proposed DTRR TS 3.7.2, "Materials," Specification c, provides a specification for the irradiation of fissionable material. However, an accompanying license condition is needed to permit the possession of fissionable material for irradiation. Provide a proposed license condition for the possession of fissionable material for irradiation including a mass (gram) limit and a description of the material form, or, if the DTRR does not intent to irradiate fissionable material, revise or delete the TS, as appropriate.

DTRR response:
TS 3.7.2 specification c has been deleted from the revised TS, DTRR TS (April 2014). Operational experience at the DTRR has shown that this specification is not necessary.

RAI-3. The proposed DTRR TS 3.7.3, "Experimental Failure and Malfunctions," provides requirements to limit the radioactivity release due to an experimental failure or malfunction to not exceed the limits in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20. Your proposed TS does not specify what section of 10 CFR 20 is limiting. However, the guidance in NUREG-1537 specifies the effluent limits not to exceed the limits listed in 10 CFR 20, Appendix B. Provide justification for the proposed TS 3.7.3, or revise to provide a limit based on 10 CFR 20, Appendix B.

DTRR response:
TS 3.7.3 has been revised by referencing the specific sub-part of the Code of Federal Regulations, in the DTRR TS (April 2014).

RAI-4. The proposed DTRR TS 3.6.2, "Effluents," Bases, states, in part, a dose value that "is 30% lower than the 10 CFR 20 Appendix B effluent concentration value...." It is not clear if the 30% reduction relates to the dose or concentration value. The dose value given is not consistent with Appendix B. Provide justification for the proposed TS 3.6.2, Bases, or revise to accurately reflect the 30% reduction.

DTRR response:
TS 3.6.2 basis has been revised by referencing the specific sub-part of the Code of Federal Regulations, consistent with the calculations provided, in the DTRR TS (April 2014).

RAI-5. The following RAIs are based on the MHA results, as provided in your response to RAI No. 52 (ADAMS Accession No. ML 12026A152):
a. The ventilation system response to a significant radiological release, such as an MHA, is not fully described. NUREG-1537 provides guidance to describe the response by the ventilation system to an MHA event. Provide a description of the response of the ventilation system to the MHA. Indicate if the ventilation system response is the result of any automatic initiation or the result of operator action, or both. For any operator action, describe the supporting procedure guidance.
b. The results of the dose calculations provided in Scenario 3 to a member of the public at the fence line boundary were based on effluent releases provided by the ventilation system operating in normal mode. However, dose calculations were not provided with the ventilation system not operating (i.e., in shutdown mode). Given that the ventilation system could be shutdown during, or shortly after the initiation of an MHA event, provide a dose calculation to the maximally exposed member of the public. List any assumptions for MHA effluent leakage from building penetrations (e.g., door seals, etc.) and for direct radiation shine from the radioactive materials retained in the reactor room.

DTRR response:

a) In the event of a significant radiological release such as that outlined in the MHA (Rupture of a fuel element), the ventilation system will be manually controlled using a switch in the control room by the reactor operator. This action will shut down the ventilation system. The operator also manually closes the exhaust with a mechanical actuator located outside the building and near the exhaust. The operator makes sure door 10 and control room door are closed, thereby essentially confining the radiation release into the reactor room only.
b) In this scenario, a fuel element ruptures and releases its contents, but the building ventilation is shut down, so the released radioactive material is contained in the reactor room. The Total Effective Dose Equivalent to an unrestricted public at the fenceline will be made of two components: 1) Direct Exposure from radionuclides contained in the reactor room and 2) Exposure from radionuclides that leak out of the reactor room.

1. Direct Exposure from Radionuclides Contained in the Reactor Room

This scenario was analyzed in response to RAI No. 52 (ADAMS Accession No. ML 12026A152) for workers who remain inside the reactor room, and it was determined that their deep dose equivalent rate from submersion in the radioactive cloud would be 131 mrem/hr. To conservatively estimate the dose rate at the fenceline (23 m to the west of the reactor room), this calculated direct exposure rate is modified by two factors:

I. the submersion dose rate is divided in half (to calculate the dose rate at the west wall of the reactor room, where the individual is exposed to only half of an infinite cloud of airborne radionuclides), and
II. the calculated dose rate is adjusted for the spread of the radiation field from the reactor room to the fenceline.

To calculate the spread of the radiation field from the reactor room to the fenceline, it will conservatively be assumed that the source of the radiation is concentrated in the center of the east wall of the reactor room, 25ft from the west wall. This will minimize the angle of spread of

the radiation that travels from the west wall of the reactor room to the fenceline, as any radiation released closer to the west wall could be released at a larger angle as compared to the direct exposure path, which would result in a lower dose estimate at the fenceline. The radiation field will be assumed to have a collimated release covering the 20ft x 12ft west wall of the reactor room, and dose rate at the fenceline will be calculated based on the projected area that this collimated beam of radiation will expand to at the fenceline. Using these parameters, the radiation incident upon the 240 ft^2 west wall of the reactor room will project to a 75.2ft x 45ft area at the fenceline. This equals a reduction in the dose rate of (240 ft^2)/(3392 ft^2), or 0.07, from the reactor room wall to the fenceline. (Note that this is equal to a $1/r^2$ calculation of the reduction of dose rate with distance with d_1 equal to 25ft and d_2 equal to 94ft).
Based on these modifications to the submersion dose, the initial dose rate to an individual located at the fenceline if all radionuclides released during a fuel rupture incident would be 131 mrem/hr * 0.5 * 0.07, or 4.6 mrem/hr. Assuming that it takes 30 minutes for the incident to be resolved or the fenceline area to be cleared of members of the public, the total deep dose equivalent received from direct exposure to the member of the public from the incident would be 2.3 mrem. This estimate of exposure is conservative because it ignores any shielding provided by the reactor room wall and the air between the reactor room and the fenceline, and also ignores any radioactive decay of the radioactive materials following their release.

2. Exposure from Radionuclides that Leak out of the Reactor Room

In addition to the direct exposure from the radionuclides trapped in the reactor room, there could be some leakage of radioactive materials to the environment that also expose members of the public at the fenceline. In Dow's prior response to RAI No. 52 (ADAMS Accession No. ML 12026A152), it was calculated that airborne radionuclides trapped in the reactor room would be leaked into the control room at a rate of 0.023 m^3/sec. Assuming a similar release rate from the reactor room to the outside environment, release rates of the radionuclides involved in the incident can be calculated (see Table 5.1). From these release rates, a fenceline concentration of radionuclides during the incident can be calculated by:

$$C = R * \chi/Q$$

In Dow's prior response to RAI No. 52 (ADAMS Accession No. ML 12026A152), a χ/Q value of 0.106 sec/m^3 was calculated for releases from the reactor building to the fenceline nearest to the reactor building. Using this formula, airborne concentrations of radionuclides at the fenceline are calculated, as shown in Table 5.1.

Using these concentrations, the Total Effective Dose Equivalent for an individual standing at the fenceline during the incident was calculated including both the inhalation and submersion pathways. These calculations were based on a breathing rate of 1.2 m^3/hr. The Total Effective Dose Equivalent Rate at the fenceline is calculated to be 4.3 mrem/hr.
Based on these calculations, and an exposure time of 30 minutes before emergency response is able to clear the area including the fenceline, the Total Effective Dose Equivalent to the maximally exposed member of the public from released radionuclides from this incident is 2.2 mrem. The assumption that the exposure time is 30 minutes is based on Emergency response time provision in the DTRR emergency plan.

In conclusion, the maximum exposure from the incident will be the sum of the Total Effective Dose Equivalent from direct exposure to radionuclides that are contained in the reactor room and the Total Effective Dose Equivalent from the exposure to radionuclides that leak out of the

reactor room and travel downwind to the fenceline. Adding these two values (2.3 mrem from contained radionuclides and 2.2 from leaked radionuclides, the total Total Effective Dose Equivalent to the maximally exposed member of the public from this incident is calculated to be 4.5 mrem. This is less than the Total Effective Dose Equivalent of 6.7 mrem, which was calculated for a member of the public at the fenceline when the ventilation system keeps running. Both values are less than the annual dose limit to members of the public in 10 CFR 20.1301(a)(1) of 100 mrem/yr.

Table 5.1. Dose Rates to a Member of the Public from the Leakage of Radioactive Materials from the Reactor Room Following a Fuel Element Rupture Event when the Ventilation System is shutdown

Radionuclide	Activity Released from Closed Reactor Room (Ci/sec)	Downwind Concentration (Ci/m3)	Inhalation Dose Rate at Fenceline (rem/hr)	Submersion Dose Rate at Fenceline (rem/hr)	Total Effective Dose Equivalent Rate at fenceline (rem/hr)
Br-82	3.40E-11	3.61E-12	6.61E-09	1.73E-12	6.62E-09
Br-83	7.00E-08	7.44E-09	7.96E-07	1.05E-11	7.96E-07
Br-84	1.28E-07	1.36E-08	1.58E-06	4.73E-09	1.58E-06
Br-84M	4.79E-09	5.09E-10	5.89E-08	1.77E-10	5.91E-08
Br-85	1.62E-07	1.72E-08	1.99E-06	5.99E-09	2.00E-06
Br-86	2.26E-07	2.40E-08	2.78E-06	8.34E-09	2.78E-06
Br-87	2.54E-07	2.69E-08	3.12E-06	9.37E-09	3.13E-06
I-131	4.03E-07	4.27E-08	1.69E-03	2.88E-09	1.69E-03
I-132	6.23E-07	6.61E-08	3.02E-05	2.74E-08	3.03E-05
I-133	9.41E-07	9.99E-08	7.01E-04	1.09E-08	7.01E-04
I-134	1.07E-06	1.14E-07	1.79E-05	5.47E-08	1.80E-05
I-135	8.80E-07	9.34E-08	1.38E-04	2.76E-08	1.38E-04
I-136	3.30E-07	3.51E-08	1.38E-03	1.69E-08	1.38E-03
Kr-83M	7.00E-08	7.44E-09	0.00E+00	1.13E-10	1.13E-10
Kr-85	3.51E-08	3.73E-09	0.00E+00	6.48E-09	6.48E-09
Kr-85M	1.61E-07	1.71E-08	0.00E+00	1.89E-06	1.89E-06
Kr-87	3.19E-07	3.39E-08	0.00E+00	1.78E-05	1.78E-05
Kr-88	4.50E-07	4.77E-08	0.00E+00	6.36E-05	6.36E-05
Kr-89	5.78E-07	6.13E-08	0.00E+00	8.17E-05	8.17E-05
Xe-131M	4.90E-09	5.20E-10	0.00E+00	2.85E-09	2.85E-09
Xe-133M	2.81E-08	2.98E-09	0.00E+00	5.93E-08	5.93E-08
Xe-133	9.41E-07	9.99E-08	0.00E+00	2.24E-06	2.24E-06
Xe-135M	1.79E-07	1.91E-08	0.00E+00	5.31E-06	5.31E-06
Xe-135	9.32E-07	9.90E-08	0.00E+00	1.71E-05	1.71E-05
Xe-137	8.51E-07	9.04E-08	0.00E+00	6.42E-05	6.42E-05
Xe-138	8.64E-07	9.17E-08	0.00E+00	6.52E-05	6.52E-05
Totals:			3.97E-03	3.19E-04	**4.29E-03**

c) Dose to the nearest member of the public, who may not be removable, during an emergency

The nearest privately owned residence to the reactor would be located more than 100 m to the west of the reactor.

In the case that the ventilation system remains running, the airborne radionuclides that are released into the reactor room will be vented very rapidly, in a much shorter duration than the 30 minutes that is assumed to be required to clear members of the public from the fenceline. Therefore, the Total Effective Dose Equivalent to the maximally exposed member of the public who may not be removable will be lower than the Total Effective Dose Equivalent to the member of the public at the fenceline calculated previously of 6.7 mrem.

If the ventilation system is not running, the leakage of airborne activity will take place over a longer period of time. Using the same leakage rate of 0.023 m3/sec as above , it will take approximately 48 hours for the reactor room to release its contents outside of the reactor building. The estimate of the Total Effective Dose Equivalent at the uncontrolled location 100 m to the west of the reactor will consist of the direct exposure to the radionuclides inside the reactor room plus the inhalation and submersion dose from the radionuclides that are released from the reactor room.

Direct Exposure from the Radionuclides Trapped in the Reactor Room

The direct exposure for the member of the public at the nearest uncontrolled area will be calculated in the same manner that was used for the member of the public at the fenceline, above. The only differences in the calculation will be the 100 m distance to the reactor room and reduction of radionuclide concentration inside the control room due to leakage and decay will be accounted for in the calculation due to the longer time frame that exposure can take place.

The rate of reduction of the dose rate due to radioactive decay will vary by radionuclides. Additionally, as the radionuclides leak out of the reactor room, the direct exposure dose rate will drop due to this leakage. Table 5-2 shows the most significant contributors to the direct exposure dose, and how the dose rate falls with time, as well as the reduction factor to account for the leakage of the radionuclides out of the building.

Table 5-2 – Calculation of the Direct Exposure Dose for a Member of the Public in an Uncontrolled Area

Time (hr)	Kr-88 Dose Rate (rem/hr)	Kr-89 Dose Rate (rem/hr)	Xe-137 Dose Rate (rem/hr)	Xe-138 Dose Rate (rem/hr)	Total Dose Rate (Decay only)(mrem/hr)	Leakage Reduction Factor	Total Dose Rate (mrem/hr)	Total Dose for Time Step (mrem) (Dose Rate * Time)
0.5	2.60E-02	3.35E-02	2.63E-02	2.67E-02	3.80E-01	1.00	3.80E-01	0.1899
1	2.30E-02	4.54E-05	1.10E-04	6.04E-03	1.27E-01	0.99	1.26E-01	0.0629
2	1.81E-02	8.39E-11	1.95E-09	3.10E-04	8.55E-02	0.97	8.29E-02	0.0829
3	1.41E-02	1.55E-16	3.44E-14	1.59E-05	6.77E-02	0.95	6.42E-02	0.0642
4	1.11E-02	2.86E-22	6.08E-19	8.15E-07	5.50E-02	0.93	5.10E-02	0.0510
5	8.68E-03	5.27E-28	1.07E-23	4.18E-08	4.53E-02	0.91	4.11E-02	0.0411
6	6.80E-03	9.73E-34	1.90E-28	2.14E-09	3.79E-02	0.89	3.35E-02	0.0335
9	3.27E-03	6.11E-51	1.04E-42	2.89E-13	2.36E-02	0.82	1.94E-02	0.0583
12	1.57E-03	3.84E-68	5.74E-57	3.89E-17	1.61E-02	0.76	1.22E-02	0.0367
18	3.64E-04	1.52E-102	1.74E-85	7.07E-25	9.09E-03	0.64	5.78E-03	0.0347
24	8.41E-05	5.99E-137	5.27E-114	1.28E-32	6.13E-03	0.51	3.13E-03	0.0188
30	1.94E-05	2.37E-171	1.60E-142	2.33E-40	4.57E-03	0.39	1.76E-03	0.0106
36	4.49E-06	9.35E-206	4.84E-171	4.24E-48	3.64E-03	0.26	9.50E-04	0.0057
42	1.04E-06	3.69E-240	1.47E-199	7.69E-56	3.05E-03	0.14	4.14E-04	0.0025
48	2.40E-07	1.46E-274	4.44E-228	1.40E-63	**2.66E-03**	0.01	2.85E-05	0.0002
							Total	**0.6928**

Summing up the total exposure of these dose rates over the 48 hours of release results in a Total Effective Dose Equivalent from direct exposure of 0.69 mrem.

Airborne Releases of Radionuclides

The methodology of calculating the dose at the uncontrolled area from airborne releases of radionuclides will be the same as the calculation of the dose at the fenceline, above, except a X/Q value for 100 m will be calculated, and decay will be accounted for due to the longer timeframe of the scenario.

At 100 m, the X/Q for transport drops from 0.106 sec/m^3 to 6.03 x 10^{-3} sec/m^3, following the calculation methodology in Reg. Guide 1.145.

The rate of reduction of the dose rate due to radioactive decay will vary by radionuclides. Table 5-3 shows the most significant contributors to the inhalation and submersion dose, and how the dose rate falls with time.

Table 5-3 – Calculation of the Direct Exposure Dose for a Member of the Public in an Uncontrolled Area

Time (min)	I-131 Dose Rate (rem/hr)	I-133 Dose Rate (rem/hr)	I-135 Dose Rate (rem/hr)	I-136 Dose Rate (rem/hr)	Total Dose Rate (mrem/hr)	Total Dose for Time Step (mrem) (Dose Rate * Time)
0.5	1.69E-03	7.01E-04	1.38E-04	1.38E-03	2.44E-01	0.122
1	1.68E-03	6.89E-04	1.31E-04	4.41E-10	1.51E-01	0.075
2	1.68E-03	6.67E-04	1.18E-04	4.47E-23	1.45E-01	0.145
3	1.67E-03	6.45E-04	1.06E-04	4.53E-36	1.42E-01	0.142
4	1.67E-03	6.24E-04	9.52E-05	4.59E-49	1.39E-01	0.139
5	1.66E-03	6.03E-04	8.57E-05	4.65E-62	1.36E-01	0.136
6	1.65E-03	5.83E-04	7.71E-05	4.71E-75	1.34E-01	0.134
9	1.64E-03	5.28E-04	5.62E-05	4.91E-114	1.27E-01	0.382
12	1.62E-03	4.78E-04	4.09E-05	5.12E-153	1.22E-01	0.367
18	1.58E-03	3.91E-04	2.17E-05	5.55E-231	1.14E-01	0.683
24	1.55E-03	3.20E-04	1.15E-05	0.00E+00	1.07E-01	0.643
30	1.52E-03	2.62E-04	6.13E-06	0.00E+00	1.02E-01	0.610
36	1.48E-03	2.15E-04	3.25E-06	0.00E+00	9.69E-02	0.581
42	1.45E-03	1.76E-04	1.73E-06	0.00E+00	9.27E-02	0.556
48	1.42E-03	1.44E-04	9.18E-07	0.00E+00	8.91E-02	0.535
Total						**5.250**

Summing up the total exposure of these dose rates over the 48 hours of release results in a Total Effective Dose Equivalent from inhalation and submersion in leaked radionuclides of 5.25 mrem.

Total Exposure in the Uncontrolled Area

Therefore, by summing the exposures over each time step in the tables above, the Total Effective Dose Equivalent to the maximally exposed member of the public in an uncontrolled area would be 0.69 mrem from direct exposure and 5.25 mrem from inhalation and submersion in released radionuclides. After the 48-hr release period, there would be no further exposure. Therefore, the Total Effective Dose Equivalent in the year following the incident would be 5.94 mrem/yr.

Note that if there was no leakage out of the reactor and all the radionuclides stayed inside the reactor room for 12 months following the incident, the Total Effective Dose Equivalent to the nearest member of the public, who may not be removable during an emergency, is still below 10 CFR 20.1301(a)(1) of 100 mrem/yr. This value is calculated by running the same calculations for direct exposure, but removing the reduction in direct exposure due to leakage (shown in the 'Decay Only' column in Table 5-2), and extending the exposure time to 365 days. Conservatively using the dose rate at 48 hours (2.66e-3mrem/hr from Table 5.2, 'decay only' column) to represent the dose rate for the remainder of the year and adding 0.69 mrem (the initial 48hr dose), the Total Effective Dose Equivalent to the maximally exposed to the nearest member of the public, who may not be removable during an emergency would be 23.9 mrem/year ($2.66e^{-3}$ X 363 X 24 + 0.69).

RAI-6. The proposed DTRR TS 1.3, "Definitions," Experiment, provides definitions for several categories of experiments (Moveable, Modified Routine, Routine, Secured, Special, and Unsecured), which appear to be based, in part, on reactivity characteristics or approval requirements. However, the definition of Routine Experiment includes an experiment, which "is not defined as any other kind of experiment." Based on this definition, is not clear as to the applicability of the various definitions of experiments to other DTRR TSs (e.g., TS 6.5.). Provide justification for the various definitions of experiments described in the proposed TS 1.3, "Definitions," and apply the definitions consistently through the applicable DTRR TSs (e.g., TS 6.5), or revise TS 1.3, "Definitions," Experiment as applicable.

DTRR response:
TS 1.3 has been revised by grouping the definition of experiments into two groups 1) based on reactivity and 2) based on the required reviews in the DTRR TS (April 2014). TS 6.5 has also been revised to consistently refer to these defined experiments without ambiguities in the revised TS, the DTRR TS (April 2014)

RAI-7. The proposed DTRR TS 1.3, "Definitions," Reference Core Condition, provides a value of -$0.30 for Xenon reactivity. Given that the reactivity required to satisfy the DTRR shutdown margin (SDM), TS 3.1, "Reactivity Limits," Specification 1, is -$0.50, in the Reference Core Condition, the resulting SDM reactivity could be as low as -$0.20, which is not consistent with the guidance in NUREG-1537, which provides a value of -$0.50 SDM reactivity. Provide justification for the proposed TS 1.3, "Definitions," Reference Core Condition by proposing a TS 3.1 SDM reactivity limit that is consistent with the SDM guidance in NUREG-1537 or revise TS 1.3, "Definitions," Reference Core Condition.

DTRR response:
TS 1.3 has been revised by removing the phrase "(<$0.30)" from the definition of the reference core condition in the DTRR TS (April 2014). The SDM in TS 3.1 is retained at $0.50 and xenon reactivity will be accounted for using procedural control whenever shutdown margin is calculated. Operational experience at the Dow TRIGA reactor shows that this procedural assurance is adequate.

RAI-8. The proposed DTRR TS 4.0, "General," Specifications 1 and 4, provide items that may not be deferred during a reactor shutdown. However, the items identified in Specification 1 do not match those items identified in Specification 4. As such, the resulting inconsistency could be subject to misinterpretation. Provide justification for the proposed TS 4.0, Specifications 1 and 4, or revise to identify a consistent list of surveillance items that may be deferred during reactor shutdown.

DTRR response:
TS 4.0 has been revised by changing the phrase "except TS4.4, items 1 and 2" to "except TS 4.4, items 1, 2 and TS 4.6, item 2" and removing TS 4.0 item 4 in the DTRR TS (April 2014) and to be consistent with the items listed in TS 4.0, TS 4.4 and TS 4.6.

RAI-9. The proposed DTRR TS 4.1, "Reactor Core Parameters," Specification, states, in part, that the reactivity measurements will be performed "following any change of reactivity by $0.25 or more from the reference core." Details describing the types of activities that could result in a reactivity change of $0.25 or more are not provided. Provide information that describes the types of activities that could result in reactivity changes of $0.25 or more.

DTRR response:
Activities that could cause a reactivity change of $0.25 are usually associated with the reactor core. These activities may include, fuel movement, control rod removal for inspection, experimental facility movement, insertion of a new fuel, replacement of a dummy fuel etc. The choice of $0.25 is because the value of reactivity change (>$0.25) is measurable and will ensure adequate coverage of the shutdown margin after taking into account the accumulation of poisons. This statement has been added to the basis of TS 4.1 in the revised TS, DTRR TS (April 2014)

RAI-10. The proposed DTRR TS 4.3, "Control and Safety Systems," Specification 4, provides for control rod inspection, but lacks any inspection criteria. NUREG-1537 provides guidance for control rods that should be inspected for damage and deterioration. Provide justification for the proposed TS 4.3, or revise to include inspection criteria consistent with the guidance in NUREG-1537.

DTRR response:
TS 4.3 has been revised by removing the phrase "at least" from items 1, 2 and4, and including control rod inspection criteria, such as damage or deteriorations, in item 4 in the DTRR TS (April 2014).

RAI-11. The proposed DTRR TS 4.4, "Reactor Coolant Systems," Specifications 3 and 5, state, in part, that "[a] channel calibration is performed if required as a result of the channel test." However, no periodic channel calibration is provided and the calibration criteria are not defined. NUREG-1537 provides guidance that manufacturer's recommendations should be considered for calibration. Provide justification for the proposed TS 4.4, Specifications 3 and 5, or revise to ensure a periodic calibration, consistent with the guidance in NUREG-1537 (e.g., manufacturer's recommendations), is provided.

DTRR response:
TS 4.4 specifications 3 and 5 have been revised to include the statement "as recommended by the manufacturer" in DTRR TS (April 2014).

RAI-12. The proposed DTRR TS 6.1.1, "Structure," Figure 6.1., indicates a communication relationship (i.e., the dashed line) between the Dow Core R&D Director (Level 1) and the Reactor Operations Committee (ROC). However, NUREG-1537 provides guidance that the review/audit group should report (i.e., a solid line) to the Level 1. Additionally, the Radiation Safety Committee (RSC) appears to have a typographical error as "ROC" is referenced in the RSC box. Provide justification for the communication relationship in proposed TS Figure 6.1, or revise to indicate a reporting relationship consistent with the guidance in NUREG-1537.

DTRR response:
TS 6.1.1, Figure 6.1, has been revised by including reporting lines between the ROC and the Level 1 manager in the DTRR TS (April 2014). Also, the typographical error has been corrected.

RAI-13. The proposed DTRR TS 6.1.3, "Staffing," Item 1, states, in part, "[t]he minimum staffing when the reactor is operating...." NUREG-1537 provides guidance that the minimum reactor staffing is required when the reactor is "not secured." Provide justification for the proposed TS 6.1.3, Item 1, or revise to ensure the minimum staffing requirements consistent with the guidance in NUREG-1537.

DTRR response:
TS 6.1.3 item 1 has been revised by changing the phrase "not operating" to "not secured" in the DTRR TS (April 2014).

RAI-14. The proposed DTRR TS 6.1.3, "Staffing," Item 3.d, states, in part, "[r]ecovery from...significant power reduction." However, the criteria for the determination of a significant power reduction are not defined in the DTRR TSs. Provide justification for the proposed TS 6.1.3, Item 3.d, or revise by incorporating criteria in order to determine a significant power reduction.

DTRR response:
TS 6.1.3 item 3.d has been revised. It now reads "Recovery from unplanned or unscheduled shutdown or unscheduled power reduction" in the DTRR TS (April 2014).

RAI-15. The proposed DTRR TS 6.2.1, "Composition and Qualification", states, in part, that "[t]he ROC shall consist of at least four members..." and "...shall be appointed by Level 1 management." However, TS 6.2.1 subsequently lists more than four members by title and the determination and composition of the ROC membership is not clearly defined. Provide justification for the proposed TS 6.2.1, or revise to provide clear guidance as the composition and determination of the ROC membership.

DTRR response:
TS 6.2.1 has been revised to include the phrase "Other ROC members shall include the following as determined by Level 1" in the DTRR TS (April 2014).

RAI-16. The proposed DTRR TS 6.2.2, "ROC Rules," Item a, states, in part, that "no more than one-half of the voting members present may be of the operating staff." However, the operating staff is not defined. Provide justification for the proposed TS 6.2.2, Item a, or revise to provide guidance as to which voting members constitute operating staff.

DTRR response:
TS 6.2.2 has been revised to clearly state what constitutes operating staff, in the DTRR TS (April 2014).

RAI-17. The proposed DTRR TS 6.2.2, "ROC Rules," Item e, states, in part, that "[t]he ROC shall report at least twice per year to the Radiation Safety Committee (RSC) through presentations by the reactor supervisor at the quarterly RSC meetings." However, it is not clear how the "report" through presentations is accomplished. Provide justification for the proposed TS 6.2.2, Item e, or revise to provide clear guidance as to how the ROC communicates the audit findings. Confirm that TS Figure 6.1 accurately depicts the relationship between the ROC and RSC.

DTRR response:
TS 6.2.2, item e, has been revised by removing the phrase "report" and replacing it with the phrase "communicate" in the DTRR TS (April 2014). TS 6.2.2, item e, is now consistent with Figure 6.1 in the DTRR TS (April 2014)

RAI-18. The proposed DTRR TS 6.2.3, "ROC Review Function," Item a, states, that the ROC shall "[r]eview all changes made under 10 CFR 50.59." NUREG-1537 provides guidance that the review should include the determinations that the proposed changes were allowed without prior NRC approval. However, TS 6.2.3, Item a, does not includes this determination. Provide justification for the proposed TS 6.2.3, Item a, or revise to include the review guidance for proposed changes consistent with the guidance in NUREG-1537.

DTRR response:
TS 6.2.3 has been revised by removing the phrase "under 10CFR-50.59" and replacing it with "in the facility" in the DTRR TS (April 2014).

RAI-19. The proposed DTRR TS 6.4, "Procedures," states, in part, that "[o]perating procedures shall be in effect..." but does not indicate if the procedures are required to be used and followed.
Provide justification for the proposed TS 6.4, or revise to include guidance for using these procedures.

DTRR response:
TS 6.4 has been revised by replacing the phrase "in place" with the phrase "used" in DTRR TS (April 2014).

RAI-20. The proposed DTRR TS 6.5, "Experimental Review and Approval," states, in part, "[a]pproved experiments..." and "[a]ll new experiments..." However, "approved experiments" and "new experiments" are not defined in the list of experiments provided in TS 1.3, "Definitions," Experiment, (which included Moveable, Modified Routine, Routine, Secured, Special, and Unsecured Experiments). It is not clear which category of experiments is applicable in proposed TS 6.5. Provide a justification for the proposed TS 6.5, or revise to delineate which experiments require review and approval.

DTRR response:
TS 6.5 has been revised by replacing the phrase "new" with the phrase "modified routine and special" and to be consistent with TS 1.3 as revised in DTRR TS (April 2014).

RAI-21. The proposed DTRR TS 6.6.1, "Actions to Be Taken in Case of Safety Limit Violation," Item b, states, in part, "[a]n immediate notification of the occurrence shall be made to the...U.S. NRC Headquarters Operations Center." This notification requirement conflicts with DTRR TS 6.7.2, "Special Reports," Item 1, which requires "[a] report not later than the following working day...." Additionally, TS 6.6.1, Item b, does not appear consistent with the guidance in NUREG-1537. Provide justification for the proposed TS 6.6.1, Specification b, or revise to be consistent with the guidance in NUREG-1537.

DTRR response:
TS 6.6.1 has been revised by removing the phrase "U.S. NRC Headquarters Operations Center" in DTRR TS (April 2014) so that TS 6.6.1 does not conflict with TS 6.7.2.

RAI-22. The proposed DTRR TS 6.7.1, "Annual Operating Reports," Item c, states, in part, the annual report shall include a tabulation of "new tests and experiments that are significantly different from those performed previously and are not described in the Safety Analysis Report." NUREG-1537 provides guidance that the annual report should include all new tests and experiments and not just significant tests and experiments to meet the requirements of 10 CFR 50.59. Provide justification for the proposed TS 6.7.1, Item c, or revise to be consistent with the guidance in NUREG-1537.

DTRR response:
TS 6.7.1 item c has been revised and replaced the phrase "a tabulation" with the phrase "brief description" in the DTRR TS (April 2014).

RAI-23. The proposed DTRR TS 6.8.1, "The following records shall be kept for a minimum period of five years or for the life of the component involved if less than five years," Item 7, states, in part, "; and" which appears to be a typographical error. Provide justification for the proposed TS 6.8.1, Item 7, or revise to remove the typographical error.

DTRR response:
TS 6.8.1 has been revised by removing the phrase "and" from the end of item 7 in the DTRR TS (April 2014).

RAI-24. The proposed DTRR TS 6.8.2, “Records to be Retained for at Least One Certification Cycle, states,” in part, that “[r]ecords...shall be retained for at least one complete requalification schedule...” The length of the time for “schedule” is not defined. The guidance in NUREG-1537 suggests the term of a cycle. Provide justification for the proposed TS 6.8.2, or revise to use the term cycle to be consistent with the guidance in NUREG-1537.

DTRR response:
TS 6.8.2 has been revised by replacing the phrase “schedule” with the phrase “cycle” in the DTRR TS (April 2014).

Attachment II

Revised DTRR Technical Specifications, April 2014

Appendix A

To

FACILITY LICENSE NO. R-108

DOCKET NO. 50-264

TECHNICAL SPECIFICATIONS AND BASES
FOR
THE DOW TRIGA RESEARCH REACTOR

April/2014

TECHNICAL SPECIFICATIONS AND BASES FOR THE DOW TRIGA RESEARCH REACTOR

1. INTRODUCTION

1.1 Scope

This document constitutes the Technical Specifications for the Facility License No. 108 as required by 10 CFR 50.36 and supersedes all prior Technical Specifications. This document includes the "Basis" to support the selection and significance of the specifications. Each basis is included for information purposes only. They are not part of the Technical Specifications, and they may not constitute limitations or requirements to which the licensee must adhere, except where they reference the DTRR SAR or a specific Technical Specification.

1.2 Format

These specifications are formatted in conformance to NUREG-1537 and ANSI/ANS15.1-2007 guidance.

1.3 Definitions

ALARA: The ALARA (As Low As Reasonably Achievable) program is a set of procedures which is intended to minimize occupational exposures to ionizing radiation and releases of radioactive materials to the environment.

Audit: An audit is a quantitative examination of records, procedures or other documents after implementation from which appropriate recommendations are made.

Channel: A channel is a combination of sensors, electronic circuits, and output devices connected by the appropriate communications network in order to measure and display the value of a parameter.

Channel Calibration: A channel calibration is an adjustment of a channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment, actuation, alarm, or trip and shall include a Channel Test.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall

include comparison of the channel with other independent channels or systems measuring the same variable.

Channel Test: A channel test is the introduction of a signal into the channel for verification that it is operable.

Control Rod: A control rod is a device fabricated from neutron absorbing material, which is used to establish neutron flux changes and to compensate for routine reactivity changes. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. Types of control rods shall include:

1. **Regulating Rod (Reg. Rod):** The regulating rod is a control rod having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.

2. **Shim/Safety Rod:** A shim/safety rod is a control rod having an electric motor drive and scram capabilities. Its position may be varied manually.

Core Lattice Position: The core lattice position is defined by a particular hole in the top grid plate of the core. It is specified by a letter indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

Excess Reactivity: Excess reactivity is that amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions.

Experiment: An experiment is any device or material, not normally part of the reactor, which is introduced into the reactor for the purpose of exposure to radiation, or any operation which is designed to investigate non-routine reactor characteristics. Specific experiments shall include:

A) Reactivity limits:

1. **Secured Experiment:** A secured experiment is any experiment or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces, which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions;

2. **Unsecured Experiment**: An unsecured experiment is any experiment or component of an experiment that does not meet the definition of a secured experiment; and

3. **Movable Experiment:** A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the core while the reactor is operating.

B) Review criteria:

1. **Routine Experiment:** A routine experiment is an approved experiment which involves operations under conditions which have been extensively examined in the course of the reactor test programs;

2. **Modified Routine Experiment:** Modified routine experiments are experiments which have not been designated as routine experiments or which have not been performed previously, but are similar to routine approved experiments in that the hazards are neither significantly different from nor greater than the hazards of the corresponding routine experiment; and

3. **Special Experiment:** Special experiments are experiments which are neither routine experiments nor modified routine experiments.

Experimental Facilities: Experimental facilities shall include the rotary specimen rack, sample containers replacing fuel elements or dummy fuel elements in the core, pneumatic transfer systems, the central thimble, and any other in-tank irradiation facilities.

Fuel Element: A fuel element is a single TRIGA® fuel rod.

Irradiation: Irradiation shall mean the insertion of any device or material that is not a part of the existing core or experimental facilities into an experimental facility so that the device or material is exposed to radiation available in that experimental facility.

Licensed Area: Rooms 51 (51, 51A, 51AA, 51B) and 52 of building 1602.

Measured Value: The measured value is the value of a parameter as it appears on the output of a channel.

Operable: A system or component shall be considered operable when it is capable of performing its intended function.

Operating: Operating means a component or system is performing its intended function.

Operational Core: An operational core shall be a fuel element core, which operates within the licensed power level and satisfies all the requirements of the Technical Specifications.

Reactivity Worth of an Experiment: The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

Reactor Operating: The reactor is operating whenever it is not secured or shutdown.

Reactor Operator (RO): An individual who is licensed to manipulate the controls of a reactor.

Reactor Safety Systems: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate, automatically or manually, a reactor scram for the primary purpose of protecting the reactor.

Reactor Secured: The reactor is secured when:

1. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material in the reactor to attain criticality under optimum available conditions of moderation and reflection; or,
2. All of the following exist:

    a. The three (3) neutron absorbing control rods are fully inserted as required by technical specifications,

    b. The reactor is shutdown,

    c. The console key switch is in the "off" position and the key is removed from the console,

    d. No experiments are being moved or serviced that have, on movement, reactivity worth exceeding the maximum value allowed for a single experiment, and

    e. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.

Reactor Shutdown: The reactor is shutdown when it is subcritical by at least one dollar both in the reference core condition and for all allowed ambient conditions, with the reactivity worth of all installed experiments and irradiation facilities included.

Reference Core Condition: The reference core condition is the condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible.

Review: A review is a qualitative examination of records, procedures or other documents prior to implementation from which appropriate recommendations are made.

Safety Channel: A safety channel is a measuring channel in the reactor safety system.

Scram Time: Scram time is the elapsed time from the initiation of a scram signal to the time the slowest scrammable control rod is fully inserted.

Senior Reactor Operator (SRO): An individual who is licensed to direct the activities of ROs. Such an individual is also an RO.

Should, Shall, and May: The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

Shutdown Margin: Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems and will remain subcritical without further operator action, starting from any permissible operating condition with the most reactive rod in its most reactive position.

Surveillance Intervals: Allowable surveillance intervals shall not exceed the following:

1. Quinquennial – interval not to exceed 70 months
2. Biennial - interval not to exceed 30 months
3. Annual - interval not to exceed 15 months
4. Semiannual - interval not to exceed 7.5 months
5. Quarterly - interval not to exceed 4 months
6. Monthly - interval not to exceed 6 weeks
7. Weekly - interval not to exceed 10 days

Unscheduled Shutdown: An unscheduled shutdown is any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

2. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit – Fuel Element Temperature

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

The objective of this specification is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specification

The temperature in any fuel element in the Dow TRIGA Research Reactor shall not exceed 500 °C under any condition of operation.

Basis

The important parameter for a TRIGA reactor is the fuel element temperature. If the fuel temperature exceeds the safety limit, a loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding. Since the pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of fuel moderator, the magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium ratio in the alloy.

According to several reports on Training Reactor and Isotope Production, General Atomics (TRIGA)-type fuels (NUREG-1282; Simnad et al., 1976 and 1981; Simnad and West et al., 1986; West et al., 1986), for stainless steel-clad $UZrH_{1.65}$ LEU 8.5 w% TRIGA fuel, GA has shown and U.S. Nuclear Regulatory Commission (U.S. NRC) has accepted that the integrity of the fuel is not compromised if the peak fuel temperature is less than 1150 °C. For aluminum-clad $UZrH_{1.0}$ LEU 8 w% TRIGA fuel, the U.S. NRC has accepted that the peak fuel temperature should not exceed 500 °C (NUREG 1537, Appendix 14.1).

2.2 Limiting Safety System Settings

Applicability

This specification applies to the reactor scram setting which prevents the reactor fuel temperature from reaching the safety limit.

Objective

The objective of this specification is to prevent the safety limit from being reached.

Specification

The LSSS shall not exceed 300 kW as measured by the calibrated power channels.

Basis

Analysis of the Dow TRIGA Research Reactor (DTRR) at 300 kW resulted in a maximum fuel temperature of less than 350 °C following a loss of coolant after infinite hours of operation. Therefore setting the LSSS not to exceed 300 kW provides assurance that the safety limit of 500 °C will not be exceeded (SAR, as supplemented by letter dated December 6, 2011).

3. LIMITING CONDITIONS FOR OPERATON (LCO)

3.1 Reactivity Limits

Applicability

These specifications shall apply to the reactor at all times that it is in operation.

Objective

The purpose of the specification is to ensure that the reactor can be controlled and shutdown at all times.

Specifications

1. The shutdown margin provided by the control rods shall be more than $0.50 with:

    a. Irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state;

    b. The most reactive control rod fully withdrawn; and

    c. The reactor in the reference core condition.

2. The excess reactivity measured at less than 10 watts in the reference core condition, with experiments in their most reactive state, shall not be greater than $3.00.

3. Positive reactivity insertion rate by control rod motion shall not exceed $.20 per second.

4. There shall be a minimum of three operable control rods in the reactor core. A control rod is considered operable when:

    a. There is no damage to the control rod or drive assembly; and

    b. The scram time meets the requirement in Technical Specification 3.3, specification c.

Bases

The value of the minimum shutdown margin assures that the reactor can be safely shutdown with the most reactive control rod withdrawn. Assigning specifications to the maximum excess reactivity and maximum reactivity insertion rates, serve as additional restrictions on the shutdown margin and limits the maximum power excursion that could

take place in the event of failure of all of the power level safety circuits and administrative controls. The requirement for three operable control rods ensures that the reactor can meet the shutdown specifications (SAR, as supplemented, by letter dated December 6, 2011).

3.2 Reactor Fuel Parameters Limits

Applicability

This specification applies to all the fuel elements.

Objective

The objective of this specification is to maintain integrity of the fuel elements.

Specification

The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and must be removed from the core if:

a. The transverse bend exceeds 0.0625 inches (0.159 cm) for stainless steel-clad $UZrH_{1.65}$ and 0.125 inches (0.318 cm) for aluminum-clad $UZrH_{1.0}$ over the length of the cladding;

b. Elongation exceeds 0.125 inches (0.318 cm) for stainless steel-clad $UZrH_{1.65}$ and 0.5 inches (1.27 cm) for aluminum-clad $UZrH_{1.0;}$

c. A clad defect exists as indicated by release of fission products; or

d. U-235 Burn-up exceeds 50% initial concentration.

Bases

Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation, bend, and burn-up limits are values that have been found acceptable to the U.S. NRC (NUREG-1537).

3.3 Reactor Control Rods and Safety Systems and Interlocks

Applicability

These specifications apply to the reactor control rods, safety system channels and interlocks.

Objective

The objective is to specify the minimum number of reactor control rods, the safety system channels and interlocks that shall be available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless:

a. The safety channels and the interlocks listed in Table 3.3A are operable;

b. The measuring channels listed in Table 3.3B are operable; and

c. The Scram Time for each of the three control rods shall not exceed one second.

TABLE 3.3A Specifications		
Minimum Reactor Safety Channels, Interlocks, and Set Points		
Scram Channels or Interlocks	**Minimum Operable**	**Scram Set Point or Interlocks**
Reactor Power Level (NM1000 & NPP1000)[1]	2	Not to exceed licensed power level (300kW)
Detector High Voltage (NPP1000)	1	Loss of the High Voltage
Detector High Voltage (NM1000)	1	Loss of the High Voltage
Manual Console Scram	1	Push Button
Watchdog (DAC to CSC) Communication Conflict	1	Not more than 10 sec delay
Startup Count Rate (Interlock)	1	Prevents control rod withdrawal when the neutron count rate is less than 2 cps
Rod Drive Control (Interlock)	1	Prevents simultaneous manual withdrawal of two control rods
Reactor Period (Interlock)	1	Prevents control rod withdrawal when the period is less than 3 seconds

Note: Bypassing of channels and interlocks in this table is not permitted.

[1] Any single power level channel may be inoperable while the reactor is operating for the purpose of performing a channel check, channel test, or channel calibration.

TABLE 3.3B Specifications	
Measuring Channel [2]	**Minimum Number Operable**
NM1000	1
NPP1000	1
Reactor Pool Water Radioactivity Monitor	1
Reactor Pool Water Temperature Monitor	1
Reactor Pool Water Level	1

[2] If any required measuring channel becomes inoperable while the reactor is operating, for reasons other than identified in this TS, the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

Bases

NUREG-1537 recommends at least two independent power level scram channels that provide diversity. This is accomplished by having one power level scram channel as an analog channel. The control rod scram time specification on the three control rods assures that the reactor can be shutdown promptly when a scram signal is initiated and that the reactor can meet the shutdown specifications (SAR, as supplemented, by letter dated December 6, 2011).

Uses of the specified reactor safety channels, set points, and interlocks given in table 3.3A assure protection against operation of the reactor outside the safety limit. The requirements for the specified measurement channels in table 3.3B provide assurance that important reactor operation parameters (power level, water radioactivity, water temperature, and water level) can be monitored during operation. The specification of maximum positive reactivity insertion rate helps assure that the Safety Limit is not exceeded (Dow SAR, as supplemented by U.S. NRC letter dated December 6, 2011).

For footnote 1, taking a single measuring channel off-line for a short duration for the purpose of a channel check, test or calibration is considered acceptable because in some cases, the reactor must be operating in order to perform the channel check, calibration or test. The redundant power level channel provides the scram function for the short period that the other power level channel is out of service. For footnote 2, events which lead to these circumstances are self-revealing to the operator.

3.4 Reactor Coolant Systems

Applicability

These specifications apply to the quality of the coolant in contact with the fuel cladding, to the level of the coolant in the pool, and to the bulk temperature of the coolant.

Objectives

The objectives of this specification include minimization of corrosion of the cladding of the fuel elements and neutron activation of dissolved materials, detection of releases of radioactive materials into the coolant before such releases become significant, ensuring the presence of an adequate quantity of cooling and shielding water in the pool, and prevention of the thermal degradation of the ion exchange resin in the purification system.

Specifications

1. The conductivity of the pool water shall not exceed 5 μmho/cm averaged over one month.

2. The pool water pH shall be in the range of 5.0 to 7.5.

3. The radioactivity of the reactor pool water shall not exceed the limits of 10 CFR 20 Appendix B Table 2 column 2 for radioisotopes with half-lives > 24 hours.

4. The water shall cover the core of the reactor to a minimum of 15 feet above the core during operation of the reactor.

5. The bulk temperature of the coolant shall not exceed 60 °C during operation of the reactor.

6. There shall be an audible alarm on the coolant level set at 15 ft 10 in above the core.

Bases

Increased levels of conductivity in aqueous systems indicate the presence of corrosion products and promote more corrosion. Experience with water quality control at many reactor facilities, including operation of the Dow TRIGA Research Reactor since 1967, has shown that maintenance within the specified limit provides acceptable corrosion control.

Maintaining low levels of dissolved electrolytes in the pool water also reduces the amount of induced radioactivity. Low levels of dissolved electrolytes in the pool water decrease the exposure of personnel to ionizing radiation during operation and maintenance. The pool water conductivity is monitored continually, except during maintenance.

Monitoring the pH of the pool water provides early detection of extreme values of pH which could enhance corrosion. Limiting the radioactivity to this level will help ensure that any disposal of pool water, either planned or inadvertent, will be within the limits of 10 CFR 20. This specification also provides verification of absence of fission product leakage.

Maintaining the specified depth of water in the pool provides shielding of the radioactive core which reduces the exposure of personnel to ionizing radiation in accordance with the ALARA program. This specification also maintains the height of water above the core used in thermal-hydraulic analyses (SAR, as supplemented by letter dated December 6, 2011).

Maintaining the bulk temperature of the coolant below the specified limit assures minimal thermal degradation of the ion exchange resin. This specification is consistent with the thermal-hydraulic analyses (SAR, as supplemented by letter dated December 6, 2011).

The alarm is audible in the control room as well as outside of the control room, and it alerts operating staff and other people in the building, when the coolant water level is low, to take appropriate action.

3.5 Ventilation

Applicability

This specification applies to the reactor room ventilation.

Objective

The objective of this specification is to mitigate the consequences of possible release of radioactive materials to unrestricted areas.

Specification

The ventilation system shall be operating whenever the reactor is operated, fuel is manipulated, any core or control rod work that can change reactivity by more than $1, or radioactive materials with the potential of airborne releases are handled in the reactor room. The ventilation system is considered operable if:

a. The exhaust and the inlet fans are operating;

b. The external door (Door 10) is closed; and

c. The exhaust louvers are open.

Basis

This specification ensures that the ventilation is operating and configured to control any releases of radioactive material during fuel handling, reactor operation, or the handling of possible airborne radioactive material in the reactor room.

3.6 Radiation Monitoring Systems and Effluents

3.6.1 Radiation Monitoring Systems

Applicability

These specifications apply to the radiation monitoring systems.

Objective

The objective is to specify the minimum radiation monitoring channels that shall be available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the minimum number of each radiation monitoring channel, listed in Table 3.6, are operating.

Table 3.6	
Radiation Measuring Channels	**Number**
Continuous Air Monitor (CAM)	1
Area Radiation Monitor (ARM)[1]	1
Environmental Monitor (Film badges)	4

[1]When the area radiation monitor channel becomes inoperable, operations may continue only if a portable gamma-sensitive ion chamber is utilized as a temporary substitute, provided that the substitute can be observed by the reactor operator, can be installed within 1 hour of discovery, and not to exceed 60 days.

Bases

The CAM provides information of existing levels of radiation and air-borne radioactive materials, including particulate alpha or beta emitters, which could endanger operating personnel or which could warn of possible malfunctions of the reactor or the experiments in the reactor.

The ARM provides information of existing levels of radiation and airborne radioactive materials which could endanger operating personnel or which could warn of possible malfunctions of the reactor or the experiments in the reactor.

The film badges or any other environmental monitors, placed in the reactor room provide historical records of radiation exposures in the reactor room. One of the four film badges is placed in the control room. Experience at the DTRR showed that the 4 film badges are adequate for monitoring environmental radiation exposures.

For footnote 1, an analysis has shown that it takes more than 60 minutes for the radiation level in the reactor room to exceed the alarm set level of 2mR/hr (SAR, as supplemented by letter dated January 20, 2012, RAI No. 56). Therefore substituting an observable ion chamber within 1 hr assures that the reactor operator has the tool to observe actionable radiation level in the reactor room.

3.6.2 Effluents

Applicability

This specification applies to the release rate of ^{41}Ar.

Objective

The objective is to ensure that the concentration of ^{41}Ar in the unrestricted areas shall be below the applicable effluent concentration value in 10 CFR 20.

Specification

The annual average concentration of ^{41}Ar discharged into the unrestricted area shall not exceed 1×10^{-9} μCi/ml.

Bases

Operational experience at DTRR shows that the use of the Rabbit System, an irradiation facility, is the main contributor to the release of ^{41}Ar. An analysis has shown that continuous operation of the Rabbit System will result in an annual release of 7.94×10^{-12} μCi/ml, with a corresponding annual TEDE to a member of the public in an unrestricted area of 0.056 mrem/yr (SAR, as supplemented by letter dated June 11, 2012, response to RAI No. 41). The TS value of 1×10^{-9} μCi/ml, which corresponds to a dose of 7.1 mrem/yr at the nearest unrestricted area release point, results in a TEDE that is 30% lower than the 10 CFR 20.1101(d) ALARA goal of 10 mrem/yr TEDE, and is therefore sufficient to meet the objective of this specification. The amount of ^{41}Ar effluent discharged, every year, will be calculated based on the actual number of hours the Rabbit System was operated and the result will be provided in the annual report.

3.7 Experiments

3.7.1 Reactivity and Position Limits

Applicability

These specifications apply to experiments installed in the reactor and the irradiation facilities.

Objective

The objective of these specifications is to prevent damage to the reactor or excessive release of radioactive materials in case of failure of an experiment.

Specification

The reactor shall not be operated unless the following conditions governing experiments exist:

a. The sum of the total absolute value of reactivity worths of all experiments shall not exceed $1.00;

b. Experiments having an absolute reactivity worth greater than $0.75 shall be securely located or fastened to prevent inadvertent movement during reactor operation; and

c. Experiments shall not occupy adjacent fuel element positions in the B- and C-rings fuel locations.

Bases

These specifications are intended to limit the reactivity of the system so that the safety limit would not be exceeded even if the experiment were in the B-ring or C-ring, or if the contribution to the total reactivity by the experiment reactivity should be suddenly moved.

The reactivity worth limit of $1.00 for all experiments is intended to prevent the reactor from becoming prompt critical during experiments.

The reactivity limit of $0.75 for movable experiments is designed to prevent an inadvertent reactor pulse from occurring and maintain a reactivity value below the shutdown margin. The specification for the unsecured experiment ($0.75) is consistent with the reactivity insertion behavior analyses (SAR, as supplemented by letter dated December 6, 2011).

The prevention of experiments from occupying adjacent fuel locations in the B and C rings helps to limit the power excursions that may arise due to experiments and to be able to control the reactor within the limits imposed by the license.

3.7.2 Materials

Applicability

These specifications apply to experiments installed in the reactor and the irradiation facilities.

Objective

The objective of these specifications is to prevent damage to the reactor or excessive release of radioactive materials in case of failure of an experiment.

Specification

The reactor shall not be operated unless the following conditions governing experiments exist:

a. Experiments containing corrosive materials shall be doubly encapsulated. The failure of an encapsulation of material that could damage the reactor shall result in removal of the sample and physical inspection of potentially damaged components;

b. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 mg TNT equivalent shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities up to 25 mg TNT equivalent may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half of the design pressure of the container.

Basis

This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials. Operation of the reactor with the reactor fuel or structure potential damages is prohibited to avoid potential release of fission products.

3.7.3 Experiment Failure and Malfunctions

Applicability

These specifications apply to experiments installed in the reactor and the irradiation facilities.

Objective

The objective of these specifications is to prevent damage to the reactor or excessive release of radioactive materials in case of failure of an experiment.

Specification

Where the possibility exists that the failure of an experiment under (a) normal operating conditions of the experiment or the reactor, (b) credible accident conditions in the reactor or (c) possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor room or the unrestricted area, the quantity and type of material in the experiment shall be limited in activity such that exposures of the reactor personnel to the gaseous activity or radioactive aerosols in the reactor room or control room will not exceed the occupational dose limits in 10CFR 20.1201. Additionally, exposures to members of the public to these releases in the unrestricted areas will not exceed the dose limits in 10CFR 20.1301, assuming that:

a. 100% of the gases or aerosols escape from the experiment;

b. If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation levels, the assumption shall be used that 10% of the gaseous activity or aerosols produced will escape;

c. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, the assumption shall be used that 10% of the aerosols produced escape;

d. For materials whose boiling point is above 55 °C and where vapors formed by boiling this material could escape only through an undisturbed column of water above the core, the assumption shall be used that 10% of these vapors escape; and

e. If an experiment container fails and releases material which could damage the reactor fuel or structure by corrosion or other means, physical inspection shall be performed to determine the consequences and the need for corrective action.

Bases

This specification is intended to ensure that released airborne radioactivity to the reactor room or unrestricted area surrounding the DTRR will not result in exceeding the total dose limits to a worker or a member of the public as specified in 10CFR 20.1201 and 10CFR 20.1301.

4. SURVEILLANCE RQUIREMENTS

4.0 General

Applicability

This specification applies to surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the operability of any system related to reactor safety.

Specifications

1. Surveillance requirements may be deferred during reactor shutdown (except TS 4.4, items 1 and 2, TS 4.6 item 2); however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.

2. Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications reviewed by the Reactor Operations Committee. A system shall not be considered operable until after it is successfully tested.

3. Required surveillances of the reactor control and safety systems, pool water level alarm and radiation monitoring systems shall be completed after maintenance of the respective items.

Basis

These specifications relate to changes in reactor systems which could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria.

4.1 Reactor Core Parameters

Applicability

These specifications apply to surveillance requirements for the reactor core parameters.

Objective

The objective of these specifications is to ensure that the specifications of section 3.1 are satisfied.

Specification

The reactivity worth of each control rod, the reactor core excess reactivity, and the reactor shutdown margin shall be measured at least annually and after each time the core fuel is moved or following any change of reactivity greater than $0.25 from a reference core.

Basis

Movement of the core fuel could change the reactivity of the core and thus affect both the core excess reactivity and the shutdown margin, as well as affecting the worth of the individual control rods. Evaluation of these parameters is therefore required after any such movement. Without any such movement, the changes of these parameters over an extended period of time and operation of the reactor have been shown to be very small so that an annual measurement is sufficient to ensure compliance with the specifications of section 3.1. The value of reactivity change (>$0.25) is measurable and will ensure adequate coverage of the shutdown margin after taking into account the accumulation of poisons.

4.2 Reactor Fuel Parameters

Applicability

This specification shall apply to the fuel elements of the Dow TRIGA Research Reactor.

Objective

The objective of this specification is to ensure that the reactor is not operated with damaged fuel elements.

Specification

Each fuel element shall be examined visually and for changes in transverse bend and length at least once each five years, with at least 20 percent of the fuel elements examined each year. If a damaged fuel element is identified, the entire inventory of fuel elements shall be inspected prior to subsequent operations.

Basis

Visual examination of the fuel elements allows early detection of signs of deterioration of the fuel elements, indicated by signs of changes of corrosion patterns or of swelling, bending, or elongation. Experience at the Dow TRIGA Research reactor and at other TRIGA reactors indicates that examination of a five-year cycle is adequate to detect problems, especially in TRIGA reactors that are not pulsed. A five-year cycle reduces the handling of the fuel elements and thus reduces the risk of accident or damage due to handling.

4.3 Control and Safety Systems

Applicability

These specifications apply to the surveillance requirements of the reactor safety systems.

Objective

The objective of these specifications is to ensure the operability of the reactor safety systems as described in Technical Specification 3.3.

Specifications

1. Control rod scram times shall be measured and reactivity insertion rates shall be calculated annually and whenever maintenance is performed or repairs are made that could affect the rods or control rod drives.

2. A channel calibration shall be performed for the NM1000 and NPP1000 power level channels by thermal power calibration annually.

3. A channel test shall be performed each day the reactor is operated and after any maintenance or repair for each of the six scram channels and each of the three interlocks listed in Table 3.3A.

4. The control rods shall be visually inspected for damage or deterioration biennially.

Bases

Measurement of the control rod scram time and compliance with the specification indicates that the control rods can perform the safety function properly. Measurement of the control rod withdrawal speed ensures that the maximum reactivity addition rate specification will not be exceeded.

Variations of the indicated power level due to minor variations of either of the two neutron detectors would be readily evident during day-to-day operation. The specification for thermal calibration of the NM1000 and NPP1000 channels provide assurance that long-term drift of both neutron detectors would be detected and that the reactor shall be operated within the authorized power range.

The channel tests performed daily before operation and after any repair or maintenance provide timely assurance that the systems will operate properly during operation of the reactor.

Visual inspection of the control rods provides opportunity to evaluate any corrosion, distortion, or damage that might occur in time to avoid malfunction of the control rods. Experience at the Dow TRIGA Reactor Facility since 1967 indicates that the surveillance specification is adequate to assure proper operation of the control rods. This surveillance complements the rod scram time measurements.

4.4 Reactor Coolant Systems

Applicability

These specifications shall apply to the surveillance requirements for the reactor coolant system.

Objective

The objective of these specifications is to ensure that the specifications of section 3.4 are satisfied.

Specifications

1. The conductivity, pH, and the radioactivity of the pool water shall be measured at least monthly.

2. A channel check of the pool water level shall be done weekly and before commencement of each day of operation.

3. A channel check of the temperature monitor shall be done during reactor operation and a channel test of the temperature monitor shall be done monthly. A channel calibration is performed as recommended by the manufacturer or if required as a result of the channel test.

4. A channel test of the pool water level alarm shall be done annually.

5. A channel check of the pool water radioactivity monitor shall be done during reactor operation and a channel test of the pool water radioactivity monitor shall be done semiannually. A channel calibration is performed as recommended by the manufacturer or if required as a result of the channel test.

Bases

Experience at the Dow TRIGA Research Reactor showed that these surveillance specifications on the conductivity, pH, and radioactivity are adequate to detect the onset of degradation of the quality of the pool water in a timely fashion. Evaluation of the radioactivity in the pool water allows the detection of fission product releases from damaged fuel elements or damaged experiments.

Experience also indicates that the surveillance specification on pool water level and pool water temperature are adequate to detect losses of pool water in a timely manner and to enable operators to take appropriate action when the coolant temperature approaches the

specified limit. The monthly test of the temperature monitor is also necessary to assure operability of the temperature channel.

The pool water level alarm system is a robust unit and therefore the specification of an annual test is sufficient to assure operability of the pool water level alarm.

The pool water radioactivity is monitored continuously, except when the unit is being repaired. Making a channel check during the reactor operation is sufficient to establish that the unit is operating. Experience shows that a semi-annual test is also sufficient to assure that the channel is operating properly.

4.5 Ventilation

Applicability

This specification applies to the surveillance of the ventilation system

Objective

The objective of these specifications is to ensure that the Technical Specification 3.5 is satisfied.

Specification

A channel check of the ventilation system shall be performed prior to each day's operation, prior to fuel manipulation, or prior to handling radioactive materials with the potential of airborne releases in the reactor room.

Basis

Experience has demonstrated that checks of the ventilation system on the prescribed basis are sufficient to assure proper operation of the system and its control over releases of radioactive material in the reactor room.

4.6 Radiation Monitoring Systems

Applicability

These specifications apply to the surveillance requirements for the Continuous Air Monitor (CAM) and the Area Radiation Monitor (ARM), both located in the reactor room.

Objective

The objective of these specifications is to ensure the quality of the data presented by these two instruments.

Specifications

1. A channel check shall be made for the CAM and the ARM before commencement of each day of operation, prior to manipulating fuel, or handling experiments or radioactive material which have a potential to become airborne.

2. A channel test shall be made for the CAM and the ARM at least weekly.

3. A channel calibration shall be made for the CAM and the ARM at least annually.

4. The environmental monitors shall be changed and evaluated at least semi-annually.

Bases

The specifications on the CAM and ARM ensure that they can perform the required functions and that deterioration of the instruments shall be detected in a timely manner when the reactor is operating, prior to manipulating fuel, or handling experiments or radioactive material which have a potential to become airborne. Experience with these instruments has shown that the surveillance intervals are adequate to provide the required assurance.

The continuous air monitor is further checked at least weekly, even if the reactor was not operating to ensure that it is performing its required function.

The frequency of changing and evaluating environmental monitors are also adequate to provide the required record based on past experience with these monitors.

4.7 Experiments

Applicability

This specification applies to the surveillance of the experiments.

Objective

The objective of these specifications is to ensure that the Technical Specification 3.7 is satisfied.

Specifications

1. The reactivity worth of an experiment shall be estimated or measured, as appropriate before reactor operation with said experiment.

2. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been reviewed and approved for compliance with Technical Specification 3.7 by the ROC.

3. ROC approved experiments shall be reviewed prior to irradiation by the Director or a designee.

4. Dose rate on contact for each sample shall be recorded when removed from the experimental facility.

Basis

Experience has shown that these specifications verify that experiments can be conducted without endangering the safety of the reactor or exceeding the limits in the technical specifications.

5. DESIGN FEATURES

5.1 Reactor Site and Building

Applicability

These specifications shall apply to the Dow TRIGA Research Reactor licensed area.

Objectives

The objectives of these specifications are to define the licensed area and characteristics of the reactor area.

Specifications

1. The minimum distance from the center of the reactor pool to the boundary of the restricted area shall be 75 feet.

2. The reactor shall be housed in a room with a minimum of 6000 cubic feet volume designed to restrict leakage.

3. All air or other gas exhausted from the reactor room and from associated experimental facilities during reactor operation shall be released to the environment at a minimum of 8 feet above ground level.

4. Emergency shutdown controls for the ventilation systems shall be located in the reactor control room.

5. The licensed area includes rooms 51 (51, 51A, 51AA, 51B) and 52 of building 1602.

Bases

The minimum distance from the pool to the boundary provides for dilution of effluents and for control of access to the reactor area.

Restriction of leakage, in the event of a release of radioactive materials, can contain the materials and reduce exposure of the public.

Release of gases at a minimum height of 8 feet reduces the possibility of exposure of personnel to such gases.

The location of emergency ventilation shutdown controls in the control room assures quick and easy access to these controls by the operator.

5.2 Reactor Coolant System

Applicability

This specification applies to the Dow TRIGA Research Reactor.

Objective

The objective of this specification is to define the characteristics of the cooling system of the reactor.

Specifications

1. The reactor core shall be cooled by natural convective water flow.

2. The water lines from the pool water to the heat exchanger shall have anti-siphon holes.

Basis

Natural convention water flow has been demonstrated to provide sufficient cooling during reactor operations (SAR, as supplemented by letter dated December 6, 2011).

Anti-siphon holes prevent siphoning of the water out of the pool, should leaks develop in the water lines.

5.3 Reactor Core and Fuel

Applicability

These specifications shall be applicable to the Dow TRIGA Research Reactor.

Objective

The objective of these specifications is to define certain characteristics of the reactor in order to assure that the design and accident analyses shall be correct.

Specifications

1. The critical core shall be an assembly of stainless-steel or aluminum-clad TRIGA fuel elements in light water.

2. The fuel shall be arranged in a close packed array for operation at full licensed power except for replacement of single individual fuel elements with in-core irradiation facilities or control rod guide tubes, or the start-up neutron source.

3. The aluminum-clad fuel elements shall be placed in the E or F ring of the core.

4. The control rods (Shim1, Shim2 and Regulating rod) shall have scram capability and shall contain borated graphite, boron carbide powder, or boron and its components in solid form as a poison in an aluminum or stainless steel cladding.

5. The reflector (excluding experiments and experimental facilities) shall be a combination of graphite and water.

6. The structural components of the core shall be limited to aluminum or stainless steel.

7. No fuel shall be inserted or removed from the core unless the reactor is subcritical by more than the worth of the most reactive fuel element.

8. No control rods shall be manually removed from the core for inspection unless it has been shown that the core is subcritical with all control rods fully withdrawn from the core.

Bases

The entire design and accident analysis is based upon the characteristics of TRIGA fuel. Any other material would invalidate the findings of these analyses.

Operation with standard U.S. NRC-approved TRIGA fuel in closed packed array ensures a conservative limitation with respect to the safety limit.

Placement of the aluminum-clad fuel element in the outer rings of the reactor core will help ensure that this element is not exposed to higher than average power levels, thus providing a greater degree of conservatism with respect to the Safety Limit for an aluminum-clad fuel element.

The control rods perform their function through the absorption of neutrons, thus affecting the reactivity of the system.

Boron has been found to be a stable and effective material for this control.

The reflector serves to conserve neutrons and to reduce the amount of fuel that shall be in the core to maintain the chain reaction.

The required conditions prior to any fuel movements ensure that an inadvertent criticality will not occur.

The required conditions prior to any control rod movements ensure that an inadvertent criticality will not occur.

5.4 Fuel Storage

Applicability

This specification applies to the Dow TRIGA Research Reactor fuel storage facilities.

Objective

The objective of this specification is the safe storage of fuel.

Specifications

1. All fuel and fueled devices not in the core of the reactor shall be stored in such a way that k_{eff} shall be less than 0.9 under all conditions of moderation, and that will permit sufficient cooling by natural convection of water or air such that temperatures shall not exceed the safety limit.

2. Fuel storage shall be limited to in pool storage only.

Basis

A value of k_{eff} of less than 0.9 precludes any possibility of inadvertent establishment of a self-sustaining nuclear chain reaction. Cooling, which maintains temperatures lower than the safety limit, prevents possible damage to the fuel elements which has a potential to release radioactive materials.

Limiting fuel storage to in-pool storage only further assures safe storage and is a practice that has been found acceptable to the U.S. NRC (NUREG-1537).

6. ADMINISTRATIVE CONTROLS

6.1 Organization

Individuals at the various management levels, in addition to being responsible for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, technical specifications, and federal regulations.

6.1.1 Structure

The reactor administration shall be related to the Core Research and Development (R&D) of the Dow Chemical Company, Midland, as shown in Figure 6.1.

6.1.2 Responsibility

The following specific organizational levels and responsibilities shall exist:

a. Dow Core R&D Director, Analytical Sciences (Level 1): The Dow Core R&D Director for Analytical Sciences is responsible for the Dow TRIGA Research Reactor's license;

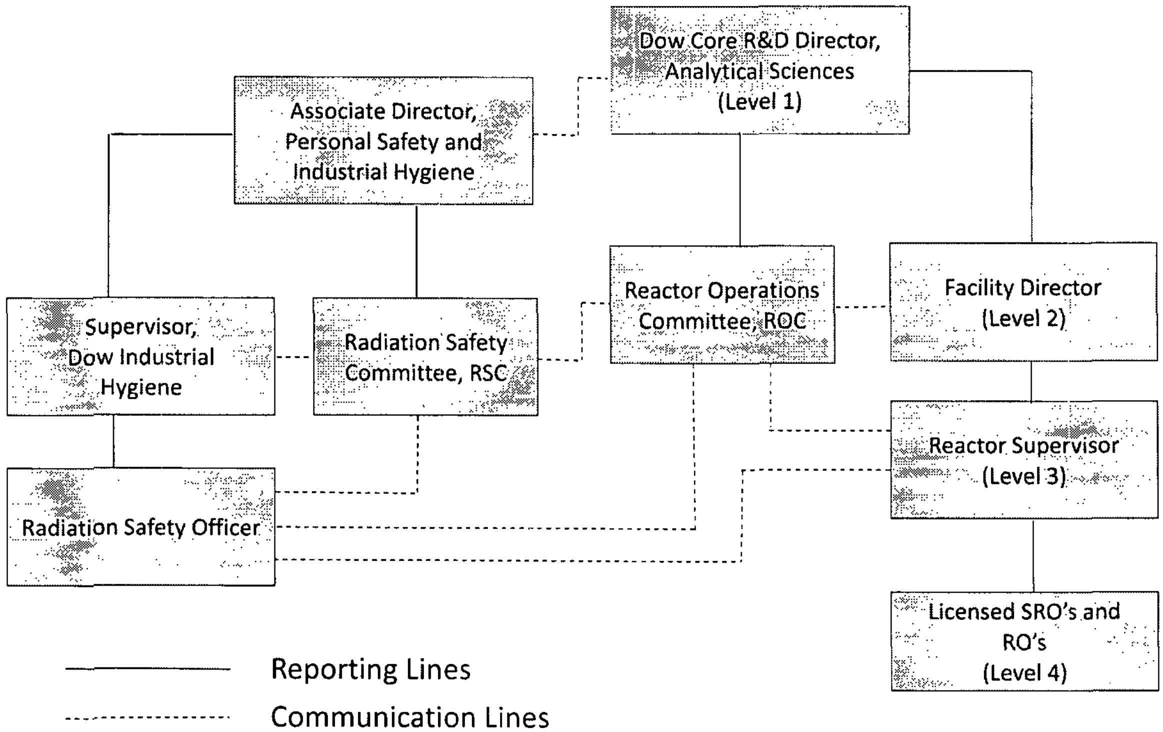
b. Dow TRIGA Research Reactor (DTRR) Director (Level 2): The DTRR Director reports to the Dow Core R&D Director, Analytical Sciences, and is accountable for the facility's operation;

c. Reactor Supervisor (Level 3): The Reactor Supervisor, who must be an SRO, reports to the DTRR Director and is responsible for directing the activities of the reactor operators and the senior operators (including training, emergency, security and requalification programs) and for the day-to-day operations and maintenance of the reactor;

d. Radiation Safety Officer, RSO, (Level 3): The RSO reports to the Supervisor, The Dow Industrial Hygiene Expertise Center, and is responsible for directing the activities of health physics personnel including implementation of the radiation safety program; and

e. Reactor Operator and Senior Reactor Operator (Level 4): The Reactor Operators and Senior Reactor Operators report to the Reactor Supervisor and are primarily involved in the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor related equipment.

Figure 6.1. Administration



6.1.3 Staffing

1. The minimum staffing when the reactor is not secured shall be:
    a. A licensed Reactor Operator or the Reactor Supervisor in the control room;
    b. A second person present in the 1602 Building able to carry out prescribed instructions; and
    c. If neither of these two individuals is the Reactor Supervisor, the Reactor Supervisor shall be readily available on call. Readily available on call means an individual who:
        I. Has been specifically designated and the designation is known to the operator on duty,
        II. Can be rapidly contacted by phone by the operator on duty, and
        III. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).
2. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:
    a. DTRR Director;
    b. Reactor Supervisor;
    c. Radiation Safety Officer; and
    d. Any Licensed Reactor Operator or Senior Reactor Operator.
3. Events requiring the direction of the Reactor Supervisor:
    a. Initial startup and approach to power of the day;
    b. All fuel or control rod relocations and maintenance within the reactor core region;
    c. Relocation of any in-core experiment or irradiation facility with a reactivity worth greater than $0.75; and
    d. Recovery from unplanned or unscheduled shutdown or unscheduled power reduction.

6.1.4 Selection and Training of Personnel

The Reactor Supervisor shall be responsible for the training and requalification of the facility Reactor Operators and Senior Reactor Operators.

The selection, training and requalification of operations personnel should be in accordance with ANSI/ANS 15.4 – 1988; R1999, "Standard for the Selection and Training of Personnel for Research Reactors."

6.2 Review and Audit

The review and audit functions shall be the responsibility of the Reactor Operations Committee (ROC).

6.2.1 Composition and Qualification

The ROC shall consist of at least four members who are knowledgeable in fields which relate to engineering and nuclear safety. The Dow Core R&D Director, Analytical Sciences, (Level 1) shall be designated the chair of the committee. Other ROC members shall include the following as determined by Level 1: Facility Director (Level 2); the Reactor Supervisor (Level 3); the Radiation Safety Officer; and one or more persons who are competent in the field of reactor operations, radiation science, or reactor/radiation engineering. The ROC shall be appointed by Level 1 management.

6.2.2 ROC Rules

The operations of the ROC shall be in accordance with written procedures including provisions for:

a. Quorums (majority of the members of the ROC, no more than one-half of the voting members present may be of the operating staff (Levels 3 and 4));

b. Meeting frequency (at least annually and as often as required to transact business);

c. Minutes of the meetings (shall be reviewed and approved within a calendar quarter of the meeting and kept as records for the facility);

d. Voting rules (Members of the ROC may be polled by telephone or email for guidance and approvals); and

e. Communications (the ROC shall communicate, at least twice per year to the Radiation Safety Committee (RSC) through presentations by the reactor supervisor at the quarterly RSC meetings). The presentations are documented as part of the RSC meeting minutes and are kept as records for the facility.

6.2.3 ROC Review Function

The ROC shall perform the following reviews:

a. Review all changes made in the facility;

b. Review of all new procedures and changes to existing procedures;

c. Review of proposed changes to the technical specifications or license;

d. Review of violations of technical specifications, license, or violations of internal procedures or instructions having safety significance;

e. Review of operating abnormalities having safety significance;

f. Review of all events from reports required by Technical Specifications 6.6.1 and 6.7.2; and

g. Review of audit reports.

6.2.4 ROC Audit Function

The ROC shall audit reactor operations at least annually. The annual audit shall include at least, the following:

a. facility operations for conformance to the technical specifications and applicable license conditions;

b. the retraining and requalification program for the operating staff;

c. the results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety;

d. the emergency response plan and implementation procedures;

e. the audit shall be performed by one or more persons appointed by the ROC. At least one of the auditors shall be familiar with reactor operations. No person directly responsible for any portion of the operation of the facility shall audit that operation; and

f. Any deficiencies that may affect reactor safety shall be immediately reported to ROC Chair, Level 1, and a written full report of the audit shall be submitted to the ROC within three months of the audit.

6.3 Radiation Safety

The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the Dow TRIGA Research Reactor. The requirements of the radiation safety program are established in 10 CFR 20. The program should use the guidelines of the ANSI/ANS-15.11-1993; R2004, "Radiation Protection at Research Reactor Facilities."

6.4 Procedures

Written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action, should the situation require such. Operating procedures shall be used for the following items:

a. Startup, operation, and shutdown of the reactor;

b. Implementation of required plans such as the emergency plan and security plan;

c. Emergency and abnormal operating events, including facility shutdown;

d. Fuel loading, unloading and movement within the reactor;

e. Maintenance of major components of systems that could have an effect on reactor control and safety.

f. Surveillance checks, tests, calibrations and inspections required by the technical specifications or those that have an effect on reactor safety;

g. Radiation protection;

h. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity; and

i. Use, receipt, and transfer of by-product material held under the reactor license.

6.5 Experimental Review and Approval

Experiments shall be carried out in accordance with 10 CFR 20, 10 CFR 50.59 and the DTRR TS, operating and administrative procedures. Procedures related to experiment review and approval shall include:

a. All modified routine and special experiments shall be reviewed and approved by the Reactor Operations Committee, and approved in writing by the Level 2 or designated alternates prior to initiation; and

b. Changes to any approved experiments shall be made only after review and approval by the Reactor Operations Committee and approved in writing by the Level 2 or designated alternates prior to initiation.

6.6 Required Actions

6.6.1. Actions to Be Taken in Case of Safety Limit Violation

In the event a safety limit (fuel temperature) is exceeded:

a. The reactor shall be shutdown and reactor operation shall not be resumed until authorized by the U.S. NRC;

b. An immediate notification of the occurrence shall be made to the Reactor Supervisor, DTRR Director, Level 1, ROC; and

c. A report, and any applicable follow-up report, shall be prepared and reviewed by the ROC. The report shall describe the following:

   i. Applicable circumstances leading to the violation including, when known, the cause and contributing factors;

   ii. Effects of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and

   iii. Corrective action to be taken to prevent recurrence.

6.6.2. Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2 Other than a Safety Limit Violation

For all events which are required by Technical Specifications to be reported to the U.S. NRC, under Section 6.7.2, except a safety limit violation, the following actions shall be taken:

a. The reactor shall be secured and the Reactor Supervisor and Director notified;

b. Operations shall not resume unless authorized by the Reactor Supervisor and the Director;

c. The Reactor Operations Committee shall review the occurrence at their next scheduled meeting; and

d. A report shall be submitted to the U.S. NRC in accordance with Section 6.7.2 of these Technical Specifications.

6.7 Reports

6.7.1. Annual Operating Reports

An annual report shall be created and submitted, by the Facility Director, to the Document Control Desk U.S. NRC, Washington, DC. by the March 31st of each year. The report shall include the following:

a. Status of the facility staff and licenses;

b. A narrative summary of reactor operating experience, including the energy produced by the reactor, or the hours the reactor was critical, or both;

c. A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;

d. The unscheduled shutdowns and reasons for them including, where applicable, corrective action taken to preclude recurrence;

e. Tabulation of major preventive and corrective maintenance operations having safety significance;

f. A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge (the summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent; if the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, only a statement to this effect is needed);

g. A summary of the radiation exposures received by facility personnel and visitors where such exposures are greater than 25% of those allowed in 10 CFR 20; and

h. A summarized result of any environmental surveys performed outside the facility.

6.7.2. Special Reports

In addition to the requirement of applicable regulations, and in no way substituting therefore, reports shall be made by the Level 1 manager to the U.S. NRC as follows:

1. A report not later than the following working day by telephone and confirmed in writing by facsimile to the U.S. NRC Headquarters Operations Center, and followed by a written report that describes the circumstances of the event within 14 days to the Document Control Desk, U.S. NRC, Washington, DC, 20555 of any of the following:

    a. Violation of the safety limit;

    b. Release of radioactivity from the site above limits;

    c. Operation with actual safety system settings for required systems less conservative than the limiting safety system settings specified in Technical Specification 2.2;

    d. Operation in violation of limiting conditions for operation established in the Technical Specifications;

    e. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is caused by maintenance, then no report is required;

    f. Any unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded;

    g. Abnormal and significant degradation in reactor fuel, cladding, or coolant boundary; or

    h. An observed inadequacy in the implementation of either administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

2. A written report shall be sent, within 30 days, to the Document Control Desk, U.S. NRC, Washington, DC, 20555, of either:

    a. Permanent changes in the facility staff involving the Level 1, 2 and 3 personnel; or

    b. Significant changes in the transient or accident analysis report as described in the Safety Analysis Report.

6.8 Records

6.8.1. The following records shall be kept for a minimum period of five years or for the life of the component involved if less than five years:

1. Normal reactor operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year);

2. Principal maintenance activities;

3. Reportable occurrences;

4. Fuel inventories, receipts, and shipments;

5. ROC meetings and audit reports;

6. Reactor facility radiation and contamination surveys;

7. Surveillance activities as required by the Technical Specifications;

8. Approved changes in the operating procedures; and

9. Experiments performed by the reactor.

6.8.2. Records to be Retained for at Least One Certification Cycle

Records of the retraining and requalification of Reactor Operators and Senior Reactor Operators shall be retained for at least one complete requalification cycle and be maintained at all times the individual is employed or until the certification is renewed. For the purpose of this technical specification, a certification is an NRC issued operator license.

6.8.3. Records to be Retained for the Lifetime of the Reactor Facility

1. Gaseous and liquid radioactive effluents released to the environment;

2. Radiation exposure of all individuals monitored;

3. Offsite environmental monitoring surveys;

4. Drawings of the reactor facility; and

5. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.