The Dow Chemical Company
Midland, Michigan 48667

July 11, 2013

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 June 26, 2013, 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 July 11, 2013

Paul O'Connor

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

Subscribed and sworn to before me this 11TH day of July, 2013

Patricia E. Brown
Notary Public
MIDLAND County, Michigan
My Commission Expires:

2-4-2020

PATRICIA E. BROWN
NOTARY PUBLIC, GLADWIN COUNTY, MI
ACTING IN THE COUNTY OF MIDLAND, MI
MY COMMISSION EXPIRES: FEBRUARY 4, 2020

PATRICIA E. BROWN
NOTARY PUBLIC, GLADWIN COUNTY, MI
ACTING IN THE COUNTY OF MIDLAND, MI
MY COMMISSION EXPIRES: FEBRUARY 4, 2020

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 June 26, 2013: Questions 1 through 22.

Today: 7/11/2013

RAI-1. In the DTRR response to NRC RAI No. 47, solid radioactive waste containing isotopes with half-lives of 15 days or less are stored for 150 days (10 half-lives) prior to disposal (incineration). This decay period reduces the radioactivity to less than 0.01 percent of the original activity. The material is surveyed with a Geiger counter to confirm no detectable activity prior to incineration. However, the detection limit of survey instrumentation was not provided in the RAI response. Provide the types of survey instrumentation used to free release materials. Describe why the survey instrumentation is appropriate for the types of radiation anticipated. Also provide the lower level of detection of your survey instrumentation and techniques. Describe how your process ensures radioactive materials are not inadvertently released to the environment.

DTRR response:

Short-lived radioactive waste consists of isotopes with half-lives of 15 days or less, and are stored for at least 150 days (10 half-lives) prior to release. At the time samples are placed in decay-in-storage, operational experience has shown that activity levels are less than 10 μCi. After 10 half-lives of decay, the activities of these short-lived isotopes will be less than 1×10^{-3} μCi per sample, and will typically be well below this level due to shorter half-lives and lower activities of most isotopes handled. Therefore, the primary purpose of the survey is to identify any long-lived isotopes that were inadvertently created during the irradiation. Samples are stored in small 1-gallon plastic waste jugs for storage during the decay-in-storage process. These waste jugs are surveyed using a handheld GM Survey Meter when they are removed from storage in order to identify those containers that contain significant quantities of residual radioactive material. According to IE Circular No. 81-07: "Control of Radioactively Contaminated Material", the minimum detectable activity of a handheld GM Survey Meter for beta contamination is about 5000dpm (2.3×10^{-3} μCi). Waste jugs that do not pass this screening process are characterized and placed into long-lived radioactive waste. Waste jugs that pass this screening survey are placed for analysis in a gamma spectroscopy system to characterize any residual radioactive material. The gamma spectroscopy system consists of two high purity germanium detectors in a proprietary shielding block. The two detectors provide twice the efficiency and twice the geometrical span of one detector. The detection limit for the gamma spectroscopy system varies by radionuclide and count durations, but a typical detection limit was determined during in a recent analysis for samples of Cs-137, which determined a minimum detectable activity of 0.2 Bq/kg above background.

Detection of any residual reactor-made radioactive material in these containers requires the isotope to be identified, and either held in storage for additional time to allow it to fully decay to background, or placed in long-lived radioactive waste storage for disposal. This process ensures radioactive materials are not inadvertently released to the environment.

RAI-2. In the DTRR response to RAI No. 52, scenario 2 provides the maximum hypothetical accident (MHA) submersion dose to occupants of the control room from leakage out of the reactor bay. However, the RAI response did not indicate if the dose included the contribution from the suspended airborne (scattered) radioactive material in the reactor bay emitting gamma rays through the intervening wall? Indicate or provide the contribution to the MHA dose to the control room from the reactor room.

DTRR response:
The response to RAI No. 52 did not include the contribution to dose to occupants of the control room from the direct exposure from airborne radio-nuclides that remain within the reactor room. The external dose to the control room occupants from direct exposure to the airborne radio-nuclides in the reactor room would be bounded by the value of one-half of the submersion dose of the reactor room occupants when the event occurs. A factor of one-half is appropriate because the exposure can only come from one side of the reactor room occupant, instead of both sides if the occupant was fully submerged in the airborne release. This conservatively ignores the presence of the reactor room wall for shielding and the distance between the reactor room wall and the location of the control room occupants, which would further reduce exposure rates. Therefore, based on the calculated submersion dose to the reactor room occupants of 0.131 rem/hr (see Table 52-2 in the response to RAI 52 submitted on January 20, 2012), the bounding exposure rate of the control room occupants from airborne radio-nuclides that remain in the reactor room would be 0.066 rem/hr. Including this contribution to the dose estimate for the control room occupants during the incident increases the calculated dose rate from 0.274 rem/hr to 0.34 rem/hr. Therefore, assuming that it takes no more than 15 minutes for the building to be evacuated, the total effective dose equivalent to members of the public in the control room would be 85 mrem.

RAI-3. In the DTRR response to RAI No. 54, the Loss of Coolant Accident (LOCA) analysis provided the potential radiological consequences to workers and members of the public directly near the reactor. However, the RAI response did not provide the results of the potential environmental consequences due to the LOCA tank breach leaking into the ground and groundwater. Provide an analysis of the consequences of a leak from the primary coolant tank into the ground. If possible, consider using assumptions based on equipment maintained by DTRR TSs or actions described in the DTRR emergency plan.

DTRR response:

Per 10 CFR Part 20, Appendix B, Column 2, releases of H-3 must be less than 0.001 μCi/ml (2220 dpm). Measurement of the concentration of tritium in the reactor pool is conducted monthly, and results average about 40 dpm (0.018×10^{-3} μCi), with a maximum result of 45 dpm (0.020×10^{-3} μCi), in the last three years of operation. Tech Spec 3.4.3 (July 2013) specifies that the concentration of radio-nuclides in the reactor pool water must be less than the allowable release limits in 10 CFR Part 20, Appendix B, Column 2. Therefore, any inadvertent releases, due to a loss of cooling water incident or a leak, would satisfy the dose requirement under 10 CFR 20.1301 and will have no significant effect on the surrounding environment.

RAI-4. The proposed DTRR TS 1.1, Scope, last sentence, indicates that the Bases do not constitute limitations or restrictions. However, the proposed DTRR TSs provide numerous TS Bases that reference sections from the DTRR Safety Analysis Report (SAR), which contains assumptions used in analyses and calculations, which form TS limits. Justify the proposed TS or propose a sentence, which does not conflict with references to SAR results.

DTRR response:
This is being addressed in the revised DTRR TS (July 2013). Bases that reference SAR or TS may constitute limitations or restrictions.

RAI-5. The proposed DTRR TS 1.3, Definitions – Reactor Secured, Item No. 2.d, contains the phrase "or of one dollar," which seems redundant to TS 3.7.1, which limits the reactivity worth of all experiments to $1.00. Justify the proposed TS or remove the phrase "or of one dollar."

DTRR response:
TS 3.7.1 has been revised by removing the phrase "or of one dollar" from it, in the revised DTRR TS (July 2013).

RAI-6. The proposed DTRR TS 1.3, Definitions – Control Rod, Item No. 2, Shim/Safety Rod describes the capabilities of the control rod, but does not indicate how its position can be changed. Justify the proposed TS or provide a description of the control of the position of the Shim/Safety Rod.

DTRR response:
TS 1.3, Definitions-Control Rod, Item No 2 has been revised by adding how the positions of the Shim/Safety Rod are changed, in the revised DTRR TS (July 2013).

RAI-7. The proposed DTRR TS 3.3, Reactor Control Rods and Safety Systems and Interlocks, Specification No. 3, does not appear to include the time for the electrical signal to initiate control rod movement as provided in the definition of the scram time. Justify the proposed TS or propose a revision to include the electrical signal time component as described in the definition of the scram time.

DTRR response:
TS 3.3 has been revised to be consistent with the definition of "Scram Time", which means time of initiation of scram signal to fully inserted control rods, in the revised DTRR TS (July 2013).

RAI-8. The proposed DTRR TS 3.5, Ventilation, does not appear to contain a provision for requiring ventilation when core or control rod work, which could cause a reactivity change of more than $1.00 as provided in the guidance of NUREG-1537 and ANSI/ANS-15.1-2007. Justify the proposed TS 3.5 or propose a revision to add the core or control rod work as provided in the guidance of NUREG-1537 and ANSI/ANS-15.1-2007.

DTRR response:
TS 3.5 has been revised to include the statement: Ventilation must be operating when core or control rod work that can change reactivity by more than $1 is being done, in the revised DTRR TS (July 2013)

RAI-9. The proposed DTRR TS 3.7.1, Specification b. does not indicate the absolute value of reactivity for an experimental worth. Justify the proposed TS or propose a revision to account for the absolute value of reactivity worth for experiments.

DTRR response:
TS 3.7.1 has been revised to include absolute value for the reactivity worth of an experiment, in the revised DTRR TS (July 2013).

RAI-10. The proposed DTRR TS 3.7.3, Bases, 1st paragraph, describes the release of fission products and the 2nd paragraph describes the effect of an undisturbed column of water. Both items do not appear consistent with the requirements in TS 3.7.3. Justify the proposed TS Bases or propose a revision to the Bases.

DTRR response:
The bases for TS 3.7.3 has been revised to address this issue in the revised DTRR TS (July 2013).

RAI-11. The proposed DTRR TS 4.3, Control and Safety Systems, does not appear to include a surveillance requirement (e.g., Channel Check, Test or Calibration) for the Reactor Pool Water Radioactivity Monitor listed in DTRR TS 3.3, Table 3.3B. Justify the proposed TS or propose a surveillance requirement for the Reactor Pool Water Radioactivity Monitor.

DTRR response:
A surveillance requirement for the pool water radioactivity, TS 4.4.5 has been added to the revised DTRR TS (July 2013)

RAI-12. The proposed DTRR TS 4.4, Reactor Coolant Systems, Bases indicates that a channel calibration of the Reactor Pool Water Temperature Monitor is performed as required as a result of the channel test. This appears to be a specification in the Bases, rather than in the TSs. Justify leaving this specification in the Bases or propose adding to TS 4.4, Specification 3.

DTRR response:
The Statement "channel calibration of the Reactor Pool Water Temperature Monitor is performed if required, as a result of the channel test" to TS 4.4.3 in the revised DTRR TS (July 2013).

RAI-13. The proposed DTRR TS 5.1, Reactor Site and Building, does not indicate the licensed area as defined in the DTRR TS Definitions. Justify the proposed TS or propose adding the licensed area to TS 5.1.

DTRR response:

The license area is now defined consistently as rooms 51(51,51A,51B,51AA) and 52, in the revised DTRR TS (July 2013)

RAI-14. The proposed DTRR TS 6.1.2, Responsibility, does not indicate the reporting relationship (solid line in TS Figure 6.1) between the Radiation Safety Officer and the Dow Core R&D Director, Analytical Sciences (Level 1). Justify the proposed TS Figure 6.1 or propose a revised figure to indicate the reporting relationship.

DTRR response:
The revised DTRR TS (July 2013) contains a revised figure 6.1 showing the reporting and communication relationships between the SRO and Director Analytical Sciences (Level1).

RAI-15. The proposed DTRR TS 6.2.1, Composition and Qualification states that the Reactor Operations Committee (ROC) shall consist of at least four members, but then lists five members by title or name. The minimum number of ROC members is not clear. Justify the proposed TS 6.2.1 or propose a revised TS, which clearly states the minimum composition of the ROC.

DTRR response:
TS 6.2.1 has been revised in the revised DTRR TS (July 2013) to indicate that the minimum member of ROC shall be four and listed possible member compositions.

RAI-16. The proposed DTRR TS 6.2.2, ROC Rules:
a. Specification c., indicates that the meeting minutes shall be reviewed and approved at the next meeting, which could be as long as a year. NUREG-1537 and ANSI/ANS-15.1-2007 guidance suggests a review in a timely manner. Justify the proposed TS or propose a revised TS with a review performed in a timely manner.
b. Specification e., does not indicate how the ROC communicates with the Radiation Safety Committee. NUREG-1537 and ANSI/ANS-15.1-2007 guidance suggests a written report or minutes of the findings. Justify the proposed TS or propose a revised TS indicating the communication method or format.

DTRR response:
TS 6.2.2 has been revised in the revised DTRR TS (July 2013) to indicate when ROC meeting minutes must be reviewed after the meeting. TS 6.2.2 has also been revised to indicate how the Reactor Operations Committee (ROC) communicates with Radiation Safety Committee (RSC).

RAI-17. The proposed DTRR TS 6.2.3, ROC Review Function, Specification c., and TS 6.2.4, Specification a, describe requirements applicable to a charter. However, a charter is not defined elsewhere in the TSs. Justify the proposed TS, or propose revised TSs 6.2.3 and 6.2.4 delineating the applicability of a charter.

DTRR response:
TS 6.2.4 has been revised by removing the word "Charter" from it, in the revised DTRR TS (July 2013).

18. The proposed DTRR TS 6.2.4, ROC Audit Function, does not appear to require deficiencies that affect reactor safety to be immediately reported to the Level 1 as provided in the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Justify the proposed TS or propose a revised TS to ensure that deficiencies that affect reactor safety are immediately reported to the Level 1.

DTRR response:
TS 6.2.4 has been revised to include the phrase: "Any deficiencies that may affect reactor safety shall be immediately reported to ROC Chair, Level 1" in the revised DTRR TS (July 2013).

RAI-19. The proposed DTRR TS 6.4, Procedures:
a. Specification e., provides control rod removal or installation as an example of major components, which could have an effect of reactor control and safety. It is not clear if control rod removal or installation is the only consideration to have an effect on reactor safety or control. Justify the proposed TS or propose a revised TS to ensure that the TS is not limited to control rod removal or installation.
b. The paragraph discussing "substantive changes to the above procedures" does not provide a definition or criteria for those changes which would be considered substantive or un-substantive. Additionally, it is unclear if the review requirements of 10 CFR 50.59 are maintained within this TS paragraph. Provide a description of substantive and un-substantive changes to procedures and how this TS paragraph ensures conformance to the requirements of 10 CFR 50.59.
c. The paragraph discussing "temporary deviations" does not appears to provide the methodology for establishing and changing procedures as provided in the guidance in NUREG-1537, Part 1, Appendix 14.1, Section 6.4, Procedures. Additionally, it is unclear if the review requirements of 10 CFR 50.59 are maintained within this TS paragraph. Provide the methodology for temporary deviations and how this TS paragraph ensures conformance to the requirements of 10CFR 50.59.

DTRR response:
TS 6.4, item e, has been revised by removing the list and therefore referring to any major maintenance in the DTRR revised TS. Also the words "substantive, un-substantive and temporary deviations" have been removed from TS 6.4 in the revised DTRR TS (July 2013).

RAI-20. The proposed DTRR TS 6.5, Experimental Review and Approval, Specification b., discusses "minor changes that do not significantly alter the experiment." It is not clear what constitutes "minor changes that do not significantly alter the experiment." Additionally, it is unclear if the review requirements of 10 CFR 50.59 are maintained within TS 6.5. Justify the proposed TS or propose a revised TS that defines "minor changes that does not significantly alter the experiment" and ensures conformance to the requirements of 10CFR50.59.

DTRR response:
TS 6.5 has been revised by removing these statements "minor changes" from it, in the revised DTRR TS (July 2013).

RAI-21. The proposed DTRR TS 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, states for all events, which are required by regulations or TSs to be reported to the NRC within 24 hours under Section 6.7.2:

a. It is not clear why "regulations" needs to be identified in the TSs since NRC regulations remain requirements as specified in Title 10 of the *Code of Federal Regulations*. Justify the proposed TS or propose a revised TS that does not refer to regulations.

b. It is not clear why "within 24 hours" needs to be identified in TS 6.6.2 when TS 6.7.2 states the report needs to be provided not later than the following day. Justify the proposed TS or propose a revised TS that do not refer to the 24 hour time.

DTRR response:
TS 6.6.2 has been revised by removing the phrases "regulations" and "within 24 hours" from it, in the revised DTRR TS (July 2013).

RAI-22. The proposed DTRR TS 6.7.1, Annual Operating Reports, states that a report shall be created and submitted annually. However, the reporting period and the time following the submittal of the report following the end of the reporting period are not specified. Justify the proposed TS or propose a revised TS that indicates the reporting period and time allowed for submittal.

DTRR response:
TS 6.7.1 has been revised to indicate how often reports are created and when they should be submitted to the U.S.NRC, in the revised DTRR TS (July 2013).

Attachment II

Revised DTRR Technical Specifications, July 2013

Appendix A

To

FACILITY LICENSE NO. R-108

DOCKET NO. 50-264

TECHNICAL SPECIFICATIONS AND BASES
FOR
THE DOW TRIGA RESEARCH REACTOR

July/2013

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:

1. **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.

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.

3. **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 and which is not defined as any other kind of experiment.

4. **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.

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

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

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 the area surrounding the core.

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

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 (<$ 0.30).

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 $x10^{-9}$ 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 $7.94x10^{-12}$ μCi/ml, with a corresponding annual TEDE to a member of the public in an unrestricted area of 0.056 mR/yr. (SAR, as supplemented by letter dated June 11, 2012, an additional information to RAI No. 41). The TS value of 1 $x10^{-9}$ μCi/ml, which corresponds to a dose of 7.1 mR/yr at the nearest unrestricted area release point, is 30% lower than the 10 CFR 20 Appendix B effluent concentration value, 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 materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials or liquid fissionable materials shall be doubly encapsulated;

b. Materials which could react in a way which could damage the components of the reactor (such as gunpowder, dynamite, TNT, nitroglycerin, or PETN) shall not be irradiated in quantities greater than 25 milligrams TNT equivalent in the reactor or experimental facilities without out-of-core tests which shall indicate that, with the containment provided, no damage to the reactor or its components shall occur upon reaction. Such materials in quantities less than 25 milligrams TNT equivalent may be irradiated provided that the pressure produced in the experiment container upon reaction shall be calculated and/or experimentally demonstrated to be less than half the design pressure of the container. Such materials shall be packaged in the appropriate containers before being brought into the reactor room or shall be in the custody of duly authorized local, state, or federal officers; and

c. Experiments containing fissionable materials shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is less than 10 micro-curies and the maximum strontium-90 inventory is less than 35 nano-curies.

Basis

The specification on corrosive, reactive, explosive and fissionable material is intended to prevent damage to reactor components resulting from failure of any experiment involving these materials. The double encapsulation requirement is also intended to prevent damage to the reactor or the experiments due to release of the listed materials.

Explosive materials such as gunpowder, dynamite, TNT, nitroglycerin, or PETN require special handling. By limiting their quantities in any experiment to within the material specifications, and by controlling handling according to authorized local, state, and federal laws, potential damages to the reactor fuel or structure are avoided.

The specification on experiments involving fissionable material is intended to reduce the severity of the results of accidental release of airborne radioactive materials to the reactor room or to the atmosphere. By placing a limit on Iodine and Sr-90, the postulated maximum doses to the workers and to members of the public would be below the limits of 10 CFR 20 (SAR, as supplemented by letter dated January 20, 2012, RAI No. 56).

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 the gaseous activity or radioactive aerosols in the reactor room or the unrestricted area, would not exceed the limits of 10 CFR 20, 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 meet the purpose of 10 CFR 20 by reducing the likelihood that released airborne radioactivity to the reactor room or unrestricted area surrounding the DTRR will result in exceeding the total dose limits to an individual as specified in 10 CFR 20.

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); 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.

4. Required surveillances of the CAM, ARM, conductivity, pH and pool level shall not be deferred during reactor shutdown.

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 by $0.25 or more from the 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.

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 at least 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 at least 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 at least 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 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 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;

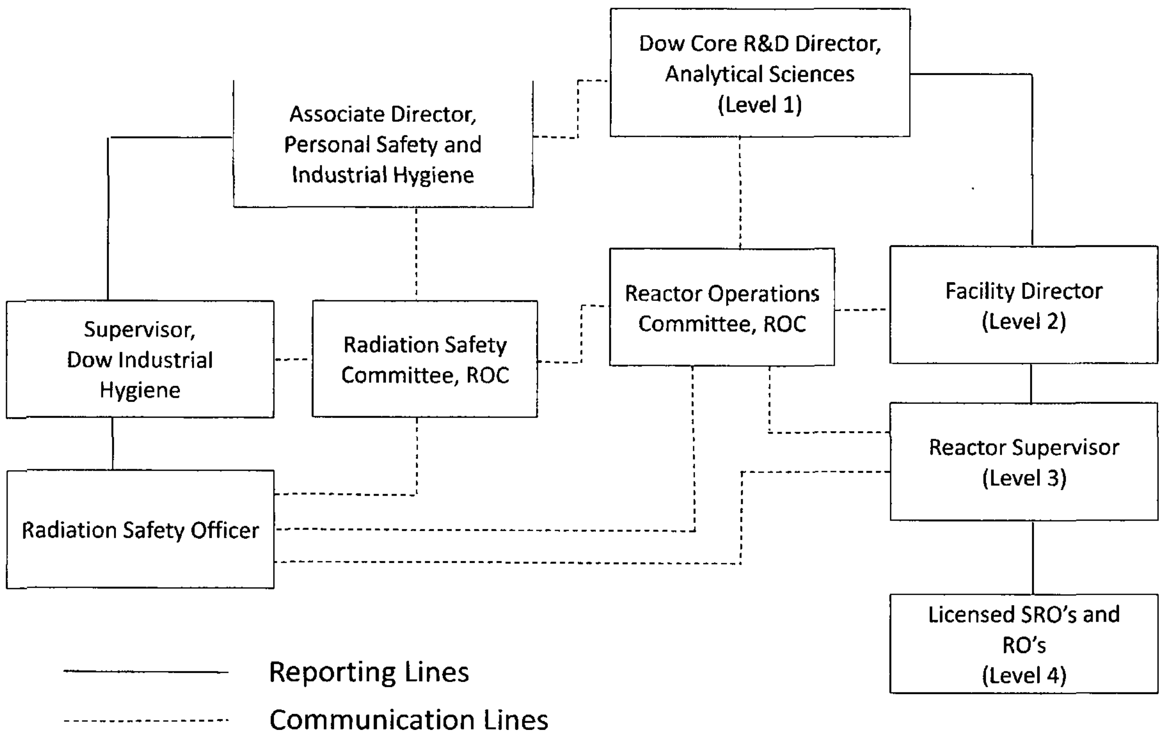
b. Dow TRIGA Research Reactor (DTRR) Director (Level 2): The DTRR Director reports to the Dow Core R&D Director, Analytical Sciences, and is accountable for the facility's operation;

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 operating 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 significant 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 necessary: 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);

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 report 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 under 10 CFR 50.59;

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 in effect 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

Approved 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 new experiments or a new class of 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 previously approved experiments or a previously approved class of 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; and

b. An immediate notification of the occurrence shall be made to the Reactor Supervisor, DTRR Director, Level 1, ROC, U.S. NRC Headquarters Operations Center; 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. Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests and experiments that are significantly different from those performed previously and are not described in the Safety Analysis Report, including a summary of the analyses leading to the conclusions that they are allowed without prior authorization by the U.S. NRC and that 10 CFR 50.59 was applicable;

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; and

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 schedule 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.