Safety Evaluation Report
Renewal of the Facility Operating License for the Dow Chemical Company Dow TRIGA Research Reactor

Docket No. 50-264

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to a timely application that the Dow Chemical Company (the licensee) filed for a 20-year renewal of Facility Operating License No. R-108 to continue operating the Dow TRIGA (Training, Research, Isotope Production, General Atomics) Research Reactor (DTRR). In its safety review, the NRC staff considered information that the licensee submitted, past operating history recorded in the licensee's annual reports to the NRC, inspection reports NRC personnel prepared, as well as firsthand observations. On the basis of its review, the NRC staff concludes that the Dow Chemical Company can continue to operate the facility for the term of the renewed facility operating license, in accordance with the license, without endangering public health and safety, facility personnel, or the environment.

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ABBREVIATIONS AND ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
ALARA	as low as reasonably achievable
Am	americium
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar	argon
ARM	area radiation monitor
Be	beryllium
Br	bromine
CAM	continuous air monitor
CEO	Chief Executive Officer
CFR	*Code of Federal Regulations*
CSC	control system computer
DAC	data acquisition computer
DCF	dose conversion factor
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DTRR	Dow TRIGA Research Reactor
EP	emergency plan
EPA	U.S. Environmental Protection Agency
FGR	Federal guidance report
FTC	fuel temperature coefficient
FY	fiscal year
GA	General Atomics
HVAC	heating, ventilation, and air conditioning
H	hydrogen
I	iodine
IR	inspection report
ISG	interim staff guidance
LCC	limiting core configuration
LCO	limiting conditions for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident

LRA	license renewal application
LSSS	limiting safety system setting
MCNP	Monte Carlo Neutron Transport
MHA	maximum hypothetical accident
N	nitrogen
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OC	operational core
OSU	Oregon State University
PDR	Public Document Room
PTS	pneumatic transfer system
RAI	request for additional information
R&D	research and development
RG	regulatory guide
RO	reactor operator
ROC	Reactor Operations Committee
RSC	Radiation Safety Committee
RSO	radiation safety officer
RTR	research and test reactor
SAR	safety analysis report
SDM	shutdown margin
SER	safety evaluation report
SL	safety limit(s)
SNM	special nuclear material
SRM	staff requirements memorandum
SRP	Standard Review Plan
SRO	Senior reactor operator
T-H	thermal-hydraulic
TEDE	total effective dose equivalent
TRIGA	Training, Research, Isotope Production, General Atomics
TS	technical specification(s)
U	uranium
U-ZrH	uranium zirconium hydride
Zr	zirconium

TECHNICAL PARAMETERS AND UNITS

±	more than or less than
$	a unit of reactivity where absolute reactivity is divided by the total effective delayed neutron fraction β_{eff}
C	temperature in degrees Celsius
μmhos	micro-mhos
μCi	micro-curies
nCi	nano-curies
Ci	Curies
cm	centimeter
cps	counts per second
ft	feet
G	gram
Hr	hour
in.	inches
kWt	kilowatts thermal
l	liter
m	meter
min	minute
MWd	megawatt days
MWt	megawatt-thermal
pH	potential of hydrogen
rem	Roentgen Equivalent in Man
mrem	millirem
W	watts
Wt	watts thermal
w%	weight percent
yr	year
α_F	fuel temperature coefficient
β_{eff}	effective delayed neutron fraction
$\Delta k/k$	absolute reactivity
ρ_E	reactivity of the experiments
ρ_R	worth of the regulating control rod

ρ_S	excess reactivity of the reactor system
ρ_{SDM}	reactivity of the shutdown margin requirement
ρ_{SH1}	worth of the Shim1 control rod
ρ_{SH2}	worth of the Shim2 control rod
ρ_X	total excess reactivity
X/Q	atmospheric relative concentration in s-cm^3

1. INTRODUCTION

1.1 Overview

In a letter dated April 1, 2009, as supplemented by letters dated September 24, 2010; January 12, February 11, April 20, May 12, May 27, August 12, August 31, October 12, November 10, and December 6, 2011; January 13, January 20, February 7, June 11, and August 10, 2012; July 11, and September 16, 2013; and April 9, 2014, the Dow Chemical Company (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for a 20-year renewal of the Class 104c Facility Operating License No. R-108, Docket No. 50-264, for the Dow TRIGA (Training, Research, Isotope Production, General Atomics) Research Reactor (DTRR) (Ref. 1).

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.51(a) states that "[e]ach license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from the date of issuance." Dow Chemical Company holds Facility Operating License No. R-108 (the license), which originally was issued on July 3, 1967. The license was renewed by License Amendment No. 3, issued on November 21, 1977. Facility Operating License No. R-108 was renewed again by License Amendment No. 5, issued on May 8, 1989, for an additional 20 years, with an expiration date of May 8, 2009. Because of the timely renewal provision contained in 10 CFR 2.109(a), the licensee is permitted to continue operation of the DTRR under the terms and conditions of the current license until the NRC staff completes action on the renewal request. A renewal would authorize continued operation of the DTRR for an additional 20 years.

The DTRR was licensed in 1967 as a research reactor facility at a maximum steady-state power level of 100 kilowatts thermal (kWt). The licensee submitted its initial license application on August 3, 1966, and the NRC issued Construction Permit No. CPRR-94, which authorized the construction of the TRIGA Mark 1 reactor at the Dow Chemical Company site in Midland, Michigan on December 20, 1966. Initial criticality was achieved on July 6, 1967. License Amendment No. 5, issued on May 8, 1989, renewed the license and authorized an increase in the licensed power level from 100 kWt to 300 kWt. The license renewal application (LRA) submitted on April 1, 2009, requested an increase in the licensed power level from 300 kWt to 750 kWt. During the NRC staff review, the licensee subsequently withdrew the power uprate portion of its LRA in its letter dated January 12, 2011 (Ref. 2). The DTRR is not licensed to perform power pulsing operations.

The NRC staff based its review of the request to renew the DTRR facility operating license on the information contained in the LRA, as well as supporting supplements and licensee responses to requests for additional information (RAIs). The initial 2009 LRA included a DTRR safety analysis report (SAR) with technical specifications (TSs), an environmental report, an operator requalification program, and an emergency plan (EP). The NRC staff conducted site visits on August 20, 2010; January 19, and July 25, 2011; and June 4, 2012, to observe facility conditions and to discuss RAIs and responses. The NRC staff requested additional information by letters dated July 12 (Ref. 44), and November 2, 2010 (Ref. 3); June 26 (Ref. 46), and September 6, 2013 (Ref 50); and March 6, 2014 (Ref. 51). By letter dated February 11, 2011, DTRR provided an update to its application with an updated final SAR and responses to certain RAIs (Ref. 4). Additional RAI responses and clarifications were provided by letters dated September 24, 2010 (Ref. 45); April 20 (Ref. 5), May 12 (Ref. 6), May 27 (Ref. 7), August 12 (Ref. 8), August 31 (Ref. 9), October 12 (Ref. 10), November 10 (Ref. 11), and December 6, 2011 (Ref. 12); January 13 (Ref. 13), January 20 (Ref. 14), February 7 (Ref. 15), June 11 (Ref. 38), and August 10, 2012 (Ref. 39); July 11 (Ref. 47), and September 16, 2013 (Ref. 49); and April 9, 2014 (Ref. 52). Updated TSs were provided by References 15, 38, 39, and 47. During the NRC staff's review of the TSs, minor typographical changes were identified and discussed with DTRR staff via telephone on January 18, 2013, and by electronic mail on August 5, 2013 (Ref. 48). The DTRR staff agreed with the proposed TS typographical corrections. Throughout this report, statements referring to the SAR shall mean the SAR provided in Ref. 4, as supplemented by these RAI responses, reports, and clarifications. The NRC staff's review also included information from DTRR annual reports and NRC inspection reports (IRs) from 2005 to 2013.

With the exception of the security plan and the EP, material pertaining to this review may be examined or copied, for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The NRC maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Documents related to this license renewal may be accessed through the NRC's Public Library on the Internet at http://www.nrc.gov. If you do not have access to ADAMS or if you experience problems accessing the documents in ADAMS, contact the NRC PDR staff by telephone at 1-800-397-4209 or 301-415-4737, or send an e-mail to the PDR at PDR.Resource@nrc.gov. The security plan is protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance requirements," and the EP is withheld from public disclosure because it is considered security-related information. Because parts of the SAR and RAI responses from the licensee contain security-related information and are protected from public disclosure, redacted versions are available to the public.

The "References" section of this document contains the dates and associated ADAMS Accession Numbers of the licensee's renewal application and associated supplements.

In conducting this review, the facility was evaluated against the requirements of the regulations, including 10 CFR Part 20, "Standards for Protection Against Radiation," 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The recommendations of applicable regulatory guides (RGs) and relevant accepted industry standards, such as those of the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series, are also considered. The NRC staff specifically referred to the recommendations contained in NUREG-1537, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 16). Because there are no specific accident-related regulations for research reactors, the NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20 (i.e., the standards for protecting employees and the public against radiation).

In SECY-08-0161, “Review of Research and Test Reactor License Renewal Applications,” dated October 24, 2008 (Ref. 17), the NRC staff provided the Commission with information on plans to streamline the review of LRAs for research and test reactors (RTRs). The Commission issued its staff requirements memorandum (SRM) for SECY-08-0161 dated March 26, 2009 (Ref. 18). The SRM directed the staff to streamline the renewal process for such reactors, using some combination of the options presented in SECY-08-0161. The SRM also directs the staff to implement a graded approach whose scope is commensurate with the risk posed by each facility. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 of SECY-08-0161. In the alternative safety review approach used in this report, the NRC staff considered the results of past NRC staff reviews. A basic requirement, as contained in the SRM, is that licensees must be in compliance with applicable regulatory requirements.

The NRC staff developed the RTR Interim Staff Guidance (ISG)-2009-001, “Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors,” to assist in the review of LRAs. The streamlined review process is a graded approach based on licensed power level. The streamlined review process divides the RTR facilities into two tiers. Facilities with a licensed power level of 2 megawatts thermal (MWt) and greater, or requesting a power level increase, undergo a full review using NUREG-1537. Facilities with a licensed power level less than 2 MWt, undergo a focused review that centers on the most safety-significant aspects of the renewal application and relies on past NRC reviews for certain findings. The NRC staff made a draft of the ISG available for public comment and considered public comments in its development of the final ISG. The NRC staff reviewed the DTRR LRA using the guidance in the final ISG, dated October 15, 2009 (Ref. 19), and because the DTRR’s licensed power level is less than 2 MWt, the NRC staff performed a focused review of the licensee’s LRA. Specifically, the review focused on reactor design and operation, accident analysis, TSs, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to the facility the licensee made after submitting the renewal application.

In the LRA, the licensee indicated that no changes to the DTRR security plan were necessary. However, the NRC staff reviewed the DTRR security plan, “Security Plan for the Dow TRIGA Reactor Facility,” in effect at the time of the LRA (April 1, 2009). As a result of this review, the NRC staff issued RAIs to the licensee in a letter dated September 6, 2013 (Ref. 50), and the licensee responded on September 16, 2013 (Ref. 49) including a revised DTRR security plan. The NRC staff reviewed the revised DTRR security plan, found that it met the applicable regulations, and based on that finding concludes that the DTRR security plan, dated September 16, 2013, is acceptable. The licensee maintains the program for the physical protection of the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73, “Physical Protection of Plants and Materials.” Changes to the physical security plan can be made by the licensee, in accordance with the requirements of 10 CFR 50.54(p), as long as those changes do not decrease the effectiveness of the plan. In addition, the NRC staff performs routine inspections of the licensee’s compliance with the requirements of the security plan. The NRC staff’s review of the DTRR IRs for the past several years identified no violations.

The NRC staff reviewed the DTRR EP provided on April 1, 2009, as part of the LRA. The NRC staff issued RAIs to the licensee in a letter dated September 16, 2010 (ADAMS Accession No. ML102520281), and the licensee responded on October 11, 2010 (non-publicly available), and provided a revised EP in a letter dated July 7, 2011 (non-publicly available). The NRC staff reviewed the DTRR EP, found that it met the applicable regulations, and based on that finding, approved the DTRR EP in a letter dated September 26, 2011 (ADAMS Accession No. ML112590262). In addition, the NRC routinely inspects the licensee's compliance with the EP requirements. The NRC staff's review of DTRR's IRs for the past several years identified no violations. The licensee maintains an EP in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," which provides reasonable assurance that the licensee will be prepared to assess and respond to emergency events.

The NRC staff reviewed the DTRR reactor operator (RO) requalification program. The NRC staff issued RAIs to the licensee in a letter dated October 27, 2010 (ADAMS Accession No. ML102790003), and the licensee responded on November 17, 2010 (ADAMS Accession No. ML103260150), and provided a revised operator requalification program by letter dated June 16, 2011 (ADAMS Accession No. ML11171A030). The NRC staff reviewed the program, found that it met the applicable regulations, and based on that finding, approved the DTRR operator requalification program by letter dated September 6, 2011 (ADAMS Accession No. ML112490009).

This safety evaluation report (SER or report) summarizes the NRC's staff's findings of the DTRR safety review of the LRA and delineates the technical details that the NRC staff considered in reviewing and evaluating the safety aspects of continued operation. This report provides the basis for renewing the DTRR license at a steady-state power level of 300 kWt.

This SER was prepared by Walter Meyer and Geoffrey A. Wertz, project managers from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking, Research and Test Reactors Licensing Branch; and Jo Ann Simpson, financial analyst from the NRC's NRR, Division of Inspection and Regional Support, Financial Analysis and International Projects Branch. Energy Research, Inc., the NRC's contractor, provided substantial input to this report.

1.2 Summary and Conclusions on Principal Safety Considerations

The DTRR SAR discusses the licensee's conclusions on principal safety considerations. This SER is the NRC staff's review and evaluation of the information in the DTRR SAR, as supplemented by past operating history recorded in the licensee's annual reports to the NRC, and the NRC's IRs. On the basis of this evaluation and resolution of the principal issues under consideration for the DTRR, the NRC staff concludes the following:

- The design and use of the reactor structures, systems, and components important to safety during normal operation discussed in the DTRR SAR, as supplemented, in accordance with the TSs, are safe, and safe operation can reasonably be expected to continue.

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA), emphasizing those that could lead to a loss of integrity of fuel element cladding and a release of fission products. The licensee performed analyses of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed doses in 10 CFR Part 20 for unrestricted areas.

- The licensee's management organization, conduct of training, and research activities, in accordance with the TSs, are adequate to ensure safe operation of the facility.

- The systems provided for the control of radiological effluents, when operated in accordance with the TSs, are adequate to ensure that releases of radioactive materials from the facility are within the limits of the Commission's regulations and are as low as is reasonably achievable (ALARA).

- The licensee's TSs, which provide limits controlling operation of the facility, are such that there is a high degree of assurance that the facility will be operated safely and reliably. There has been no significant degradation of the reactor, as discussed in Chapter D of the SAR, and the TSs will continue to ensure that there will be no significant degradation of safety-related equipment.

- The licensee has reasonable access to sufficient resources to cover operating costs and eventually to decommission the reactor facility.

- The licensee maintains a program for providing for the physical protection of the facility and its SNM, in accordance with the requirements of 10 CFR Part 73. Changes to the security plan have been made in accordance with 10 CFR 50.54(p).

- The licensee maintains an emergency plan in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, which provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events. All changes to the emergency plan have been made in accordance with 10 CFR 50.54(q).

- The licensee's procedures for training its reactor operators and the operator requalification plan give reasonable assurance that the licensee will continue to have qualified personnel who can safely operate the reactor

On the basis of these findings, the NRC staff concludes that the licensee can continue to operate the DTRR in accordance with the renewed license without endangering the health and safety of the public, facility personnel, or the environment. The issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Description

The DTRR is located on the Dow Chemical Company Midland site, within the city limits of Midland, Michigan, as documented in the licensee's response to RAI-2 (Ref. 4). The city of Midland has approximately 42,000 people.

The DTRR facility is located within a general purpose laboratory building (Building 1602) enclosed behind a security fence that surrounds the Dow Chemical Company research and manufacturing facilities at the Dow site. The building is constructed as a laboratory with fireproof construction consisting of a steel frame, concrete panels, and block walls. The reactor installation used fuel and components manufactured by General Atomics (GA) and the reactor structures were built to specifications that were in use at the time by GA. According to the licensee's response to RAI-7 (Ref. 8), the building code in place and used at the time of construction of the DTRR building was "The Uniform Building Code, 1958 Edition."

The DTRR is a heterogeneous pool-type nuclear reactor fueled with TRIGA fuel. The coolant is demineralized water, which circulates through the core by natural convection. The maximum licensed steady-state power level is 300 kWt. The fuel is nominally 8.5 weight percent (w%) uranium (U), enriched to less than 20 w% in U-235, which is known as low enriched uranium (LEU).

The reactor tank is a cylindrical-shaped structure with an approximately 6.5 feet (ft) (1.9 meter (m)) inner diameter and approximately 21.5 ft (6.6 m) in depth. The reactor is installed in a pit below grade so that the top of the tank is at the surface of the reactor bay floor. The tank is made of 0.25 inches (in.) (0.6 centimeter (cm)) thick aluminum. The tank is wrapped on the outside with three layers of pitch and felt to retard corrosion. This tank and liner are contained in a poured concrete shell 3 ft (0.9 m) wide, which is contained in a corrugated steel shell. The aluminum liner rests on a concrete pad that is 3.5 ft (1.1 m) wide.

The reactor core is located at the bottom of the reactor tank and is submerged under approximately 20 ft (6.1 m) of water. Heat generated from the reactor core is directly transferred to the pool water. The reactor assembly is cooled by natural convection using the pool water and the water in the primary cooling circuit. Heat is removed from the primary circuit by natural convection to the air of the reactor room at the surface of the pool, through the tank walls by conduction, and through two heat exchangers. One heat exchanger is the Huron system, which has a capacity of 100 kWt and uses a once-through water system. The other heat exchanger is the SRX-1 system, which rejects heat to a heat exchanger that has a capacity of 1 MWt. These systems may be operated independently or together for a combined capacity of 1.1 MWt. When the Huron system is used, the ultimate heat sink is the service water. When the SRX-1 system is used, the ultimate heat sink is the chiller. According to the licensee's response to RAI-23 (Ref. 4), the SRX-1 system can be operated manually or set to automatically start when the pool water reaches 27 degrees Celsius (C), and shuts down at 22 degrees C. The system provides cooling only to the reactor pool and operates a temperature measurement uncertainty of plus or minus (±) 2 degrees C.

Entry to the reactor room is restricted to a single door from the control room. The reactor room also has direct access to the outside loading area through an overhead door referred to in the DTRR SAR as Door 10. This overhead door and the door between the reactor and control room form the confinement enclosure for the DTRR. In this report, and in all conclusions supporting the LRA, the reactor room and the laboratories are treated as a single area called the "reactor bay."

The DTRR reactor room uses a heating, ventilating, and air conditioning (HVAC) system that is dedicated solely to the reactor room. Fresh air is supplied through a single fresh air inlet that is heated and then passed into the reactor bay of Building 1602. This air is then exhausted from the reactor bay by another fan through a louver to the outside of the building that is centered about 8 ft (2.4 m) above the ground level. This system is described in SAR Section I.1 and the supplemental information provided in the responses to RAI-34 (Ref. 4). The operation of this system maintains the reactor bay under a slight negative pressure relative to atmospheric conditions. This is referred to as the normal mode of operation. In response to a radiation alarm indication, the system can switch to the isolation mode of operation, where both the inlet and exhaust fans are de-energized and the inlet louver is closed. The exhaust louver can be closed manually by the operator from outside the reactor building to provide further isolation of any airborne radiation. The operator can manually change to the isolation mode using a switch on the console.

1.4 Shared Facilities and Equipment

The DTRR is located in a building that contains minimal shared facilities which include electrical power, heating, cooling, water, and sewerage. The shared facilities remain under the control of the DTRR staff. During the site visit on August 20, 2010, the NRC staff observed no other shared facilities or equipment.

The electrical power for the DTRR is supplied from the site electrical power system. The design of the safety equipment of the DTRR does not require building electrical power to safely shut down the reactor, nor does the DTRR require building electrical power to maintain the reactor shutdown.

The water supplied to the DTRR as primary cooling water is purified by equipment that the DTRR staff operates and maintains. The safety equipment of the DTRR does not require building service water to safely shut down the reactor, nor does the DTRR require building service water to maintain the reactor shutdown. There is no safety function for this system specified in the DTRR SAR.

Similarly, the sewage, heating, and cooling facilities are not required for safe operation, shutdown, or maintaining DTRR shut down.

1.5 Comparison with Similar Facilities

In the DTRR SAR, Section A.5 (Ref. 4), the licensee provides general statements about the number of TRIGA-type reactors in service or being built. The DTRR is a Mark 1 reactor of the TRIGA design that uses original TRIGA reactor fuel and has both stainless-steel and aluminum cladding. The fuel is arranged in a circular pattern, similar to several other TRIGA reactors. Similarly, the experimental facilities are typical of other TRIGA reactors.

The TRIGA fuel typically has no established performance-related issues as long as well-established operating limits and water quality are maintained.

1.6 Summary of Operations

Dow Chemical Company uses the DTRR for a variety of research purposes. The reactor can use a rotating specimen rack (lazy susan), a pneumatic transfer system (PTS), and a central thimble for incore irradiation of specimens. The DTRR also has a radiochemistry laboratory for experiment analysis. The licensee calculated the DTRR total burnup for the period from 2005-2011, and the result was approximately 8,700 kW-days, which translates to an average full power operation of about 30 days per year. The DTRR operational workload is not expected to increase significantly from this level.

This review considered DTRR annual reports and NRC IRs from 2005 through 2013. The annual report summaries did not indicate any significant degradation of fuel element integrity, control rod operability issues, or radiological exposure concerns. The fuel temperature and scram circuits required for operation are calibrated regularly. The IRs identified no findings of significance.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the licensee has entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. In a contract (DE-CR01-83-NE-44483), the Dow Chemical Company obtained a commitment from DOE to accept the fuel at cessation of operation (Ref. 1, DTRR SAR, Section Q). By entering into such a contract with DOE, the DTRR has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

This review considered all of the changes made to the DTRR since the renewed license was issued on May 8, 1989. The DTRR SAR and licensee’s response to RAI-4 (Ref. 4) provided a comprehensive list of the major facility changes (summarized in Table 1-1 below). The most recent change was the installation of the closed loop heat exchanger in 2005. NRC IR, “Dow Chemical–NRC Inspection Report No. 50-264/2006-201” (Ref. 20), documented the NRC staff review that concluded that the change was performed in accordance with the requirements of 10 CFR 50.59, “Changes, tests and experiments,” and was acceptably implemented. The NRC staff reviewed IRs from 2005–2013 and, based on those results, concludes that the licensee performed changes in conformance with the requirements of 10 CFR 50.59, or submitted license amendment requests. The NRC staff concludes that all DTRR facility changes appear to be reasonable and the licensing actions taken over the years seem appropriate.

Table 1-1 Modifications to the DTRR Facility

Period of Activity	Modifications Performed
2004–2005	Added a closed loop heat exchanger to the operational open loop cooling system
1990s	Replaced the console using General Atomics as the vendor
1989	Increased the licensed power level from 100 kWt to 300 kWt
1982	Modified the terminal of the pneumatic transport system in the hot-lab hood to include an automatic feed mechanism and an online gamma ray detector

1.9 <u>Financial Considerations</u>

1.9.1 Financial Ability to Operate a Non-power Reactor

Regulation 10 CFR 50.33(f) states the following:

> Except for an electric utility applicant for a license to operate a utilization facility of the type described in § 50.21(b) or § 50.22, [an application shall state] information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with the regulations in this chapter, the activities for which the permit or license is sought.

Dow Chemical Company does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Further, 10 CFR 50.33(f)(2) states the following:

> [A]pplicants to renew or extend the term of an operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license.

The NRC staff determined that Dow Chemical Company must meet the financial qualification requirements pursuant to 10 CFR 50.33(f) and is subject to a full financial qualifications review. Dow Chemical Company is thus required to provide information to demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover the estimated operating costs for the period of the license. Pursuant to 10 CFR 50.33(f), Dow Chemical Company was required to submit estimates of the total annual operating costs for each of the first 5 years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover these costs.

In a letter dated December 7, 2010 (Ref. 40), Dow Chemical Company submitted its projected operating costs for the DTRR for each of the fiscal years (FYs), FY 2012-FY 2016. The projected operating costs for the reactor are estimated to range from $810,000 in FY 2012 to $911,662 in FY 2016. Dow Chemical Company's primary source of funding to cover the operating costs for the FYs cited above will be from their Analytical Sciences department budget, which is, in turn, part of the annual operating budget for Dow Chemical Company's Global Core Research and Development. According to Dow Chemical Company, funding is allocated annually. In a letter dated April 4, 2011 (Ref. 41), Dow Chemical Company provided historical budgetary information and stated that the Global Core Research and Development 2009 budget was $1.6 billion dollars, and the Analytical Sciences 2009 budget was $50 million dollars. Dow Chemical Company expects that this funding source will continue for FY 2012-FY 2016. The NRC staff reviewed Dow Chemical Company's estimated operating costs and its projected source of funds to cover those costs, and finds them to be reasonable.

The NRC staff finds that Dow Chemical Company has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs for the period of the license. Accordingly, the NRC staff determined that Dow Chemical Company met the financial qualification requirements pursuant to 10 CFR 50.33(f) and is financially qualified to engage in activities related to the DTRR.

1.9.2 Financial Ability to Decommission the Facility

The NRC has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to ensure the adequate protection of public health and safety. Regulation 10 CFR 50.33(k)(1) requires that:

> [A]n application for an operating license…for a production or utilization facility, [must provide] information in the form of a report, as described in § 50.75, indicating how reasonable assurance will be provided that funds will be available to decommission the facility.

Regulation 10 CFR 50.75(d)(1) requires that:

> [E]ach non-power reactor applicant for or holder of an operating license for a production or utilization facility shall submit a decommissioning report as required by § 50.33(k)….

The decommissioning report must contain a cost estimate for decommissioning the facility, indicate the funding method(s) to be used to provide funding assurance for decommissioning, and describe the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing decommissioning funding assurance are specified in 10 CFR 50.75(e)(1).

In its letter dated August 10, 2011 (Ref. 42), Dow Chemical Company provided an updated decommissioning cost estimate of $5,516,000 in 2011 dollars. The decommissioning cost estimate summarizes costs by labor, waste disposal, fuel removal and transportation, materials, equipment and supplies, and includes a contingency factor of 25 percent. According to Dow Chemical Company, the updated DTRR decommissioning cost estimate is comparable to more recent decommissioning cost estimates for the following facilities: University of Wisconsin, Washington State University, Reed College, and University of Utah. Dow Chemical Company stated that the Reed College reactor is the unit most similar to the DTRR and may provide the most comparative decommissioning cost estimate. Based on the average estimated decommissioning cost per kW, the decommissioning cost estimate for the DTRR is more conservative relative to the aforementioned facilities.

In its letter dated December 7, 2010, Dow Chemical Company states that it will continue to update the decommissioning cost estimate for the DTRR annually using the implicit price deflator for gross national product, which is published in July each year by the U.S. Department of Commerce's Bureau of Economic Analysis. In reviewing the decommissioning cost estimate for the DTRR ($5,516,000 in 2011 dollars), the NRC staff reviewed the decommissioning method and cost estimate for the DTRR, which were provided in letters dated December 7, 2010; April 4, and August 10, 2011. Based on its review of the information provided above, the NRC staff concludes that the decommissioning approach and the decommissioning cost estimate submitted are reasonable and conservative compared to other recent research reactor decommissioning cost estimates.

Dow Chemical Company currently is providing financial assurance for decommissioning with a surety bond, as allowed by 10 CFR 50.75(e)(1)(iii)(A), which states that, "[A] surety method may be in the form of a surety bond or letter of credit." To support its use of the surety bond for providing decommissioning funding assurance, by letter dated April 15, 2011 (Ref. 43), Dow Chemical Company submitted corroborating documentation that it had updated the value of the surety bond to $5,521,813, increased the surety bond rider, and updated the standby trust agreement Schedules A and B. According to Dow Chemical Company, it increased the surety bond to $5,521,813 because of the annual inflation factor, which exceeds the estimated decommissioning cost estimate of $5,516,000 (in 2011 dollars). As stated in the application, Dow Chemical Company will review and update the surety bond rider annually to reflect the DTRR decommissioning cost estimate of the current year.

The NRC staff finds that any adjustment of the decommissioning cost estimate must incorporate, among other things, changes in costs due to the availability of disposal facilities. Furthermore, Dow Chemical Company has an obligation under 10 CFR 50.9, "Completeness and accuracy of information," to update any changes in the projected cost, including changes in costs resulting from increased disposal options.

The NRC reviewed Dow Chemical Company's information on decommissioning funding assurance as described above and finds that the surety bond is acceptable, the decommissioning cost estimate is reasonable, and Dow Chemical Company's means of adjusting the decommissioning cost estimate periodically over the life of the facility is reasonable and, therefore, satisfies the requirements of 10 CFR 50.33(k) and 10 CFR 50.75(d).

1.9.3 Foreign Ownership, Control, or Domination

1.9.3.1 Background

Sections 103d and 104d of the Atomic Energy Act of 1954, as amended (AEA), provide, in relevant part, that:

> No license may be issued to any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation or a foreign government. In any event, no license may be issued to any person within the United States if, in the opinion of the Commission, the issuance of a license to such person would be inimical to the common defense and security or to the health and safety of the public.

The NRC's regulation, 10 CFR 50.38, "Ineligibility of certain applicants," contains language to implement this statutory prohibition. The NRC staff evaluated the application and supplements to the application in a manner that is consistent with the guidance provided in the Final Standard Review Plan [SRP] on Foreign Ownership, Control, and Domination, dated September 28, 1999 (see *Federal Register* Notice 64 FR 52355-52359, or https://federalregister.gov/a/99-25182), and hereafter referred to as the SRP, to determine if Dow Chemical Company is owned, controlled, or dominated by an alien, a foreign corporation, or foreign government.

The NRC position outlined in the SRP states that "[t]he foreign control determination is to be made with an orientation toward the common defense and security." Further, the SRP outlines how the effects of foreign ownership may be mitigated through implementation of a negation action plan to ensure that any foreign interest is effectively denied control or domination over the applicant.

1.9.3.2 Discussion

Dow Chemical Company is organized as a corporation incorporated in the State of Delaware. The principal location where it does business is 2030 Dow Center, Midland, Michigan, 48640. The LRA included the names, addresses, and citizenship of Dow Chemical Company's directors and officers. Upon review, the NRC staff identified that 8 of the 32 directors or officers are non-U.S. citizens, including the Chief Executive Officer (CEO) and Chairman. According to the LRA, Dow Chemical Company is free of alien, foreign corporation, or foreign government control or domination.

The Commission previously has stated that, "[a]n applicant is considered to be foreign owned, controlled, or dominated whenever a foreign interest has the 'power,' direct or indirect, whether or not exercised, to direct or decide matters affecting the management or operations of the applicant" (*Federal Register* Notice 64 FR 52355- 52358, or https://federalregister.gov/a/99-25182). As a result, the NRC staff's foreign ownership, control, or domination determination is based on the totality of facts since a foreign interest may exert indirect control through factors other than ownership and voting interests, including, but not limited to, management positions held by non-U.S. citizens, and financial arrangements. In its letter dated December 7, 2010, Dow Chemical Company stated that physical access to the reactor and SNM is restricted to U.S. citizens and that physical security is discussed, planned, and approved with individuals on the Reactor Operations Committee (ROC), which is restricted to U.S. citizens. In its letter dated August 10, 2011, Dow Chemical Company provided updated DTRR organization information in a chart format, which also included the names and citizenship of the ROC and its management. In addition, the organizational chart shows the chain of command from the DTRR to the CEO, and indicated that all of the individuals in the chain are U.S. citizens with the exception of the Chairman and CEO of Dow Chemical Company, who is an Australian citizen. Finally, Dow Chemical Company also provided information on budgetary matters and stated that all budgetary decisions for the reactor are made at the global director, research and development (R&D) level. As previously mentioned in Section 1.9.1 of the SER, the projected operating cost for the reactor for FY 2012 is $810,000. This information will form the basis for the NRC staff's conclusions that follow.

1.9.3.3 Negation Action Plan

Dow Chemical Company submitted a negation action plan to ensure that Dow Chemical Company will segregate decisions relating to safety and security of the DTRR and SNM from non-U.S. citizen directors and officers. Dow Chemical Company executed a resolution entitled, "Assignment of Responsibility for Research Nuclear Reactor and its Special Nuclear Materials," which was adopted on December 15, 2011, at a meeting of the Board of Directors of Dow Chemical Company, held in Midland, Michigan, at which a quorum of the board of directors was present. The adopted resolution is in full force and effect, and was submitted to the NRC by letter dated January 13, 2012 (Ref. 13).

Under the resolution, exclusive authority over the DTRR and its SNM will be vested in the Vice-President over R&D, so long as such person is a citizen of the U.S. The current executive Vice-President over R&D is a U.S. citizen. If at any time the Vice -President over R&D is not a U.S. citizen, then at such time the exclusive authority over the DTRR and its SNM will be transferred to an executive of at least the rank of Vice-President who is a U.S. citizen. As stated in the resolution, if at any time, action by the board of directors is necessary for any matter involving the DTRR or its SNM, only directors and officers who are U.S. citizens may participate in deliberations or decisions, or have a vote, concerning such action. This unanimous consent resolution applies to all current and future officers and directors of Dow Chemical Company, so long as Dow Chemical Company continues to own or operate the DTRR and its SNM. Finally, according to Dow Chemical Company, nothing herein shall require any decision to be made at executive or board levels if action at that level is not otherwise required by applicable policies, including but not limited to the authorization policy.

However, to ensure that the proposed negation action plan Dow Chemical Company will be implemented and will operate to segregate decisions relating to safety, security, and reliability of the DTRR and its SNM from non-U.S. citizen directors and officers, the NRC staff will require, as conditions of the renewed facility operating license, the following:

1. The Dow Chemical Company Resolution included with the supplement dated January 13, 2012, and the representations made in the application regarding reporting relationships and authority over safety and security issues, shall be adhered to and may not be modified in any respect concerning the decision making authority of the Dow Chemical Company over the Dow TRIGA Research Reactor without the prior written consent of the Director, Office of Nuclear Reactor Regulation.

2. The Vice-President over Research and Development, who is a U.S. citizen, shall have exclusive executive authority over the Dow TRIGA Research Reactor. If at any time the Vice-President over Research and Development is not a U.S. citizen, then at such time the exclusive authority over the nuclear reactor and its special nuclear materials will be transferred to an executive of at least the rank of Vice-President who is a U.S. citizen. This individual shall ensure that the business and activities of the Dow Chemical Company with respect to the Dow TRIGA Research Reactor are at all times conducted in a manner consistent with the public health and safety and common defense and security of the United States.

In light of the above, the NRC staff does not know or have reason to believe that Dow Chemical Company is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government within the meaning of the AEA, and has satisfied the requirements of 10 CFR 50.38.

1.9.4 Nuclear Indemnity

Dow Chemical Company currently has an indemnity agreement with the Commission that does not have a termination date. Therefore, Dow Chemical Company will continue to be a party to the present indemnity agreement following issuance of the renewed license. Dow Chemical Company will be indemnified for any claims arising out of a nuclear incident under the Price-Anderson Act, Section 170 of AEA, and in accordance with the provisions of its indemnity agreement of Appendix B, “Form of indemnity agreement with licensees furnishing insurance policies as proof of financial protection,” to 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements,” up to $500 million.

1.9.5 Conclusion

The NRC staff reviewed the financial status of the licensee and concludes that there is reasonable assurance that the necessary funds will be available to support the continued safe operation of the DTRR and, when necessary, to shut down the facility and carry out the decommissioning activities. In addition, the NRC staff concludes that there are no problematic foreign ownership or control issues or insurance issues that would preclude the issuance of a renewed license.

2. REACTOR DESCRIPTION

2.1 Summary Description

2.1.1 Introduction

The DTRR is a GA TRIGA reactor licensed for a maximum power level of 300 kWt and nonpulsed operation. It is a standard Mark 1 design that provides a variety of irradiation facilities, including a rotary specimen rack, a central thimble, a PTS, a single-element replacement, and a gamma irradiation facility.

The reactor core is located near the bottom of a cylindrically shaped water-filled aluminum tank that is 6.5 ft (2 m) wide and 21.5 ft (6.6 m) high. The tank is bolted at the bottom to a 3.5 ft (1.1 m) thick poured concrete slab. The tank has a minimum wall thickness of 0.25 in. (0.6 cm) and is surrounded by approximately 3 ft (0.9 m) of concrete. The tank and the water provide shielding for personnel. The tank is located below ground level. The column of water above the core also provides a coolant source. The control rod drives are mounted above the tank on a bridge structure spanning the diameter of the tank.

The DTRR uses uranium zirconium hydride (U-ZrH) fuel elements containing 8.5 w% U enriched to less than 20 w% U-235 arranged in a circular array. The reactor power is regulated by inserting or withdrawing neutron-absorbing control rods. Many TRIGA reactors are designed and instrumented to operate in the pulse mode; however, the DTRR has no pulsing equipment or supporting analysis.

The inherent safety of TRIGA reactors has been demonstrated by the extensive experience gained from TRIGA designs used throughout the world. TRIGA fuel is characterized by a strongly negative prompt temperature coefficient characteristic of U-ZrH fuel moderator elements that contributes to safe operation. A series of GA and NRC reports discuss such features as reactor kinetic behavior (GA-7882, "Kinetic Behavior of TRIGA Reactors, dated March 31, 1967 (Ref. 21)); fission product retention (NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," issued August 1987 (Ref. 22), and "The U-Zr_xH Alloy: Its Properties and Use in TRIGA Fuel," M.T. Simnad, 1980 (Ref. 23)); and accident analysis (NUREG/CR-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," issued April 1982 (Ref. 24)).

2.1.2 Summary of Reactor Data

The licensee provided updated neutronics and T-H analyses (Ref. 12). In the neutronics report, the licensee addresses the guidance in NUREG-1537, Section 4.5.1, requesting a limiting core configuration (LCC). The LCC is defined in NUREG-1537 as the core that provides the highest power density. All other core configurations therefore are encompassed within the safety analysis of the LCC. A description of the LCC can be found in the DTRR neutronics report (Ref. 12) and some of the attributes that differentiate the LCC from the current operational core are listed below:

- fuel in the B-ring is replaced with fresh fuel; in addition four low-power elements on the periphery of the core are replaced with graphite reflector elements;

- the reactor power is kept at 300 kWt; and
- the coolant inlet temperature is assumed to be 60 degrees C.

The LCC has a higher power density than the operational core (OC). The licensee states that this helps ensure that the analysis of normal operating conditions and accidents is bounding for the DTRR.

Table 2-1 presents the basic design parameters and results of the DTRR LCC as provided in the neutronics report (Ref. 12).

Table 2-1 Reactor Parameters for the DTRR LCC Core

Parameter	Result
Licensed reactor power kilowatts thermal (kWt)	300 kWt
Number of fuel elements in core	76
Number of control rods in core	3
Maximum fuel temperature at 300 kWt degrees Celsius (C)	246.70 C
Fuel temperature coefficient, 0-327 C reactivity per degree ($/C)	-0.0181 $/C
Maximum rod power at 300 kWt	6.08 kWt
Average rod power at 300 kWt	3.95 kWt
Departure from nucleate boiling ratio (DNBR) at 300 kWt	6.76
Effective delayed neutron fraction	0.0070
Linear power trip setpoint	300 kWt

The NRC staff's review included a comprehensive examination of the supporting reports. Based on its review, the NRC staff finds that the parameters cited above are reasonable and the differences between calculated and measured values also are reasonable.

2.1.3 Experimental Facilities

The DTRR experimental facilities are described in the DTRR SAR, Section J. DTRR has multiple incore irradiation facilities to facilitate a broad range of potential experimental activities. These facilities include a rotary specimen rack, a central thimble, a pneumatic transfer system (PTS), and individual fuel element locations.

The central thimble is located in the central fuel element position. A special tube is provided to accommodate samples and can be placed in the central fuel element position through a cable. The dimensions of the central thimble are the same as a fuel element.

A PTS is available for use at the DTRR. The specimen capsule is installed within a tube and is driven by the force of dry, compressed helium. The PTS has a slight curve in its tube to prevent direct streaming of neutrons from the core to the surface of the pool. The DTRR PTS is designed to quickly transfer individual specimens in and out of the reactor core. The specimens are placed in a small enclosed polyethylene holder, called a rabbit, which is then placed into the receiver. It travels through aluminum and plastic tubing to the terminus at reactor core centerline, and returns along the same path to the receiver. Directional gas flow moves the rabbit between the receiver and terminus. A compressed gas system supplies helium to the PTS and a solenoid valve directs flow. Controls to operate the compressed gas and solenoid valve are located on the control console.

The DTRR has a rotary specimen rack, commonly called a "lazy susan," which is integral to the radial graphite reflector assembly. The rack may be rotated (repositioned) manually from the top of the reactor, and a motor allows continuous rotation at about 1.17 revolutions per minute.

2.1.3.1 TS 3.7.1 Reactivity and Position Limits

TS 3.7.1 states the following:

> 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 of 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.

TS 3.7.1, Specification a, helps ensure that the $1.00 reactivity worth limit imposed on the total worth of all experiments will not result in an inadvertent reactor pulse. The DTRR is not licensed for pulsing operation. The results of the supporting analysis are described in Section 4.1.2 of this SER, including the NRC staff's independent confirmatory analysis, which demonstrate that as much as $1.50 of reactivity inadvertently inserted into the DTRR reactor would not result in a maximum fuel temperature in excess of either the TS safety limit (SL) or TS limiting safety system setting (LSSS).

TS 3.7.1, Specification b, helps ensure that experiments with the potential to add a reactivity insertion into the reactor greater than $0.75 are secured from inadvertent movement. The analysis presented in the DTRR T-H report, Section 3 (Ref. 12), indicated that a $0.75 increase in reactivity to the reactor at full power would not result in the fuel temperature exceeding the TS SL or the LSSS. The NRC staff reviewed the licensee's analysis and methodology, including a confirmatory calculation. The results of the NRC staff's review are provided in Section 4.1.2 of this SER and, are acceptable.

TS 3.7.1, Specification c, helps ensure that experiments may not occupy adjacent positions in the B- and C-rings. Experiment reactivity often is estimated from past experimental experience; and TS 3.7.1, Specification c, helps ensure that the licensee will not introduce multiple experiments of estimated value in adjacent positions that could result in an unexpected net effect on the DTRR reactivity insertion. Thus, TS 3.7.1, Specification c, helps prevent experimental reactivity combinations in the B- and C-rings and helps to maintain the accuracy of the DTRR estimated experimental reactivity.

The NRC staff reviewed the reactivity limits established in TS 3.7.1, Specifications a through c above, and determined that the specifications are based on adequate evaluations of reactivity insertions for the DTRR. The NRC staff has evaluated the reactivity insertion scenario described in Section 4.1.2 of this SER, including a confirmatory calculation, and concluded that the results were acceptable. The NRC staff finds that TS 3.7.1 helps ensure that excess reactivity introduced by experiments are properly controlled by the DTRR staff. The NRC staff also finds that TS 3.7.1 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors" (Ref. 25). Therefore, based on the information provided above, the NRC staff concludes that TS 3.7.1, Specifications a through c, are acceptable.

2.1.3.2 TS 3.7.2 Materials

TS 3.7.2 states the following:

> <u>Specification</u>
>
> 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.

TS 3.7.2, Specification a, follows the guidance provided in NUREG-1537, Appendix 14.1, Section 3.8.2, and helps ensure that double encapsulation is required for the stated materials as a method to reduce the likelihood of encapsulation failure which could allow the corrosive materials to potentially damage the fuel cladding.

TS 3.7.2, Specification b, helps ensure that potentially explosive material will not produce a detonation that could damage the reactor or reactor components. Explosive material greater than 25 milligrams may not be irradiated. Explosive material up to 25 milligrams may be irradiated, provided the pressure produced on detonation of the explosive has been calculated or experimentally demonstrated to be less than half the design pressure of the irradiation container. For this small amount of explosive, a calculation can be used instead of an experiment. This specification helps ensure that no damage to the fuel cladding will result because of an experiment containing explosive material. The NRC staff finds that this specification is consistent with the recommendations of NRC RG 2.2, "Development of Technical Specifications for Experiments in Research Reactors," issued November 1973 (Ref. 26), and in NUREG-1537, Appendix 14.1, Section 3.8.2. Therefore, the NRC staff concludes that TS 3.7.2, Specification b, is acceptable.

The NRC staff finds that TS 3.7.2 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that TS 3.7.2 helps ensure that potentially corrosive, explosive, and fissionable materials introduced into the reactor by experiments are properly controlled by DTRR staff in order to minimize the potential likelihood that a failure could damage the fuel cladding or result in unacceptable doses to workers or the public. Therefore, based on the information provided above, the NRC staff concludes that TS 3.7.2 is acceptable.

2.1.3.3 TS 3.7.3 Experiment Failure and Malfunctions

TS 3.7.3 states the following:

> 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 10 CFR 20.1201. Additionally, exposures to members of the public to these releases in the unrestricted areas will not exceed the dose limits in 10 CFR 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.

TS 3.7.3 addresses the potential for failures and malfunctions of experiments by requiring assumptions for experiments that will help ensure that the source term calculations are conservative such that if an experiment failure or malfunction should occur, the gases or aerosols released will not result in exceeding limits of 10 CFR Part 20. The NRC staff finds that TS 3.7.3 helps ensure that the radiological consequences of experiment failure are adequately considered and the quantity of material introduced is limited and properly controlled by the DTRR staff. In addition, the NRC staff finds that the assumptions cited in TS 3.7.3, Specifications a through e, follow the guidance in NUREG-1537, Appendix 14.1, Section 3.8.3. Therefore, based on the information provided above, the NRC staff concludes that TS 3.7.3 is acceptable.

The NRC staff finds that the DTRR experimental facilities are typical of TRIGA reactors and that their use is properly controlled by TSs 3.7.1, 3.7.2, and 3.7.3. Furthermore, based on the information provided above, the NRC staff concludes that the DTRR experimental facilities and TSs 3.7.1, 3.7.2, and 3.7.3, are acceptable.

2.2 Reactor Core

The DTRR core is described in the DTRR SAR, Section D (Ref. 4), and in the neutronics report, Section 2 (Ref. 12). The DTRR core assembly is a right circular cylinder consisting of a compact array of cylindrical fuel-moderator elements, a central thimble, a neutron source, and control rods, all positioned vertically between two grid plates that are fastened to the reflector assembly and then to the support structure. The outer region of the core may contain some graphite reflector elements. A radial reflector surrounds the core and is composed of graphite with a radial thickness of about 12 in. (30.5 cm) encased in an aluminum can. The control rods pass through guide tubes inserted through the top grid plate and attached to the bottom grid plate through a locking device. Natural convection of the water cools the core.

The DTRR core components are contained between top and bottom aluminum grid plates. The top grid plate has 91 lattice positions in 5 concentric rings around a central thimble, as illustrated in Figure 2-1 and provided as Figure 8 in the DTRR neutronics report. The fuel elements consist of a zirconium-hydride moderator that is homogeneously combined with enriched uranium.

The DTRR core fuel consists of two types: there are 79 stainless-steel clad $U\text{-}ZrH_{1.6}$ fuel elements and 1 aluminum-clad $U\text{-}ZrH_{1.0}$ fuel element, as described in the neutronics report as the “2011 core configuration” and shown below as Figure 2-1. The location of the individual elements is described in greater detail in Table A1 of the neutronics report (Ref. 12). The hydrogen (H) to zirconium (Zr) ratio is represented by the “x” in the $U\text{-}ZrH_x$ nomenclature, and is referred to as the fuel stoichiometry ratio. The H content is important as it influences many of the temperature characteristics of the fuel and is explained in Section 2.2.2 of this SER.

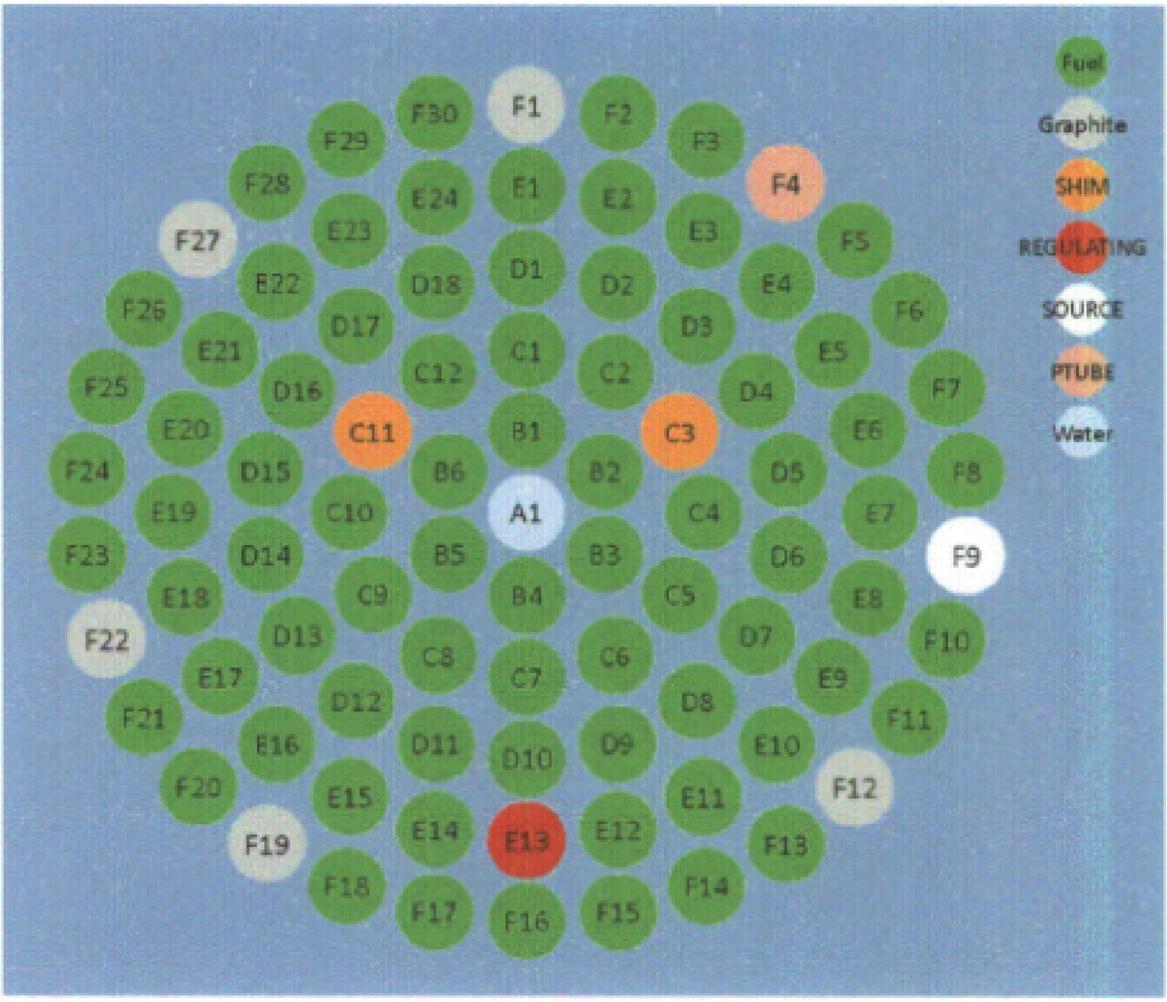

Figure 2-1 The DTRR lattice and typical component arrangement

The DTRR TRIGA fuel elements nominally have a 1.475 in. (3.75 cm) clad outer diameter, except for the upper and lower end fixtures, and are approximately 28.375 in. (72 cm) long. The active fuel length is 15 in. (38.1 cm) for stainless-steel fuel elements and 14 in. (35.5 cm) for the one (1) aluminum-clad fuel element. Axial reflector pellets are located within the cladding, at each end of the fuel element, and consist of a graphite segment that is 3.45 in. (8.8 cm) long. The aluminum-clad graphite reflector, which is part of the rotary specimen rack, and the graphite reflector elements that occupy certain lattice positions provide the neutron reflection in the radial direction. The TRIGA fuel elements are described in detail in the licensee's neutronics report, Section 2 (Ref. 12).

The DTRR power level is controlled using three control rods called the Regulating rod, Shim1 rod, and Shim2 rod. These control rods use a boron carbide powder that provides for strong neutron absorption. The licensee provided a detailed description of the control rods in its neutronics report, Section 2 (Ref. 12), and in SAR, Section D (Ref. 4).

The DTRR is operated with a 2.0 curie americium (Am)-beryllium (Be) neutron startup source located in DTRR core lattice position F9. Any loss of integrity of the source cladding would be indicated by the radiation detector located in the water treatment line and by the monthly samples of the reactor pool water (RAI-10, Ref. 4).

2.2.1 TS 5.3 Reactor Core and Fuel

TS 5.3 states the following:

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.

TS 5.3, Specifications 1 and 2, help ensure that only TRIGA fuel elements, either stainless-steel or aluminum-clad, are used in the DTRR TRIGA and placed in a close packed array. These specifications are consistent with the neutronics report (Ref. 12), and help ensure that the fuel elements used in any operating core (OC) configurations remain consistent with the assumptions provided in the LCC analysis.

TS 5.3, Specification 3, helps ensure that any use of aluminum-clad fuel elements, which have lower temperature limits than the stainless-steel clad fuel elements, are only located in the two outer rings of the DTRR core lattice. This location will provide a lower power density and thus a lower fuel temperature, as demonstrated in the DTRR neutronics report (Ref. 12). This specification requirement provides an additional level of conservatism for the use of the aluminum-clad fuel elements.

TS 5.3, Specification 4, helps ensure that the Shim1, Shim2, and the Regulating control rods have scram capability and are constructed with materials that have well-established nuclear characteristics for the absorption of neutrons. These control rod characteristics are consistent with TRIGA facilities and the assumptions used in the DTRR neutronics report (Ref. 12).

TS 5.3, Specification 5, helps ensure that the reflector employed in any OC configuration consists of graphite and water. This specification is consistent with the assumptions in the DTRR neutronics report (Ref. 12), which states that DTRR TRIGA fuel was used in conjunction with a radial graphite reflector block and interstitial water.

TS 5.3, Specification 6, helps ensure that the DTRR core structural components are composed of aluminum or stainless steel. This specification ensures that the structural materials used in the DTRR core system will minimize the introduction of corrosion products into the pool water, which could then become radioactive and result in elevated radiation levels and worker radiation dose exposure.

TS 5.3, Specifications 7 and 8, help ensure that the reactivity requirements established for the reactor system before removal or insertion of fuel elements or removal of control rods are acceptable. Fuel elements shall not be inserted or removed unless the reactor is subcritical by more than the worth of the most reactive fuel element, and control rods shall not be removed unless the reactor is subcritical with all control rods removed from the core. These specifications help ensure that such activities are preceded by placing the core in a subcritical condition that prevents inadvertent criticality.

The NRC staff finds that TS 5.3, Specifications 1 through 8, characterize the DTRR core design features and help ensure that core loading conforms and is limited to the analysis provided in the DTRR SAR (Ref. 4) and the neutronics report (Ref. 12). TS 5.3 helps ensure that excessive power densities will not result from any allowed core loading, core components are adequate to control the power level of the DTRR, core components do not create unnecessary radioactivity, and the reactivity of the DTRR core is controlled during core alterations so that the core remains subcritical. The NRC staff finds that TS 5.3 is consistent with the guidance provided in NUREG-1537, Section 4.5.1, which recommends that the licensee identify the highest power density of any possible core arrangement. The NRC staff finds that TS 5.3 is consistent with the reactor core and fuel as described in the DTRR SAR and the neutronics report. Furthermore, TS 5.3 characterizes the DTRR design features for the reactor core and helps ensure that the operating core loading conforms to the LCC. Therefore, based on the information provided above, the NRC staff concludes that TS 5.3 is acceptable.

The DTRR LCC and normal operating core characteristics, from the DTRR neutronics report (Ref. 12), are provided in Table 2-2.

Table 2-2 DTRR Reactor Core Elements

Core Item	Number in DTRR OC	Number in DTRR LCC	Core Location
Stainless-steel fuel elements	79	75	various
Aluminum fuel element	1	1	F28
Shim1 control rod	1	1	C11
Shim2 control rod	1	1	C3
Regulating rod	1	1	E13
Graphite reflector elements	5	9	F1,F12,F19,F22,F27
Source	1	1	F9
Central thimble	1	1	A1
Pneumatic transfer system (rabbit)	1	1	F4
Total	91	91	
Operational Parameters			
Hot fuel element position	B6	C6	
Average predicted fuel element (kWt)	3.75	3.95	
Maximum predicted fuel element (kWt)	5.91	6.08	

The NRC staff reviewed the DTRR LCC, as shown in Figure 2-2 of this SER, and based on the DTRR neutronics report, Figure 15 was used to determine the results of the DTRR accident analyses. The DTRR LCC used 75 fuel elements in the 300 kWt core. Since four fewer fuel elements were used in the analysis, the average power in each fuel element increased from 3.75 kWt (300 kWt/79 fuel elements) for the 2011 core configuration to 3.95 kWt for the LCC. The peak fuel element power was located in lattice position C6. The DTRR neutronics report, Section 3.6, established the maximum fuel element power that could be expected for the DTRR core. The result for the DTRR LCC was a limiting fuel element power in the lead fuel element of 6.08 kWt for DTRR under any steady-state circumstances. The NRC staff concludes that the DTRR LCC is consistent with the guidance in NUREG-1537, Section 4.5.1.

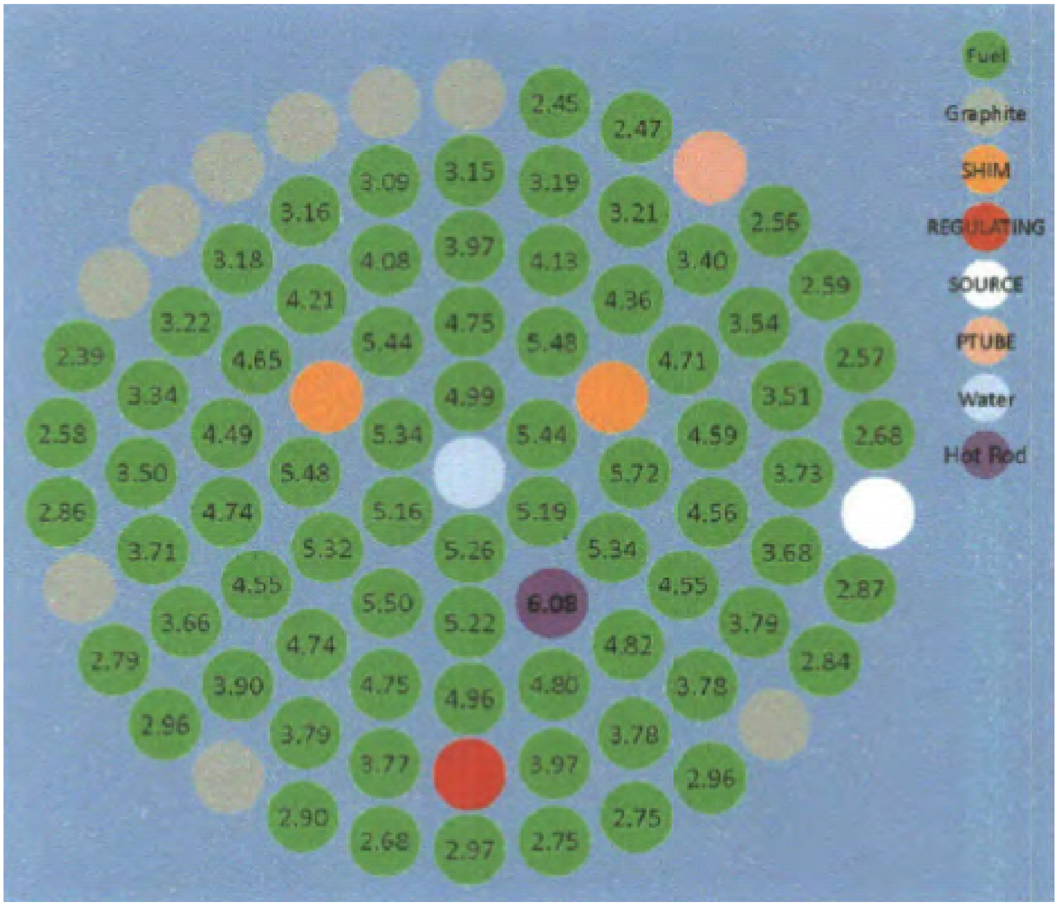

Figure 2-2 The DTRR LCC

The NRC staff reviewed the Monte Carlo Neutron Transport (MCNP) code results, provided in the DTRR neutronics report as Figure 15 (Ref. 12), which was used to estimate the fuel depletion in the DTRR core. MCNP code established the average intra-rod power distribution for the nine designated fuel segments within each element. Using this distribution, the U-235 of each fuel element was reduced until agreement was achieved between MCNP and the as-measured core excess reactivity for the original critical DTRR core. The MCNP code resulted in a U-235 depletion of 10.55 percent for each element, which provided a satisfactory agreement between the calculated \$0.21 system reactivity and the measured \$0.13 system reactivity. The fuel depletion history is provided in Table 2-3.

Table 2-3 DTRR Fuel Depletion History

Applicable Years	Energy (MWd)
1967–1991	9.60
1991–1997	8.51
1997–2001	6.59
2001–2005	6.94
2005–2011	8.70
Total to date	40.34

A schematic of the DTRR MCNP model is provided in Figure 2-3 (reproduced from the DTRR neutronics report, Figure 1). The NRC staff reviewed the schematic and concludes that it provides an accurate portrayal of the DTRR core size and position of the reactor features, such as the lazy susan and the reflector assembly.

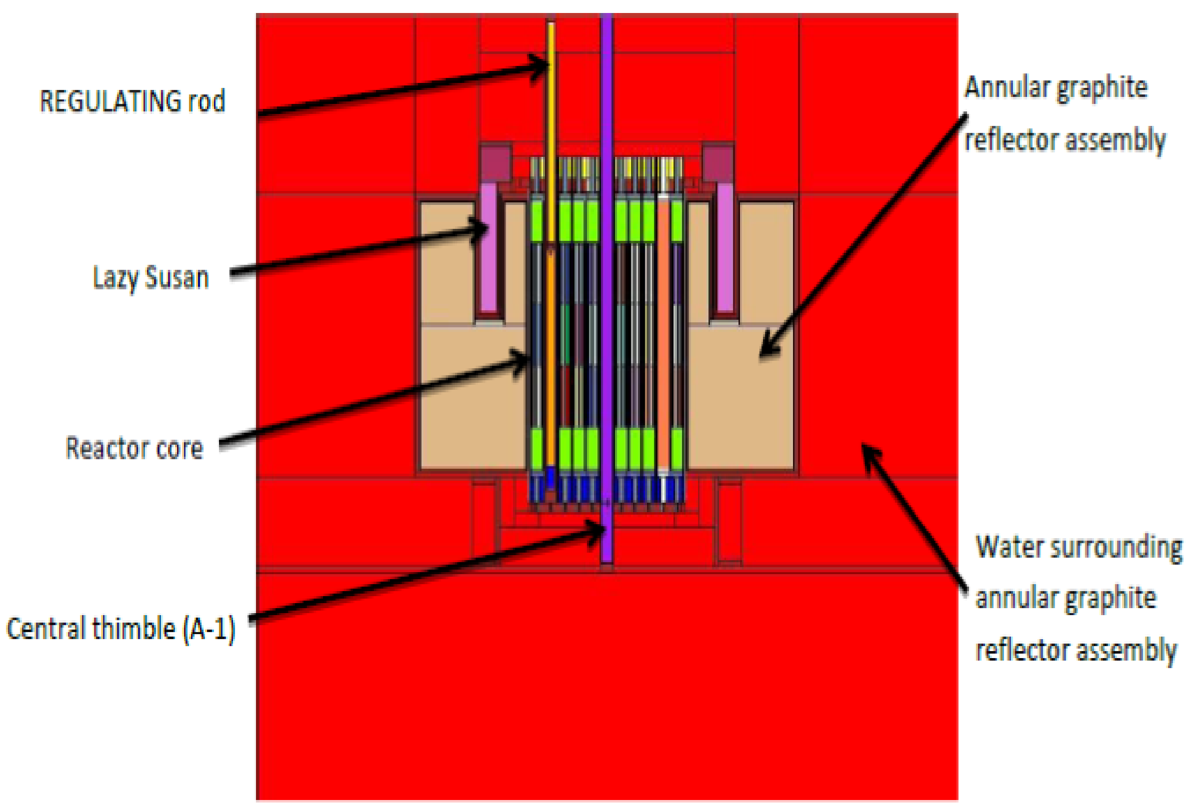

Figure 1. Vertical cross section of the MCNP5 model developed for the DTRR.

Figure 2-3 Schematic of the MCNP model of DTRR

Based on a review of the information that the licensee provided in the DTRR SAR (Ref. 4) and neutronics report (Ref. 12), the NRC staff finds that the licensee has accurately described the LCC used in the DTRR, including design limits, and the bases for these limits. The licensee has also adequately provided and discussed the constituents, materials, and components for the OC, and the design features of the DTRR LCC. The LCC MCNP analysis properly reduces the amount of U-235 to replicate core operational features associated with fuel depletion. The MCNP model incorporated all of the required features cited in the neutronics report that provided the results of the LCC analysis. The NRC staff finds that compliance with the applicable TS will ensure uniform core operational characteristics and operation in compliance with design bases and safety-related requirements. The NRC staff also finds that the licensee's analysis of the LCC provides sufficient margin of fuel element power density for DTRR OC configurations to function safely for the renewal period. Therefore, based on the information provided above, the NRC staff concludes that the licensee's analysis of the LCC and TS 5.3 is acceptable.

2.2.2 Reactor Fuel

The DTRR uses cylindrical stainless-steel and aluminum-clad fuel elements in which the fuel is a solid homogeneous mixture of ZrH-hydride alloy containing nominally 8.5 w% uranium enriched to less than 20 w% U-235. The design details of both fuel elements are described in Section 2 of the neutronics report (Ref. 12). The active part of the fuel element is shown in Figure 1 of the neutronics report (Ref. 12). Therein, the stainless-steel fuel is described as having a 0.25-in. (0.64 cm) hole in the center that is filled with a Zr rod.

NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," August 1987 (Ref. 22) summarizes the characteristics of TRIGA fuel types. As indicated in the DTRR neutronics report, DTRR uses only the original fuel type whose characteristics are given in Table 2-4.

Table 2-4 TRIGA Fuel Characteristics

Type of Fuel	w% Uranium	w% Erbium	U-235 (w%)	$\alpha_F \times 10^5$ ($\Delta k/k$- C)	Core Lifetime (MWd)	Uranium (volume %)
Original	8.5	0.0	20	9.5	100	2.6

The licensee indicated in the 2011 neutronics report that the total DTRR fuel depletion was approximately 40.34 megawatt days (MWd), which is within the characteristics provided in NUREG-1282. In addition, DTRR fuel elements do not contain burnable absorbers (poisons), such as erbium, which makes the calculation of excess reactivity and shutdown margin (SDM) more linear with burnup.

In Section 2 of the neutronics report (Ref. 12), the licensee stated that the DTRR fuel currently consists of two types: 1 aluminum-clad fuel element containing U-$ZrH_{1.0}$ fuel and 79 stainless-steel clad fuel elements containing U-$ZrH_{1.6}$ fuel. The H-to-Zr stoichiometry ratio of the fuel is represented by the "x" in the U-ZrH_x nomenclature (i.e., the ratio of H to Zr). The H content is important because it influences many attributes of fuel behavior. Figure 2-4 illustrates the U-ZrH fuel matrix phase diagram for a range of fuel stoichiometry referenced in NUREG-1282 and presented in the Simnad report (Ref. 23). The LCC for the DTRR used both stainless-steel (with a nominal stoichiometry ratio of 1.6) and aluminum-clad fuel (with a nominal stoichiometry ratio of 1.0). The significance of the stoichiometry is discussed below.

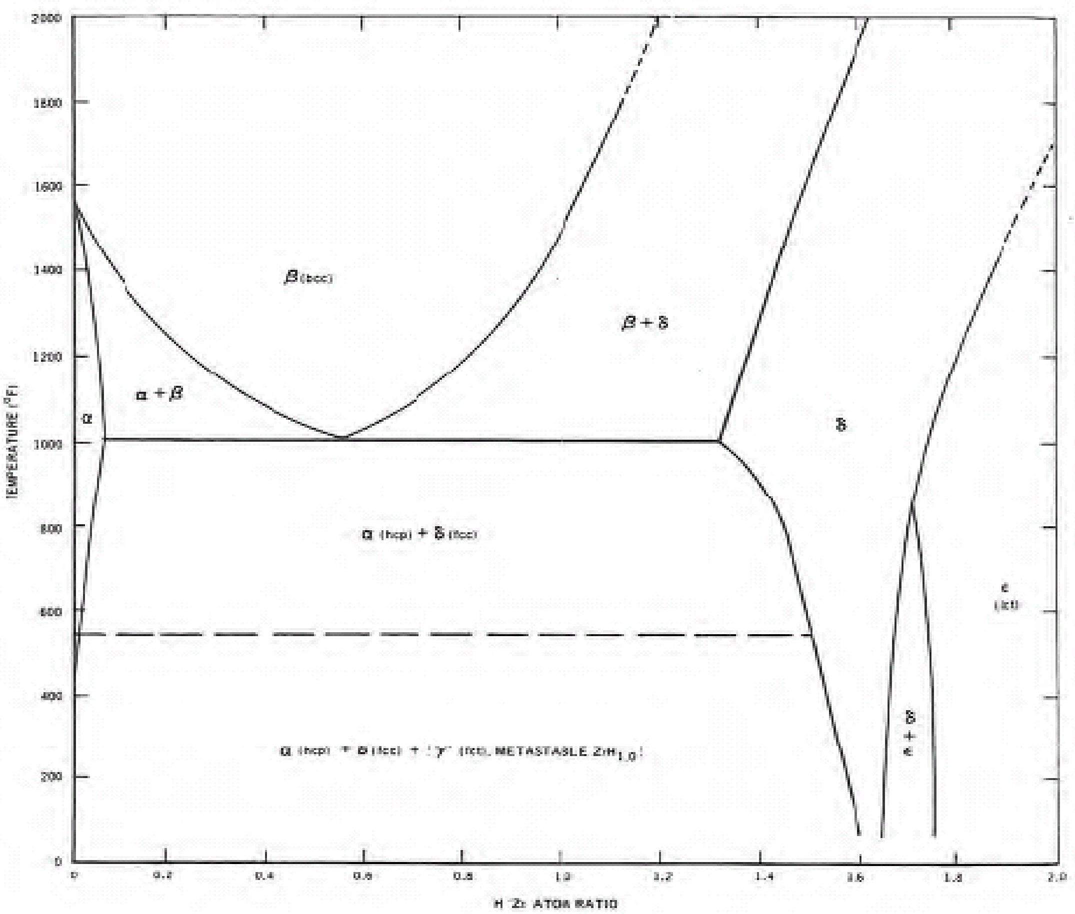

Figure 2-4 Phase diagram for zirconium-hydride fuel

2.2.2.1 TS 2.1 Safety Limit – Fuel Element Temperature

TS 2.1 states the following:

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

TS 2.1 provides the SL for the DTRR fuel elements, which is consistent with the guidance provided in NUREG-1537, Appendix 14.1, for TRIGA fuel elements. NUREG-1537 provides guidance that states that a peak fuel temperature limit of 500 degrees C for aluminum-clad fuel elements and 1,150 degrees C for stainless-steel clad fuel elements are acceptable.

These different peak fuel temperature limits exist because of different clad failure mechanisms for aluminum-clad U-$ZrH_{1.0}$ and stainless-steel clad U-$ZrH_{1.6}$ fuel. For aluminum-clad U-$ZrH_{1.0}$ fuel, the 500 degrees C SL provides a safety margin to the temperature at which a phase change takes place in the aluminum-clad U-$ZrH_{1.0}$ fuel (alpha plus delta to beta plus delta phase change at approximately 537 degrees C). The higher temperature beta plus delta fuel phase occupies a greater volume than the lower temperature alpha plus delta fuel phase. The force on the clad of the "swelling" of the fuel leads to cladding failure (similar to the results caused by the phase change of water freezing in a closed container).

For high-hydride-type (U-$ZrH_{1.6}$), stainless-steel-clad elements calculations performed by General Atomics and confirmed by experiments indicate that no cladding damage occurs at peak fuel temperatures as high as approximately 1175 degrees C (2150 degrees Fahrenheit (F)). The fundamental consideration in limiting fuel temperature to 1150 degrees C is to provide a safety margin to limit the fuel element internal pressure caused by the buildup of hydrogen gas (resulting from the diffusion of the hydrogen out of the zirconium-uranium matrix) and fission products within the fuel element gap. Limiting the maximum fuel temperature prevents generating excessive internal pressures from the gas, which will increase the stress on the clad beyond the yield point, thereby causing a clad rupture and release of fission products.

The DTRR fuel inventory consists of 79 stainless-steel and 1 aluminum-clad fuel elements. The SL proposed by the licensee is the more limiting of the aluminum-clad or stainless-steel clad fuel. Therefore, the selection of 500 degrees C as the SL for DTRR is adequate to ensure that the aluminum-clad fuel element temperature limit is maintained to protect the integrity of the fuel cladding. The SL provides a safety margin to the temperature at which phase changes could take place in the aluminum-clad U-$ZrH_{1.0}$ fuel (approximately 537 degrees C) and the temperature at which internal gas pressure in stainless-steel clad U-$ZrH_{1.6}$ fuel could cause clad failure (approximately 1,175 degrees C). The safety margin is enhanced further by the implementation of DTRR TS 5.3, Specification 3, which helps ensure that the aluminum clad fuel element is placed in low power, and thus lower temperature locations by placement in the outer rings of the DTRR core lattice.

The NRC staff finds that TS 2.1 is consistent with the guidance in NUREG-1537, Section 2.1 of Appendix 14.1, and therefore, helps ensure that the fuel element cladding integrity is maintained to lessen the likelihood for the potential for cladding failure and the release of fission products. Therefore, based on the information provided above, the NRC staff concludes that TS 2.1 is acceptable.

2.2.2.2 TS 3.2 Reactor Fuel Parameters Limits

TS 3.2 states the following:

<u>Specification</u>

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.

TS 3.2, Specifications a through d, establish inspection requirements to detect gross failure or visual deterioration of the DTRR fuel. The fuel element attributes inspected include fuel element transverse bend and length, and a visual inspection for bulges or other cladding defects. The NRC staff finds that TS 3.2, Specifications a through d, provide limits on transverse bend and length, and fuel burnup, which are consistent with the guidance provided in NUREG-1537, Appendix 14.1. Based on the information provided above, the NRC staff concludes that TS 3.2, Specifications a through d, are acceptable.

The NRC staff reviewed the DTRR SAR (Ref. 4) and the neutronics report (Ref. 12), which described the fuel elements used in the DTRR, their design limits, and the technological and safety-related bases for these limits. The NRC staff finds that the licensee also adequately discussed the constituents, materials, and components for the fuel elements. The NRC staff also finds that compliance with the applicable TS will help ensure uniform core operating characteristics and adherence to the design bases and safety-related requirements. Based on the information provided above, the NRC staff concludes that the DTRR fuel elements and their associated TS are acceptable.

2.2.3 Control Rods

The DTRR control rods are described in the DTRR SAR (Ref. 4) and neutronics report (Ref. 12). The control rods consist of boron carbide powder neutron-absorbing material and are consistent with the control rods used at other TRIGA reactors. The boron carbide powder neutron absorber is enclosed in aluminum tubes that are 0.875 in. (2.2 cm) (Regulating) and 1.25 in. (3.2 cm) (Shim1 and Shim2) in diameter, as described in the DTRR SAR, Section D.2.2. The Regulating rod is used for fine control during the DTRR operation. The control rod drive's safety function is to accurately position the rod vertically on console operator command and to release the control rod on scram conditions.

The control rods pass through normal fuel positions in the DTRR core on the top and bottom of the grid plates. Guide tubes ensure that the control rods remain properly aligned. The Shim1, Shim2, and Regulating control rods are located at core locations: C11, C3, and E13, respectively.

Each control rod has a drive that consists of a stepping motor, a magnet rod-coupler, a rack and pinion gear system, and a potentiometer used to provide an indication of rod position. The pinion gear engages a rack attached to a draw-tube that supports an electromagnet. The magnet engages a chrome-plated armature attached above the water level to the end of a connecting rod that fits into the connecting tube. The connecting tube extends down to the control rod. The magnet, its drawtube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the control rod drive mounting plate with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod drive assembly. The lower section of the barrel contains an air snubber to dampen the shock of the scrammed rod. In the snubber section, the control rods are decelerated through the last 3 in. (7.6 cm) of their travel.

Each control rod is held in place by electromagnets. When a scram is initiated, the current to the electromagnets is cut, the armature is released, and the control rod drops by gravity into the core. If there is a loss of power event, the control rods are released independently of operator action as a result of the action of the safety circuits, as explained in the response to RAI-37 (Ref. 4). All of the DTRR control rods are scrammable. A control rod can be withdrawn from the reactor core only when the electromagnet is energized. The withdrawal speed of the rod is adjustable. The position of each control rod is displayed on the operator console. The response to RAI-9 (Ref. 4) provides additional detail regarding control rod positions to supplement the description provided in the DTRR SAR, Section D.2.2.

The DTRR has an automatic power control system as described in DTRR SAR, Section G.3.3. The system consists of a servo amplifier that uses a signal from the wide-range nuclear channel, a power-demand signal set by the reactor operator (RO), and the derivative signal from the period circuit of the log power channel. The servo compares the reactor power with the power demand set by the RO and adjusts the regulating rod position in or out, thereby maintaining reactor power. The system also can be used to change power level. During operation of this system, the reactor power increase is automatically limited to the minimum allowed by a reactor period of 7 seconds.

The licensee has incorporated the applicable control rod design features in TS 3.1, TS 4.3, and TS 5.3 to help ensure that the control rods are capable of performing their intended functions. The NRC staff's review of these TSs is provided in Sections 2.5, 5.4, and 5.2 of this SER.

The NRC staff reviewed the design and performance of the control rods and determined that the licensee has demonstrated that the control rods provide adequate reactivity worth, structural rigidity, and reliability to ensure reliable operation under all operating conditions. The NRC staff finds that the control rods have the ability to scram without challenging the integrity of other reactor systems. The control rod materials have been used in many other TRIGA reactors and have demonstrated reliable operation and service life. The design of these control rods meets the DTRR design requirements.

Based on a review of the information that the licensee provided in the SAR, and the results of the NRC staff review provided above, the NRC staff finds that the control rods conform to the applicable design bases and can shut down the DTRR from any operating condition or applicable accident scenario. The control rod design for the DTRR includes reactivity worths that can control the excess reactivity planned for the DTRR, including assurance of acceptable shutdown reactivity and margin. The licensee has acceptably described the control rods used in the DTRR, including design limits, and the technological and safety-related bases for these limits. The licensee has also acceptably discussed the constituents, materials, and components for the control rods. Compliance with the applicable TSs will ensure uniform characteristics and compliance with design bases and safety-related requirements. Based on the information provided above, the NRC staff concludes that the DTRR control rods are acceptable.

2.2.4 Neutron Moderator and Reflector

The moderator and reflector in the DTRR are described in DTRR SAR, Section D.2.3 (Ref. 4), in the licensee's response to RAI-9 (Ref. 4), and in the neutronics report (Ref. 12). The moderator is described as consisting of both the contributions from the H and Zr in the fuel matrix and the water present in the core region. This combination of materials is sufficient to moderate the neutron fission spectra and thermalize the neutrons. The reflector surrounding the core includes the rotary specimen rack (lazy susan), which is a ring-shaped block of graphite with an inside diameter of approximately 18 in. (46 cm), a radial thickness of 12 in. (30 cm), and a height of 22 in. (56 cm). A welded aluminum container keeps water from contact with the graphite. The reflector ring assembly rests on the reflector platform. Additionally, the outer row of the core lattice contains graphite reflector elements. These elements are of the same dimensions as the fuel moderator elements, but they are filled entirely with graphite. These graphite elements also serve as radial reflectors. The fuel elements contain two sections of graphite, one above and one below the fuel, which serve as top and bottom reflectors for the core. The reflector elements are defined in the MCNP model described in Section 2 of the DTRR neutronics report. The NRC staff finds that the contribution to system reactivity is described and accounted for appropriately in reactor measurements and calculations provided in the SAR.

Based on its review of the information provided in the DTRR SAR and discussed above, the NRC staff finds that the moderator and reflector elements used in the DTRR are consistent with other TRIGA reactors. The NRC staff reviewed the constituents, materials, and components for the reflector elements and concludes that they are in agreement with the description provided in the DTRR SAR and modeled adequately by the licensee's MCNP analysis. Based on the information provided above, the NRC staff concludes that the DTRR moderator and reflector elements are acceptable.

2.2.5 Neutron Startup Source

The DTRR Am-Be neutron startup source was described in the DTRR SAR, Section D.2.4 (Ref. 4), and in licensee's response to RAI-10 (Ref. 4). The startup source has an activity of 2 curie (Ci) and is similar to those used in other TRIGA reactors. The Am-241 has a half-life of 432 years and emits an alpha particle through radioactive decay. The alpha particle is absorbed by the Be-9 nucleus, which then emits a neutron. The source is clad in stainless-steel and designed to withstand the chemical, thermal, and radiation environment presented in a TRIGA reactor. The licensee stated that the loss of integrity of the source cladding would be indicated by an increased count rate from the Geiger counter installed in the water treatment line, or from monthly pool water sampling as required by TS 4.4, Specification 1.

TS 3.3, Table 3.3.A, contains an operational interlock that restricts control rod withdrawal if there is an associated neutron count rate of less than 2 counts per second (cps). One of the primary functions of a neutron source is to provide sufficient counts so that instrumentation will function properly during reactor startup. The NRC staff finds that the DTRR TS 3.3, Table 3.3.A requirement, helps ensure that the detector provides sufficient indication of neutrons to provide controlled reactor startup conditions.

The NRC staff reviewed the information described above and finds that the licensee has acceptably described the Am-Be neutron startup source used in the DTRR. Therefore, the NRC staff concludes that the DTRR neutron startup source is appropriate for use in the DTRR and, is acceptable.

2.2.6 Core Support Structure

The DTRR core support structure is described in the DTRR SAR and in the licensee's response to RAI-11 (Ref. 10). The core components are contained between top and bottom aluminum grid plates. The plates have 91 total positions; 5 concentric rings around a central port. The coolant flow is provided by 36 smaller coolant passages in the bottom core plate. The NRC staff finds that the core support arrangement helps ensure a stable and reproducible DTRR core reactivity. Sufficient coolant flow is provided as demonstrated in the DTRR T-H report (Ref. 12), and the confirmatory analysis that the NRC staff provided, which are discussed in Section 2.6 of this SER.

Based on the information provided in the DTRR SAR, and described above, the NRC staff finds that the DTRR reactor core components are typical of TRIGA reactors, will be capable of maintaining the DTRR fuel element geometry acceptable for all anticipated operating and accident conditions, and will provide adequate coolant flow to the fuel elements. Based on the information provided above, the NRC staff concludes that the DTRR core support structure is acceptable.

2.3 Reactor Tank or Pool

The DTRR SAR, Section D.3, provides a description of the DTRR reactor tank. The DTRR core is located near the bottom of a cylindrically shaped water-filled aluminum tank that is 6.5 ft (2 m) in diameter and 21.5 ft (6.6 m) deep. The tank is placed on a 3.5 ft (1.1 m) thick concrete slab. The tank is located within a 3 ft (0.9 m) concrete shell that is encased within a corrugated steel shell. The tank is filled with demineralized water to a depth of 16 ft (4.9 m), which provides shielding above the top of the core. An alarm is provided that indicates if the water level falls to a depth of 15 ft, 10 in. (4.8 m). The DTRR pool is open to the atmosphere of the reactor bay. Natural circulation of the reactor tank water cools the DTRR core. The volume of the tank is approximately 5,000 gallons (18,927 liters (l)). There are no beam ports or other penetrations in the tank. In the event that the reactor tank should develop a leak, the reactor pool level alarm provided by TS 3.3, Table 3.3B, would alert the operators. In the licensee's response to RAI-3 (Ref. 47), the licensee proposed that TS 3.4, Specification 3, limits the radioactivity of the reactor pool water not to exceed the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, for radioisotopes with half-lives greater than 24 hours. This requirement helps ensure that any potential dose resulting from an inadvertent leak of reactor pool coolant would not result in a radiation exposure to any member of the public in excess of 50 millirem (mrem).

The DTRR is a natural convection water-cooled pool type reactor. Based on the size and low power rating of the DTRR (300 kWt), operation of the primary coolant system to remove heat from the pool water is not required as a safety system for the facility, but it is used to maintain efficient reactor operation and water quality. TS 3.4, Specification 5, helps to limit the reactor coolant system temperature limit of 60 degrees C in order to preserve the assumption used in the T-H report (see SER Section 2.6). The water in the reactor pool is used to moderate the reactor, cool the fuel elements during reactor operation, and shield against radiation coming from the operating reactor core. The primary cooling system is used to remove the heat generated during operation, remove any particulate and soluble impurities, maintain low conductivity, control the H ion activity (pH), and maintain optical clarity.

Based on a review of the information provided in the SAR, Section D.3, and described above, the NRC staff finds that the design of the DTRR reactor tank will provide adequate cooling and shielding for the DTRR. The NRC staff also finds that the DTRR reactor tank, including the constituents, materials, and components of the reactor tank, are typical of other TRIGA reactors. In the unlikely event of a leak of primary coolant from the reactor tank, TS 3.4, Specification 3 helps ensure that any potential radiation doses to any members of the public would be within regulatory limits. Therefore, based on the information provided above, the NRC staff concludes that the design of the DTRR reactor tank is acceptable.

2.4 Biological Shield

The DTRR biological shield was described in the DTRR SAR and the licensee's response to RAI-13 (Ref. 4). The shield consists of the reactor pool water, the tank, and the surrounding concrete. The DTRR core is at the bottom of the reactor tank, which is covered by at least 16 ft (4.9 m) of water. The pool is surrounded by reinforced concrete and is almost totally underground. There are no accessible rooms or facilities below the floor level of the reactor. This combination of water, tank, and concrete shell, act as a biological shield. The licensee's response to RAI-13 (Ref. 4) provided information on typical measurements of exposure detected at various locations within the reactor bay resulting from DTRR operation and is reproduced in Table 2-5.

Table 2-5 Typical Dose Rates at 300 kWt

Location	Dose Rate
1 ft above the DTRR pool surface	18–30 mrem/hr
At the handrail; edge of the tank	3–6 mrem/hr
Outside the handrail	0.15–3 mrem/hr

The NRC staff reviewed DTRR annual reports and NRC IRs from 2005 through 2013, and finds that the annual releases and worker doses were below the limits of 10 CFR Part 20. The corresponding NRC IRs contained no contradictory findings.

Based on a review of the information provided in the DTRR SAR and the operational experience described above, the NRC staff finds that there is reasonable assurance that the DTRR biological shield design will limit exposures from the reactor and reactor-related sources of radiation so that the limits of 10 CFR Part 20 will not be exceeded. The NRC staff finds that the DTRR biological shield components are typical of TRIGA reactors and adequately described in the DTRR SAR, Section D.4. Based on the information provided above, the NRC staff concludes that the DTRR biological shield design is acceptable.

2.5 Nuclear Design

The reactor design bases, as described in the DTRR SAR, are established by the maximum operational capability for the fuel elements and fuel element configurations. The TRIGA reactor system has five major areas that define the reactor design bases:

- fuel temperature
- prompt temperature coefficient
- control rod worths
- thermal-hydraulics and heat transfer (pool water temperature)
- reactor power

NUREG-1537 indicates that the SL is based on the fuel temperature, which, because of the large negative temperature coefficient of reactivity of the TRIGA fuel, contributes to the safety of the TRIGA reactor. A limit on fuel temperature ensures DTRR operation within the assumptions described in the SAR, as well as below the fuel temperature SL, and helps ensure that the fuel cladding integrity is maintained.

The information discussed in this SER section establishes the design bases for other chapters, the safety analyses, and the DTRR TSs. The following sections provide an assessment of the nuclear design analysis of the DTRR core operating at the licensed power of 300 kWt. The material contained in this review includes the neutronics and T-H reports (Ref. 12) and supplemental information provided in the licensee's responses to RAI-14, RAI-15, RAI-16, and RAI-17 (Ref. 10).

2.5.1 Normal Operating Conditions

As previously described in this SER in Table 2-2, the LCC for the DTRR consists of 1 aluminum-clad fuel element, 75 stainless-steel clad fuel elements, 3 control rods, and 9 graphite reflector elements. Figure 2-2 provides an illustration of the LCC.

The neutronics methods and design analysis were described in the DTRR neutronics report (Ref. 12). The licensee used the MCNP computer program to model the LCC. The MCNP model solves the Boltzman transport equation using a Monte Carlo technique. This technique is extensively benchmarked and widely used in the RTR community for neutronic evaluations. The DTRR core is represented with the core loading diagram shown in Figure 2-2, with dimensions and fuel parameters obtained from drawings supplied with the fuel, and with the reactor documentation as cited in the neutronics report.

The NRC staff reviewed the description of the modeling techniques described in the DTRR neutronics report and noted that significant information from the report was used as input to the DTRR T-H analysis. The T-H methods and design analysis were described in the T-H report (Ref. 12). The licensee used the RELAP-3D computer program to model the limiting fuel element in the LCC. RELAP-3D was also used to model the limiting fuel element under varying power level conditions. The DTRR core was represented with the parameters necessary for determining the limiting fuel element in the LCC as provided by the DTRR neutronics report. The dimensions and fuel parameters are obtained from drawings supplied with the fuel and the reactor documentation.

The NRC staff reviewed the licensee's use of MCNP and RELAP-3D for the DTRR neutronic and T-H core analyses and concluded that the analysis in the DTRR SAR, as supplemented, satisfied all TRIGA operational limits as described in the guidance in NUREG-1537. In addition, the DTRR normal operating conditions were bounded by the limits imposed by the DTRR LCC. Therefore, based on the information provided above, the NRC staff concludes that the DTRR LCC is acceptable.

2.5.2 Reactor Core Physics Parameters

Calculation Methodology

As stated previously, the DTRR core performance parameters were evaluated by the licensee by modeling the DTRR LCC using the MCNP code. The calculated values of the control rod worths, shutdown margin (SDM), and core excess reactivity, were obtained at the licensed power level of 300 kWt. The licensee provided the information in its responses to RAI-14, RAI-15, and RAI-16 (Ref. 10), and in the DTRR neutronics report (Ref. 12).

The NRC staff reviewed the DTRR modeling techniques and finds they were adequate and properly implemented. The NRC staff also finds that the level of accuracy the licensee used to model the DTRR core physical attributes and the level of detail presented in the modeling was acceptable.

The control rod worths, excess reactivity, and SDM evaluations are provided in greater detail below:

Control Rod Worths

The MCNP core model for the DTRR was validated by the licensee by comparing the calculated excess reactivity, SDM, and control rod worths with the corresponding measured values for these parameters in the DTRR 2011 core. The DTRR neutronics report, Section 3 (Ref. 12), summarizes several calculated and measured DTRR reactivity parameters, which are presented in Table 2-6. The comparisons indicate general agreement in that they vary from 4 percent to 12 percent. The main sources of measurement error are attributed to the curve-fitting technique used, the inaccuracy in reading the control rod positions, the power fluctuations occurring during the measurement, and the timing of the measurement process. In consideration of these factors and the MCNP simulation confidence interval, the NRC staff finds that the calculated worths compare acceptably with the measured values as presented in Table 2-6.

Table 2-6 Comparisons of Measured and Calculated 2011 Control Rod Worths

Component	January 2011 Measurement (M) ($)	MCNP Calculation (C) ($)	% Difference [(M-C)/average]
Shim1 control rod	-3.25	-3.65	-11.6
Shim2 control rod	-3.10	-3.35	-7.8
Regulating control rod	-1.15	-1.20	-4.3

<u>Excess Reactivity</u>

The purpose of monitoring excess reactivity is two-fold. First, as a component of the SDM calculation, it is a basic safety requirement. Second, the change in excess reactivity with burnup is predictable and consistent; it is virtually linear if no burnable poisons are used, as in the case of DTRR. This change can be reviewed over time to monitor for reactivity anomalies.

2.5.2.1 TS 3.1 Reactivity Limits

TS 3.1 states the following:

<u>Specifications</u>

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.

TS 3.1, Specification 1, helps ensure that the SDM limit will provide an acceptable amount of negative reactivity to shut down the reactor with the most reactive control rod fully withdrawn, and all non-secured experiments are in their most reactive position. The NRC staff finds that the $0.50 negative reactivity required for the SDM is consistent with the guidance provided in NUREG-1537, Appendix 14.1.

TS 3.1, Specification 2, helps establish an appropriate limit on the DTRR core excess reactivity and requires the licensee to include the reactivity contributions from experiments in any excess reactivity assessments. The $3.00 core excess reactivity value that the licensee has chosen allows the DTRR facility to have operational flexibility to load fuel and experiments, while limiting the reactivity available for reactivity addition accident scenarios. The licensee has calculated (+$2.81) and measured (+$2.28) values for this parameter, as provided in the neutronics report (Ref. 12). The NRC staff finds that these values are less than the TS limit (+$3.00); and furthermore, they provide an acceptable measure of agreement between calculated and measured values.

TS 3.1, Specification 3, helps ensure that a limit on control rod reactivity insertion rates exists. The licensee has chosen a reactivity insertion rate limit of $0.20 per second, and provided as the basis, its analysis of the uncontrolled rod withdrawal presented in the DTRR T-H report, Section 3. The NRC staff's evaluation, including confirmatory calculation, of this analysis is provided in Section 2.5.4 of this SER. Therefore, based on the NRC staff's review of the uncontrolled rod withdraw scenario, the NRC staff concludes that the TS limit of $0.20 per second reactivity insertion rate is acceptable.

TS 3.1, Specification 4, helps ensure that the appropriate conditions exist to establish control rod operability. These conditions include verification that no visible damage to the control rods is evident and that the control rod scram time meets TS 3.3, Specification c, requirements. The 1 second scram insertion time is typical of TRIGAs as provided in the guidance in NUREG-1537, Appendix 14.1, Section 3.2(1), and it was used as an input parameter for the uncontrolled rod withdrawal transient analysis presented in the T-H report (Ref. 12). The acceptability of the scram time was established by the analysis discussed in Section 2.5.4 of this SER. The NRC staff concludes that based on the review provided in Section 2.5.4 of this SER, TS 3.1, Specification 4, is acceptable.

<u>Shutdown Margin–Confirmatory Analysis</u>

The NRC staff performed a confirmatory analysis of the DTRR SDM using information provided in the DTRR SAR (Ref. 4), as supplemented in the neutronics report (Ref. 12). The licensee supplied both measured and calculated control rod worths. Table 2-7 provides the results of the SDM confirmatory analysis. These results show that the actual core shutdown reactivity is less than or equal to the SDM requirement (ρ_{SDM}) with the contribution from the maximum worth control rod removed. In the case of the DTRR, where the Shim1 control rod is the largest worth control rod in either measurements or calculations this can be expressed as:

$$\rho_S=\rho_X-\rho_{SH2}-\rho_R-\rho_E$$

$$\rho_S \leq -\$0.50$$

where the applicable values are contained in Table 2-6.

Table 2-7 DTRR Shutdown Margin Confirmatory Calculations

Calculation Number (strongest rod withdrawn)	Initial Excess Reactivity (ρ_x) TS 3.1(2)	Control Rod Worths			Exp. (ρ_E) TS 3.7.1(a)	Calculated Shutdown Reactivity (ρ_S)	SDM Req. (ρ_{SDM}) TS 3.1(1)
		Shim1 (ρ_{SH1})	Shim2 (ρ_{SH2})	Reg. (ρ_R)			
LCC Core							
1	+\$3.00	stuck out*	-\$3.35	-\$1.20	+\$1.00	-\$0.55	-\$0.50
Operational Core							
2	+\$2.28	stuck out*	-\$3.10	-\$1.15	+\$1.00	-\$0.97	-\$0.50

LCC Core

In the SDM calculation above, the positive reactivity worth of the core excess, as provided by TS 3.1, Specification 2, and of an experiment, as provided by TS 3.7.1, Specification a, is offset by the calculated negative reactivity worths of the control rods minus the maximum worth rod (Shim1). The calculated shutdown reactivity (-$0.55) confirms that the maximum experimental reactivity worth ($1.00) is acceptable in all configurations, and TS 3.1, Specifications 1 and 2, and TS 3.7.1, Specification a, are acceptable.

Operational Core

In this reactivity calculation, the measured values of the excess reactivity (+$2.28) and control rod worths are used (Shim2: -$3.10, and Reg: -$1.15), and the resulting shutdown reactivity is -$0.97, which satisfies the proposed SDM requirement of TS 3.1, Specification 1, of -$0.50.

Based on the information provided and calculation results in Table 2-7 above, the NRC staff finds that the licensee's values for measured and calculated excess reactivity, SDM, and control rod worths are acceptable. Based on the information provided above, the NRC staff concludes that the requirements in TS 3.1, Specifications 1 through 4, are acceptable.

2.5.3 Reactivity Coefficients

<u>Prompt Negative Fuel Temperature Coefficient</u>

In the DTRR SAR, the licensee states that a significant feature of a TRIGA reactor is the large, prompt, negative fuel temperature coefficient (FTC, or α_F) of reactivity, resulting from the intrinsic characteristics of the molecular shape of the U-ZrH fuel matrix at elevated temperatures. The negative temperature coefficient results primarily from the neutron spectrum hardening (faster neutrons) at elevated temperatures, which increases the leakage of neutrons from the fuel-bearing material into the water moderator material, where they are absorbed and lost to the nuclear fission reaction. This results in a reactivity decrease, which is a prompt effect and occurs much more rapidly than any change to the moderator temperature. An additional contribution to the prompt negative temperature coefficient is the Doppler broadening of fuel resonances, which increases neutron capture and provides additional means for the loss of fission neutrons.

Because of the large, prompt, negative FTC, a step insertion of reactivity resulting in an increasing fuel temperature will be rapidly compensated for by this feedback. This dampens any power excursion before the electronic or mechanical reactor safety systems or the actions of the RO can take place. Also, changes of reactivity resulting in a change in fuel temperature during steady-state operation can be rapidly compensated for by this feedback effect, thus limiting the reactor steady-state power level. More details on the physics described above are discussed in the GA report, GA-4314 (Ref. 21).

The FTC represents the change in fuel reactivity versus the change in the fuel temperature. It is calculated by varying the fuel temperature while keeping all other core parameters fixed and using the resulting eigenvalues to calculate an effective coefficient. The licensee used the MCNP model with all rods removed and provided the results in the neutronics report. The licensee’s results are provided in Figure 2-5.

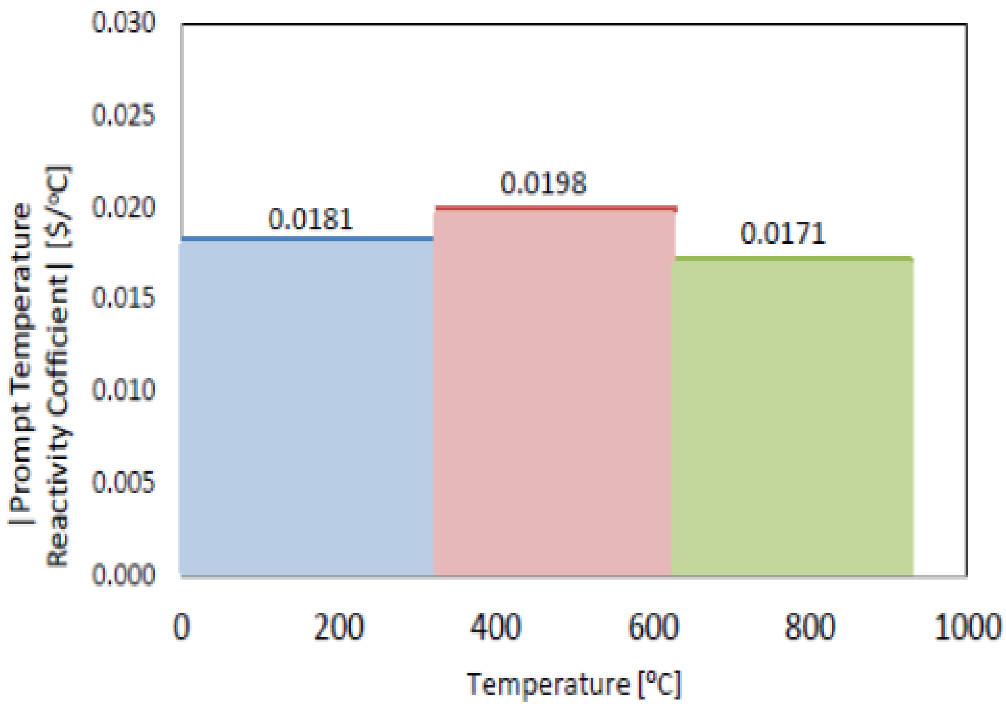

Figure 2-5 DTRR fuel temperature coefficient results

Prompt Negative Fuel Temperature Coefficient–Confirmatory Analysis

The NRC staff performed a detailed series of calculations of the FTC using a unit cell in an infinite lattice. The general model used is presented in Figure 2-6. In this model, a central rod region is provided and used for stainless-steel clad fuel elements. The physical dimensions of the model are taken from the DTRR neutronics report (Ref. 12).

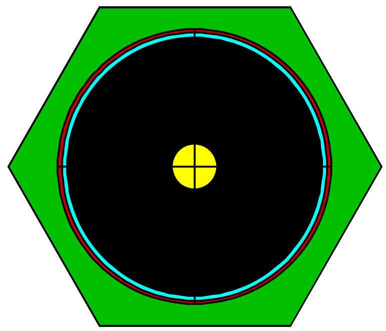

Figure 2-6 WIMS model of DTRR fuel elements

The program WIMS-ANL (Ref. 27) is used to perform the confirmatory analysis. This program uses a 69-group library specifically developed for RTR analysis. This library has cross-sections at a wide range of temperatures from 300 to 1,600 degrees Kelvin. The condensation of the cross-sections was performed using representative spectra for LEU TRIGA fuel. The confirmatory calculations were performed at seven temperatures of interest (31 degrees C, 150 degrees C, 300 degrees C, 400 degrees C, 600 degrees C, 800 degrees C, and 1,000 degrees C). At each temperature of interest, a pair of calculations was performed (e.g., for 150 degrees C, calculations were performed at 145 degrees C and 155 degrees C). Values were selected to provide the reactivity critical conditions at 31 degrees C and are then held constant in the other calculations.

The reactivity coefficient is calculated using:

$$\alpha_F = \frac{(k_2 - k_1)}{k_2 \times k_1} / \beta_{eff} \, (T_2 - T_1)$$

The units for α_F are $/C.

Figure 2-7 provides the referenced GA results, the DTRR results (Figure 2-5), and the NRC staff's confirmatory results. The NRC staff's confirmatory calculations validated the value and trend of the GA results. The values that the licensee demonstrated are consistent (same order and trend) with the referenced GA results. The NRC staff used the FTC values calculated by the NRC staff for the confirmatory calculations of the Reactivity Insertion Accident (see SER Section 4.1.2) and the Uncontrolled Rod Withdrawal Transient (see SER Section 2.5.4).

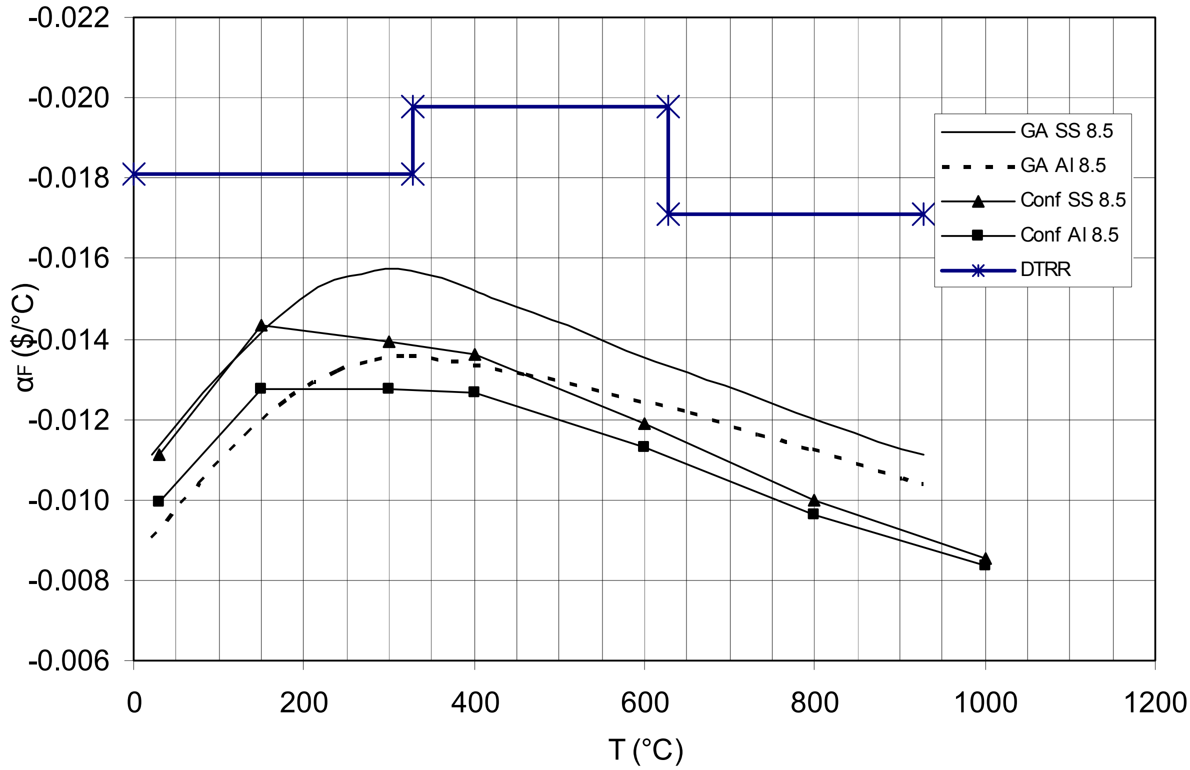

Figure 2-7 Fuel temperature coefficients

2.5.4 Transient Analysis of an Uncontrolled Rod Withdrawal

In the DTRR T-H report (Ref. 12), the licensee provided an analysis of the point reactor kinetics model. This model uses DTRR operating characteristics, limits, and the FTC determined previously. The delayed neutron fraction and the 6-group delayed neutron precursors are used. The NRC staff reviewed the methodology provided and finds that the assumptions and techniques were appropriate for the analysis of stepped reactivity insertions for the DTRR.

The uncontrolled withdrawal of a control rod at the DTRR maximum reactivity insertion rate of $0.20 per second is modeled by the licensee for the LCC core. In the DTRR neutronics report (Ref. 12), the most reactive rod was Shim1 with an integral rod worth of $3.65. During the most recent rod calibrations at the DTRR, the rod withdrawal times were measured, and the fastest rod withdrawal time measured was 41.53 seconds, corresponding to a rod speed of 21.67 in. per minute. Using a conservatively faster rod speed of 22 in. per minute and the calculated rod worth for Shim1, the maximum reactivity insertion rate for the 2011 DTRR core configuration is $0.14 per second. Upon insertion of the remaining two control rods at the completion of the transient, the reactor core must be subcritical by the TS shutdown margin of at least $0.50.

The limiting transient was determined to occur for low initial reactor power levels because a higher startup rate was achieved before reaching the scram setpoint. The model parameters were the same as those described above, with the following exceptions: 1) the reactivity insertion was modeled as a linear ramp at a reactivity addition rate of $0.20 per second; 2) a reactor scram was initiated at 6.825 seconds after the initiation of the uncontrolled rod withdrawal (this corresponds to the time at which power reaches 300 kWt plus an additional second to account for control rod insertion time); and, 3) the rod experiencing the uncontrolled withdrawal was assumed not to scram and continued its withdrawal, which assumes multiple failures and makes the analysis quite conservative. The use of a constant reactivity insertion rate of $0.20 per second was also conservative since the differential rod worth curve for the affected rod will have its largest value near the center of the core and will decrease as the rod approaches the end of its travel.

For the transient described, the peak power attained is 10.3 MWt, producing a core average fuel temperature increase of 72.15 degrees C. Using a peaking factor of 2.613, this resulted in a maximum temperature in the highest power rod of 213.5 degrees C, which is well below the DTRR fuel temperature SL of 500 degrees C.

<u>Uncontrolled Rod Withdrawal–Confirmatory Analysis</u>

The NRC staff performed a confirmatory calculation using the TRACE model developed for the DTRR (Ref. 28) for the LCC core. A further discussion of this model is provided in Section 2.6 of this SER. In this analysis, the peak power density fuel element is modeled to determine the maximum fuel element temperature resulting from the transient.

The confirmatory calculation used the DTRR calculated rod worths of $1.20 (Regulating), $3.65 (Shim1), and $3.35 (Shim2). The primary assumption in the analysis was that the maximum worth control rod (Shim1) is withdrawn at a critical condition with the two remaining rods already fully withdrawn. An initial power 0.1 kWt with an initial fuel temperature of 25 degrees C was assumed. The scram occurs at 300 kWt and the rod continues to move out of the core momentarily. A conservative 1-second delay time is employed between the initiation of the scram signal and the movement of the Shim2 and the regulating control rods before these two remaining control rods are fully inserted. The TS 3.3 scram time of 1 second and withdrawal rate of $0.20 per second are used with the worth of the two inserted control rods being linear over the insertion time. The reactor power and peak fuel element temperatures, for both the licensee's and NRC staff's calculations, are displayed in Figures 2-8 and 2-9.

The NRC staff's analysis used the confirmatory value of the FTC, which provides conservative results (by a factor of 2 greater than the licensee's results as shown in Figures 2-8 and 2-9), and determined that the scram signal occurs at 5.71 seconds at a power of 300 kWt. The power level continues to rise until the FTC overcomes the transient at 16.55 MWt occurring at 5.97 seconds. The control rods begin inserting at 6.71 seconds and finish inserting 1 second later. Since nuclear processes terminate much more rapidly than T-H processes, the peak fuel temperature, which begins the transient at ambient temperature of 25 degrees C, peaks at 268 degrees C.

The NRC staff finds that the peak power reached, for a very short time, does not pose a challenge to the fuel element safety limit of 500 degrees C. Consequently, the confirmatory calculation established the acceptability of the FTC, TS scram time value of 1 second, and the rod withdrawal rate of $0.20 per second. The NRC staff finds that the reactivity insertion rate ($0.14 per second) is below the TS maximum reactivity insertion rate of $0.20 per second used in the present analysis, and therefore, acceptable.

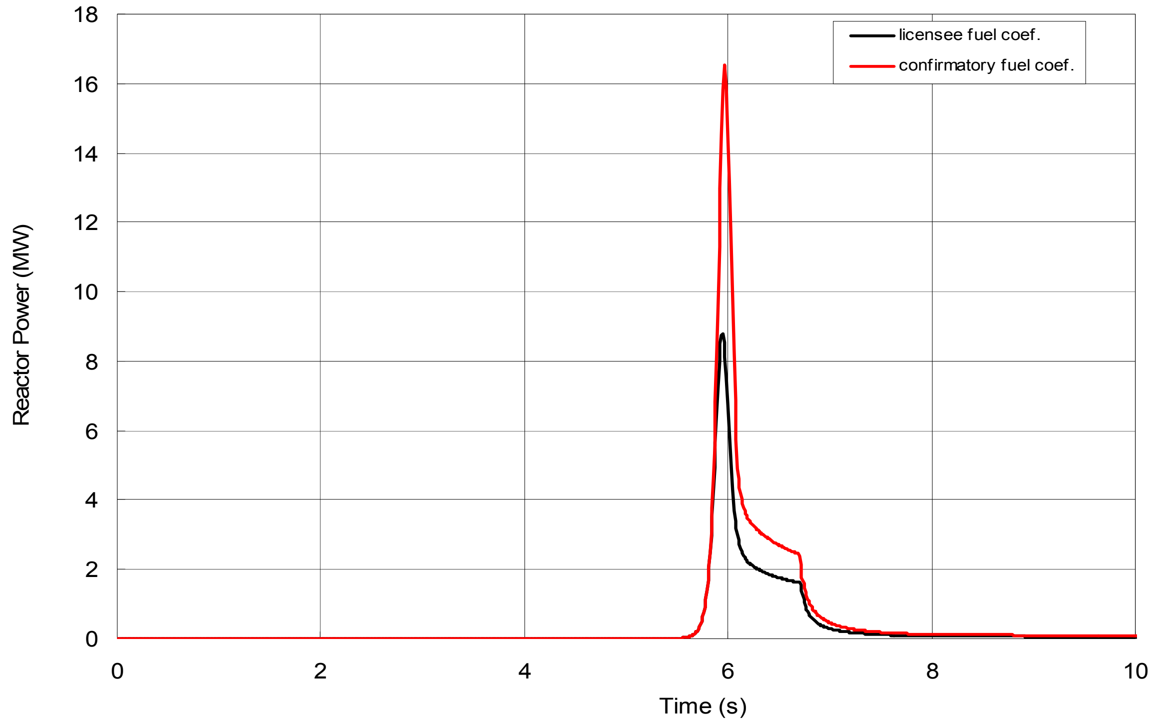

Figure 2-8 Uncontrolled rod withdrawal–total reactor power

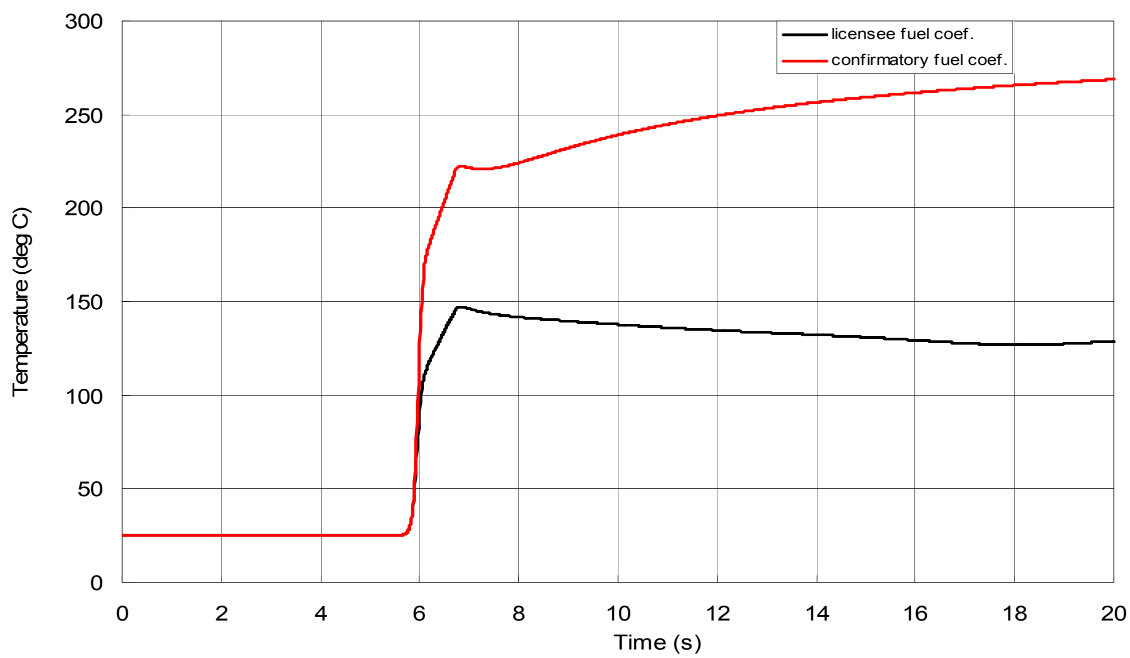

Figure 2-9 Uncontrolled rod withdrawal–peak fuel element temperature

Based on the information provided above, including the NRC staff's confirmatory calculation results, the NRC staff finds that the consequences of the uncontrolled rod withdraw event are acceptable, as are the values provided in DTRR TS 3.3, Specification c, and TS 3.1, Specification 3. Furthermore, based on the information discussed above, the NRC staff concludes that the licensee's results for the uncontrolled rod withdrawal accident scenario are, therefore, acceptable.

2.5.5 Operating Limits

Regulations in 10 CFR 50.36(d)(1) require reactors to specify an SL and an LSSS. These regulations define an SL as a limit upon important process variables necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity. LSSS for nuclear reactors are defined as settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is given for a variable on which an SL is placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before an SL is exceeded.

The principal physical barrier to the release of fission products for TRIGA reactors is the fuel element cladding. The most important parameter to maintain the fuel cladding integrity is the fuel element temperature. A loss in the integrity of the fuel rod cladding may occur if the fuel temperature exceeds the SL. The fuel clad temperature and the ratio of H to Zr in the alloy determine the failure mechanism.

The SL for the stainless-steel clad, high-hydride TRIGA fuel is based primarily on experimental data obtained during high-performance reactor tests on this fuel (Ref. 22). These data indicate that the stress in the cladding caused by H pressure from the disassociation of ZrH will remain below the stress limit, provided that the temperature of the fuel does not exceed 1,150 degrees C and the fuel cladding is water cooled.

The SL for the aluminum-clad, low-hydride TRIGA fuel elements depends on avoiding the phase change in the ZrH that might cause failure of fuel element cladding. This phase change takes place at 530 degrees C (Ref. 16).

The SL for the DTRR provided in TS 2.1 limits the fuel element temperature to a maximum of 500 degrees C for all fuel elements under any operating condition. This is the most limiting of the SLs for the two fuel types used in the DTRR. The DTRR SL is imposed to prevent excessive stress on the cladding caused by a phase change of the U-ZrH fuel. Based on theoretical and experimental data (Ref. 23), this SL represents a conservative value to provide confidence that the integrity of the fuel elements will be maintained and that no damage to cladding will occur. The evaluation of the SL TS 2.1 was presented in Section 2.2.2 of this SER.

The DTRR TSs include an LSSS to ensure that there is a considerable margin of safety before the TS SL specified above is reached. The LSSS is required for the operation of the reactor under 10 CFR 50.36, "Technical specifications," and represents the limiting values for settings of the safety channels by which point protective action must be initiated. The LSSS TS must be chosen so that automatic protective action will terminate the abnormal situation before the SL is reached. Because the LSSSs are analytical limits, the protective channels may be set to actuate at more conservative values. The more conservative values may be established as limiting conditions for operation (LCOs).

The licensee's response to RAI-17 (Ref. 10) provides an explanation of the operating limits for the DTRR using the LCC in the neutronics and T-H reports (Ref. 12). The NRC staff reviewed these reports and confirmed an acceptable analysis to support DTRR operation at 300 kWt, including any postulated transient conditions. The licensee's response to RAI-30 (Ref. 4) explained that the operating range for the nuclear instrumentation covered the range of 3×10^{-4} watts thermal (Wt) to $3\times10^{+5}$ Wt, which was sufficient for the licensed operating range of the DTRR.

2.5.5.1 TS 2.2 Limiting Safety System Settings

TS 2.2 states the following:

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

As discussed in the DTRR T-H report, the RELAP5-3D code was used to calculate the maximum fuel centerline temperature for 300 kWt for the DTRR LCC using an assumed reactor pool water coolant inlet temperature of 60 degrees C and a reactor pool water level of 15 ft (4.6 m). The maximum fuel element power was calculated to be 6.08 kWt with a maximum fuel centerline temperature of 246.7 degrees C; the maximum fuel element cladding temperature was 122.8 degrees C, and the DNBR was 6.76 using the Bernath correlation. The NRC staff reviewed the licensee's methodology and calculated results and concluded that the DTRR TS 2.2 LSSS of 300 kWt provided acceptable margin to the SL of 500 degrees C.

Based on the information provided above, the NRC staff finds that TS 2.2 helps ensure that the power level provided by the LSSS will prevent the fuel temperature from exceeding the SL specified in TS 2.1. The NRC staff finds that the analysis in the DTRR SAR, as supplemented, provided adequate calculation of the LSSS using techniques and assumptions consistent with TRIGA reactors and the guidance provided in NUREG-1537. Therefore, the NRC staff concludes that TS 2.2 is acceptable.

2.5.5.2 TS 3.3 Reactor Control Rods and Safety Systems and Interlocks

TS 3.3 states the following:

> 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.

TS 3.3, Specification a, helps ensure the operability of the Safety Channels and Interlocks listed in TS Table 3.3A. This includes establishing the scram setpoint at 300 kWt, which helps ensure that the licensed power limit and the value of the LSSS (TS 2.2) are maintained during operation. The NRC staff finds that TS 3.3, Specification a, is consistent with the assumptions used in the insertion of excess reactivity and loss-of-coolant accident (LOCA) analysis presented in the DTRR SAR and the NRC staff's review in Section 4.1.3 of this SER. The two NM1000 and NPP1000 power scram channels provide a diverse and redundant measuring system for ensuring the DTRR power level does not exceed 300 kWt. The accuracy of the NM1000 and NPP1000 instruments are between 1.0 and 0.1 percent, per the GA TRIGA Operators Manual for Instrumentation and Control. The NM1000 and NPP1000 are consistent with the guidance provided in NUREG-1537, Appendix 14.1, Section 3.2(4).

TS 3.3, Specification b, helps ensure the operability of the Measuring Channels listed in TS Table 3.3B. The NRC staff finds that the Measuring Channels support the safe operation of the DTRR by providing sufficient information to the operator to control the operation of the reactor. The Measuring Channels provide console instrumentation indicating reactor power and reactor pool water radioactivity, level and temperature, and that are consistent with the assumptions used in various accident scenarios as described in the DTRR SAR.

TS 3.3, Specification c, helps ensure that the Control Rod Scram Time is less than 1 second. The NRC staff finds that the scram time value was demonstrated to be acceptable by the analysis presented in the DTRR SAR, as supplemented, for the uncontrolled rod withdrawal accident, which was further discussed in Section 2.5.4 of this SER. The analysis demonstrated that the scram time contributed to terminating accident scenarios rapidly enough to maintain fuel temperatures below the TS SL.

A detailed review of each scram and interlock channel (except the manual console scram), as well as the minimum number of channels, follows:

TS Table 3.3A:

Power Level Scram Setpoint

The Power Level Scram Setpoint LCO is established at the DTRR licensed power of 300 kWt. The analysis provided in the T-H report demonstrated the acceptability of this setpoint. This power level was used consistently for safety analysis, DNBR analysis, and transient analysis. Each scenario resulted in acceptable consequences with the fuel temperature remaining below the TS SL of 500 degrees C. The calculated maximum centerline fuel temperature provided in T-H report was 246.7 degrees C for the LCC at 300 kWt. Therefore, based on the information provided above, the NRC staff concludes that TS Table 3.3A, Power Level Scram Setpoint of 300 kWt is acceptable. Footnote 1 provides the necessary conditions in order to perform surveillance for performing channel checks, channel tests, and channel calibrations with the reactor in operation. This is acceptable for the period of time required to perform the necessary surveillance as a redundant scram channel is available to provide a safety signal, if necessary.

Detector High-Voltage Scram Setpoint

The Detector High-Voltage Scram Setpoint LCO actuates an automatic scram upon loss of high voltage power to the DTRR core instruments. This LCO helps ensure that the accuracy of reactor core measuring instruments that provide an input to the power level scram are maintained. The NRC staff finds that the LCO setpoint for the automatic trip on loss of high voltage is consistent with the guidance in NUREG-1537. Therefore, based on the information provided above, the NRC staff concludes that TS Table 3.3A, High-Voltage Scram Setpoint, is acceptable.

Watchdog (DAC to CSC) Communication Conflict

The Watchdog Scram pertains to the communications that take place between the data acquisition computer (DAC) and the control system computer (CSC) and provides a scram if communications are interrupted for more than 10 seconds. This scram is consistent with vendor recommendation and design (GA TRIGA). Based on the information provided, the NRC staff concludes that TS Table 3.3A, the Watchdog Communication Conflict LCO scram setpoint, is acceptable.

Interlocks

The DTRR Startup Count Rate, Control Rod Withdrawal, and Reactor Period Interlock LCOs are typical of TRIGA facilities, and consistent with the guidance in NUREG-1537, Appendix 14.1, Section 3.2(5). The NRC staff finds that the note to Table 3.3A is appropriate to DTRR operation, has been properly considered in the DTRR SAR, and is acceptable. Based on the information provided above, the NRC staff finds that the interlock LCOs and note specified in TS Table 3.3A are acceptable.

TS Table 3.3B:

Number of Operable Channels

The guidance provided in NUREG-1537, Appendix 14.1, Table 14.3, describes the recommended minimum number of measuring channels for operation, which includes two redundant channels for power level scrams. DTRR LCO Table 3.3B helps ensure that measuring channels, NM1000 and NPP1000 are operable, which provides two diverse and redundant channels. The two instrumentation channels monitor and indicate the reactor neutron flux and power level on the console. The bulk water temperature and the reactor tank outlet and inlet water temperatures are also displayed on the console. The water conductivity, measured at the inlet and outlet of the demineralizer, is displayed on a panel near the console. In addition, primary reactor water is routinely monitored to identify any significant increase in radioactivity. The response to RAI-40 (Ref. 4) states that experimental systems cannot affect the function of safety systems. The reactor pool water level provides an indication of the pool inventory to help ensure an adequate heat sink is available for reactor operation. Therefore, based on the information provided above, the NRC staff concludes that TS Table 3.3B is acceptable. Footnote 2 provides a 5-minute period for the operator to diagnose the cause of a loss of indication from a required measuring channel. This TS footnote helps to avoid unnecessary reactor shutdowns if the cause of the loss of indication can be determined and corrected within the 5-minute period.

Reactor Pool Water Radioactivity Monitor

The DTRR Reactor Pool Water Radioactivity Monitor LCO provides to the operator console information concerning any radioactivity detected in the pool water. The NRC staff finds that the information would provide early indication of fuel cladding failure and require subsequent corrective actions. Based on the information provided above, the NRC staff concludes that TS Table 3.3B, Pool Water Radioactivity Monitor LCO, is acceptable.

Reactor Pool Water Temperature Monitor

The Reactor Pool Water Temperature Monitor LCO provides its information at the operator console to help ensure that the reactor is operated in conformance with the safety-basis assumption for pool water temperature. The NRC staff finds that a high temperature would alert the RO to take corrective action. Therefore, based on the information provided above, the NRC staff concludes that TS Table 3.3B, Reactor Pool Water Temperature LCO, is acceptable.

Based on the information provided above the NRC staff finds that TS 3.3 establishes operability requirements for safety channels, measuring channels, interlocks, and control rod scram time that are consistent with the analyses in the DTRR SAR. Therefore, the NRC staff concludes that DTRR TS 3.3 is acceptable.

Furthermore, the NRC staff finds that the DTRR nuclear design as described in the SAR is typical of TRIGA reactors, was properly documented in the SAR, and important design features were properly implemented in the appropriate portions of the DTRR TS as described above. Therefore, based on the information provided above, the NRC staff concludes that the DTRR nuclear design is acceptable.

2.6 Thermal-Hydraulic Design

The licensee states that the important parameter in T-H design is the critical heat flux, which describes the heat flux associated with the departure from nucleate boiling. The parameter of interest is the DNBR, which is the ratio of the critical heat flux to the maximum heat flux at full power. The guidance provided in NUREG-1537, Appendix 14.1, Section 2.1.2, indicates that this ratio should be greater than 2.0 for forced cooled systems and this value is often applied to natural convection systems in TRIGA reactors.

The licensee presented a detailed T-H analysis of the DTRR DNBR using RELAP5-3D in the T-H report (Ref. 12). The RELAP5-3D model is presented in Figure 2-10 and illustrates the basic features of the DTRR system, which uses a single-flow channel divided into axial and radial segments. The model consists of a coolant source, cold leg, horizontal connector, hot channel, and coolant sink. The coolant source is modeled as a time-dependent volume that allows inlet pressure and temperature conditions to change with time. The cold leg is incorporated to create a pressure differential between the cold coolant entering the subchannel and the heated coolant passing through the subchannel. This pressure differential drives the natural circulation flow. The horizontal connector is simply a connection between the cold leg and hot channel to allow for communication between the cold leg and the hot channel during the computational process. The hot channel is the volume that contains the fuel rod of a single DTRR subchannel. By using the peaking factors from the neutronic report (Ref. 12), the channel has been adjusted to represent the hot channel having the most conservative T-H parameters found in the DTRR core. The DTRR model assumes no cross flow between adjacent fuel element channels, which is conservative since higher values of temperature and lower margins to the DNBR are predicted when cross flow between adjacent channels is not allowed.

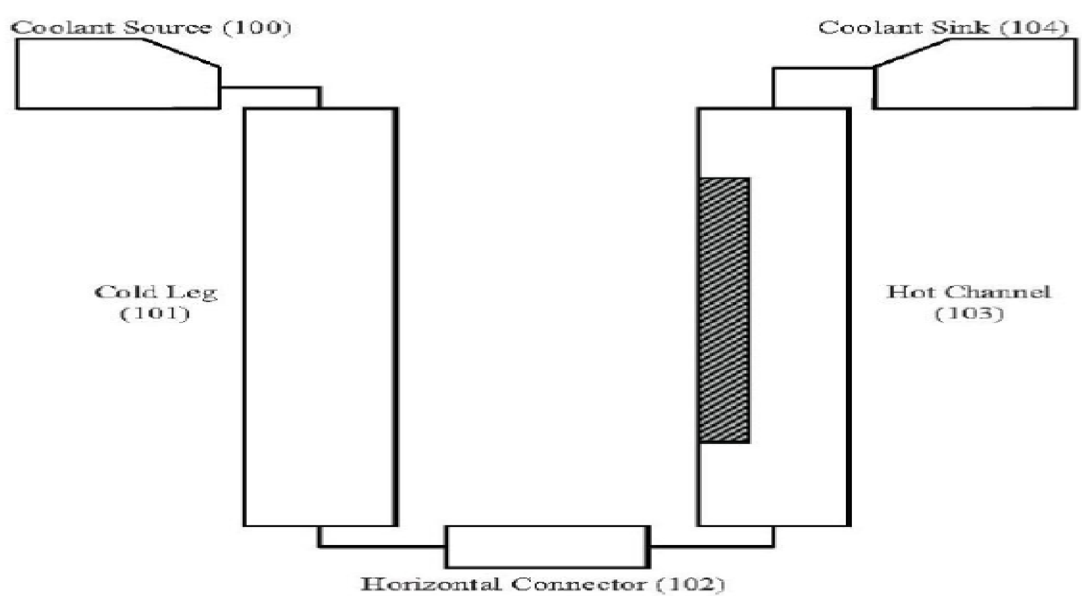

Figure 2-10 The DTRR RELAP-3D model

The licensee analyzed the DTRR system operating with cooling provided by natural convection water flow around the fuel elements. The predicted steady state T-H performance of the DTRR is determined for the reactor operating at 300 kWt with a water inlet temperature of 60 degrees C that is bounded by TS limit for the pool temperature, which is 40 degrees C. The maximum power fuel rod and maximum power heated subchannel were analyzed for the DTRR under steady-state conditions. The power in the hottest rod at which critical heat flux is predicted to occur is used to characterize the DNBR. Since the limiting locations in the core contain only stainless-steel clad elements, only the stainless-steel configuration was analyzed. The fuel rod, 1.47 in. (3.7 cm) diameter and 15 in. (38.1 cm) long, was represented in the hot channel by 24 discrete radial and axial nodes each. The mesh points within the fuel region that the licensee used in the RELAP5-3D model correspond to 1 node for the central zirconium pin, 20 nodes of equal radial thickness for the fuel material, 1 node for the fuel to clad gap and 1 node for the clad. The radial model of the subchannel is illustrated in Figure 2-11 (provided by the T-H report as Figure 1).

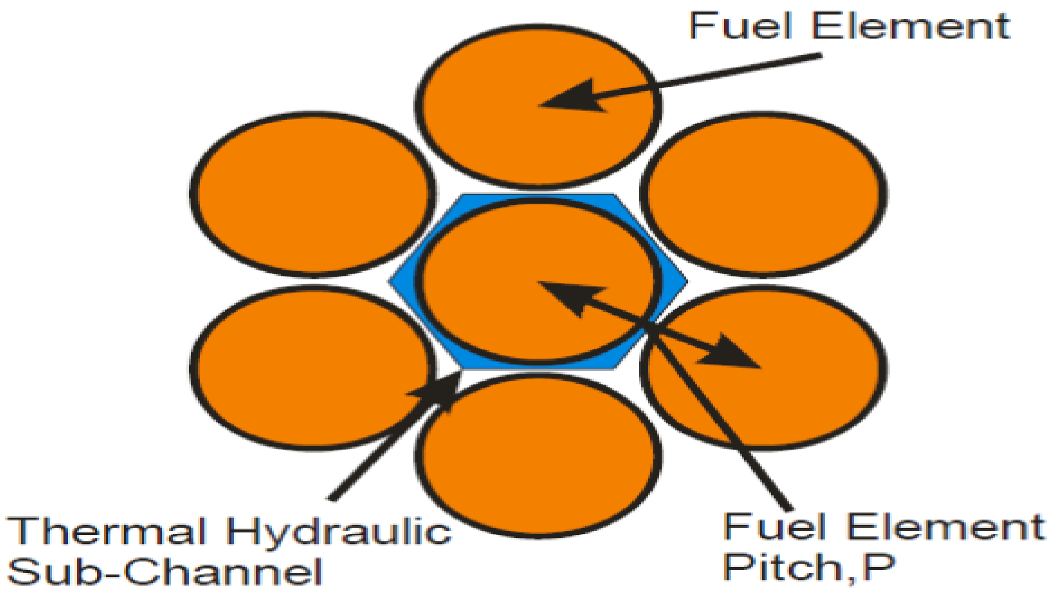

Figure 2-11 Schematic of the subchannel used for DNBR analysis

The licensee analyzed power peaking factors for each core configuration using the MCNP code. The highest power rod for each configuration was determined by calculating the total power produced in each fuel element present in the configuration. After the highest power fuel element is determined, further analyses were performed to find the detailed axial and radial power shapes associated with that fuel element. The axial and radial power shapes were determined for 20 equally spaced nodes in both the axial and radial directions. The MCNP results are used to calculate three peaking factors:

- Hot channel fuel peak factor–(maximum fuel element power)/(core average fuel element power)
- Hot channel fuel axial peak factor–(maximum axial power in the hot fuel element)/(average axial power in the hot fuel element)

- Hot channel fuel radial peak factor–(maximum radial power in the hot fuel element)/(average radial power in the hot fuel element)

A summary of the peaking factor results, at a power level of 300 kWt, were provided in Table 5 of DTRR T-H report (Ref. 12) and are provided in Table 2-8.

Table 2-8 LCC Hot Channel Fuel Power Summary

Rod Location	Hot Channel Fuel Thermal Power [kWt]	Hot Channel Fuel Peaking Factor	Hot Channel Fuel Axial Peaking Factor	Hot Channel Fuel Radial Peaking Factor	Effective Total Peaking Factor
C6	6.08	1.539	1.312	1.196	2.415

The DTRR LCC average fuel element power was 3.95 kWt (300 kWt/76 fuel elements). The ratio of the power in the maximum power fuel element to the average fuel element was 1.539. The MCNP analysis determined that the ratio of the axial peak power in the maximum power fuel element to the average axial power in that element was 1.312, which was applicable to any fuel element in the core. Similarly, the ratio of the maximum fuel element power at the peak radial location to the average radial power at that location was 1.196. This factor also was applicable to any fuel element in the core.

The NRC staff reviewed the licensee's methodology for the peaking factor analysis and finds that the licensee's peaking factor results are typical of TRIGAs, and acceptable.

To evaluate DNBR for the DTRR core, the licensee used Bernath's correlation, which GA, the developer of the TRIGA reactor, also used for its evaluation of the DNBR in other TRIGA reactors. This correlation produces the most limiting critical heat flux ratio values when compared to other correlations or lookup tables. The DTRR LCC minimum DNBR was determined by the licensee to be 6.76 at a maximum fuel temperature of 246 degrees C at 300 kWt power level operation. This ratio is consistent with the guidance provided in NUREG-1537, which indicates that a value of two or greater provides an acceptable safety margin.

<u>DNBR–Confirmatory Analysis</u>

The NRC staff performed a confirmatory calculation of the DTRR DNBR using the TRACE program (Ref. 28). The model used by the NRC staff is displayed in Figure 2-12.

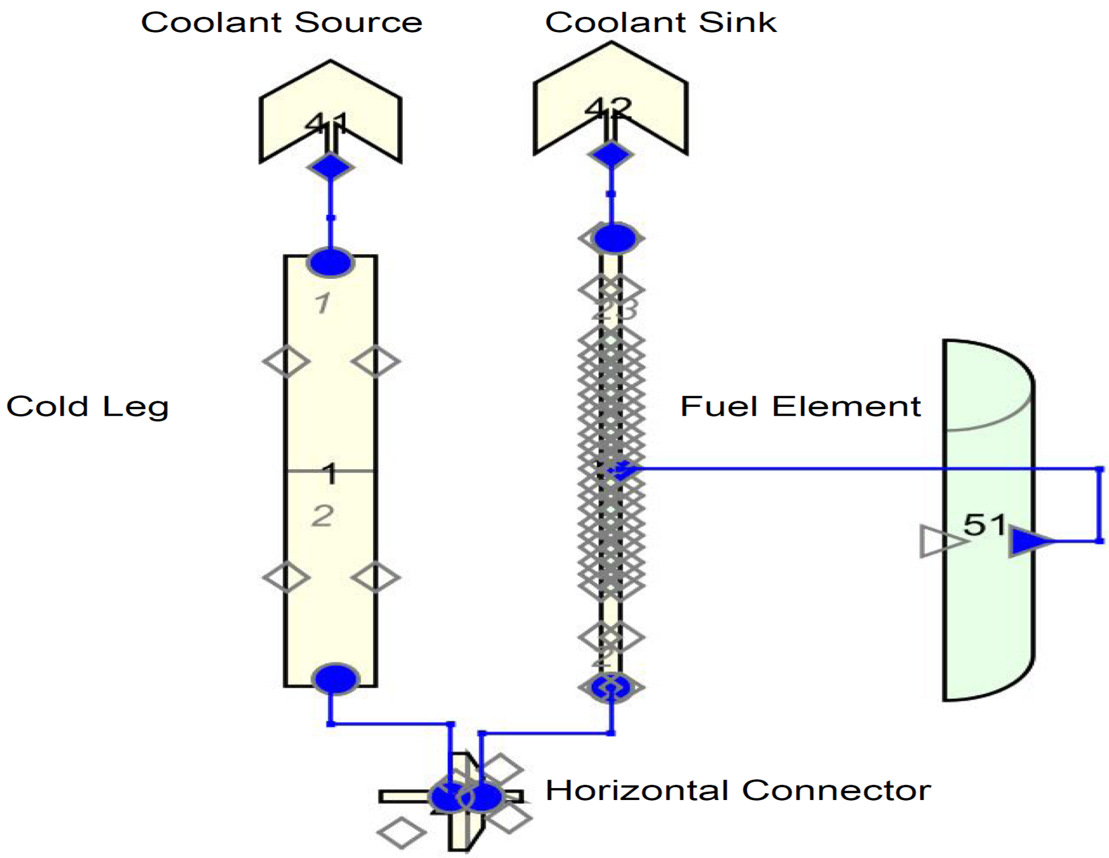

Figure 2-12 TRACE model of the DTRR

The TRACE model was developed using the physical characteristics from the DTRR T-H report. The DTRR inlet loss coefficients were used without modification and the Bernath correlation was used. The DTRR DNBR value provided by the licensee for the LCC was 6.76. The corresponding TRACE value calculated by the NRC staff using the Bernath correlation was 6.08. These values are in good agreement and significantly greater than the guidance in NUREG-1537 (DNBR greater than 2.0). Additional calculations were performed by the NRC staff to provide more details of the DNBR at various power levels of the DTRR. Figure 2-13 illustrates how the reactor power was increased and the time allowed at each power level to provide a stable reactor condition in order to calculate the DNBR. These power levels were based on the power level used in the DTRR T-H report. The DNBR values calculated at higher power levels than the DTRR licensed power of 300 kW shows the additional margin available to the DTRR.

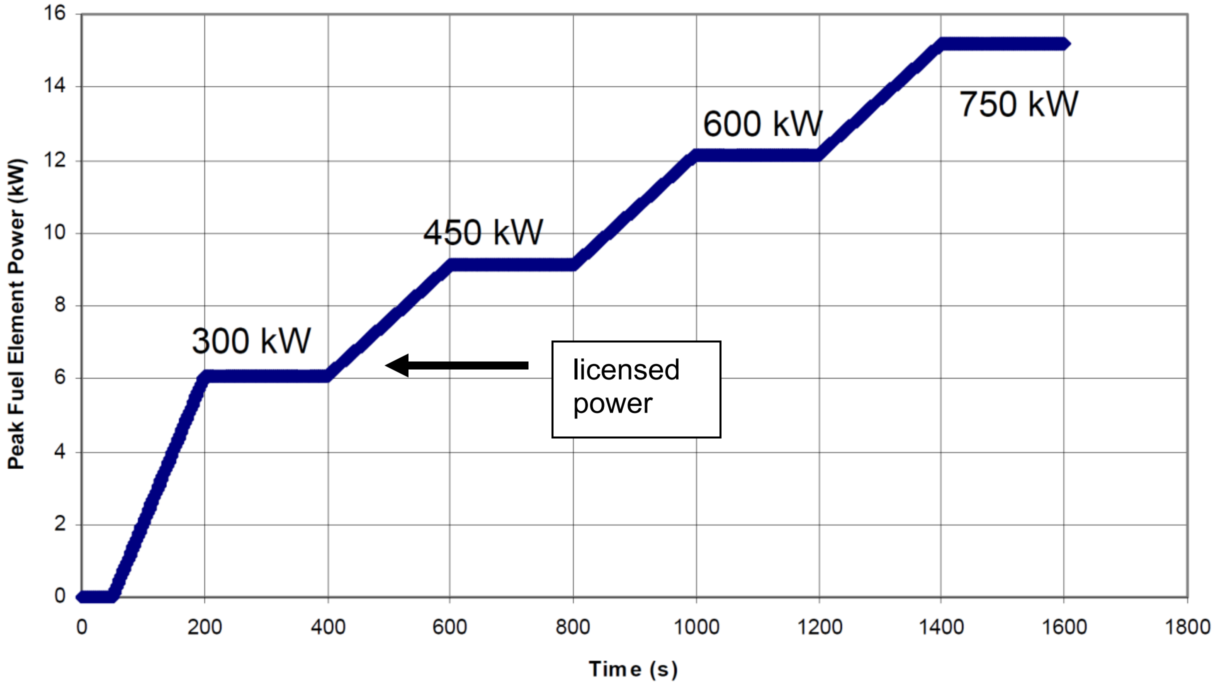

Figure 2-13 TRACE model power ramp

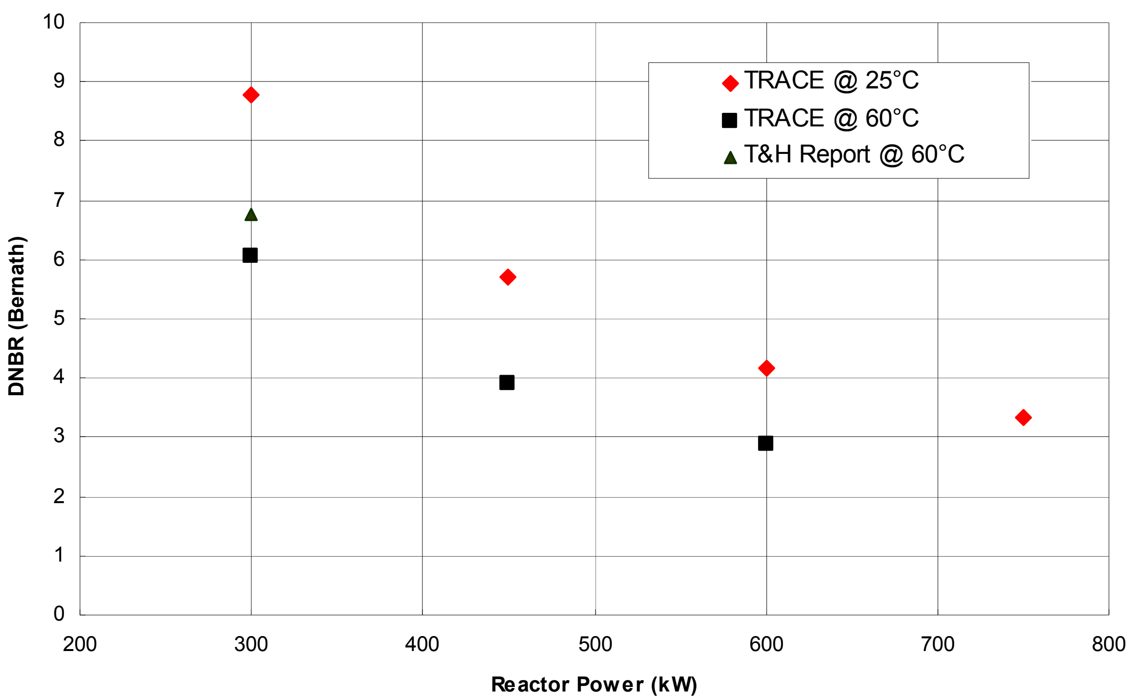

Figure 2-14 Comparison of confirmatory and DTRR DNBR results

Figure 2-14 displays the results of the NRC staff's TRACE calculated DNBR values as a function of reactor power at the various power levels described in Figure 2-13. The calculations were performed using a nominal reactor pool water inlet temperature of 25 degrees C and the TS limit on bulk pool temperature of 60 degrees C from TS 3.4, Specification 5. Figure 2-14 provides an illustration of the DTRR and NRC staff DNBR calculations. The NRC staff's methods used for the TRACE confirmatory analysis were equivalent to those used in the DTRR T-H report (Ref. 12). The NRC staff finds that the DTRR DNBR value is consistent with the NRC staff's confirmatory calculation.

The NRC staff finds that the DTRR T-H design analysis was consistent with other TRIGA reactors, adequately described in the DTRR SAR, and properly controlled and implemented in the DTRR TS. The NRC staff finds that the DTRR-supplied T-H analyses use qualified calculation methods and acceptable assumptions. In addition, the NRC staff confirmatory calculation of the DTRR DNBR provided comparable results. Based on the information provided above, the NRC staff concludes that the T-H analysis for the DTRR adequately demonstrates that the reactor can operate at 300 kWt with sufficient T-H safety margins.
.

2.6.1 TS 5.2 Reactor Coolant System

TS 5.2 states the following:

> 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.

TS 5.2, Specification 1, helps ensure effective core cooling is maintained as a design feature consistent with the assumption used in the DTRR SAR. The NRC staff finds, as provided in the T-H report for power levels as high as 300 kWt, and confirmed by the NRC staff's confirmatory analysis, and illustrated in Figure 2-14 of this SER, that natural convection provides acceptable cooling to the DTRR under all intended modes of operation, including anticipated transients and accidents (reference Section 4, Accident Analysis for additional information).

TS 5.2, Specification 2, helps ensure an adequate supply of reactor coolant is available for cooling by requiring anti-siphon holes in water lines so that reactor coolant water cannot be inadvertently drained by siphon action (gravity) in the case of a primary coolant system failure. The NRC staff finds that this specification is accomplished as described in SAR, Section M.1.1.

The NRC staff finds that TS 5.2 establishes effective design features that are consistent with the assumptions used by the licensee in the DTRR SAR and confirmed by the NRC staff's confirmatory calculations of the DTRR fuel temperature. Based on the information provided above, the NRC staff concludes that TS 5.2 is acceptable.

2.6.2 TS 3.4 Reactor Coolant Systems

TS 3.4 states the following:

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.

TS 3.4, Specification 1, helps ensure that the pool water conductivity is less than 5 micro mhos per centimeter (µmhos/cm). Limiting this conductivity helps to extend the longevity and integrity of the fuel clad. It also helps promote better heat transfer between the clad and coolant by minimizing the oxide buildup. The NRC staff finds that experience at many research reactor facilities has shown that maintaining the conductivity within the specified limit provides acceptable control of corrosion. A small rate of corrosion continuously occurs in any water-metal system. A conductivity limit of 5 micro mhos per centimeter is consistent with the conductivity limits used at other TRIGA reactors. The NRC staff also finds that TS 3.4, Specification 1, is consistent with the guidance in NUREG-1537, Section 5.4 and the design assumptions described in the DTRR SAR Section E.2. Based on the information provided above, the NRC staff concludes that TS 3.4, Specification 1, is acceptable.

TS 3.4, Specification 2, helps ensure an acceptable LCO range on the pool water pH. The NRC staff finds that the proposed pH range is consistent with the DTRR SAR, Section E.2, and the guidance provided in NUREG-1537, Section 5.4. Maintaining the pH of the tank pool water between 5.0 and 7.5 helps to ensure that the water is kept chemically neutral and the generation of corrosion products is minimized. Based on the information provided above, the NRC staff concludes that TS 3.4, Specification 2, is acceptable.

TS 3.4, Specification 3, helps ensure that the radioactivity of the pool water is controlled, and that the radiological conditions in the reactor coolant are monitored. The NRC staff finds that TS 3.4, Specification 3 is consistent with the guidance in NUREG-1537, Section 5.2. The NRC staff also finds that pool water monitoring will likely detect a fuel element failure long before the continuous air monitor (CAM) or area radiation monitor (ARM) would alarm. Based on the information provided above, the NRC staff concludes that TS 3.4, Specification 3, is acceptable.

TS 3.4, Specification 4, helps ensure that the minimum reactor pool water level covering the core will be at least 15 ft of water to provide core cooling and shielding. The channel has a visual and audible alarm (RAI-21, Ref. 4). The NRC staff finds that the TS is consistent with assumptions described in the DTRR SAR, Sections D.4, and L3.2. An analysis of radiation doses was provided in the response to RAI-13 (Ref. 4), which demonstrated that acceptable doses are measured for this depth of water at reactor operation at 300 kWt. An acceptable DNBR analysis was provided in the T-H report (Ref. 12), which supported the adequacy of 15 ft (4.6 m) of water for cooling purposes. Based on the information provided above, the NRC staff concludes that TS 3.4, Specification 4, is acceptable.

TS 3.4, Specification 5, helps ensure that the reactor bulk water temperature limit used in the DTRR SAR is maintained. The channel has a visual and audible alarm (RAI-21, Ref. 4). The NRC staff finds that this limit ensures the assumptions used in the SAR accident analyses, DNBR analysis, and also serves to prevent the breakdown of water treatment resins important to maintaining water chemistry and purity. This value is consistent with statements in the SAR in Sections D.5 and E.2 and the T-H report (Ref.12). Based on the information provided above, the NRC staff concludes that TS 3.4, Specification 5, is acceptable.

TS 3.4, Specification 6, helps ensure that an audible alarm will occur if the reactor pool water level decreases to a level of 15 ft, 10 in. (4.8 m). The NRC staff finds that TS 3.4, Specification 6, and the associated monitoring equipment, provide the ROs with an acceptable way to detect a low water level condition and provide corrective actions to ensure that adequate shielding and cooling are maintained. TS 3.4, Specification 6, also maintains the assumptions used in the reactor shielding and T-H analyses. Based on the information provided above, the NRC staff concludes that TS 3.4, Specification 6, is acceptable.

The NRC staff finds that TS 3.4 establishes adequate reactor coolant system setpoints and conditions that are consistent with the safety analysis as outlined in the SAR. Based on the information provided above, the NRC staff concludes that TS 3.4, Specifications 1 through 6, are acceptable.

2.7 <u>Reactor Description Conclusions</u>

Based on the above considerations, the NRC staff concludes that the licensee has presented adequate information and analyses to demonstrate the technical ability to configure and operate the DTRR core without undue risk to public health and safety or the environment. The NRC staff's review of this information includes studying its design and installation, its controls and safety instrumentation, its operating procedures, and its operating limitations, as identified in the TSs. The NRC staff concludes that the DTRR SAR, as supplemented, demonstrated that the DTRR core results in acceptable safety margins. The NRC staff reviewed the design and installation, controls and safety instrumentation, and operational limitations as identified in the DTRR TSs.

The NRC staff finds that the licensee's analyses used qualified calculation methods and conservative or justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with independent confirmatory analysis that the NRC staff performed. The NRC staff reviewed the analysis of the steady-state operation of the DTRR core at a power level of 300 kWt and finds that the maximum core fuel temperature remains below the limit set by the known mechanical and thermal properties of the fuel. The

NRC staff finds that the DTRR TSs on the reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely in accordance with the TSs. Therefore, based on this review, the NRC staff concludes that there is reasonable assurance that continued operation of the DTRR, up to 300 kWt, as limited by the TSs, would not pose undue radiological risk to the facility staff, the public, or the environment.

3. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

Activities involving radiation at the DTRR are controlled under the radiation protection program, which must meet the requirements of 10 CFR 20.1101, "Radiation protection programs." The regulations in 10 CFR 20.1101 specify, in part, that each licensee shall develop, document, and implement a radiation protection program, and shall use, to the extent practical, procedures and controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. The licensee shall periodically (at least annually) review the radiation protection program content and implementation.

The NRC inspection program routinely reviews radiation protection and radioactive waste management at the DTRR facility. The licensee's historic performance in this area is reviewed by considering the information from the DTRR annual reports and NRC IRs from 2005 through 2013.

3.1.1 Radiation Sources

The NRC staff review considered the descriptions provided of potential radiation sources, including the inventories of each physical form and their locations. The radiation sources at the DTRR are categorized as airborne, liquid, and solids.

<u>Airborne Radiation Sources</u>

The DTRR SAR and the licensee's responses to RAI-41 (Ref. 38) and RAI-43 (Ref. 4) provided updated calculations of the production and release of routine airborne radioactive effluents and the resultant doses to the DTRR staff and members of the public. During normal operations of the DTRR, the primary airborne sources of radiation are argon-41 (Ar-41) and nitrogen-16 (N-16). As described in the DTRR SAR, Section E.6, the licensee uses a N-16 control system to reduce the N-16 released to the atmosphere. N-16 results from the irradiation of oxygen in the reactor water coolant, has a 7-second half-life, and the normal transport time from the reactor core to the pool surface is 42 seconds. When the pool cooling system is in operation, the transport time is further increased by the vertical flow of water downward over the core from the N-16 diffuser resulting in a negligible release of N-16 to the atmosphere. The flow of water from diffuser breaks up the thermal plume of buoyant hot water raising from the reactor core, which increases the transport time of the N-16 to the pool surface. Thus, Ar-41 is the main radiological dose contributor. Ar-41 results from irradiation of the air in experimental facilities and dissolved air in the reactor pool water. At the DTRR, Ar-41 is primarily produced in the pneumatic transfer system and the rotating specimen rack. The NRC staff reviewed the licensee's calculations of the production and release of routine airborne radioactive effluents and the resultant doses to the DTRR staff and members of the public.

<u>Occupational Dose</u>

The licensee indicated that the only significant source of Ar-41 that contributed to occupational radiation exposure was that which was generated in, and released from, the reactor tank, the lazy susan, and the PTS into the reactor room. The NRC staff finds that the licensee provided a conservative estimate of the Ar-41 production by assuming that the reactor operates continuously for 8 hours (hrs) per day with the PTS in operation. This is reasonable given that a normal working shift is 8 hours per day. The estimated Ar-41 effluent discharged was 5.52×10^{-9} microcuries (μCi)/milliliter (ml). If the effluent is conservatively assumed to be released directly into the reactor room, instead of being discharged into a hood exhaust duct, the Ar-41 concentration would reach 9.8×10^{-9} μCi/ml. The total effective dose equivalent (TEDE) to a worker in the reactor room for an entire year would be 17 mrem. Note that the reactor operates for an average of about 15 minutes per day instead of the 8 hours per day conservatively assumed as the basis for the calculations. However, the reactor license contains no restriction on operating hours. The NRC staff finds that the licensee's dose estimates are conservative and satisfy the requirements of 10 CFR Part 20. Therefore, based on the information provided above, the NRC staff concludes that the licensee's occupational airborne dose estimates for the operation of the DTRR are acceptable.

<u>Dose to Member of the Public</u>

The NRC staff finds that the licensee calculated the potential dose to the public from the release of Ar-41 during normal reactor operations by using a ratio of the estimated stack concentration to the 10 CFR Part 20 allowable effluent concentration, and 50 mrem/yr (consistent with Appendix B to 10 CFR Part 20, concentration, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage"), as well as a calculation based on continuous operation of the DTRR at full power for one year. The NRC staff also confirmed the acceptability of the calculations and finds the discussion provided in the updated response to RAI-41 (Ref. 38) and RAI-43 (Ref. 4) to be acceptable. The licensee concluded that the resulting dose was 0.65 mrem/yr at the release point (stack) and 0.056 mrem/yr at the fence line. The NRC staff reviewed the licensee's calculation and determined that this was an acceptable conclusion. The dose provided was below the 10 CFR Part 20 limit, and the licensee's ALARA goal of 10 mrem/yr, therefore, on this basis, is acceptable to the NRC staff.

3.1.1.1 TS 3.6.2 Effluents

TS 3.6.2 states the following:

> <u>Specification</u>
>
> The annual average concentration of ^{41}Ar discharged into the unrestricted area shall not exceed 1 $\times10^{-9}$ uCi/ml.

Based on the analysis provided by the licensee and discussed above, the NRC staff finds that the production and control of the DTRR routine airborne radiation sources and atmospheric effluent releases of Ar-41 and N-16 meet the requirements in 10 CFR Part 20. The NRC staff finds that TS 3.6.2, and the dose information provided in the DTRR SAR, as supplemented, provides a reasonable assurance that during continued normal operation of the DTRR, airborne radioactive releases will result in doses to the maximally exposed member of the public of 1 mrem/yr or less, are in compliance with 10 CFR Part 20, and will not pose a significant risk to public health and safety or the environment. Based on the information provided above, the NRC staff concludes that normal operation of the DTRR is within the limits of 10 CFR Part 20 and TS 3.6.2, is, therefore, acceptable.

Liquid Radiation Sources

The DTRR SAR, Section K (Ref. 4) indicates that impurities in the primary coolant become activated by neutrons as they pass through the reactor core. Most of this material is captured by either mechanical filtration or ion exchange in demineralizer resins and, therefore, is dealt with as solid radioactive waste.

As provided in the licensee's response to RAI-44 (Ref. 4), the licensee indicated that liquid experimental solutions are kept in their primary container and are less than 7 ml per sample, with a total activity at the time of removal from the reactor of less than 1 mCi. The experimental solutions are sorted as short-lived (or temporary) or long-lived waste based on their radioactive decay period, and further sorted by solvent (organic or aqueous). Short-lived liquid waste is stored for decay, and long-lived liquid waste is disposed of through the facility radiation safety program. Long-lived radioactive materials are solidified prior to disposal as solid radioactive waste. The primary reactor coolant is not released to the waste-water treatment system, and no liquid radioactive wastes are discharged to the sanitary sewer. The DTRR SAR, Section K.2, states that all floor drains in the laboratory area have been sealed to minimize the possibility of loss of radioactive materials to the municipal sewer system. The NRC staff reviewed the information provided by the licensee and finds that the operation of the DTRR facility, from the perspective of liquid radiation sources, is in compliance with 10 CFR 20.2003, "Disposal by release into sanitary sewerage."

Based on the information provided above, the NRC staff finds that all DTRR liquid radioactive sources from continued normal operation of the DTRR are adequately controlled and will not pose a significant hazard to the public health or safety or the environment. Therefore, the NRC staff concludes that the handling of liquid radioactive sources from normal operation of the DTRR, are acceptable.

Solid Radiation Sources

The DTRR SAR, Section K (Ref. 4), indicates that solid radioactive sources include reactor fuel, a startup neutron source, and fixed radioisotope sources such as those used for instrumentation calibration. Solid wastes include ion exchange resin used in reactor water cleanup, irradiated samples, lab equipment, and anticontamination clothing associated with reactor experiments, surveillance, or maintenance operations.

As provided in the DTRR SAR, solid radioactive waste is sorted by half-life as being short-lived radioactive waste consisting of activation products with half-lives not exceeding 15 days, or long-lived radioactive waste consisting of activation products with half-lives exceeding 15 days. For temporary radioactive waste, fiber packs are provided for disposal. Each pack is classified by a specific number and includes a waste sample log sheet. Full fiber packs are sealed, dated, and affixed with radioactive materials labels and a log sheet. These full packs are stored for 150 days (at least 10 half-lives, or at least 99.9 percent decay), at which time the labels and log sheets are removed, the waste is surveyed to determine that no detectable radioactivity above background is present, and is sent for disposal.

Based on the licensee's response to RAI-1 (Ref. 47), operational experience indicates that the short-lived radioactive waste activity levels are approximately 10 µCi and decay (10 half-lives) to less than 1×10^{-3} µCi. The waste is surveyed for residue long-lived isotopes by measurement in a gamma spectroscopy system which consists of two high purity germanium detectors in a shielded block. The typical minimum detectable activity is 5.4×10^{-6} μCi above background. If any residual radioactivity is detected, the waste is allowed to decay to background activity levels, or the long-lived isotope is separated and transferred to long-lived radioactive waste.

Long-lived solid radioactive wastes are collected in an approved steel drum, with the type of waste, radionuclides, and quantities recorded on a waste sample log sheet. When the drums are full, they are sealed and health physics personnel are notified to arrange for proper disposal.

The NRC staff reviewed the DTRR SAR, as supplemented, and finds that solid radioactive sources and wastes from normal operation of the DTRR are controlled and have not resulted in any significant exposures. Therefore, based on the information provided above, the NRC staff concludes that the control of solid radioactive sources and wastes at DTRR is acceptable.

3.1.2 Radiation Protection Program

The regulations in 10 CFR 20.1101(a) require each licensee to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities. As indicated in the DTRR SAR, Section K (Ref. 4), the production and use of radioactive materials within the reactor laboratory are subject to the DTRR radiation protection program under the direction of the radiation safety officer (RSO). The program includes, among other topics, radiation fundamentals, pertinent Federal and State regulations, contamination control, inventory control, and monitoring. The licensee states that procedures are in place for radiation protection during normal operation and reactor experiments.

3.1.2.1 TS 6.3 Radiation Safety

TS 6.3 states the following:

> 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."

The licensee states that the DTRR radiation protection program is prepared by the RSO, who is also responsible for the implementation of the program. The program is designed to be compliant with NRC and state regulations, and to follow the guidelines in ANSI/ANS-15.11-1993 (R2004), "Radiation Protection at Research Reactor Facilities" (Ref. 29). The goal of the program is to limit radiation exposures to employees and the public, and radioactivity releases, to a level that is ALARA without unnecessarily restricting the operation of the facility. The RSO is overseen by the Radiation Safety Committee (RSC), which is chaired by a representative of management from the Dow environmental health, and safety organization. This committee has sufficient authority to influence changes in operations necessary to protect employees and the public from the hazards of ionizing radiation. The RSO is also a member of the ROC and is responsible for providing an annual audit of the radiation protection program for content and implementation. A summary of the audit findings is provided to the ROC. The reactor supervisor is responsible for ensuring that the requirements of the radiation protection program are followed and that exposure to radiation for workers and the public are kept ALARA. The RSO is responsible for training and instructing the operation staff for radiation protection, including the use of personnel and area monitoring equipment. Personnel who need access to the facility, but who are not reactor staff, are either escorted by trained personnel or provided facility access through training. Radiation training for licensed operators and operating staff is integrated with the training and requalification program, which includes training on the ALARA principle. Specific training requirements of 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspections and Investigations," and 10 CFR Part 20, the DTRR radiation protection program and the emergency plan are included in the radiation training.

The NRC staff reviewed the DTRR radiation protection program and finds that it complies with NRC regulations and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.11-1993 (R2004). The NRC staff concludes that TS 6.3 is acceptable.

3.1.3 ALARA Program

The licensee states that it established a program designed to keep radiation exposures to personnel ALARA so as to comply with 10 CFR 20.1101. This includes using methods and procedures that shield radiation sources and personnel; increase the distance between an exposure point and a radiation source; reduce the time a person might be exposed to a given dose rate; containing sources; and the use of careful, thoughtful, advanced planning when working in an area that might contain a radiation field. Various administrative controls have been put into place to accomplish the ALARA goals. The ALARA program is controlled by the RSC to ensure that unnecessary exposures to radiation are avoided. Individuals are not allowed to occupy areas where radiation levels are above background unless it is necessary in the performance of their job. Good housekeeping and contamination control are practiced in all radioactive materials use and storage areas. Any personnel exposure problems are investigated at levels well below regulatory dose limits and that trigger a formal investigation by the RSO. All personnel exposures are reviewed annually to identify trends in exposure and opportunities for further dose reductions.

The NRC staff reviewed the DTRR ALARA program and finds that, as part of the licensee's commitment to ALARA, specific goals are established to help ensure that actual exposures are no greater than 10 percent of the occupational limits and no greater than 50 percent of the public limits in 10 CFR Part 20. The NRC staff finds that the DTRR ALARA program is consistent with TS 6.3 on radiation safety and uses the guidance of ANSI/ANS-15.11-1993 (R2004). The NRC staff finds that the DTRR ALARA program also defines and requires surveys, monitoring, radiation records, and personnel dosimetry. The NRC staff finds that the DTRR ALARA program complies with the guidance of ANSI/ANS-15.11-1993 (R2004) and with 10 CFR 20.1101, and provides reasonable assurance that radiation exposures will be maintained ALARA for all activities at the DTRR facility. Furthermore, the NRC staff reviewed the results of the NRC IRs for the years 2005 through 2013, which indicated that the licensee has been consistent in meeting its ALARA goals. Therefore, based on the information provided above, the NRC staff concludes that the DTRR ALARA program is acceptable.

3.1.4 Radiation Monitoring and Surveying

The DTRR SAR, Section K (Ref. 4) indicates that radiation levels at the DTRR are measured in unrestricted areas at locations in closest proximity to the strongest radiation sources to confirm that acceptably low dose rates exist in those areas. Zones of potential radiation hazards are clearly marked, and areas in which the dose rate exceeds 5 mrem/hr are posted as "Radiation Areas," in accordance with 10 CFR 20.1901, "Caution signs," and 10 CFR 20.1902, "Posting requirements." Although areas with elevated airborne radioactivity levels generally are not encountered, if there is a potential for air concentrations in excess of 10 percent of the values in Table 1 of Appendix B to 10 CFR Part 20, the areas are posted as an "Airborne Radioactivity Area." In addition to radiation levels, monthly swipes are taken at 20 critical locations in the facility and counted for alpha and beta radiation for contamination control determinations. The DTRR SAR, Section K.1.4 states that dosimetry are placed on the walls of the reactor room and console. The dosimeters are exchanged quarterly, and additional independent surveys of the facility are conducted twice a year by Dow Chemical Company's industrial hygiene department.

The regulations in 10 CFR 20.1501(a) requires each licensee to make (or cause to be made) radiation surveys that have the following characteristics:

1. may be necessary for the licensee to comply with the regulations; and

2. are reasonable under the circumstances to evaluate the following:

    i. the magnitude and extent of radiation levels;
    ii. concentrations or quantities of radioactive material; and
    iii. the potential radiological hazards.

The regulations in 10 CFR 20.1501(b) requires licensees to ensure that instruments and equipment used for quantitative radiation measurements (e.g., dose rate and effluent monitoring) are calibrated periodically for the radiation measured.

3.1.4.1 TS 3.6.1 Radiation Monitoring

TS 3.6.1 states the following:

> 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.

According to DTRR SAR, Section L.3, the ARM setpoint is 2.0 mR/hr, the setpoint for the CAM alarm level is 4,000 counts/min, and the water radioactivity alarm setpoint (see SER section 2.5.5.2) is 300 counts/second. These values represent typical background readings and the instrument setpoints may be adjusted accordingly to ensure any change above background is identified, but nuisance alarms are prevented from unnecessarily distracting the reactor operators.

TS 3.6.1, helps ensure that at an ARM and an CAM are operable to support reactor operations. The DTRR SAR, Section K.1.4 provides an overview of radiation monitoring and surveillance at the DTRR facility. There are fixed radiation monitors in the facility (CAM and ARM) and monitors in the coolant line. Portable detectors are available for use and are checked for operability daily. Monthly wipes are taken at 20 critical locations in the facility including doorknobs, floors, and phones. Any identified contamination is immediately cleaned. The NRC staff finds that TS 3.6.1 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and helps ensure that the radiation monitoring systems will alert the operator in the event that alarm setpoints are exceeded during reactor operations. Footnote 1 provides an alternate monitoring method should the ARM become inoperable. The analysis provided by the licensee in its response to RAI-56 (Ref. 14) indicates that a 1-hour period to install a comparable gamma-sensitive measuring instrument will ensure that any potential doses to the workers or members of the public will remain below the limits in 10 CFR Part 20.

The licensee indicated that the RSO is responsible for calibration of the instruments using written procedures. Calibration is indicated through labels on each instrument, and the records are maintained by the DTRR staff. In addition to the monitors required by TS 3.6, the licensee has an ARM, a CAM, and a set of portable radiation survey instrumentation that covers, with sufficient ranges, the various types of radiation that may be encountered at the DTRR. Twice a year, with no more than 7.5 months between each review, the RSO is responsible for calibrating the ARM and CAM. Portable radiation monitors are checked for calibration as part of the daily startup and shutdown procedures, and the RSO coordinates calibration, when required. The NRC staff reviewed the NRC IRs and finds that the calibration of portable survey meters and friskers were performed by a contracted company, and that the calibration of fixed area radiation detectors were performed by the DTRR staff. The NRC staff finds that the calibration frequency and records are maintained as required.

The NRC staff finds that the licensee's equipment is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to help ensure compliance with 10 CFR 20.1501(a) and (b). Therefore, the NRC staff concludes that the DTRR radiation monitoring and surveying programs are acceptable.

3.1.5 Radiation Exposure Control and Dosimetry

The DTRR SAR, Section K (Ref. 12) indicates that radiation exposure control depends on factors such as the facility design features, operating procedures, training, and equipment. Design features include shielding, ventilation, containment of the inventory within the fuel, entry control, protective equipment, personnel dosimetry, and annual dose versus location estimates. The shielding for the DTRR is similar to shield designs used successfully at many other similar reactors. The principal design feature for control of radiation exposure during operation is the column of water around and above the reactor core, plus the location of the reactor tank being partially below ground level. The DTRR is designed so that radiation from the core area can be accessed by a vertical tube for research. The radiation exposure is controlled by restricting access to areas of elevated radiation fields, including restricting access for maintenance to the roof of the reactor building during reactor operations.

The regulations in 10 CFR 20.1502 require monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the limits described in 10 CFR 20.1501. The regulation requires monitoring of individuals entering a high or very high radiation field in which an individual could receive a dose equivalent of 0.1 rem in 1 hour. The licensee has established two investigation levels in the radiation protection program for TEDE, skin or extremities, lens of the eye, and other organs or tissue (other than the lens of the eye). For doses below Investigation Level I (with a quarterly TEDE greater than 125 mrem), no further action is required since these doses are well below the regulatory limits, unless deemed necessary by the RSO. For doses in excess of Investigation Level I, but less than Investigation Level II (with a quarterly TEDE greater than 375 mrem), the RSO conducts an investigation of the root cause, and a copy of the individual's NRC Form 5 is presented at the next scheduled RSC meeting. The RSC reviews the dose compared with the doses for other individuals performing similar tasks, and the review is recorded in the committee meeting minutes. No further action is taken since these doses are below regulatory limits. For doses equal or greater than Investigation Level II (with a quarterly TEDE greater than 375 mrem), the RSO will investigate the root cause and, if warranted, take action. A report of the investigation, any actions taken, and a copy of the individual's NRC Form 5 (or its equivalent) will be presented to the RSC upon completion of the investigation. The DTRR maintains a radiation dosimetry program, and restricts access to areas of elevated radiation fields to demonstrate compliance and helps ensure personnel protection.

The DTRR SAR, Section K (Ref. 4) indicates that the licensee collects and maintains records of occupational exposure information, using the appropriate NRC forms. Records of self-reading dosimeters are kept in a logbook that the DTRR staff maintains as permanent records, as are measurement results of accidental releases to the environment. Radiation levels outside the restricted areas are monitored using dosimeters. Air samples outside the reactor facility are taken, as necessary, to detect radioactive materials that may be dispersed in the air, or if there is reason to believe that environmental releases will exceed 10 percent of the allowable concentrations in Appendix B, Table 2 to 10 CFR Part 20.

The NRC staff reviewed the information provided in the DTRR SAR, Section K.1.5, and the licensee's responses to RAI-46 (Ref. 4), and finds that the DTRR radiation exposure and control program is acceptable. The NRC staff finds historically low radiation doses and the application of the equipment and procedures used to be acceptable. The personnel exposures at DTRR facility are controlled through satisfactory radiation protection and ALARA programs. The NRC staff finds these conclusions are consistent with information provided in DTRR annual reports for the period 2005 through 2013, and its review of the health physics program in NRC IRs from 2005 through 2013. The annual reports indicate that occupational exposures have been consistently maintained below 25 percent of the annual regulatory limits in 10 CFR Part 20, which is the licensee's quarterly Investigation Level I. The NRC IRs indicate that the radiation protection program satisfies regulatory requirements and that the personnel dosimetry results comply with 10 CFR Part 20 and the applicable TSs. The NRC IRs indicate that highest individual exposures in 2010 were 70 mrem whole body and 10 mrem to extremities, which meets the licensee's ALARA goal of 10 percent of the applicable NRC limits. Therefore, based on the information provided above, the NRC staff concludes that the licensee's control of personnel exposures and dosimetry is acceptable.

3.1.6 Contamination Control

The DTRR SAR, Section K (Ref. 4) indicates that radioactive contamination is controlled at the DTRR facility by using written procedures for radioactive material handling, using trained personnel, and implementing a monitoring program designed to detect contamination in a timely manner. The DTRR SAR, Section K.1.6 indicates that contamination is identified by observation and daily or monthly surveillance. Work involving the use of radioactive materials is performed in a manner to minimize the spread of radioactive contamination in the facility, and in areas that are properly posted as radiation or contamination areas. Most activated samples remain in sealed vials throughout the counting process. No approved experiments require post-irradiation processing, which minimizes the potential for spreading removable contamination. Personnel wear protective clothing to prevent skin contamination and dispose of waste in appropriate containers per established procedures. Geiger-Mueller survey meters are used to detect contamination on personnel and equipment in the facility. Contamination surveys of work areas are conducted monthly, and any areas of elevated contamination (above background) are promptly cleaned. The cleanup of minor contamination is performed by the ROs and research staff, and larger areas of contamination are cleaned under the direction of the RSO and environment health and safety staff. Fixed contamination is clearly labeled.

The NRC staff finds that the DTRR radiation protection program to be robust and mature, helping to minimize the spread of radioactive contamination. Based on the NRC staff's review of the DTRR SAR and historic performance of the facility's program for contamination control, the NRC staff concludes that adequate contamination controls exist to prevent the spread of contamination at the DTRR facility.

3.1.7 Environmental Monitoring

The DTRR SAR, Section K.1.7, provides information about the DTRR facility environmental monitoring program. Environmental monitoring is conducted at DTRR to ensure compliance with Subpart F of 10 CFR Part 20, "Surveys and Monitoring," and applicable DTRR TSs. Installed monitoring systems include an ARM and a CAM, which have been managed and maintained in a comprehensive program. An ARM is required by TS 3.6 within the reactor bay.

With the exception of Ar-41, there are no pathways for radioactive materials from the DTRR to enter the unrestricted environment during normal operations. DTRR TS 3.6 requires one CAM in the reactor bay, which alarms in the reactor bay and control room. Calibration of the CAM is accomplished as required by the TS and in accordance with facility procedures. Several optically stimulated luminescence dosimeters are placed around the inside walls of the reactor facility. These dosimeters historically have shown minimal radiation exposure to the environment. There are no liquid discharges from the reactor facility. As required by 10 CFR 20.1501, "General," airborne radiation and contamination surveys are conducted to ensure compliance with regulations and to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive material, and potential radiological hazards.

Based on its review of the information provided in the DTRR SAR, Section K, as described above, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the DTRR on the environment.

3.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to ensure that radioactive waste materials are identified, assessed, controlled, and disposed of in conformance with all applicable regulations and in a manner to protect the DTRR staff, the health and safety of the public and the environment. The DTRR SAR, Section K.2, provides an overview of the radioactive waste management program for the DTRR facility.

3.2.1 Radioactive Waste Management Program

The objectives of the DTRR radioactive waste management program are to minimize, properly handle, store, and dispose of the radioactive waste. The DTRR radioactive waste management program is reviewed during NRC inspections. The NRC staff's review of the NRC IRs confirmed that acceptable controls are in place to prevent uncontrolled personnel exposures from radioactive waste operations, and that the necessary accountability is provided to prevent unauthorized release of radioactive waste. Therefore, based on the information provided above, the NRC staff concludes that the DTRR radioactive waste management program is acceptable.

3.2.2 Radioactive Waste Controls

The DTRR SAR, Section K (Ref. 4), indicates that radioactive materials produced in the reactor are segregated according to half-life. Radioactive materials with half-lives less than 15 days are stored for longer than 6 months, then monitored for residual radioactivity and discharged as chemical waste if no radioactivity above background is found. If residual radioactivity is detected, the radionuclides are identified and the material is treated as radioactive waste. Radioactive materials with half-lives greater than 15 days are stored until they can be disposed of as low-level radioactive waste through the industrial hygiene function of Dow Chemical Company. The radioactive waste historically has been sent to licensed commercial facilities for disposal using a waste broker. Long-lived radioactive materials are solidified before disposal, and spent ion-exchange resins from the water purification system are dried and disposed of as long-lived radioactive waste. All floor drains in the laboratory area have been sealed to minimize the possibility of the loss of radioactive materials to the sewer system, which leads to the municipal waste water treatment system. Although disposal of liquids to the sanitary sewer system is permitted under 10 CFR 20.2003, according to the DTRR SAR, the licensee states that liquid wastes are not released from DTRR facility.

Based on its review of the information in the DTRR SAR, as described above, the NRC staff finds that acceptable procedures are in place to monitor the radiation exposure from radioactive waste, and to perform required handling operations. Furthermore, the NRC staff concludes that the DTRR facility has adequate radioactive waste controls in place to monitor the radiation exposure from radioactive waste, to perform required handling operations, and prepare the material for transfer to offsite disposal.

3.2.3 Release of Radioactive Waste

The DTRR SAR, Section K (Ref. 4), states that the DTRR does not allow the release of radioactive waste into the environment other than gaseous radioactive effluents, notably Ar-41, which is regulated in accordance with 10 CFR Part 20 discharge limits. Gaseous effluents are monitored using a CAM in the exhaust to ensure compliance with the regulatory limits (i.e., the allowable effluent concentration for Ar-41 is 10^{-8} μCi/cm^3) and the facility TSs.

The licensee states that if contaminated liquids are produced, they are contained locally, added to an absorbent or solidified, and transferred into a solid radioactive waste disposal drum in preparation for transfer to a licensed burial facility. The NRC staff reviewed the information provided in the DTRR SAR, Section K.2, and NRC IRs, and finds that controls are in place to eliminate releases of radioactive material into the sanitary sewer system. Therefore, liquid releases from the DTRR do not pose a significant risk to public health and safety.

The NRC staff reviewed the information provided in the DTRR SAR, and concludes that the DTRR has adequate controls in place to control or eliminate releases of radioactive material into the environment.

3.3 Radiation Protection Program and Waste Management Conclusions

Based on the evaluation of the information presented in the DTRR SAR, observations of the licensee's operations during site visits, information in the licensee's annual reports for the period of 2005 through 2013, and results of the NRC inspections as documented in IRs, the NRC staff concludes the following regarding the DTRR radiation protection and waste management:

- The DTRR radiation protection program complies with the requirements in 10 CFR 20.1101(a). The program is acceptably staffed and implemented, and provides reasonable assurance that the facility staff, the environment, and the public are protected from unacceptable radiation exposures.

- Radiation sources and effluents are acceptably characterized and controlled. The radiation protection organization has acceptable lines of authority and communication to carry out the program.

- The programs provided for the control of radiological effluents, when operated in accordance with the TSs, are acceptable to ensure that releases of radioactive materials from the facility are within the limits of the Commission's regulations and are ALARA.

- The DTRR ALARA radiation protection program complies with the requirements of 10 CFR 20.1101(b) and uses the guidelines of ANSI/ANS-15.11-1993 (R2004) implementing time, distance, and shielding to reduce radiation exposures. A review of historical radiation doses and current controls for radioactive material in the DTRR facility provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA.

- The results of radiation surveys carried out at the DTRR facility, doses to the persons issued dosimetry, and results of the environmental monitoring program help verify that the radiation protection and ALARA programs are effective.

- The licensee properly identifies, describes and controls potential radiation sources.

- Facility design and operational procedures limit the production and release of Ar-41 and N-16 and control the potential for facility staff and public radiation exposures. The licensee provided conservative calculations of the quantities of these gases released into restricted and unrestricted areas, which provides reasonable assurance that doses to the DTRR staff and public will be below applicable 10 CFR Part 20 limits.

- The radioactive waste management program provides reasonable assurance that radioactive waste released from the facility will neither exceed applicable regulations nor pose an unacceptable radiation risk to the environment and the public.

The NRC staff reviewed the DTRR radiation protection program and waste management summary as described in the DTRR SAR, as supplemented. The NRC staff finds that the licensee implemented adequate and sufficient measures to minimize radiation exposure to workers and the public. Furthermore, the NRC staff concludes that there is reasonable assurance that the DTRR radiation protection and waste management programs will provide acceptable radiation protection to its workers, the public, and the environment.

4. ACCIDENT ANALYSIS

The DTRR SAR, Chapter M, in conjunction with the licensee’s response to RAI-52 through RAI-57, provided accident analyses to demonstrate that the health and safety of the public and workers are protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analyses provide the basis to establish the DTRR TSs described in this report. The accident analysis helps ensure that no credible accident could lead to unacceptable radiological consequences to the DTRR staff, the public, or the environment. Additionally, the licensee analyzes the consequences of the MHA, which is an event involving the rupture of the cladding of an irradiated fuel element in air. The MHA is considered the worst-case fuel failure scenario for a TRIGA reactor that would lead to the maximum potential radiation hazard to facility personnel and members of the public. The results of the MHA are used to evaluate the ability of the licensee to respond and mitigate the consequences of this postulated radioactive release.

NUREG-1537 recommends licensees consider the applicability of each of the following accident scenarios:

- the MHA
- insertion of excess reactivity
- LOCA
- loss-of-coolant flow accident
- mishandling or malfunction of fuel
- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment

4.1 Accident Analysis Initiating Events and Determination of Consequences

4.1.1 The Maximum Hypothetical Accident

For the DTRR, the MHA is defined as the rupture of the clad of one fuel element in air. The scenario assumes that such an accident occurs after a long period of operation at full licensed power so that the inventories of all the radiologically consequential radionuclides in the scenario are near saturation values. The licensee assumes at the time of clad failure that all of the gaseous fission products that have accumulated in the gap are released abruptly into the air with no radioactive decay; this includes the release of noble gases and halogens. The information used to evaluate the MHA was provided by the licensee’s response to RAI-52 (Refs. 14 and 52), which replaces the description in the DTRR SAR, Chapter M (Ref. 4).

Nuclide Inventory

For determining the radionuclide inventories, the licensee assumes the reactor had been operating continuously for 1 year at 300 kWt. The licensee indicated that DTRR contains fuel with characteristics that are similar to the Oregon State University (OSU) TRIGA reactor LEU fuel. The licensee uses the fission product inventory of 1-year continuous operation in the fuel element of the OSU reactor with a peak power density of 18.2 kWt, and apportions this inventory to estimate the inventory in a fuel element in the DTRR, based on the highest fuel element power of 6.08 kWt determined for the DTRR LCC.

The NRC staff performed confirmatory calculations using the highest DTRR fuel element power of 6.08 kWt to determine the saturation fission gas inventory (fuel element power x fissions x fission yield). The NRC staff compared these calculations to estimates of radionuclide inventories for select halogens and noble gases at shutdown as provided by the licensee and those estimated from information in NUREG/CR–2387 “Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors,” (Ref. 24). Therefore, based on the results of the confirmatory calculations, and the information provided above, the NRC staff concluded that the licensee’s MHA radioactive source estimates are acceptable.

Release Fractions

The licensee calculated the releases of noble gases and halogens from the fuel matrix to the fuel gap using the GA-developed correlation for fission product release in “Fission Product Release From TRIGA-LEU Reactor Fuels,” issued October 1980 (Ref. 30). This correlation estimates the release based on the average fuel temperature. GA experiments determined that the release fraction was constant at 1.5×10^{-5} for fuel temperatures below 300 degrees C. The licensee calculated a peak fuel temperature of 247 degrees C for the peak power of 6.08 kWt, and therefore, used the release fraction of 1.5×10^{-5} for fission produce release to the fuel gap.

The licensee assumed that the fuel clad failure occurs in the air, and the fuel gap inventory of both the noble gases and halogens release directly to the reactor room. This assumption is conservative for halogens (e.g., iodines) because they are chemically active and typically become trapped by materials with which they come into contact. The guidance typically used (Technical Information Document (TID) 14844, “The Calculations of Distance Factors for Power and Test Reactor Sites,” March 1962 (Ref. 31) and RG 3.33, “Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant,” issued April 1977 (Ref. 32)) indicates that most iodines either will not become airborne, or will not remain airborne after they are released. This guidance provides that only 25 percent of the halogens in the gap are released to the reactor room and to the outside environment (50 percent of the gap inventory is released and about 50 percent of halogens will stay in the reactor building). If the cladding fails in the water, which is the most likely scenario, then the fraction available for release to the air and environment will be even lower.

Table 4-1 provides a comparison of the licensee and the NRC staff’s confirmatory calculations of the release fractions of noble gases and halogens to the reactor room. Based on the information provided above, the NRC staff concludes that the licensee’s assumption regarding the release of halogens to the reactor room and the environment is conservative and will lead to computed halogen doses that are higher than expected in the actual MHA scenario. The NRC staff calculation for halogens includes the 25 percent release fraction described above.

Table 4-1 Total Release Fractions

Release Fractions	Noble gas	Halogens
Release to air (DTRR)	1.5×10^{-5}	1.5×10^{-5}
Release to air (NRC staff confirmatory calculation)	1.5×10^{-5}	3.8×10^{-6}

Atmospheric Dispersion Factor χ/Q

The licensee used the Gaussian plume diffusion model to calculate nuclide concentrations at two selected downwind distances (the DTRR fence line at 23 m (75 ft), and at the property boundary at 100 m (328 ft)). The licensee assumed a stable atmospheric class F, with a wind speed of 2 m (6.6 ft) per second at a height of 10 m (32 ft). In addition, the licensee estimated the dispersion factors in lateral (y-axis) and axial (z-axis) from Figures 1 and 2 of RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, issued February 1983. The licensee assumed the downwind condition was affected by fumigation conditions and used Equation 2 of RG 1.145, Regulatory Position C.1.3.1. The NRC staff used the RG 1.145, Revision 1, method to determine the atmospheric dispersion factors at the selected distance from the reactor building for comparisons and confirmatory analysis. The NRC staff verified by comparison that the licensee's method and data used in the atmospheric dispersion factor calculations for the DTRR MHA dose calculations were reasonable, conservative, and acceptable.

Dose Calculations

The licensee calculated the occupational dose for an individual in the reactor room. Boundary conditions for these calculations included assuming the failure of the hottest fuel element, incorporating conservative release fractions, and using the reactor room volume of 130 cubic meters (m^3) (4591 ft^3). Other parameters used in the dose calculations include a breathing rate of 0.02 m^3/min (0.7 ft^3/min), consistent with the value given in Appendix B to 10 CFR Part 20; and a ventilation rate of 50 m^3/min (1766 ft^3/min.). In addition, the licensee used dose conversion factors (DCFs) for the inhalation and submersion external exposure pathways from the U.S. Environmental Protection Agency (EPA) Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988 (Ref. 33). The licensee used DCFs for submersion in the air to the halogen isotopes from FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," issued September 1993 (Ref. 34). If a DCF value is not given for an isotope in the FGR No. 11, or No. 12, the licensee uses a pseudo value from the nearest isotope. For example, because of very low half-lives, FGR 11 does not provide inhalation DCF values for bromine (Br)-85, or iodine (I)-136. The licensee uses Br-84 and I-131 DCF values to represent those of Br-85 and I-136, respectively. The NRC staff reviewed this practice and concluded that it was acceptable.

The licensee considered the following locations, as described in Scenarios 1 through 4 below, (and provided in their response to RAI-52 (Refs. 14 and 52)), to provide the bounding dose estimates for an MHA event. The ventilation system is required to be operating during reactor operation in accordance with DTRR TS 3.5. If a significant radiological release occurred, reactor operators would manually isolate (shutdown) the ventilation system to reduce the potential release of any radioactive material (the ventilation system does not have any automatic response to a radioactive event). The scenarios below assume the DTRR ROs respond in accordance with procedures to the radiological event and isolate the ventilation system. This results in a more conservative (higher) dose calculation to persons in the reactor bay and the control room than if the ventilation system remains in operation. However, In Scenarios 3 and 4, it was more conservative to leave the ventilation system in operation. Therefore, the operator's actions to isolate the ventilation system results in the lowest dose to members of the public in unrestricted areas.

- Scenario 1–Reactor Bay Room: the licensee considers this area a radiological restricted area and assumes the occupants are radiation workers (occupational exposure limits apply) who will be exposed to the airborne gaseous fission products with no credit for radionuclide decay. The ventilation system is assumed to be off. It is estimated that the workers will spend about 30 minutes to bring the reactor to a safe condition before evacuating the reactor room.

- Scenario 2–Control Room: the licensee considers this a radiological unrestricted area and assumes that some occupants may not be radiation workers (i.e., 10 CFR 20 public dose limits apply), the ventilation system is off, and air leakage into the control room will be through narrow openings around the door. The licensee calculated a leak rate of 0.23 m^3/second (8.1 ft^3/second) into the control room. The control room has a volume of 66.3 m^3 (2341 ft^3) and is assumed to be evacuated within 15 minutes. No air is released from the control room, and no credit is taken for radionuclide decay. Radiation exposure from gamma rays emanating from the reactor bay room into the control room is included in the dose calculation.

- Scenario 3–Outside the Dow Chemical Company facility security fence boundary at 23 m (75 ft) from the DTRR: the licensee considers this location as the nearest radiological unrestricted area with occupants who are not radiation workers (i.e., 10 CFR 20 public dose limits apply). This location is within the owner controlled area of the Dow Chemical Company and evacuation in accordance with the DTRR emergency plan is expected to occur within 30 minutes. Dose calculations were performed with the ventilation system on and off in order to assess each condition. With the ventilation system on, the radioactive gases are released to the outside atmosphere at a rate of 50 m^3/min (1766 ft^3/min). The licensee assumes the entire MHA radioactive source to be exhausted to the outside with one complete room air exchange, which would occur in 2.6 minutes. The occupant would be exposed (submersion and inhalation) by the plume. With the ventilation system shutdown, the gases are generally contained within the reactor bay. Since some building leakage occurs, and the total radiation dose includes exposure from both the plume and the gamma rays (shine) from the reactor bay.

- Scenario 4 – Outside the Dow Chemical Company property, nearest location at 100 m (328 ft): the licensee considers this location nearest to the DTRR and outside the jurisdiction of the Dow Chemical Company's emergency plan for evacuation of members of the public. As such, the potential MHA exposure duration is not limited by any evacuation (e.g., 30 minutes). The licensee assumes the ventilation system is isolated (shutdown), and the building leakage rate remains 0.23 m^3/second (8.1 ft^3/second), as calculated above, which results in all radioactivity effluent being released within 48 hours.

The potential radiation exposure was limited by the evacuation times footnoted in the tables below. Evacuation of Scenarios 1, 2, and 3 are included in the DTRR emergency plan and are expected to occur within 15 or 30 minutes from the initiation of the event. In addition, the licensee indicates that exposure in other locations within the building will be bounded by dose estimates from air leakage into the control room. Furthermore, the DTRR emergency plan provides specific evacuation criteria, including annual exercises and drills, and alarm and communication testing. The NRC staff reviewed IRs from 2005 through 2013 and no issues or concerns were identified. The evacuation time noted in the emergency plan from the past performance of emergency exercises was significantly less than the evacuation times assumed in the scenarios. Based on the information provided above, the NRC staff concluded that the assumptions provided in Scenarios 1 through 4 are acceptable.

The NRC staff reviewed the licensee's scenarios and methodology described above for estimating the doses within and beyond the confines of the reactor facility in case of an MHA fission product release and therefore, concludes that the scenarios and methodology are acceptable.

<u>MHA Dose - Confirmatory Analysis</u>

The NRC staff performed confirmatory calculations of the MHA TEDE, for those scenarios nearest the limits in 10 CFR 20, in order to demonstrate the adequacy of the licensee's MHA results. These calculations were performed using the assumptions, geometry, and radiological source terms that were verified to be consistent with those used by the licensee.

The occupational and public TEDE doses were calculated for the three locations described in Scenarios 1 through 3 above. In these calculations, the licensee used the DCFs from FGR No. 11 (Ref. 33) and FGR No. 12 (Ref. 34). The confirmatory analysis also used the DCFs from FGR No. 11 and a combination of DCFs from FGR No. 11 and FGR No. 12. FGR No. 12 provides submersion DCFs for all radionuclides, whereas FGR No. 11 submersion dose factors were limited to noble gases only. The NRC staff calculated the TEDE for the above scenarios using the licensee's provided isotope inventory and the DCFs from FGR No. 11 and FGR No. 12. In addition, the doses were also calculated using the isotopes at their saturation inventories along with the DCFs from FGR No. 11 and FGR No. 12. These two confirmatory analyses determined the adequacy of the results presented by the licensee. They also provided insights into the significance of differences between the methods used to determine the initial inventory. These analyses confirmed that the differences in doses related to differences in inventories between saturation values versus values provided by the licensee to be less than 2 percent. Therefore, the NRC staff concludes, based on the information provided above, that the licensee's estimated radionuclide inventories were acceptable.

For the evaluations of the dose results, the NRC staff used the saturated isotope inventories. Table 4-2 through Table 4-4, summarize the MHA dose results the licensee provided in its response to RAI-52 (Refs. 14 and 52), along with the results from the NRC staff's confirmatory calculations using a combination of FGR No. 11 and FGR No. 12 DCFs. These results indicate that the occupational and public doses remain below the regulatory limits in 10 CFR 20.1201, “Occupational dose limits for adults,” and 10 CFR 20.1301, “Occupational dose limits for individual members of the public.” Table 4.3 was updated to include the direct radiation contribution from the reactor bay of 66 mrem/hr (RAI-2, Ref. 47).

Table 4-2 Scenario 1: MHA Occupational Dose Estimates–Reactor Bay

Time (min)	**Total Effective Dose Equivalent (mrem)**		
	DTRR Results	**NRC Staff Confirmatory Calculation Results**	**Dose Limit**
	TEDE	TEDE	5000
15-min	438	502	
30-min*	875	1,004	
60-min	1,750	2,007	
***expected exposure duration**			

Table 4-3 Scenario 2: MHA Public Dose Estimates–Control Room

Time (min)	**Total Effective Dose Equivalent (mrem)**		
	DTRR Results	**NRC Staff Confirmatory Calculation Results**	**Dose Limit**
	TEDE	TEDE	100
15-min*	85	94	
30-min	170	189	
60-min	340	380	
***expected exposure duration**			

Table 4-4 Scenarios 3 and 4: MHA Public Dose Estimates–Offsite

<table>
<tr><th rowspan="2">Downwind Distance (m)</th><th colspan="3">Total Effective Dose Equivalent (mrem)</th></tr>
<tr><th>DTRR Results</th><th>NRC Staff Confirmatory Calculation Results</th><th>Dose Limit</th></tr>
<tr><td></td><td>TEDE</td><td>TEDE</td><td rowspan="6">100</td></tr>
<tr><td rowspan="2">Scenario 3 – 23 m
(30 minute dose)</td><td>6.7*</td><td>7.7*</td></tr>
<tr><td>4.5**</td><td>N/A</td></tr>
<tr><td>Scenario 4 – 100 m
(48 hour dose)</td><td>5.9**</td><td>N/A</td></tr>
<tr><td colspan="3">*** ventilation system on / ** ventilation system off**</td></tr>
</table>

The NRC staff considers the dose calculations applicable to the control room to be representative of a “bounding” or potential maximum dose to other individuals within the connected buildings, because there are no direct air leakages into other locations from the reactor room, and any air leakage from the control room will result in a lower exposure. As indicated above, the NRC staff finds that the potential MHA dose to the worker and public was accurately calculated and the result was within the regulatory limit for each scenario evaluated.

The NRC staff concludes, based on its review of the licensee’s dose calculations, and the results of the NRC staff’s confirmatory calculation, that the MHA dose results (provided above) clearly demonstrate that the maximum TEDE doses are well below the occupational limit in 10 CFR 20.1201 and the public dose limit in 10 CFR 20.1301.

MHA Dose Calculation Conclusions

The NRC staff reviewed the MHA analysis for all three scenarios that the licensee provided, as well as the dose calculation results, and concluded that the licensee used appropriate assumptions and analytical techniques and that its conclusions were appropriate and acceptable. The independent confirmatory dose calculations that the NRC staff performed demonstrate that the licensee properly evaluated the postulated doses from the MHA scenarios. The results of the NRC staff’s confirmatory dose calculations are consistent with the dose results that the licensee provided. In addition, the doses from the postulated scenarios provided above demonstrate that the maximum TEDE doses were below the occupational limits in 10 CFR 20.1201, and the public exposure limits in 10 CFR 20.1301. Based on the results of the estimated doses provided above, and confirmed by the NRC staff’s independent calculations, the NRC staff concludes that the results of the licensee’s analysis of the MHA doses are acceptable.

4.1.2 Insertion of Excess Reactivity

The DTRR insertion of excess reactivity accident scenario was provided in the licensee’s response to RAI-53 (Ref. 10) and supplemented in the T-H report (Ref. 12). The licensee’s method used the generalized modeling of reactivity insertion events to determine the resulting maximum reactor power and peak fuel temperature. The licensee analyzed a step insertion of the maximum worth of an unsecured experiment ($0.75), as stated in TS 3.7.1, Specification b. The initial conditions were: fuel temperature of 200 degrees C, β_{eff}=0.0070, the prompt-neutron lifetime of 60 μsec, and FTC of -$0.0181/degrees C. With an initial power of 300 kWt, the licensee's results were an average fuel temperature of 255 degrees C, with a corresponding temperature increase of 55 degrees C. The results, presented in Section 2 of this SER, include the maximum peaking factor for the highest power fuel element of 2.613 for the 2011 OC configuration. The calculated temperature rise at the hottest point in the highest power fuel element was calculated to be (2.613×55 degrees C) 143.7 degrees C. The addition of this temperature increase (to the maximum fuel temperature of 246.7 degrees C for any of the T-H conditions considered in the SER, Section 2) results in the maximum predicted fuel temperature for the step removal of an experiment with the maximum unsecured reactivity worth of $0.75 of 390.4 degrees C. This temperature is significantly less than the fuel temperature SL of 500 degrees C. Initiation of this event from lower power levels results in correspondingly lower peak fuel temperatures.

The NRC staff finds the licensee's insertion of excess reactivity analysis to be very conservative because there is no credible means by which the control rods can be manipulated so as to add the assumed reactivity ($0.75) without violating several TS requirements and operating procedures. The NRC staff also finds that the licensee's evaluation of the insertion of excess reactivity scenario to be conservative and the results are within the fuel temperature SL. Based on the information provided above, the NRC staff concludes, that the licensee's evaluation of the insertion of excess reactivity is acceptable.

<u>Insertion of Excess Reactivity - Confirmatory Analysis</u>

The NRC staff performed a confirmatory calculation of the insertion of excess reactivity using a maximum experiment reactivity of $1.50. This is conservative with respect to TS 3.7.1, Specification a, which limits the reactivity worth of all experiments to $1.00. The initial reactor temperature was conservatively assumed to be 200 degrees C. Initial reactor conditions were assumed to be critical at a maximum licensed power of 300 kWt. The FTC used was -0.01432 $/C from Section 2.5.3 of this SER, the value of β_{eff} was .0070, and the prompt neutron lifetime is 60 μs from the DTRR neutronics report (Ref. 12). The peaking factors for the fuel temperature and power were taken from the DTRR neutronics report for the LCC (Ref. 12). The method used was the Fuchs-Nordheim technique documented in GA-7882 (Ref. 21), which is commonly used as the basis for TRIGA analysis of instantaneous reactivity insertion events.

Table 4-5 Step Reactivity Insertion Effect on Fuel Temperature

Parameter	DTRR Results	NRC Staff Confirmatory Calculation Results
Initial reactor power (kWt)	300	300
Initial fuel temperature (C)	200	200
Amount of reactivity inserted ($)	0.75	1.50
Total peaking factor	2.613	2.415
Energy released during pulse (MWt-s)	N/A	5.83
Peak power attained (MWt)	N/A	83.55
Peak final fuel temperature (C)	390	365

The results of the NRC staff's confirmatory analysis indicate that an insertion of $1.50 does not result in the DTRR maximum fuel temperature exceeding the design basis SL of 500 degrees C. Therefore, based on the information provided above, the NRC staff concludes that the results of the DTRR insertion of excess reactivity scenario are acceptable.

4.1.3 Loss-of-Coolant Accident

The DTRR SAR, Section M.1.1, as supplemented by the response to RAI-54 (Ref. 14), provided the results of the licensee's LOCA analysis. The licensee identified three possible scenarios for the loss of coolant: 1) the siphoning of the reactor tank; 2) the evaporation of the reactor tank coolant; and 3) a breach of the reactor tank. The licensee states that provisions have been included in the design that prevent siphoning of water below 15 ft (4.6 m) above the top of the core, should there be any pipe failure through which reactor pool water is pumped from the reactor tank to the water treatment system. This requirement is provided in DTRR TS 5.2, Specification 2. In addition, evaporation of the reactor tank coolant would occur over a long period, allowing sufficient time for cooling replenishment by ROs. Since the reactor tank resides below ground level, with the ground water level at about 8 ft (2.4 m) below the surrounding soil, a tank breach would lead to a slow draining of the tank and would eventually reach equilibrium with about 7 ft (2.1 m) of water in the tank. This scenario would still allow for the replenishment of the reactor tank coolant. See Section 2.3 of this SER for a discussion of the environmental consequences of a tank leak.

In addition, the licensee has analyzed a case of complete and instantaneous LOCA at the DTRR, which is not credible given the reactor tank anti-siphon design. This analysis was performed in support of the previous license renewal, which the NRC staff reviewed and documented in the DTRR SER, NUREG-1312, "Safety Evaluation Report Related to the Renewal of the Facility License for the Research Reactor at the Dow Chemical Company," issued April 1989 (Ref. 35). In this analysis, the licensee assumed the reactor had operated at a power of 300 kWt for a period sufficient to build up the maximum inventory of radioactive fission products, when the tank lost all water instantly. The loss of moderator terminated the neutron chain reaction, but the decay heat continues to heat the fuel. In this analysis, the licensee assumed that the fuel was cooled by natural thermal convection of air up through the core. The licensee calculated that a maximum fuel temperature of 307 degrees C was reached in about an hour. This maximum temperature is well below the SL; therefore, the results of the LOCA analysis did not result in loss of cladding integrity.

With the current LRA, the licensee provided direct and scattered dose rates for two positions inside the reactor building, as described in the DTRR SAR, Section M.1.1, and supplemented by the response to RAI-54. One position was directly over the tank about 18 ft (5.5 m) above the uncovered reactor core, and the other position was at the shielded top edge of the tank, which was shielded from the core by concrete but subject to scatter radiation. Table 4-6 provides the results of the current analysis.

Table 4-6 Radiation Dose from Uncovered Core at DTRR Following a LOCA

Time After Complete Loss of Coolant	Direct Radiation–18 ft Directly Above the Core (R/hr)	Indirect Radiation–at Shield Top Edge of the Tank (R/hr)
10 seconds	3,000	0.78
1 day	360	0.090
1 week	130	0.042
1 month	35	0.012

In the response to RAI-54 (Ref. 14), the licensee used the measured scattered dose rates at various locations inside and outside of the reactor room from the past use of a neutron beam tube in 1991 (no longer in use), in conjunction with the calculated scatter dose rates from the uncovered core to estimate the potential dose to individuals inside the control room and outside the reactor room after a LOCA event. Given the ratio of the measured scattered doses at the edge of the tank and a location immediately outside the east wall of the reactor room, the licensee estimated the dose at this location to be about 554 mrem/hr immediately after the core was exposed to air (10 seconds). In this scenario, the exposed reactor core would result in actuation of the radiation alarm and an immediate evacuation in accordance with the emergency plan.

The licensee stated that It should take less than 10 minutes to evacuate the area (this 10 minutes includes 5 minutes for the operators to identify the radiological condition, sound the alarm, or respond to the alarm, and less than 5 minutes to implement the evacuation of the building and surrounding area). Therefore, the dose to an individual standing immediately outside the east wall of the reactor room for the full 10 minutes would be approximately 92 mrem. The potential occupational dose to an individual who provides mitigation actions in refilling the tank is estimated to be about 565 mrem.

The licensee also performed a more realistic assessment assuming that the reactor pool water was not instantaneously voided, but was inadvertently pumped from the reactor tank at a rate of 10 gallons (37.8 l) per minute (which is the capacity of the reactor room pump). In this scenario, the dose rate outside the reactor room door would be approximately 100 mrem/hr and the

NRC staff calculated that this would result in a dose rate at the fence (public dose) of approximately 1 mrem/hr.

The NRC staff reviewed the direct and scatter radiation doses and found that the doses can be proportioned based on the reactor power. The scattered dose rate was acceptable because of the reactor building and structural interference. Therefore, the NRC staff concludes that using a ratio of known doses to estimate the scattered dose at locations outside of the reactor room was acceptable.

Although the direct and scattered dose rates from the unshielded core are high, evacuation of the reactor room and the building, and exclusion of the public from the vicinity of the facility boundary, would ensure that the 10 CFR Part 20 dose limits to the workers, building occupants, and the public are maintained. Based on the information provided above, the NRC staff concludes that the DTRR LOCA analysis used appropriate assumptions and analytical techniques, and that the results are acceptable.

4.1.4 Loss-of-Coolant Flow

The DTRR SAR describes the loss-of-coolant flow in Section M.1.1. Since the DTRR uses natural convection cooling, the geometry and design of the reactor flow components make it highly unlikely that local fuel element flow blockages would occur within the reactor core.

The DTRR is located in a water-filled tank and is cooled by natural convection flow. Heat generated from the reactor core is directly transferred to the pool water. The reactor assembly is cooled by natural convection using the pool water and the water in the primary cooling circuit. Heat is removed from the primary circuit by natural convection to the air of the reactor room at the surface of the pool, through the tank walls by conduction, and through two heat exchangers. One heat exchanger is the Huron system, which has a capacity of 100 kWt and uses a once-through water system. The other heat exchanger is the SRX-1 system, which rejects heat to a heat exchanger that has a capacity of 1 MWt. These systems may be operated independently or together for a combined capacity of 1.1 MWt. When the Huron system is used, the ultimate heat sink is the service water. When the SRX-1 system is used, the ultimate heat sink is the chiller. According to the licensee’s response to RAI-23 (Ref. 4), the SRX-1 system can be operated manually or set to automatically start when the pool water reaches 27 degrees C, and shuts down at 22 degrees C. The system provides cooling only to the reactor pool and operates a temperature measurement uncertainty of ±2 degrees C.

The DTRR was designed to operate without any additional cooling capacity, such as an external heat exchanger. A loss of cooling would result in a slow increase in the temperature of the pool water, which is monitored by the ROs, and result in the termination of reactor operation before exceeding the limit in TS 3.4, Specification 5, of 60 degrees C.

The NRC staff reviewed the description of the DTRR grid plates provided in the licensee’s response to RAI-11 (Ref. 4), which states that 36 holes are provided on the lower grid plate for natural convection cooling. Cooling water passes through the differential area between the triangular spacer block on the top of each fuel element and the round holes in the grid plate. In addition, the grid plate provides spacing between the fuel elements. The NRC staff’s evaluation of the potential for coolant channel blockage finds that the open fuel element lattice would ensure sufficient continuing cooling of all fuel elements as a result of cross flow. Similarly, the

NRC staff finds that the loss of cooling would result in a slow temperature increase, which would be corrected by the DTRR reactor operators.

Based on the information provided above, the NRC staff concludes that the results of the licensee's postulated loss-of-coolant flow accident scenario would not result in any fuel failure or radiological release and, therefore, are acceptable.

4.1.5 Mishandling or Malfunction of Fuel

In its response to RAI-55 (Ref. 9), the licensee stated that the mishandling and malfunction of fuel could lead to fuel rupture, which would result in releases of fission product into the reactor pool. In the unlikely occurrence of such events, however, the consequences would be bounded by the results of MHA analysis evaluated in Section 4.1.1 of this SER. The effect of such a failure within the normal pool water level would result in significantly lower releases of halogens to the reactor room air, and therefore, much lower doses than those stated in the results of the MHA analysis. Additionally, TS 3.2, provides fuel parameter limits, which helps ensure that damaged or degraded fuel elements are removed from the reactor, thereby, reducing the potential for fuel barrier failure.

The NRC staff finds that the results of the mishandling or malfunction of fuel accident scenario would be bounded by those consequences in the MHA as discussed in Section 4.1.1 of this SER. Therefore, based on the information provided above, the NRC staff concludes that the results of the licensee's mishandling or malfunction of fuel scenario are acceptable.

4.1.6 Experiment Malfunction

In the DTRR SAR, Section M.1.4 (Ref. 4), as supplemented with the licensee's response to RAI-56 (Ref.14), the licensee indicates that the experiment malfunction accident scenario can occur from three principal causes: 1) an unexpected reactivity insertion, 2) a release of material from the specimen container, and 3) a detonation. The licensee controls and limits experiment reactivity in accordance with the requirements of TS 3.7.1 to prevent a step change in reactivity greater than $0.75 for unsecured experiments. The NRC staff finds that the neutronic analysis supplied in response to RAI-53 (Ref. 12) demonstrates the acceptability of this limitation. Accordingly, the NRC staff finds that the licensee's experiment malfunction scenario related to unexpected reactivity is, therefore, acceptable.

The licensee limits the introduction of corrosive materials, as required by TS 3.7.2, which requires double encapsulation. This TS helps ensure that it is highly unlikely that a failure of a double encapsulation device could occur and release of corrosive material into the coolant system. The NRC staff finds that the TS helps ensure that an anticipated failure of a double capsule is highly unlikely and, therefore, is acceptable.

TS 3.7.2 establishes the requirement to limit the irradiation of explosive material in DTRR to 25 milligrams TNT equivalent, and states that quantities less than 25 milligrams of TNT equivalent may be irradiated provided that the pressure produced in the experiment container shall be demonstrated to be less than the design pressure of the container. The NRC staff finds that this requirement is consistent with the guidance in RG 2.2, "Development of Technical Specifications for Experiments in Research Reactors," issued November 1973 (Ref. 26), and NUREG-1537, and, therefore, is acceptable.

With regard to experiment malfunction of experiments containing fissionable materials leading to materials release, based on TS 3.7.2, Specification c, the licensee limits the consequences of releases in terms of iodine isotopes I-131 to I-135 of 10 μCi. The licensee conservatively assumed that this release could occur in air. Assuming that it would take a worker about 60 minutes to resolve this incident, and operation of the reactor room ventilation is terminated by the RO a worker would potentially receive a total dose of approximately 3 mrem, which is well below the limits in 10 CFR 20.1201. The 10 μCi iodine release is about 200 times lower than the amount of I-131 assumed to be released in the reactor room in the MHA. Therefore, the NRC staff finds that the consequences of experiment malfunction leading to release would result in very small radiation doses and are bounded by the MHA for which acceptable analyses were evaluated in Section 4.1.1 of this SER.

TS 3.7.3 addresses experiment failures and malfunctions. Where the possibility exists that the failure of an experiment could, under normal operating condition of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment, release radioactive gases or aerosols to the reactor room or the unrestricted area, the quantity and type of material in the experiment is limited such that the airborne radioactivity in the reactor room or unrestricted area will not result in exceeding the applicable dose limits of 10 CFR Part 20.

Based on the DTRR TS limits for quantity and type of materials allowed in an experiment, the NRC staff concludes that the licensee's evaluation of the consequences of experiment malfunction leading to a radiological release are consistent with the guidance provided in NUREG-1537 and bounded by the MHA analysis evaluated in Section 4.1.1 of this SER. Therefore, based on the information provided above, the NRC staff concludes that the licensee's evaluation of the consequences of the postulated experimental malfunction is acceptable.

4.1.7 Loss of Normal Electrical Power

In the DTRR SAR, Section M.1.5, the licensee evaluated the loss of normal electrical power scenario. The facility does not have emergency power. The loss of normal electrical power will cause the reactor to shut down as the loss of voltage to the control rod drive mechanism will de-energize the magnets and result in a reactor scram. Confirmation of control rod insertion into the reactor core can be visually performed. The licensee indicated that the loss of normal electrical power will not result in any scenario that could cause the release of radioactive material. The loss of normal electrical power does not affect the radiation safety as radiological surveillances can be performed using the portable battery-powered instruments. The loss of electrical power would result in stopping the primary and secondary coolant pumps and the HVAC. However, reactor decay heat would be dissipated through natural circulation in the reactor pool and the loss of HVAC function was acceptably analyzed in the MHA. The HVAC is manually operated.

The NRC staff has reviewed the results of the licensee's postulated loss of electrical power and finds the analysis to be acceptable. Based on this information, the NRC staff concludes the results of the DTRR's analysis of the loss of normal electrical power event are acceptable.

4.1.8 External Events

In the DTRR SAR, Section M.1.6, Sections B.3 through B.5, and Sections C.2 through C.4, the licensee describes the analysis of the potential impact to the DTRR from external events. Floods and extreme winds were not considered to pose a threat to the reactor. In DTRR SAR, Section B.3, the licensee states that the facility history indicates that the probability of occurrence of tornadoes is very low. Furthermore, the NRC staff finds that the reactor area is protected by a reactor building, which is constructed of a steel frame with concrete block. Additional protection from high wind damage is provided by the below-grade location of the reactor.

Seismic activity in the State of Michigan and adjacent areas typically are moderate with minor consequences. In recent history, seismic occurrences have been low-intensity events with little or no damage. The DTRR SAR, Section C.4, states that the building that houses the reactor was designed and built to withstand the seismic activities predicted for the region. The vast majority of seismic activity is low (less than 4.0 magnitude on the Richter scale and located greater than 50 miles from the DTRR site.). In an earthquake with significant severity, the consequences to the DTRR facility are not expected to cause events more severe than the MHA. A severe earthquake accident may result in loss of electric power as discussed in Section 4.1.7, which in turn results in a reactor trip and a loss-of-coolant from the shutdown of the pool cooling system as discussed in Section 4.1.4. The consequence of a LOCA with the core intact is not expected to result in clad failure, as discussed in Section 4.1.3, and should clad damage occur, its consequences would be bounded by the analysis for the MHA. In DTRR SAR, Sections C.2 and C.3, the licensee states that the building is built to withstand the weather extremes and is above the 100-year flood plain.

The NRC staff finds that severe storms, floods, and extreme winds do not pose a threat to the DTRR building or structure. The seismic activity in the area is low, and the building was designed and built for the expected seismicity in the region. If an earthquake with significant severity could cause loss of shielding, cooling, or fuel element cladding failure, the consequences to the DTRR facility are not expected to result in events more severe than the events analyzed in the MHA. The consequence of a LOCA with the reactor core intact is not expected to result in clad failure, and should one occur, its consequences would be bounded by the analysis for the MHA. Based on this information, the NRC staff concludes that the DTRR's analysis of the consequences of external events is bounded by the MHA analysis and is acceptable.

4.1.9 Mishandling or Malfunction of Equipment

In the DTRR SAR, Section M.1.7, as supplemented in the licensee's response to RAI-57 (Ref. 9), the licensee evaluated the mishandling or malfunction of equipment. The licensee stated that mishandling and malfunction of equipment was not expected to result in any damage to the reactor. In an unlikely event, it could result in fuel cladding breach of one or more fuel elements, and result in the possible release of radioactive material. However, such a release would have consequences similar to those analyzed and would be bounded by the radiological results provided in the MHA scenarios. Mitigating this possibility, the licensee states that the DTRR reactor design includes appropriate control system interlocks and automatic protective circuits. TRIGA fuel is designed to accept large-step reactivity insertion events without the loss of clad integrity. Therefore, events caused by operator errors during reactor operation most likely would result in reactor scram and no fuel damage. Failure of the confinement building could have the greatest impact and could lead to an accidental release if it occurred concurrently with a fuel failure accident. However, the loss of confinement (i.e., the ventilation system shutdown) is considered in the MHA analysis for the doses to an unrestricted area, as discussed above in Section 4.1.1 of this SER, the NRC staff concludes that it is acceptable.

Therefore, based on its review as described above, the NRC staff concludes that the results of the licensee's analysis of the mishandling or malfunction of equipment are acceptable.

4.2 Accident Analyses and Determination of Consequences

The NRC staff reviewed the licensee's postulated and analyzed accident scenarios. On the basis of its evaluation of the information presented in the licensee's SAR, as supplemented, the NRC staff concludes the following:

- The licensee considered the expected consequences of a sufficiently broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a loss of integrity of fuel-element clad and a release of fission products.

- The licensee performed analyses of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed doses in 10 CFR Part 20 for unrestricted areas.

- The licensee has employed appropriate methods for accident analysis and consequence analysis.

- Doses from the MHA and all credible accidents are below the limits of 10 CFR Part 20.

- The licensee used conservative assumptions in evaluating occupational and public exposure from releases in an MHA. The MHA will not result in an occupational radiation exposure to the facility staff or radiation exposure to the public in excess of the applicable NRC limits in 10 CFR Part 20.

- For accidents involving insertions of excess reactivity, the licensee has demonstrated that a reactivity insertion of $0.75 will not result in a peak fuel temperature above the SL. An insertion of excess reactivity resulting from the uncontrolled withdrawal of an experiment is limited to $0.75 by TS 3.7.1 and, therefore, does not pose a threat to fuel integrity. The licensee did not identify any other accident scenarios involving a reactivity addition that were not bounded by the DTRR SAR.

- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures.

- The accident analysis for the DTRR establishes the acceptability of the limiting core configuration defined and analyzed in the DTRR SAR.

- The accident analysis confirms the acceptability of the licensed power of 300 kWt, including the response to anticipated transients and accidents.

- The accident analysis confirms the acceptability of the assumptions stated in the individual analyses provided in the DTRR SAR.

4.3 Accident Analyses Conclusions

The NRC staff reviewed the radiation source term and MHA calculations for the DTRR. The NRC staff finds the calculations, including the assumptions, demonstrated that the source term assumed and other boundary conditions used in the analysis are acceptable. The radiological consequences to the public and occupational workers at the DTRR are in conformance with the requirements in 10 CFR Part 20. The licensee reviewed the postulated accident scenarios provided in NUREG-1537 and did not identify any other accidents with consequences not bounded by the MHA. The DTRR design features and administrative restrictions found in the TSs help to prevent the initiation of accidents and mitigate associated consequences. Therefore, on the basis of its review, the NRC staff concludes that there is reasonable assurance that no credible accident would cause significant radiological risk and the continued operation of the DTRR poses no undue risk to the facility staff, the environment, or the public.

5. TECHNICAL SPECIFICATIONS

In this section of the SER, the NRC staff provides its evaluation of the licensee's TSs. The DTRR TSs define specific features, characteristics, and conditions governing the safe operation of the DTRR facility. TSs are explicitly included in the renewal license as Appendix A. The NRC staff reviewed the format and content of the TSs for consistency with the guidance in NUREG-1537, Part 1, Chapter 14, and Appendix 14.1, and ANSI/ANS-15.1-2007 (Ref. 25). The NRC staff specifically evaluated the content of the TSs to determine if the TSs meet the requirements in 10 CFR 50.36. The NRC staff also relied on the references provided in NUREG-1537 and the ISG (Ref. 19) to perform this review.

5.1 Technical Specification Definitions

5.1.1 TS 1.3 Definitions

The licensee proposed to add or modify the TS definitions to be consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 1.3 states the following:

> **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. **Moveable 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

6. **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.

The NRC staff finds that the proposed DTRR TS definitions are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's proposed additions and modifications of these TS definitions are acceptable. Therefore, the NRC staff finds TS 1.3 is acceptable.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 TS 2.1 Safety Limit – Fuel Element Temperature

See the evaluation in Section 2.2.2 of this SER.

5.2.2 TS 2.2 Limiting Safety System Settings

See the evaluation in Section 2.5.5 of this SER.

5.3 <u>Limiting Conditions for Operation</u>

5.3.1 TS 3.1 Reactivity Limits

See the evaluation in Section 2.5.2 of this SER.

5.3.2 TS 3.2 Reactor Fuel Parameters Limits

See the evaluation in Section 2.2.2 of this SER.

5.3.3 TS 3.3 Reactor Control Rods and Safety Systems and Interlocks

See the evaluation in Section 2.5.5.2 of this SER.

5.3.4 TS 3.4 Reactor Coolant Systems

See the evaluation in Section 2.6 of this SER.

5.3.5 TS 3.5 Ventilation

TS 3.5 states the following:

> <u>Specification</u>
>
> 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.

TS 3.5 helps ensure that the ventilation system is maintained operable when the potential exists for the release of airborne radioactivity. The NRC staff finds that TS 3.5, Specifications a through c, help to ensure the operability of the ventilation system. These specifications are consistent with the assumptions used in the DTRR SAR for dose calculations for both occupational and public doses. Additionally, TS 3.5 supports the DTRR ALARA program by reducing the potential exposure to Ar-41. The NRC staff finds that TS 3.5 establishes conditions required to consider the ventilation system operable that are reasonable and consistent with the analysis outlined in the DTRR SAR, as supplemented. Therefore, based on the information provided above, the NRC staff concludes that TS 3.5 is acceptable.

5.3.6 TS 3.6 Radiation Monitoring Systems and Effluents

See the evaluation in Section 3.1 of this SER.

5.3.7 TS 3.7 Experiments

See the evaluation in Section 2.1.3 of this SER.

5.4 <u>Surveillance Requirements</u>

NUREG-1537 and ANSI/ANS-15.1-2007 recommend surveillance requirements that prescribe the frequency and scope of the surveillance activities required to ensure that the limiting conditions for operation are acceptably maintained.

5.4.1 TS 4.0 General

TS 4.0 states the following:

<u>Specifications</u>

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.

TS 4.0, Specification 1, helps ensure that deferred surveillances are accomplished in a planned and organized manner. Surveillances of the CAM, ARM, and pool water conductivity, pH, and level are not deferred during reactor shutdown. The NRC staff finds that this specification helps ensure that these systems and measurements, which are important to maintaining the integrity of the DTRR systems during shutdown conditions, are properly maintained. The NRC staff finds that this specification is consistent with the guidance in NUREG-1537, Appendix 14.1, Section 4.0.

TS 4.0, Specification 2, helps ensure that additions, modifications, or maintenance are completed in accordance with original specifications of the DTRR TRIGA. This TS helps to maintain the design basis of the DTRR from which the components are designed and analyzed.

TS 4.0, Specification 3, helps ensure that surveillances of specified systems are completed after maintenance. The NRC staff finds this specification is consistent with the guidance provided in NUREG-1537, Appendix 14.1, Section 4.0.

TS 4.0, Specifications 1 through 3, help ensure that the quality of systems and components will be maintained to their original design specifications. The NRC staff finds that TS 4.0, Specifications 1 through 3, provide appropriate DTRR surveillance practices, and are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. Furthermore, TS 4.0 helps ensure that the quality of systems and components are maintained, the DTRR facility operation will be conducted within SLs, and the LCOs will be satisfied. Therefore, based on the information provided above, the NRC staff concludes that TS 4.0 is acceptable.

5.4.2 TS 4.1 Reactor Core Parameters

TS 4.1 states the following:

> 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.

TS 4.1 helps ensure that the reactivity components that are important, support the safe operation of the DTRR, are verified at appropriate time intervals such that any changes in core behavior are identified. TS 4.1 requires the surveillance of TS 3.1, Specifications 1 and 2 be performed at specified intervals. The NRC staff finds that TS 4.1 is consistent with the guidance in NUREG-1537, Appendix 14.1, Section 4.1(2), and, that the core reactivity worth parameters necessary for the determination of SDM are determined on an appropriate schedule. Therefore, based on the information provided above, the NRC staff concludes that TS 4.1 is acceptable.

5.4.3 TS 4.2 Reactor Fuel Parameters

TS 4.2 states the following:

> 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.

TS 4.2 helps ensure that the DTRR fuel elements are not operated in a damaged condition, which could result in the potential loss of the cladding integrity and release of radioactive gases. The NRC staff finds that TS 4.2 is typical of TRIGA fueled reactors, and consistent with the guidance in NUREG-1537, Appendix 14.1. TS 4.2 implements the surveillance for TS 3.2. The NRC staff review of DTRR annual reports from 2005 through 2013 has not identified any reported fuel failures, damaged fuel, or fuel out-of-tolerance conditions. The NRC staff finds that TS 4.2 is consistent with the guidance in NUREG-1537, Appendix 14.1, and helps ensure that the quality of the DTRR TRIGA reactor fuel is maintained. Therefore, based on the information provided above, the NRC staff concludes that TS 4.2 is acceptable.

5.4.4 TS 4.3 Control and Safety Systems

TS 4.3 states the following:

<u>Specifications</u>

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.

TS 4.3, Specification 1, helps ensure that the control rod withdrawal times and scram times are maintained in accordance with the limit provided by TS 3.1, Specifications 3 and 4, and TS 3.3 Specification c, requiring the measurement of control rod scram and withdrawal speed. The NRC staff finds that TS 4.3, Specification 1, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the operability of the control rod drives, and preserves the design assumptions used in the DTRR SAR for control rod scram and withdraw times.

TS 4.3, Specification 2, helps ensure the operability of the NM1000 and NPP1000 nuclear instrumentation channels by requiring a calibration. The NRC staff finds that TS 4.3, Specification 2, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the scram and power measuring channels accurately indicate the DTRR reactor power level.

TS 4.3 Specification 3, helps ensure the operability of the scram and interlock channels described in TS 3.3, Specification 1, by requiring a channel test of the six scram channels and three interlocks. The NRC staff finds that TS 4.3, Specification 3, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the operability of the scram and power measuring channels are maintained.

TS 4.3, Specification 4, helps ensure the control rods are operable by requiring a visual inspection biennially. The NRC staff finds that TS 4.3, Specification 4, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps to ensure that degradation or damage to the control rods is identified.

The NRC staff reviewed TS 4.3, Specifications 1 through 4, for control and safety systems. The NRC staff finds that TS 4.3 is consistent with the guidance in NUREG-1537 and ANSI/ANS-

15.1-2007. Therefore, based on the information provided above, the NRC staff concludes that TS 4.3 is acceptable.

5.4.5 TS 4.4 Reactor Coolant Systems

TS 4.4 states the following:

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.

TS 4.4, Specification 1, helps ensure the operability of the pool water conductivity, pH, and radioactivity parameters of the reactor coolant system. This TS helps to maintain a suitable chemical environment for the use of core components. The NRC staff finds that TS 4.4, Specification 1, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the quality of the reactor water is maintained.

TS 4.4, Specification 2, helps ensure adequate pool water inventory by measuring the reactor pool water level. This helps ensure that the assumptions in the safety analysis, Ar-41 analysis, ALARA commitments, and transient analysis are maintained. The NRC staff finds that TS 4.4, Specification 2, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the acceptable inventory of the reactor water is maintained.

TS 4.4, Specification 3, helps ensure that the pool water bulk temperature is maintained. The NRC staff finds that TS 4.4, Specification 3, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the acceptable reactor water temperature is maintained (Ref. No. 47).

TS 4.4, Specification 4, helps ensure that the pool-level alarm is maintained. This TS helps ensure that the RO is notified of a loss of reactor pool water and has an opportunity to correct the water level before decreasing below the 15 ft used as an assumption in the DTRR SAR safety analyses. The NRC staff finds that TS 4.4, Specification 4, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the acceptable performance of the level alarm system is maintained.

TS 4.4, Specification 5, helps ensure that the pool water radioactivity monitor is maintained. The NRC staff finds that TS 4.4, Specification 5, is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the performance of the radioactivity monitor is maintained (Ref. No. 47).

The NRC staff reviewed TS 4.4, Specifications 1 through 5, for the reactor coolant system. Therefore, based on the information provided above, the NRC staff concludes that TS 4.4 is acceptable.

5.4.6 TS 4.5 Ventilation

TS 4.5 states the following:

> <u>Specification</u>
>
> 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.

TS 4.5 helps ensure the operability of the ventilation system by requiring that TS 3.5, Specifications a, b, and c, are maintained. The NRC staff finds that TS 4.5 is consistent with the guidance in NUREG-1537, Appendix 14.1, and that it helps ensure that the performance of the ventilation system is maintained. Therefore based on the information provided above, the NRC staff concludes that TS 4.5 is acceptable.

5.4.7 TS 4.6 Radiation Monitoring Systems

TS 4.6 states the following:

> <u>Specifications</u>
>
> 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.

TS 4.6, Specifications 1, 2, and 3, help ensure the operability of the CAM and ARM channels by requiring channel checks, channel calibrations, and channel tests. The NRC staff finds that TS 4.6, Specifications 1, 2, and 3, are consistent with the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS 15.1-2007, and that TS 4.6 helps ensure that the operability of the radiation monitoring systems is maintained.

TS 4.6, Specification 4, helps ensure the performance of effluent measurements by requiring the replacement of thermoluminescent dosimeters in the reactor room semi-annually. This provides an indication of the radiological impact of facility operation and any anomalous measurements can be identified and the cause investigated. The NRC staff finds that TS 4.6, Specification 4, is consistent with the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS 15.1-2007.

The NRC staff reviewed TS 4.6, Specifications 1 through 4, for the radiation monitoring systems. The NRC staff finds that TS 4.6 is consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Therefore, based on the information above, the NRC staff concludes that TS 4.6, Specifications 1 through 4, are acceptable.

5.4.8 TS 4.7 Experiments

TS 4.7 states the following:

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.

TS 4.7, Specification 1, helps ensure that the reactivity worth of an experiment is estimated or measured before use in DTRR as required to support TS 3.1, Specification 1, and TS 3.7.1, Specifications a and b, and helps maintain adequate SDM.

TS 4.7, Specifications 2 and 3, help ensure that experiments are not inserted into the reactor unless a valid safety analysis has been reviewed and approved to support the specifications of TS 3.7.1, TS 3.7.2, and TS 3.7.3. The NRC staff finds that TS 4.7, Specifications 2 and 3, are consistent with the guidance in ANSI/ANS 15.1-2007.

TS 4.7, Specification 4, helps ensure that the dose rates attributable to a given sample are recorded for further use in dose assessments required to support the radiation protection program described in the DTRR SAR, Section K.1. The NRC staff finds that TS 4.7, Specification 4, is consistent with the provisions of the DTRR radiation protection program.

The NRC staff reviewed TS 4.7, Specifications 1 through 4, controlling experiments. The NRC staff finds that TS 4.7, supports TS 3.1 and TS 3.7, and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on the information provided above, the NRC staff concludes that TS 4.7 is acceptable.

5.5 Design Features

The DTRR design features are described and evaluated by the NRC staff as follows:

5.5.1 TS 5.1 Reactor Site and Building

TS 5.1 states the following:

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.

TS 5.1, Specifications 1 through 5, help ensure important features of the physical design of the facility used to house the DTRR. These specifications support the accident analysis by providing parameters necessary to support the assumptions used to demonstrate compliance with 10 CFR Part 20 requirements. The NRC staff finds that this TS is consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 5.1 is acceptable (Ref. No. 47).

5.5.2 TS 5.2 Reactor Coolant System

See the evaluation in Section 2.6 of this SER.

5.5.3 TS 5.3 Reactor Core and Fuel

See the evaluation in Section 2.2 of this SER.

5.5.4 TS 5.4 Fuel Storage

TS 5.4 states the following:

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.

The NRC staff reviewed TS 5.4, Specifications 1 and 2, noting that the aspects of safety need to be reviewed when addressing fuel storage. For new or unirradiated fuel, the main safety issue is to avoid an unintended nuclear criticality. For used or irradiated fuel, the main safety issue is to maintain adequate cooling and radiation shielding.

The licensee references the analysis performed by Foushee, "Storage of TRIGA Fuel Elements," March 1, 1966 (Ref. 36), which provides a comprehensive review of TRIGA fuel element criticality for the in-tank storage racks mounted on the inside surface of the tank at an elevation similar to the core. The DTRR SAR describes the detailed analysis and concludes that the fuel stored in this manner cannot violate the subcritical value of k_{eff} cited in TS 5.4, Specification 1, under all normal or accident conditions. Since the fuel is located with a fuel element-to-fuel element spacing greater than spacing provided in the DTRR core, and the elevation is the same as the core, cooling of the fuel is assured. For shielding, the NRC staff noted that the DTRR facility implements the requirements in the DTRR radiation protection program, as described in the SAR, Section K.1. In its response to RAI-35 (Ref. 9), the licensee indicated that the storage of fuel was limited to the pool only. Since the licensee has adopted the guidelines described in the Foushee analysis, the NRC staff concludes that the licensee's storage of fuel in storage racks is acceptable.

The NRC staff finds that TS 5.4, Specifications 1 and 2, for fuel storage, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on the information provided above, the NRC staff concludes that TS 5.4 is acceptable.

5.6 Administrative Controls

TS 6.0, Administrative Controls, provides requirements for the conduct of DTRR operations. The administrative controls presented in TS 6.0 include responsibilities, facility organization, staff qualifications, training, the safety committee, operational reviews and audits, procedures, required actions, reports, and records.

The primary guidance for the development of administrative controls for research reactor operation is NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's TSs are based on this guidance. In some cases, the wording proposed was not identical to that provided in NUREG-1537 and ANSI/ANS-15.1-2007. However, this review considered these cases and determined that the licensee's proposed administrative controls met the intent of the guidance and were acceptable.

5.6.1 TS 6.1 Organization

TS 6.1 states the following:

> 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.

TS 6.1 helps ensure that DTRR organizational responsibilities are delineated. The NRC staff finds that the organizational responsibilities described in TS 6.1 is consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537. Therefore, based on this information, the NRC staff concludes that TS 6.1 is acceptable.

5.6.1.1 TS 6.1.1 Structure

TS 6.1.1 states the following:

> 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.

TS 6.1.1 helps ensure that the DTRR organization structure is delineated. The DTRR organizational structure described in TS 6.1.1, and shown in Figure 5-1 (which reproduces DTRR TS Figure 6.1), is consistent with the guidance in ANSI/ANS-15.1-2007 Section 6.1.1. Therefore, based on this information, the NRC staff concludes that TS 6.1.1 is acceptable.

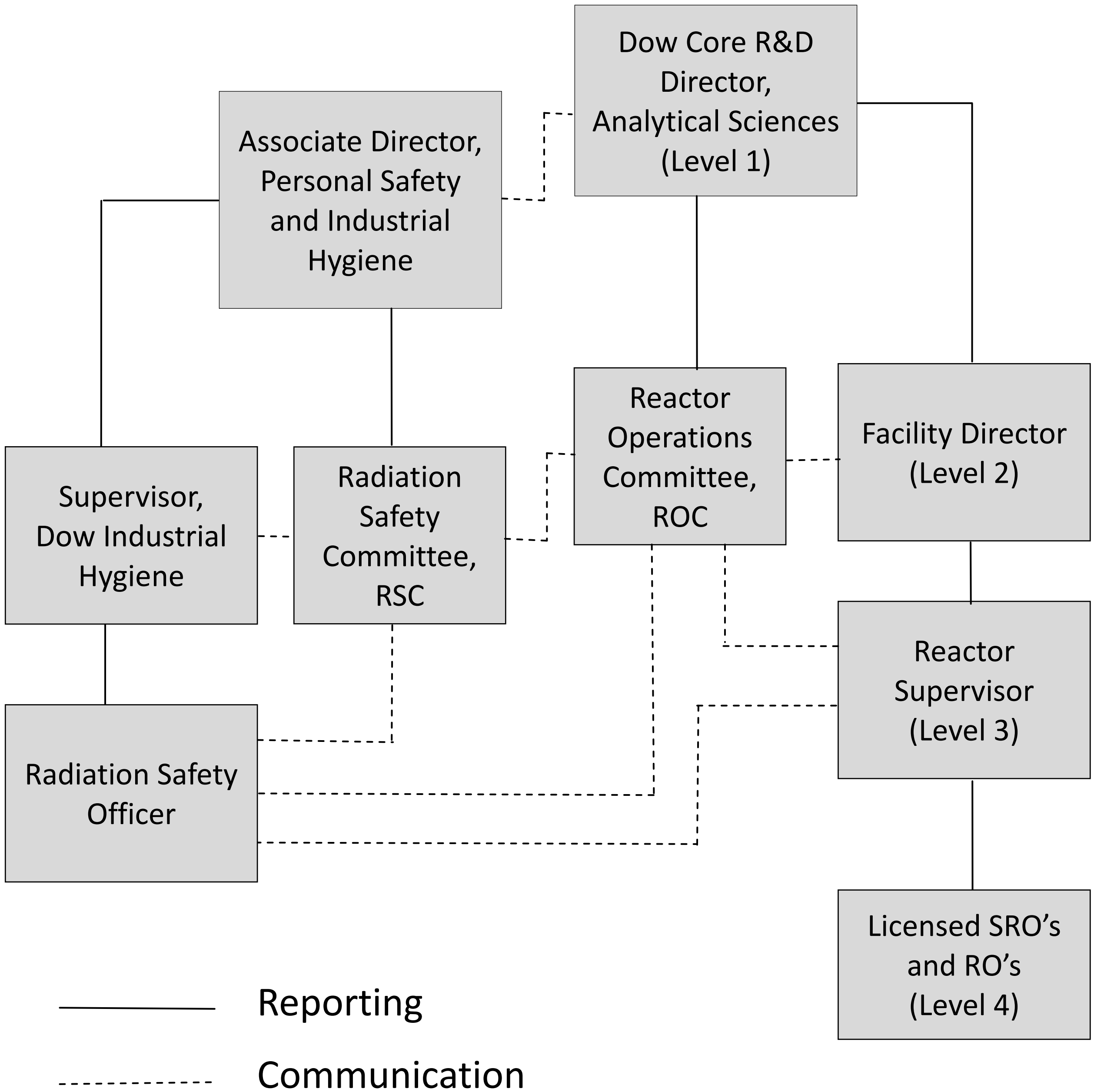

Figure 5-1 DTRR administrative organization for nuclear reactor operations

5.6.1.2 TS 6.1.2 Responsibility

TS 6.1.2 states the following:

> 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.

TS 6.1.2 helps ensure that the DTRR specific organization levels and responsibilities are delineated. The NRC staff finds that the organizational responsibilities stated in TS 6.1.2 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.1.2 is acceptable.

5.6.1.3 TS 6.1.3 Staffing

TS 6.1.3 states the following:

> 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.

TS 6.1.3, Specification 1, describes the minimum staffing necessary to safely operate the DTRR. The regulation in 10 CFR 50.54(k) states, “An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility.”

TS 6.1.3, Specification 2, describes those key personnel whose name and telephone numbers must be readily available in the control room to the operating staff.

TS 6.1.3, Specification 3, requires the reactor supervisor, who must be an SRO, to be present for certain reactor operations. The regulation in 10 CFR 50.54(m)(1) states, “A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or unscheduled reduction in power, and refueling, or as otherwise prescribed in the facility license."

The NRC staff finds that the requirements of TS 6.1.3 are consistent with the requirements of 10 CFR 50.54(k) and 10 CFR 50.54(m) and the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.1.3 is acceptable.

5.6.1.4 TS 6.1.4 Selection and Training of Personnel

TS 6.1.4 states the following:

> 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."

TS 6.1.4 helps ensure acceptable criteria for the training and requalification program for operations personnel. The licensee used ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," 1988; R1999 (Ref. 37), as guidance for selecting and training personnel. The NRC staff finds that the requirements in TS 6.1.4 are consistent with the guidance in NUREG-1537. Therefore, based on this information, the NRC staff concludes that TS 6.1.4 is acceptable.

5.6.2 TS 6.2 Review and Audit

TS 6.2 states the following:

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

TS 6.2 helps ensure that the review and audit function is properly delineated. The function of the ROC, as described in TS 6.2, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, Section 6.2. Therefore, based on this information, the NRC staff concludes that TS 6.2 is acceptable.

5.6.2.1 TS 6.2.1 Composition and Qualification

TS 6.2.1 states the following:

> 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.

TS 6.2.1 helps ensure that the ROC composition, qualifications, and operation, are properly delineated (Refs. 47 and 52). The NRC staff finds that the requirements in TS 6.2.1 are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.2.1 is acceptable.

5.6.2.2 TS 6.2.2 ROC Rules

TS 6.2.2 states the following:

> 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 of the facility.

TS 6.2.2, Specifications a through e, help establish the ROC rules. The NRC staff finds that the composition and qualifications for the ROC, as stated in TS 6.2.2, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.2.2 is acceptable (Ref. No. 47).

5.6.2.3 TS 6.2.3 ROC Review Function

TS 6.2.3 states the following:

> 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.

TS 6.2.3, Specifications a through g, help ensure the ROC review functions are properly delineated. The NRC staff finds that the ROC review functions as specified in TS 6.2.3 are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.2.3 is acceptable.

5.6.2.4 TS 6.2.4 ROC Audit Function

TS 6.2.4 states the following:

> 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.

TS 6.2.4, Specifications a through f, help ensure that the ROC audit functions are properly delineated. The NRC staff finds that the ROC audit functions, as stated in TS 6.2.4, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore based on this information, the NRC staff concludes that TS 6.2.4 is acceptable.

5.6.3 TS 6.3 Radiation Safety

See the evaluation in Section 3.1.2.1 of this report.

5.6.4 TS 6.4 Procedures

TS 6.4 states the following:

> 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.

TS 6.4, Specifications a through i, help ensure that the operational procedures for the DTRR are properly delineated. The NRC staff finds that the specifications provided in TS 6.4 are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.4 is acceptable.

5.6.5 TS 6.5 Experimental Review and Approval

TS 6.5 states the following:

> 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.

TS 6.5, Specifications a and b, help ensure acceptable management control over DTRR experiments. TS 6.5 provides requirements for the review and approval of different types of experiments before being performed at the DTRR. The NRC staff finds that TS 6.5 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.5 is acceptable.

5.6.6 TS 6.6 Required Actions

5.6.6.1 TS 6.6.1 Actions to Be Taken in Case of Safety Limit Violation

TS 6.6.1 states the following:

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.

TS 6.6.1, Specifications a through c, help ensure that the proper actions are taken if a SL violation occurs. TS 6.6.1 requires the facility to shut down in the event that an SL is exceeded. The facility may not resume operation without authorization from the NRC. The violation also must be reported to the ROC and NRC. The reporting requirement is detailed in TS 6.7.2, specifying that the NRC must be notified within 24 hours by telephone and a report is required to be submitted to the NRC within 14 days. TS 6.6.1, Specification c.iii, specifies that corrective actions are to be taken to prevent recurrence. The NRC staff finds that TS 6.6.1 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and in conformance to the requirements provided in 10CFR50.36(d)(1) for actions to be taken when a SL is exceeded. Therefore, based on this information, the NRC staff concludes that TS 6.6.1 is acceptable.

5.6.6.2 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

TS 6.6.2 states the following:

> 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 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.

TS 6.6.2 helps ensure that the proper actions are taken following an event identified in TS 6.7.2 other than a SL violation. TS 6.6.2 requires the DTRR to be shut down in the event of a reportable occurrence. The event and corrective actions taken also must be reported to the facility director, who notifies the ROC Chairman. The reporting requirement is also detailed in TS 6.7.2, specifying that the NRC must be notified no later than the following working day by telephone and a report must be submitted to the NRC within 14 days. The NRC staff finds that the actions the licensee proposes are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.6.2 is acceptable.

5.6.7 TS 6.7 Reports

5.6.7.1 TS 6.7.1 Annual Operating Reports

TS 6.7.1 states the following:

> 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.

TS 6.7.1, Specifications a through h, help ensure that adequate annual reporting information is delineated. TS 6.7.1 provide requirements for the status of the facility, major changes, radiation exposures, and other pertinent information to be provided to the NRC. The NRC staff finds that TS 6.7.1, Specifications a through h, provide DTRR annual operating report requirements that are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.7.1 is acceptable.

5.6.7.2 TS 6.7.2 Special Reports

TS 6.7.2 states the following:

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.

TS 6.7.2, Specifications 1 and 2, help ensure that special reporting requirements are adequately delineated. The NRC staff reviewed TS 6.7.2, Specifications 1 and 2, and finds that the special report requirements are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on this information, the NRC staff concludes that TS 6.7.2 is acceptable.

5.6.8 TS 6.8 Records

TS 6.8.1 states the following:

5.6.8.1 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:

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.

TS 6.8.1, Specifications 1 through 9, help ensure that record retention requirements for records to be retained for at least five years, are properly delineated in the DTRR TSs. The NRC staff finds TS 6.8.1, Specifications 1 through 9, record requirements, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.8.1 is acceptable.

TS 6.8.2 states the following:

5.6.8.2 TS 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.

TS 6.8.2 helps ensure that records required to be retained for one certification cycle are delineated in the TSs. The NRC staff finds the record retention requirements stated in TS 6.8.2 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.8.2 is acceptable.

TS 6.8.3 states the following:

5.6.8.3 TS 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.

TS 6.8.3, Specifications 1 through 5, help ensure that the records retention requirements for records that need to be retained for the lifetime of the DTRR facility are properly delineated in the TSs. The NRC staff finds that TS 6.8.3, Specifications 1 through 5, for the lifetime record retention requirements are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, based on this information, the NRC staff concludes that TS 6.8.3 is acceptable.

5.7 Technical Specifications Conclusions

The NRC staff reviewed and evaluated the DTRR TSs as part of its review of the LRA. The DTRR TSs define certain features, characteristics, and conditions governing the operation of the facility. The NRC staff specifically evaluated the content of the TSs to determine if they meet the requirements of 10 CFR 50.36. Based on its review, the NRC staff concludes that the DTRR TSs are acceptable for the following reasons:

- To satisfy the requirements of 10 CFR 50.36(a), DTRR provided proposed TSs with the LRA. As required by the regulations, the proposed TSs included their appropriate summary bases.

- The DTRR is a facility of the type described in 10 CFR 50.21(c), and, therefore, as required by 10 CFR 50.36(b), the facility operating license will include the TSs. To satisfy the requirements of 10 CFR 50.36(b), DTRR provided TSs derived from analyses in the DTRR SAR.

- The TSs follow the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007, by using definitions that are acceptable.

- To satisfy the requirements of 10 CFR 50.36(c)(1), DTRR provided TSs specifying a SL on the fuel temperature and an LSSS for the reactor protection system to preclude reaching the SL.

- The TS contain LIMITING CONDITIONS FOR OPERATION on each item that meets one or more of the criteria specified in 10 CFR 50.36(c)(2)(ii).

- The TS contain SURVEILLANCE REQUIREMENTS that satisfy the requirements of 10 CFR 50.36(c)(3).

- The TSs contain DESIGN FEATURES that satisfy the requirements of 10 CFR 50.36(c)(4).

- The TSs contain ADMINISTRATIVE CONTROLS that satisfy the requirements of 10 CFR 50.36(c)(5). The DTRR's administrative controls contain requirements for initial notification, written reports, and records that meet the requirements specified in 10 CFR 50.36(c)(1), (2), (7), and (8).

The NRC staff finds that the DTRR TSs are acceptable and concludes that normal operation of the DTRR within the limits of the TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or occupational exposures. The NRC staff also finds that the DTRR TSs provide reasonable assurance that the facility will be operated as analyzed in the DTRR SAR, as supplemented, and that adherence to the TSs will limit the likelihood of malfunctions and the potential accident scenarios evaluated in Chapter 4, "Accident Analysis," of this SER.

6. CONCLUSIONS

Based on its evaluation of the LRA, as discussed in the previous chapters of this SER, the NRC staff concludes the following:

- The application for license renewal, dated April 1, 2009, as supplemented on September 24, 2010; January 12, February 11, April 20, May 12, May 27, August 12, August 31, October 12, November 10, and December 6, 2011; January 13, January 20, February 7, June 11, and August 10, 2012; July 11, and September 16, 2013; and April 9, 2014, complies with the standards and requirements of the AEA and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations*.

- The facility will operate in conformity with the application, as well as the provisions of AEA and the rules and regulations of the Commission.

- There is reasonable assurance that (1) the activities authorized by the renewed facility operating license can be conducted at the designated location without endangering public health and safety, and (2) such activities will be conducted in compliance with the rules and regulations of the Commission.

- The licensee is technically and financially qualified to engage in the activities authorized by the renewed facility operating license, in accordance with the rules and regulations of the Commission.

- The issuance of the renewed facility operating license will not be inimical to the common defense and security or to public health and safety.

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