Safety Evaluation Report
Renewal of the Facility Operating License for the TRIGA Nuclear Reactor at Reed College

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 2012

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 Reed College (the licensee) filed for a 20-year renewal of Facility Operating License No. R-112 to continue operating the Reed Research Reactor. In its safety review, the NRC staff considered information the licensee submitted, past operating history recorded in the licensee's annual reports to the NRC, and inspection reports NRC personnel prepared, as well as firsthand observations. On the basis of its review, the NRC staff concludes that Reed College can continue to operate the facility for the term of the renewed facility 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 Is Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar	argon
ARI-1	all control rods inserted into the core minus the maximum worth rod
ARM	area radiation monitor
CAM	continuous air monitor
CFR	*Code of Federal Regulations*
CHF	critical heat flux
DAC	derived air concentration
DCF	dose conversion factors
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
EHS	environmental health and safety
EP	emergency plan
EPA	U.S. Environmental Protection Agency
FGR	Federal guidance report
FTC	fuel temperature coefficient
GA	General Atomics
H	hydrogen
HEPA	high-efficiency particulate air
HVAC	heating, ventilation, and air conditioning
IR	inspection report
Kr	krypton
ISG	interim staff guidance
LCC	limiting core configuration
LEU	low enrichment uranium
LOCA	loss-of-coolant accident
LRA	license renewal application
LSSS	limiting safety system setting
MCNP	Monte Carlo neutron transport
M	measured

MHA	maximum hypothetical accident
N	nitrogen
NRC	U.S. Nuclear Regulatory Commission
PDR	Public Document Room
PRKE	Oregon State point reactor kinetics model
PTS	pneumatic transfer system
RAI	request for additional information
RAM	radiation area monitor
RG	regulatory guide
ROC	Reactor Operations Committee
RRC	Reactor Review Committee
RRR	Reed Research Reactor
RTR	research and test reactor
SAR	safety analysis report
SDM	shutdown margin
SER	safety evaluation report
SL	safety limit
SNM	special nuclear material
SRM	staff requirements memorandum
TEDE	total effective dose equivalent
TRIGA	Training, Research, Isotopes, General Atomics
TS	technical specification
U	uranium
U-ZrH	uranium-zirconium hydride
Zr	zirconium

TECHNICAL PARAMETERS AND UNITS

$	a unit of reactivity where absolute reactivity is divided by the total effective delayed neutron fraction β_{eff}
$/%	reactivity in $ per % void in coolant or moderator
% burnup	the change in fuel composition due to depletion expressed as a % of the original U-235 content
C	degrees Celsius
µmhos	micromhos
µS	microsiemens
CFM	cubic feet per minute
Ci	Curies
cm	centimeter
cps	counts per second
cm^3	cubic centimeters
dpm	disintegrations per minute
ft	feet
g	gram
hr	hour
K	temperature in Kelvin
keff	k-effective
kg	kilogram
kW(t)	kilowatts thermal
m	meter
min	minute
mPa	mega Pascals (10^6 Pa)
MWd	mega-Watt days (10^6 Watt-days)
MW(t)	megawatt thermal
µCi	microcuries
ml	milliliters
pH	potential of hydrogen
psi	pounds per square inch
rem	Roentgen Equivalent Man
mrem	millirem
s	second

W	watts
w%	weight percent
yr	year
α_F	fuel temperature coefficient
β_{eff}	effective delayed neutron fraction
$\Delta k/k$	absolute reactivity
ρ_{EXP}	reactivity of the experiments
ρ_R	worth of the regulating control rod
ρ_{SDM}	reactivity of the shutdown margin requirement
ρ_{SH}	worth of the shim control rod
ρ_{SA}	worth of the safety control rod
ρ_X	total excess reactivity
X/Q	atmospheric relative concentration, s-cm^3

1. INTRODUCTION

1.1 Overview

In a letter dated August 29, 2007, as supplemented by letters dated January 26, July 30, and October 15, 2010, May 20, August 3, and December 12, 2011; and January 27, and March 26, 2012, Reed College (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for a 20-year renewal of the Class 104c Facility License No. R-112, Docket No. 50-288, for the Reed Research Reactor (RRR) (Ref. 1).

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.51(a) states that "each license will be issued for a period of time to be specified in the license but in no case to exceed 40 years from the date of issuance." Reed College holds Facility License No. R-112 (the license), which was originally issued on July 2, 1968. The term of the license was for a period of 40 years until October 3, 2007. Because of the timely renewal provision contained in 10 CFR Part 2.109(a), the licensee is permitted to continue operating RRR under the terms and conditions of the current license until the NRC staff completes action on the renewal request. A renewal would authorize the licensee to continue operation of the RRR for an additional 20 years.

The RRR was licensed in 1968 as a teaching and research facility at a maximum steady-state power level of 250 kilowatts thermal (kW(t)). The RRR was not licensed to have pulsing capability. The RRR submitted the initial license application on April 15, 1967, and the NRC authorized the construction of the RRR with Construction Permit No. CPRR-101. The license renewal application (LRA) submitted on August 29, 2007, requested an increase in the licensed power level from 250 kW(t) to 500 kW(t). During the NRC staff review, RRR subsequently withdrew the power uprate portion of the LRA in its letter, "Amendment to License Renewal," dated January 26, 2010 (Ref. 2).

The NRC based its review of the request to renew the RRR facility operating license on the information contained in the LRA, as well as supporting supplements and the licensee's responses to requests for additional information (RAIs). The initial 2007 LRA included an RRR safety analysis report (SAR) with technical specifications (TS), financial qualifications and decommissioning information, an environmental report, a radiation protection plan, and an operator requalification plan. The security plan and emergency plan were not revised as a result of the LRA request, and additional discussion on these plans is provided below. The NRC staff conducted site visits on November 10, 2009, and December 7, 2011, to observe facility conditions and to discuss RAIs and RAI responses. The licensee provided its responses to the RAIs in letters dated July 30, 2010 (Ref. 3), May 20, 2011 (Ref. 4), and December 12, 2011 (Ref. 5). The letter dated July 30, 2010, included an update to the final SAR, and the letters dated May 20, 2011, and December 12, 2011, included updated neutronics and thermal-hydraulic reports. As a result of the NRC staff LRA review, the licensee updated its TS and provided them to the NRC in a letter dated January 27, 2012 (Ref. 6). Throughout this report, statements referring to the RRR SAR shall mean the application SAR (Ref. 1), as updated by the licensee's letter dated July 30, 2010 (Ref. 3), and supplemented by RAI responses in letters dated May 20, 2011, and December 12, 2011 (Ref. 4 and Ref. 5), and the updated TS in a letter dated January 27, 2012 (Ref. 6). The licensee requested a change to TS 3.4 in a letter dated March 26, 2012 (Ref. 30). The NRC LRA review also included information from RRR annual reports for 2005 to 2011 and NRC inspection reports (IRs) for 2005 to 2011.

With the exception of the security plan and the emergency plan (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, MD. 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 Resources@nrc.gov. The physical security plan is protected from public disclosure under 10 CFR 73.21, "Requirements for the Protection of Safeguards Information," 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 its safety review, the NRC evaluates the facility against the requirements of the regulations, including 10 CFR Part 20, "Standards for Protection against Radiation," Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," Part 50, "Domestic Licensing of Production and Utilization Facilities," Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," and Part 70, "Domestic Licensing of Special Nuclear Material." The NRC also considers the recommendations of applicable regulatory guides (RGs) and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff also considers the recommendations contained in NUREG–1537, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (Ref. 7). Because no specific accident-related regulations exist for research reactors, dose values for accidents are compared 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. 8), the NRC staff gave the Commission information about plans to revise the review of license renewal applications for research and test reactors. The Commission issued a Staff Requirements Memorandum (SRM) for SECY-08-0161, dated March 26, 2009 (Ref. 9), which directed the NRC staff to streamline the renewal process for such reactors, using some combination of the options presented in SECY-08-0161. The SRM also directed the NRC staff to implement a graded approach with a scope commensurate with the risk each facility poses. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 to SECY-08-0161. In the alternative safety review approach, the NRC staff should consider the results of past NRC staff reviews when determining the scope of the review. A basic requirement, as contained in the SRM, is that licensees must be in compliance with applicable regulatory requirements.

The NRC staff developed the research and test reactor (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 facilities into two tiers. Facilities with a licensed power level of 2 megawatts thermal (MW(t)) and greater, or those requesting a power level increase, undergo a full review using NUREG–1537. Facilities with a licensed power level less than 2 MW(t) undergo a focused review of the most safety-significant aspects of the renewal application that relies on past NRC reviews for certain safety 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 coordinated the RRR LRA review using the guidance in the final ISG, dated October 15, 2009 (Ref. 10), and, because the RRR's licensed power level is less than 2 MW(t), the NRC staff performed a focused review of the licensee's LRA. Specifically, the review focused on reactor design and operation, accident analysis, TS, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to the facility after submitting the application.

With respect to the security plan, the EP, and the reactor operator requalification plan, the ISG states that if the licensee has proposed no changes to these plans or procedures as part of license renewal, then the NRC-approved plan or procedures remain in place and any review of these plans or procedures is outside the scope of a focused renewal review.

The NRC staff approved RRR's physical security plan, "Physical Security Plan for Reed College Reactor Facility," issued in June 1983, submitted by a letter dated November 10, 1983, and supplemented by letters dated November 10, 1983, and February 22, 1984. The licensee maintains a program for providing 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." All changes to the physical security plan have been made in accordance with 10 CFR 50.54(p) and therefore, according to the licensee, these changes will 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 inspections for the past several years identified no violations. Furthermore, in their LRA letter dated August 29, 2007, the licensee indicated that no changes to the RRR security plan were needed as a result of the LRA. For the reasons stated above, the NRC-approved plan remains in place.

As a result of the licensee's initial request for a power uprate, the NRC staff reviewed the RRR EP. An updated version of the RRR EP was provided in a letter dated December 15, 2009 (non-publicly available). The NRC staff issued RAIs to the licensee in a letter dated April 8, 2010 (ADAMS Accession No. ML100970471), and the licensee responded with a revised EP in a letter dated May 25, 2010 (non-publicly available). The NRC staff approved the RRR EP in a letter dated September 9, 2010 (ADAMS Accession No. ML102440002). In addition, the NRC routinely inspects the licensee's compliance with the EP requirements. The NRC staff's review of 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, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, which provides reasonable assurance that the licensee will be prepared to assess and respond to emergency events.

As a result of the licensee's initial request for a power uprate, the NRC staff reviewed the RRR operator requalification plan. The licensee provided an updated version in a letter dated December 15, 2009 (ADAMS Accession No. ML100040244). The NRC staff issued RAIs to the licensee in a letter dated June 30, 2010 (ADAMS Accession No. ML101520062), and the licensee responded with a revised requalification plan in a letter dated July 22, 2010 (ADAMS Accession No. ML102100189). The NRC staff approved the RRR requalification plan in a letter dated August 20, 2010 (ADAMS Accession No. ML102230296).

The purpose of this safety evaluation report (SER) is to summarize the findings of the RRR safety review and to delineate the technical details the NRC staff considered in evaluating the radiological safety aspects of continued operation. This report provides the basis for renewing the RRR license at a steady-state power level of 250 kW(t).

This SER was prepared by Francis DiMegilo, and Geoffrey A. Wertz, Project Managers from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking (DPR), Research and Test Reactors Licensing Branch; and Jo Ann Simpson, Financial Analyst from the NRC's NRR/DPR, Financial Analyst Branch. Energy Research, Inc., the NRC's contractor, gave substantial input to this report.

1.2 Summary and Conclusions on Principal Safety Considerations

The RRR SAR discusses the licensee's understanding of this topic and that information is included in this evaluation. This evaluation includes a review of the information in the RRR SAR, 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 RRR, the NRC staff concludes the following:

- The design and use of the reactor structures, systems, and components important to safety during normal operation, as discussed in Chapter 4 of the RRR SAR and in accordance with the TS, are safe, and safe operation can reasonably be expected to continue.

- The licensee considers 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's analysis of the most serious credible accidents and the MHA demonstrates that the calculated potential radiation doses outside the reactor room would not exceed the dose limits specified in 10 CFR Part 20 for unrestricted areas.

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

- The systems provided for the control of radiological effluents, when operated in accordance with the TS, 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 TS, which specify limits that control operation of the facility, offer a high degree of assurance that the facility will be operated safely and reliably. No significant degradation of the reactor has occurred, as discussed in Chapter 4 of the RRR SAR, and the TS will continue to ensure that no significant degradation of safety-related equipment will occur.

On the basis of these findings, the NRC staff concludes that the licensee can continue to operate the RRR 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 RRR is located in the RRR building on the campus of Reed College, in a suburban area within the city limits of Portland, OR. The reactor is housed in a separate building constructed for that purpose and is adjacent to the psychology building near the southeast corner of the Reed College campus. The campus has approximately 1,300 students and the city of Portland has approximately 560,000 people.

The RRR building is near a body of water referred to as Reed Lake. The building conforms to Seismic Zone 2 requirements under the Uniform Building Code. The original reactor installation in 1968 used fuel and components manufactured by General Atomics (GA), and the structures are built under GA specifications. All building modifications and equipment additions are in conformance with the State of Oregon and City of Portland building codes, in existence at the time, to meet fire, safety, seismic, and flood requirements. The architect for the reactor installation was Farnham & Peck.

The reactor is a heterogeneous pool-type nuclear TRIGA (Training, Research, Isotopes, General Atomics) reactor fueled with TRIGA fuel. The coolant is demineralized light water, which circulates through the core by natural convection. The maximum licensed steady-state power level is 250 kilowatts (kW). The fuel is nominally 8.5 weight percent (w%) uranium (U), enriched to less than 20 w% in U-235.

The reactor tank is an oval-shaped cylindrical structure that measures 10 feet (ft) by 15 ft at the base with a 5-ft radius on each long end. The reactor core is submerged under this column of water. Heat generated from the reactor core is directly transferred to the pool water by natural convection. The reactor assembly is cooled by natural convection using the available 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 via a 250 kW heat exchanger that connects the primary cooling circuit to the secondary cooling circuit. The secondary cooling system uses a cooling tower as the ultimate heat sink.

Entry to the reactor room from inside the building is restricted to a single door exiting the control room. The reactor room also has direct access to the outside loading area through an overhead door. This overhead door and the doors between reactor and control room form the confinement enclosure for the RRR. In this SER and in all conclusions supporting the LRA, the reactor room and the adjacent spaces are treated as a single area called the reactor bay.

The walls in the reactor bay are constructed of 1-hour fire-resistant plaster and metal stud construction with the exception of the west wall, which is an exterior reinforced concrete wall. Large windows offer visibility to the reactor room from the administrative offices, control room, and computational lab. Nonporous enamel paint finishes are used on all walls and ceilings of the reactor area.

The reactor room of the RRR uses a heating, ventilation, and air conditioning (HVAC) system that is independent of all other buildings. A single fresh air inlet that feeds into the loft space supplies fresh air, as described in the RRR SAR, Section 9.1 and the supplemental information in responses to RAI No. 1 and No. 23 (Ref. 4). The HVAC exhausts through an elevated stack that is 12 ft above ground level. This maintains the reactor bay under a slight negative pressure relative to atmospheric conditions. This is referred to as the normal mode of ventilation operation.

On a high radiation alarm, the damper on the system closes and reduces the intake of outside air. Air exhausted from this HVAC system is then passed through high-efficiency particulate air (HEPA) filters. The system continues to exhaust through the stack. This is referred to as the isolation mode of ventilation operation.

1.4 Shared Facilities and Equipment

The RRR is located in a separate building that contains minimal penetrations. Shared facilities include electrical power, heating, cooling, water, and sewerage. The RRR separately controls electrical power, heating, cooling, and water distribution systems from the distribution junctions. During site visits, the NRC staff did not identify any other shared facilities or equipment.

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

The water supplied to the RRR as primary circuit water is purified by equipment operated and maintained by the RRR staff. The RRR cooling tower provides the ultimate heat sink for the heat generated by the reactor. No safety function for this system is specified in the safety analysis.

1.5 Comparison with Similar Facilities

In Section 1.5 of the RRR SAR, the licensee provides general statements about the number of TRIGA-type reactors in service or being built. The RRR is a Mark 1 TRIGA reactor that uses TRIGA reactor fuel. RRR received License Amendment No. 8 on January 4, 2011, which allowed an increase in the SNM possession limit (Ref. 11). This allowed the RRR to receive and use stainless-steel cladded TRIGA fuel from the decommissioned University of Arizona TRIGA reactor. The licensee subsequently replaced all the aluminum-clad fuel with stainless-steel cladded fuel and notified the NRC in a letter, dated August 3, 2011 (ADAMS Accession No. ML11222A026). The fuel is arranged in a circular pattern similar to several other TRIGA reactors. Similarly, the experimental facilities are typical of other TRIGAs.

The unique feature of the RRR is the oval-shaped tank, which affords much greater total volume and provides additional space for the fuel shipping cask and disassembly of equipment.

The TRIGA fuel typically has no performance-related issues as long as the operator maintains established operating limits and water quality.

1.6 Summary of Operations

The RRR is used for a variety of research and educational projects in the Portland area. The reactor can use a rotating specimen rack (lazy susan) pneumatic transfer system and a central thimble for in-core irradiation of specimens. The RRR also has a radiochemistry laboratory for experiments and training in nuclear science education. The RRR is noted for the number of students receiving reactor operator training and NRC-issued operator licenses. This review considers annual reports and IRs from the period of 2005 to 2011. The annual report summaries did not indicate any significant degradation of fuel element integrity, control rod operability issues, excessive inadvertent scrams, or radiological exposure concerns. The fuel temperature circuits are regularly calibrated. The IRs identified no findings of significance.

The RRR received a Severity Level III violation in a letter from the NRC dated December 19, 2008 (ADAMS Accession No. ML083540027), for operation of the RRR in excess of the licensed power limit. The cause was instrument miscalibration that coincided with the installation of a new fuel element. The NRC staff concluded that the licensee's corrective actions were appropriate.

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 have 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 letter dated May 3, 1983, R.L. Morgan, of DOE, informed H. Denton, of the NRC, that universities and other government agencies operating nonpower reactors had entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to take the spent fuel, or high-level waste, or both, for storage or reprocessing. An e-mail sent from James Wade of DOE to Paul Doyle (NRC) (Ref. 12) reconfirms this obligation for the fuel at RRR (DOE Contract No. 78311, valid August 25, 2008–August 31, 2013). By entering into such a contract with DOE, the licensee for RRR 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 RRR since construction. The RRR SAR, Section 1.8, Table 1.3, "Major Facility Modifications," gives a comprehensive list of the major facility changes (summarized in Table 1-1, below). The most recent change, in 2008, pertains to replacing a fume hood and moving the ventilation ducts and vent pipe. The NRC inspection report documents that this change was performed under 10 CFR 50.59, "Changes, Tests and Experiments," and that it was acceptably accomplished. The NRC staff reviewed NRC IRs and annual reports for the period from 2005 to 2011 and concludes that the licensee performed changes in conformance with the requirements of 10 CFR 50.59. The NRC staff concludes that all changes appear to be reasonable and the licensing actions taken over the years seem appropriate.

Table 0-1 Modifications to the RRR Facility

Year	Activity*
1968	completed construction and loaded fuel; initial criticality
1994	replaced heat exchanger with plate-type system and installed new secondary pump and cooling tower to replace lake-based cooling
1995	added supplemental high-voltage power to linear and log-n channels
1998	replaced linear channel display with new Sorrento NMP-1000 meter (m)
2000	replaced percent power and log-n meters with Sorrento NP-1000 and NLW-1000
2001	upgraded facility security system
2003	installed a Honeywell Multitrend for data logging to replace chart recorders
2008	fume hood replacement and ventilation piping modifications

* Note: Reviews under 10 CFR 50.59 were performed for all of these activities.

1.9 Financial Considerations

1.9.1 Financial Ability To Operate a Nonpower Reactor

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 regulations of this chapter, the activities for which the permit or license is sought.

Reed College 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 has determined that Reed College must meet the financial qualification requirements pursuant to 10 CFR 50.33(f) and is subject to a full review of its financial qualifications. Reed College must demonstrate that it possesses, or has reasonable assurance of obtaining, the funds necessary to cover estimated operating costs for the period of the license. Therefore, Reed College must 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 August 29, 2007 (ADAMS Accession No. ML092200010), as supplemented on March 15, 2010 (ADAMS Accession No. ML100820438), Reed College submitted a table of its projected operating costs for the RRR for each of the fiscal years (FY) 2011–2012 through FY 2016–2017. The projected operating costs for the RRR are estimated to range from $191,800 in FY 2011–2012 to $217,200 in FY 2016–2017. According to Reed College, the Board of Trustees of Reed College approved funding for the operating costs. The primary sources of funds are tuition and the college's endowment. The NRC staff reviewed Reed College's estimated operating costs and projected sources of funds provided in the March 15, 2010, submittal and finds them to be reasonable.

The NRC staff finds that Reed College has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs for the RRR for the period of the renewed license. Accordingly, the NRC staff has determined that Reed College has met the financial qualifications pursuant to 10 CFR 50.33(f) and is financially qualified to engage in the proposed activities regarding the RRR.

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. According to 10 CFR 50.33(k), an application for an operating license for a production or utilization facility must contain information that demonstrates how reasonable assurance will be provided that funds will be available to decommission the facility. Under 10 CFR 50.75(d), each nonpower reactor applicant for or holder of an operating license is required to submit a decommissioning report that contains a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding assurance for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are specified in 10 CFR 50.75(e)(1).

In supplements to the application dated March 15, 2010, and October 5, 2010, (ADAMS Accession No. ML102861079), Reed College updated its decommissioning cost estimate for the RRR to approximately $2,250,000 in 2010 dollars. The decommissioning cost estimate for the RRR summarizes costs by labor, burial, support costs for spent fuel shipment, and includes a 25 percent contingency factor, assuming the decommissioning method that will be used is DECON (the RRR will be decontaminated to meet the requirements for unrestricted use, as stated by Reed College). According to Reed College, its labor costs are broken down by staff (e.g., Director, Associate Director, Health Physicist) and labor (e.g., Shift Engineer, Craftsmen, Crew Leader), work years, rate (in dollars per hour) and total costs and burial costs are broken down based on disposal costs at Richland, Washington, and the volume of material to be disposed of. Reed College states that it will update the decommissioning cost estimate periodically over the life of the facility using the Portland-Salem Consumer Price Index. The NRC staff reviewed the material Reed College provided on decommissioning the RRR, as well as the cost estimate, and concludes that the decommissioning cost estimate is reasonable.

Reed College is currently using a self-guarantee to provide financial assurance for decommissioning, as allowed by 10 CFR 50.75(e)(1)(iii) for nonprofit entities such as colleges and universities. The regulation states that "…a guarantee of funds by the applicant or licensee may be used if the guarantee and test are as contained in Appendix E to 10 CFR Part 30." Reed College submitted a self-guarantee agreement and information showing that the guarantor meets or exceeds the financial test criteria for a nonprofit university that issues bonds.

The NRC staff reviewed Reed College's information on decommissioning funding assurance as described above and finds that the decommissioning cost estimate for the planned DECON option is reasonable, the self-guarantee is acceptable, and Reed College's means of adjusting the cost estimate and associated level periodically over the life of the facility is reasonable. The

NRC staff notes that any cost estimate adjustments must incorporate, among other things, changes in costs due to the availability of disposal facilities.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d of the Atomic Energy Act of 1954, as amended (AEA), prohibits the NRC from issuing a license under Section 104 of the AEA 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." The NRC regulation 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, and supplements to the application, Reed College is incorporated as a public benefit corporation in the State of Oregon, principally doing business within the State of Oregon. Reed College provided the names, addresses, and citizenship of its trustees and officers, and all 37 trustees and officers are U.S. citizens. Reed College did state that one trustee has dual citizenship of both the United States and the United Kingdom. According to the application, Reed College is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff does not know or have reason to believe otherwise.

1.9.4 Nuclear Indemnity

The NRC staff notes that Reed College currently has an indemnity agreement with the Commission, which does not have a termination date. Therefore, Reed College will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, "Scope," Reed College, as a nonprofit corporation licensee, is not required to provide nuclear liability insurance. The Commission will indemnify Reed College for any claims arising out of a nuclear incident under the Price-Anderson Act, Section 170 of the AEA, and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, "Appendix E—Form of Indemnity Agreement with Nonprofit Educational Institutions," for up to $500 million and above $250,000. Also, Reed College is not required to purchase property insurance under 10 CFR 50.54(w).

1.9.5 Conclusions

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 RRR and, when necessary, to shut down the facility and carry out decommissioning activities. In addition, the NRC staff concludes 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 RRR is a General Atomics TRIGA Mark I reactor that is licensed for a maximum power level of 250 kW and non-pulsed operation. It is a standard design providing a variety of irradiation facilities, including a central thimble, pneumatic transfer system, single-element replacement, and a gamma irradiation facility.

The reactor core is located near the bottom and at one end of an oval-shaped, water-filled aluminum tank that is 10 ft wide and 15 ft long with a 5-ft radius at each end. The tank is 25 ft deep and is bolted at the bottom to a 24-inch-thick poured concrete slab. The tank minimum wall thickness is 0.25 inches and is surrounded by approximately 2.5 ft of concrete. The tank and the water provide shielding for personnel. The approximately 22-ft column of water above the core also provides coolant. The control rod drives are mounted above the tank on a bridge structure spanning the diameter of the tank.

The RRR uses solid U-zirconium (Zr) hydride (U-ZrH) fuel containing 8.5 w% U enriched to less than 20 w% U-235 stainless-steel cladded TRIGA fuel 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 RRR has no pulsing equipment or supporting analysis.

The inherent safety of TRIGA reactors has been demonstrated by the extensive experience gained from similar 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. 13)); 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. 14)), and GA-4314, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," issued in 1980 (Ref. 15)); and accident analysis (NUREG/CR–2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," issued April 1982 (Ref. 16))

2.1.2 Summary of Reactor Data

The RRR provided updated neutronics and thermal-hydraulics analyses (Ref. 4 and Ref. 5) following the replacement of the aluminum-cladded fuel with stainless-steel cladded fuel. The updated RRR neutronics analysis conformed to the guidance in NUREG–1537, Section 4.5.1, which requested the licensee identify the limiting core configuration (LCC). The LCC is defined in NUREG-1537 as the core that provides the highest power density. All other core configurations are therefore encompassed within the safety analysis of the LCC. A description of the RRR LCC can be found in the response to RAI No. 14 (Ref. 5), and some of the attributes which differentiate the LCC from the current operational core are listed below:

- The LCC uses a mixture of aluminum and stainless-steel cladded fuel elements in an arrangement operated under the previous license that has 15 fewer fuel elements.

- The reactor power is kept at 250 kW.

- The coolant inlet temperature is assumed to be 60 degrees Celsius (C).

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

Table 2-1 presents the basic design parameters and results that typify the RRR LCC as provided in the RRR SAR (Ref. 4 and Ref. 5). Because the LCC core is the previously loaded RRR configuration, certain of the supplied calculated parameters have measured counterparts.

Table 2-1 Reactor Parameters for the RRR LCC Core

Parameter	**Result**	
Licensed reactor power	250 kW	
Number of fuel elements in core	64	
Number of control rods in core	3	
Maximum fuel temperature at 250 kW	264 ºC	
Fuel temperature coefficient	-0.0103 $/K	
Maximum rod power at 250 kW	7.24 kW	
Average rod power at 250 kW	3.91 kW	
Departure from nucleate boiling ratio (DNBR)	6.33	
Effective delayed neutron fraction	0.0075 (used) .00778 (calc.)	
	Calculated ($)	**Measured ($) (8/13/10)**
Safety control rod worth	-3.52	-3.10
Shim control rod worth	-3.53	-3.05
Regulating control rod worth	-1.06	-1.31
Excess reactivity	+1.65	+1.54
Shutdown reactivity	-2.93	-2.82
Linear power trip setpoint	250 kW	

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

2.1.3 Experimental Facilities

The RRR experimental facilities are described in the RRR SAR. The RRR has been designed with multiple in-core irradiation facilities to facilitate a broad range of potential experimental activities. These facilities include a rotary specimen rack, central thimble, the pneumatic transfer tube, and individual fuel element locations.

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

A pneumatic transfer system (PTS) is available for use at the RRR facility. 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 pool surface. The RRR PTS is designed to quickly transfer individual specimens into and out of the reactor core. The specimens are placed in a small polyethylene holder (the rabbit), which, in turn, is placed into the receiver. The rabbit is an enclosed polyethylene holder. 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 receiver and terminus. A compressed gas system supplies helium and a solenoid valve directs flow. Controls to operate the compressed gas and solenoid valve are on the console.

The RRR 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 (min).

2.1.3.1 TS 3.6.1 Reactivity Limits

TS 3.6.1 states the following:

> Specifications. The reactor shall not be operated unless the following conditions governing experiments exist:
>
> a. The absolute value of the reactivity worth of any single unsecured experiment shall be less than \$1.00; and
>
> b. The sum of the absolute values of the reactivity worths of all experiments shall be less than \$2.00.

TS 3.6.1, Specification a, establishes a limit on the reactivity worth of unsecured experiments. If this amount of reactivity is inadvertently removed, it will not have an unacceptable effect on the reactor system as demonstrated in the analysis of excess reactivity insertion in the RRR SAR response to RAI No. 42 (Ref. 4).

TS 3.6.1, Specification b, establishes a limit on the total worth of all experiments. Similar to Specification a, if this amount of reactivity is inadvertently removed, it will not have an unacceptable effect on the reactor system as demonstrated in the analysis of excess reactivity insertion in the RRR SAR response to RAI No. 42 (Ref. 4). See Section 4.1.2 of this SER for the results of the excessive reactivity insertion scenario.

The NRC staff reviewed the reactivity limits established in TS 3.6.1, Specifications a and b above, and determined that the specifications include the determination of SDM and excess reactivity, as provided in TS 3.1.2 and TS 3.1.3. TS 3.6.1 helps to ensure that reactivity insertion events are properly controlled by RRR staff during experiments. The NRC staff finds that TS 3.6.1, Specifications a and b, are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors," (Ref. 17). On the basis of the information provided above, the NRC staff concludes that TS 3.6.1 is therefore, acceptable.

2.1.3.2 TS 3.6.2 Materials

TS 3.6.2 states the following:

> Specifications. The reactor shall not be operated unless the following conditions governing experiments exist:
>
> a. 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 less than 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; and
>
> b. Experiments containing corrosive materials shall be doubly encapsulated. If the encapsulation of material that could damage the reactor fails, it shall be removed from the reactor and a physical inspection of potentially damaged components shall be performed.

TS 3.6.2, Specification a, limits the quantity of explosive material to 25 milligrams or less. 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. 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 RG 2.2, "Development of Technical Specifications for Experiments in Research Reactors," issued November 1973 (Ref. 18). TS 3.6.2, Specification a, also implements the recommendation in NUREG–1537, Appendix 14.1, Section 3.8.2, regarding experiments that have explosive content by limiting the amount to less than 25 grams TNT equivalent.

TS 3.6.2, Specification b, follows the guidance provided in NUREG–1537, Appendix 14.1, Section 3.8.2, and requires the double encapsulation of corrosive materials as a means to reduce the likelihood that the encapsulation will fail and the corrosive material could damage the fuel cladding.

The NRC staff finds that TS 3.6.2 is consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 3.6.2 is therefore, acceptable.

2.1.3.3 TS 3.6.3 Experiment Failures and Malfunctions

TS 3.6.3 states the following:

> Specifications.
>
> Where the possibility exists that the failure of an experiment under normal operating conditions of the experiment and reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor bay or the unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor bay or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR 20, assuming that:
>
> a. 100% of the gases or aerosols escape from the experiment;
>
> b. If the effluent from an irradiation facility exhausts through a holdup tank, which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape;
>
> c. If the effluent from an irradiation facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these aerosols can escape; and
>
> d. For materials whose boiling point is above 54.4 ºC (130 ºF) and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, 10% of these vapors can escape.

TS 3.6.3 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 under 10 CFR Part 20. The NRC staff finds that the specific assumptions cited in Specifications a through d adequately implement the guidance provided in NUREG–1537, Appendix 14.1, Section 3.8.3.

The NRC staff finds that the RRR experimental facilities are typical of TRIGA reactors, and their use is properly controlled by the TSs 3.6.1, 3.6.2, and 3.6.2. Furthermore, on the basis of the information provided above, the NRC staff concludes that the RRR experimental facilities and TSs 3.6.1, 3.6.2, and 3.6.3, are acceptable.

2.2 Reactor Core

The RRR core is described in the RRR SAR. The RRR 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 fastened to the reflector assembly. The outer region of the core may contain some graphite reflector elements. The reflector surrounds the core and is composed of graphite with a radial thickness of about 12 inches 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 with a locking device. The core is cooled by natural convection of the water.

The core components are contained between top and bottom aluminum grid plates. The top grid plate has 91 positions for core elements in 5 concentric rings around a central thimble. The fuel elements consist of a ZrH moderator homogeneously combined with enriched uranium. The fuel elements are stainless-steel cladded U-$ZrH_{1.6}$ fuel elements. The hydrogen (H) to Zr stoichiometry ratio is represented by the “x” in the U-ZrH_x nomenclature. Figure 2-1 shows the physical arrangement of the LCC, as described in the RRR SAR.

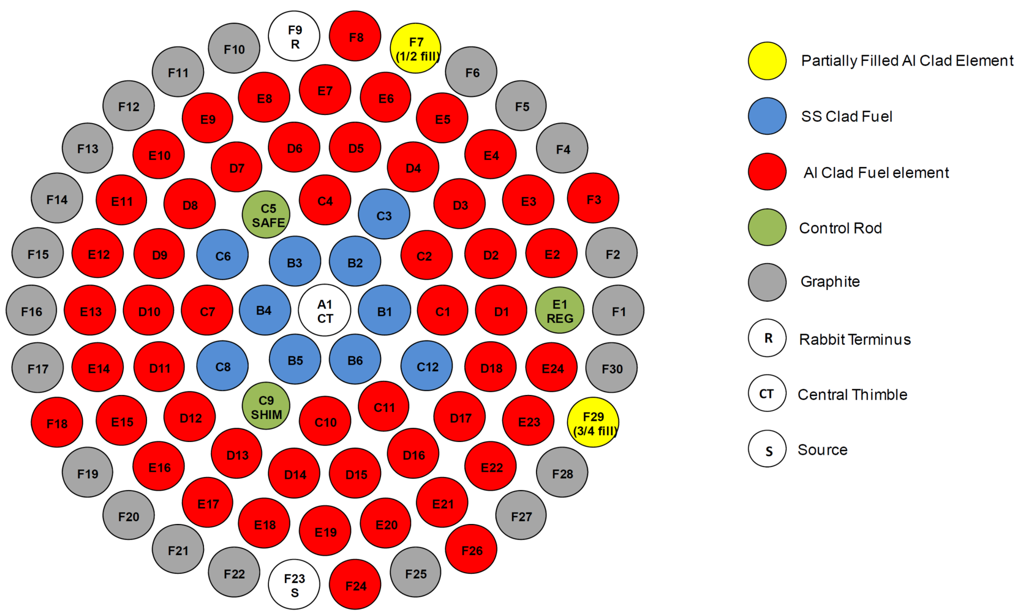

Figure 2-1 RRR core lattice positions and fuel as loaded in the LCC

The stainless-steel fuel core elements have a 1.475-inch nominal outer diameter, except for the upper and lower end fixtures, and are approximately 28.44 inches long. The fuel meat length of the stainless-steel fuel elements is 15 inches; the fuel meat of the aluminum fuel elements is 14 inches in length. Solid graphite reflector slugs are at either end of the fuel meat inside the cladding.

The reactivity and the power level of the RRR reactor is controlled with three control rods. Two instrumentation channels monitor and indicate the reactor neutron flux and power level on the console.

2.2.1 TS 5.3.1 Reactor Core

TS 5.3.1 states the following:

<u>Specifications</u>.

a. The core assembly shall consist of stainless steel clad 8.5/20 TRIGA® fuel elements.

b. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, irradiation facilities, graphite dummies, control rods, startup sources, or central thimble.

c. The reflector, excluding experiments and irradiation facilities, shall be water and graphite.

d. Fuel shall not be removed from or inserted into the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel element.

e. Control rods shall not be removed manually from the core unless the core has been shown to be subcritical with all control rods fully withdrawn from the core.

TS 5.3.1, Specification a, helps ensure that only TRIGA fuel elements are authorized to be used in the RRR. This design feature information is important to ensure that the LCC for the RRR consists of core elements approved for use.

TS 5.3.1, Specification b, helps ensure that the physical arrangement is close-packed and has no open internal positions except as identified (e.g., the central thimble).

TS 5.3.1, Specification c, helps ensure that reflectors used are identified and approved for use by the RRR SAR so that predictable reflection of the core is provided.

TS 5.3.1, Specification d, is precautionary and requires the core to be more subcritical than the most reactive fuel element before performing any fuel element insertion or removal. As stated in the basis, this precaution helps prevent inadvertent criticality.

TS 5.3.1, Specification e, is precautionary and requires the core to be subcritical without crediting the position of the control rods before performing any control rod insertion or removal. As stated in the basis, this precaution helps prevent inadvertent criticality.

The NRC staff finds that TS 5.3.1, Specifications a through e, characterize the RRR design features for the reactor core and help ensure that the core loading conforms and is limited to the analysis in the RRR SAR. TS 5.3.1 helps ensure that excessive power densities will not result from any allowed core loading. The NRC staff finds that TS 5.3.1 is consistent with the guidance provided in NUREG–1537, Section 4.5.1, which recommends the applicant to identify the highest power density of any possible core arrangement. The core configuration used in the accident analysis is presented in Chapter 4 of this report. On the basis of the information provided above, the NRC staff concludes that TS 5.3.1 is acceptable.

The RRR LCC reactor core elements from the RRR SAR (Ref. 1, Ref. 4, and Ref. 5), are listed in Table 2-2.

Table 2-2 The RRR LCC Reactor Core Elements

Core item	Number in the LCC	Number in the operational core	Location
Stainless steel fuel elements	10	79	various
Aluminum fuel element	54	0	various
Shim control rod	1	1	C9
Safety control rod	1	1	C5
Regulating rod	1	1	E1
Graphite reflector elements	21	6	various
Source	1	1	F23
Central thimble	1	1	A1
Pneumatic transfer system	1	1	F9
Total	91	91	
Operational parameters			
Hot fuel element position	B5	B5	
Average predicted fuel element (kW)	3.91	3.16	
Maximum predicted fuel element (kW)	7.20	5.07	

The licensee designated the LCC (shown in Figure 2-2) which was then used by the licensee to develop a Monte Carlo Neutron Transport (MCNP) model (shown in Figure 2-3). This model included all experimental positions and a description of the major reflector elements as provided in the SAR. The NRC staff reviewed this model and concluded that it was an accurate representation of the RRR LCC.

Figure 2-2 illustrates the power distribution for each fuel element location in terms of kW/fuel element. The peak power fuel element is 7.24 kW in location B5.

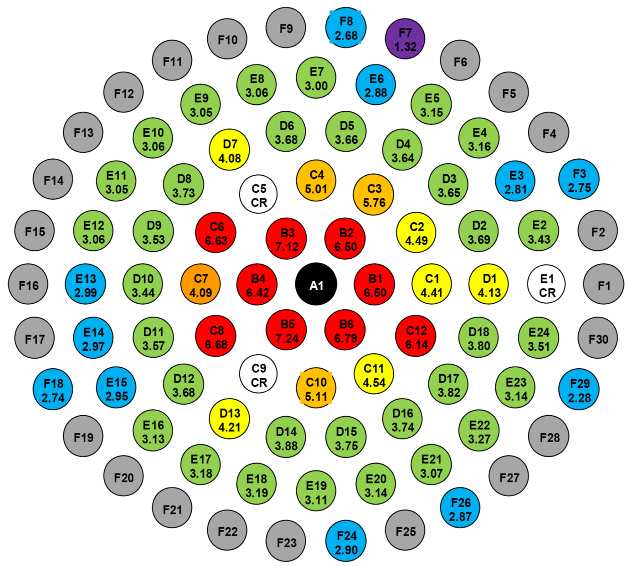

Figure 2-2 RRR LCC power distribution

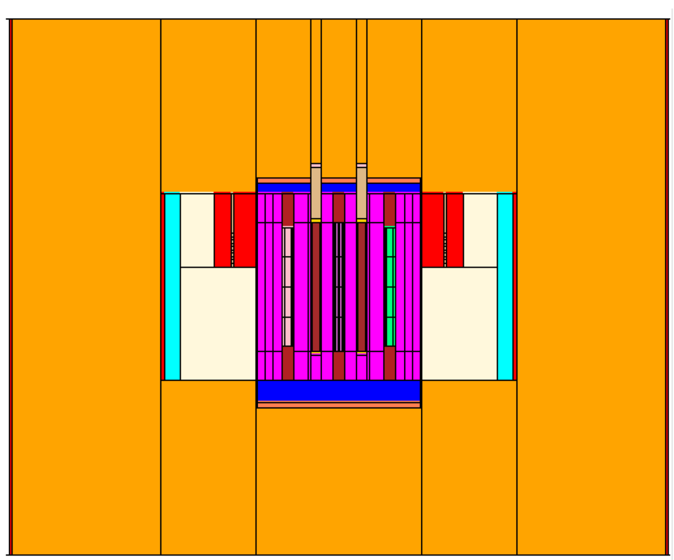

Figure 2-3 RRR cross-section of the MCNP model

Based upon a review of the information provided by the licensee in the RRR SAR and described above, the NRC staff finds that the licensee has accurately described the LCC used in the RRR SAR, including design limits and the bases for these limits. The licensee has also adequately provided and discussed the constituents, materials, and components of the LCC. Compliance with the applicable TS will help ensure uniform reactor core operational characteristics and compliance with the design bases and safety-related requirements. The NRC staff also finds that the licensee's analysis of the LCC provides sufficient margin with respect to fuel element power density for RRR operational core configurations to function safely for the renewal period. On the basis of the information provided above, the NRC staff concludes that the licensee's analysis of the LCC, and TS 5.3.1, are acceptable.

2.2.2 Reactor Fuel

The RRR fuel is described in the RRR SAR. The RRR uses cylindrical stainless-steel cladded fuel elements in which the fuel is a solid homogeneous mixture of U-ZrH alloy containing nominally 8.5 w% uranium enriched to less than 20 w% in U-235. The RRR SAR describes the design details of the fuel elements. The stainless-steel fuel has a 0.25-inch hole in the center that is filled with a Zr rod. This rod is used during fabrication to control the ZrH stoichiometry ratio of the fuel to be approximately 1.6.

NUREG–1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," provides regulatory approval for the fuel types listed in Table 2-3. RRR stated in the response to RAI-7 (Ref. 5) that only original fuel is used.

Table 2-3 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 that the total RRR fuel depletion through 2011 is approximately 61.1 megawatt-days (MWd) which is less than the guidance provided in NUREG-1282. In addition, RRR fuel elements do not contain burnable absorbers (poisons) such as erbium which provides makes the calculation of excess reactivity and shutdown margin (SDM) more linear with burnup. The burnup of the RRR LCC is very small (~61.1 MWd) and this small change is reflected in the RRR models by reducing the U-235 content.

Figure 2-4 illustrates the U-ZrH fuel matrix phase diagram for a range of fuel stoichiometries provided in GA-4314 (Ref. 15). A vertical arrow has been added to indicate the stoichiometry of the stainless-steel fuel used in the RRR. The horizontal arrow corresponds to the maximum temperature that fuel of this stoichiometry may attain without changing the fuel matrix phase, which is the face-centered cubic (δ) phase. The RRR analysis of the LCC used the two types of fuel elements in use at the onset of the license renewal: (1) stainless-steel cladded, high-hydride U-$ZrH_{1.6}$ fuel elements and (2) aluminum-clad, low-hydride U-$ZrH_{1.0}$ fuel elements. (All aluminum-clad, low-hydride $ZrH_{1.0}$ fuel elements were replaced with stainless- steel fuel elements and the aluminum-clad fuel elements removed from the RRR facility in 2012.) The H-to-Zr stoichiometry ratio is represented by the "x" in the U-ZrH_x nomenclature. The H content is important because it influences many attributes of fuel behavior. The vertical arrow indicates the range of temperatures that the stainless-steel cladded fuel (U-$ZrH_{1.6}$) may be subjected to without incurring a change in phase of the fuel matrix. Operation within this range helps ensure that unacceptable changes that could lead to fuel clad breach are avoided.

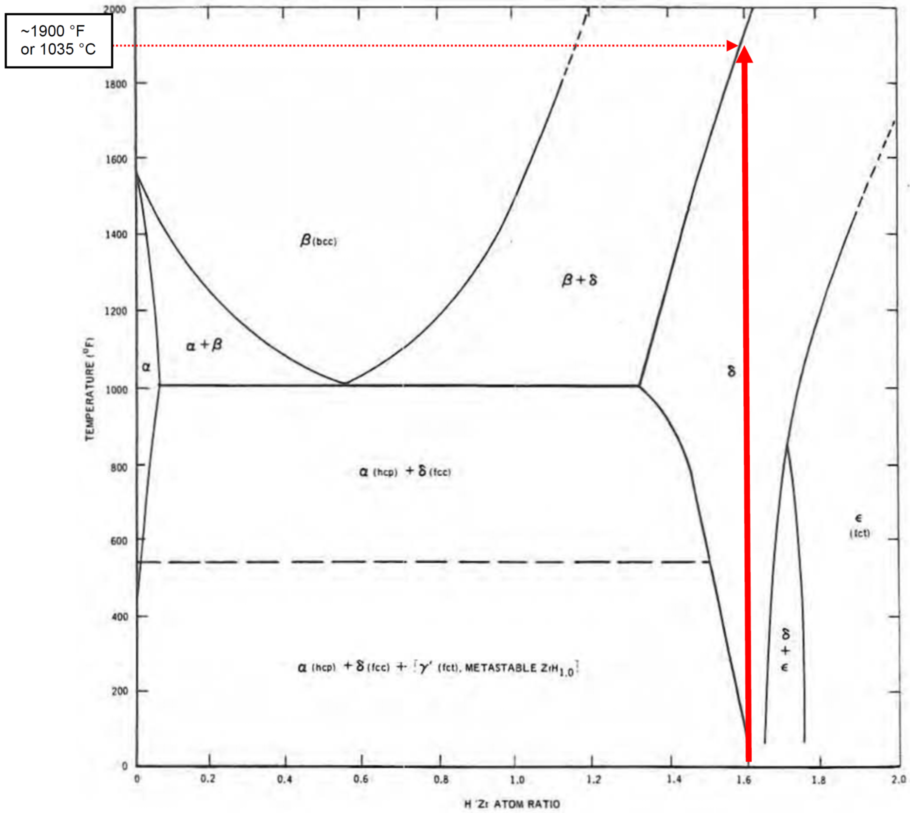

Figure 2-4 Phase diagram for zirconium-hydride fuel

2.2.2.1 TS 2.1 Safety Limit: Fuel Temperature

TS 2.1 states the following:

> <u>Specification</u>.
>
> The maximum fuel temperature shall not exceed 1000°C.

TS 2.1 provides the SL for the RRR fuel elements. The RRR SAR references NUREG–1282, which identifies the safety limit (SL) for TRIGA fuel elements with stainless-steel cladding based on the stress to the cladding that results from H pressure from the dissociation of the ZrH. This stress will remain below the yield strength of the stainless-steel cladding if the fuel temperature is below 1,150 degrees C. During operation, fission product gases and dissociation of the H and Zr build up a gas inventory in internal components and spaces of the fuel elements. Limiting the maximum fuel temperature prevents an excessive internal pressure that heating the gases could generate. Fuel growth and deformation can occur during normal operation, as described in GA-4314. The RRR TS 2.1 limit of 1,000 degrees C provides a margin of safety to the established limit of 1,150 degrees C.

The NRC staff finds that TS 2.1 is consistent with the guidance provided in NUREG–1537, Appendix 14.1, Section 2.1. On the basis of the information provided above, the NRC staff concludes that TS 2.1 is, therefore, acceptable.

2.2.2.2 TS 5.3.3 Reactor Fuel

TS 5.3.3 states the following:

> Specifications.
>
> The individual unirradiated TRIGA® fuel elements shall have the following characteristics:
>
> a. Uranium content: nominal 8.5 weight percent enriched to less than 20% in U-235;
>
> b. Hydrogen-to-zirconium atom ratio (in the ZrHx): between 1.5 and 1.65;
>
> c. Cladding: stainless steel, nominally 0.020 inches thick; and
>
> d. Identification: each element shall have a unique identification number.

TS 5.3.3, Specification a, provides the nominal w% and maximum enrichment of the TRIGA fuel and helps ensure that the fuel requirement is consistent with the analysis supplied in the RRR SAR.

TS 5.3.3, Specification b, provides the fuel stoichiometry to help ensure that it is consistent with the fuel used to develop the SL established in TS 2.1 and the LSSS in TS 2.2. The RRR SAR provides analyses based on the nominal stoichiometry of 1.6 for the RRR fuel, which support the thermal margins developed for the bases to TSs 2.1 and 2.2. The NRC staff finds the allowable fuel stoichiometry in TS 5.3.3, Specification b, is therefore acceptable.

TS 5.3.3, Specification c, represents the fuel described in the RRR SAR and helps ensure that the fuel cladding material and thickness is consistent with the analyses described in the RRR SAR and used to establish the basis for the SL in TS 2.1.

TS 5.3.3, Specification d, helps ensure that the fuel elements in use are properly labeled and located for purposes of fuel accountability.

TS 5.3.3, Specifications a through d, help ensure that important design features of the RRR reactor fuel are maintained as described in the RRR SAR. TS 5.3.3, Specifications a through d, support the bases for the SL in TS 2.1 and TS 2.2, are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 5.3.3, is therefore, acceptable.

2.2.2.3 TS 3.1.4 Fuel Parameters

TS 3.1.4 states the following:

> Specifications.
>
> The reactor shall not be operated 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. A cladding defect exists as indicated by release of fission products;
>
> b. Visual inspection identifies bulges, gross pitting, or corrosion;
>
> c. The sagitta (traverse bend) exceeds 0.0625 inches over the length of the cladding;
>
> d. The length exceeds its original length by 0.125 inches; or
>
> e. The burn-up of U-235 in the fuel matrix exceeds 50% of the initial concentration.

TS 3.1.4, Specifications a through e, establish inspection requirements to detect gross failure or visual deterioration of the fuel. The fuel element attributes inspected include the fuel element transverse bend and length and a visual inspection for bulges or other cladding defects. The NRC staff finds that the TS 3.1.4 limits on transverse bend and length, and fuel burnup are consistent with the values provided in NUREG–1537. On the basis of the information provided above, the NRC staff concludes that TS 3.1.4, Specifications a through e, are acceptable.

The NRC staff reviewed the RRR SAR, which described the fuel elements used in the RRR, 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. On the basis of the information provided above, the NRC staff concludes that the RRR fuel elements and their associated TS are acceptable.

2.2.3 Control Rods

The RRR control rods are described in the RRR SAR. RRR uses boron carbide (B_4C) control rods that are characteristic of most TRIGA reactors. The boron carbide neutron absorbers are enclosed in aluminum tubes of nominally 1.25 inches in diameter. One control rod is designated as a regulating rod and is used for minor changes to reactor power during RRR operation. The regulating control rod can be placed into automatic rod control, which disables the manual rod control and engages a rod control servo that moves the rod to keep the power level on the current decade of the multirange linear power channel within 2 percent of the percent demand set by the operator. The other control rods are designated as the shim and safety control rods. The control rods pass through normal fuel positions in the RRR core on the top and bottom of the grid plates. Guide tubes ensure that the control rods remain properly aligned.

Each control rod is coupled to a drive that consists of a stepping motor, a magnet rod-coupler, a rack-and-pinion gear system, and a potentiometer; the latter is used to provide an indication of rod position. The pinion gear engages a rack mounted on a drawtube extending approximately 12 inches below the center channel. At the bottom of the draw tube is an electromagnet which, when actuated, connects the draw tube to the control rod armature and allows rod withdrawal and insertion. The draw tube and top of the armature are housed in a tubular barrel that extends below the water surface. Just below the connection to the magnet on the control rod armature is a piston that travels within the barrel assembly. Vents in the top portion of the barrel enable the water to escape, allowing the piston to move freely, but the bottom 2 inches restrain the motion by dashpot action, providing cushioning for the control-rod mechanism in the event of a scram.

Electromagnets hold each control rod in place. When a scram is initiated, the electrical 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 or reaction of the safety circuits. All of the RRR control rods have scram capabilities in accordance with RRR TS 5.3.2. 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 vertical position of each control rod is displayed on the operator console.

The withdrawal speed of the rods is adjustable, and the RRR control rod drives are limited by TS for a maximum insertion or withdrawal rate of $0.16/s.

2.2.3.1 TS 5.3.2 Control Rods

TS 5.3.2 states the following:

> Specification.
>
> The control rods shall have scram capabilities and the poison section shall contain borated graphite, B_4C powder, or boron and its compounds in solid form as poison in an aluminum or stainless steel cladding.

TS 5.3.2 requires that the control rods employ a B_4C absorber. This material has well-established nuclear and material characteristics. It also requires that all of the RRR control rods be scrammable. The NRC staff finds that TS 5.3.2 characterizes important design features of the RRR, is consistent with the guidance provided in NUREG–1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that the RRR TS 5.3.2 is, therefore, acceptable.

2.2.3.2 TS 3.2.1 Control Rods

TS 3.2.1 states the following:

> Specifications.
>
> The reactor shall not be operated if any control rod is not operable. Control rods shall not be considered operable if:
>
> a. Damage is apparent to the rod or rod drive assembly;

b. The scram time exceeds 1 second; or

c. The reactivity addition rate exceeds \$0.16 per second.

TS 3.2.1, Specification a, helps ensure that control rods are free of any apparent damage.

TS 3.2.1, Specification b, helps ensure that the control rod insertion time is 1 second or less. The NRC staff concludes that the 1-second scram insertion time is typical of TRIGAs as documented in NUREG–1537, Appendix 14.1, Section 3.2(1), and is correctly used as an assumption in the RRR safety analysis for the reactivity insertion accident described in Section 2.5.4 of this report.

TS 3.2.1, Specification c, helps ensure that reactivity insertion rates are maintained consistent with the assumptions described in the safety analysis in Section 2.5.4 of this report.

TS 3.2.1 helps ensure that during the normal operation of the RRR, the time required for the control rods to be fully inserted, from the instant that a safety channel variable reaches the safety system setting, is rapid enough to prevent fuel damage. Analysis evaluated in Section 2.5.4 of this report indicates that, for the range of transients anticipated for the RRR, the specified scram time is adequate to ensure safety. TS 3.2.1 supports the basis design requirements provided in the RRR SAR to prevent fuel damage, and is therefore, acceptable to the NRC staff.

The NRC staff reviewed the design and performance of the control rods and finds that the control rods provide adequate reactivity worth, structural rigidity, and reliability to ensure reliable operation under all operating conditions. 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 similar TRIGA reactors and demonstrated reliable operation and a long service life. The design of these control elements is consistent with the requirements provided in the safety analyses.

Based upon its review of the information provided in the RRR 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 RRR from any operating condition or applicable accident scenario. The control rod design for the RRR includes reactivity worths that can control the excess reactivity planned for the RRR, including the assurance of an acceptable shutdown reactivity and margin. The licensee has justified appropriate TS design limits, LCOs, and surveillance requirements for the control rods. On the basis of the information provided above, the NRC staff concludes that the RRR control rods and applicable TSs are acceptable.

2.2.4 Neutron Moderator and Reflector

The RRR SAR response to RAI No. 9 (Ref. 4) describes the RRR moderator and reflector. The moderator consists of both the contributions from the hydrogen and zirconium 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 neutrons. The water in the RRR pool acts as both a moderator and reflector, in addition to being used as a coolant. The reflector surrounding the core consists of the 21 aluminum-clad graphite reflector elements and the ring-shaped block of graphite.

The reflector ring assembly rests on the reflector platform. The inside diameter of the reflector ring is approximately 18 inches, with a radial thickness of 12 inches and a height of 22 inches. A welded aluminum container that encases the graphite keeps water from contact with the graphite. The rotary specimen rack (lazy susan) is an integral component to the reflector ring.

Between the reflector ring and the outer row of lattice positions, a gap filled with water also provides reflection. A total of 21 lattice positions in the outer ring contain graphite reflector elements. These elements are of the same dimensions as the fuel elements, but are filled entirely with graphite.

The fuel elements contain two sections of graphite, one above and one below the fuel, which serve as axial reflectors for the core. The fuel itself provides a significant contribution to moderation of the neutrons through scattering with the fuel matrix.

Based upon its review of the information in the RRR SAR and discussed above, the NRC staff finds that the moderator and reflector elements used in the RRR are consistent with other TRIGA reactors. The NRC staff has reviewed the constituents, materials, and components for the reflector elements and concluded that they are in agreement with the description provided in the RRR SAR. On the basis of the information provided above, the NRC staff concludes that the RRR moderator and reflector elements are acceptable.

2.2.5 Neutron Startup Source

The RRR SAR response to RAI No. 10 (Ref. 4) describes the facility's startup source. The neutron source is a 1.64-Curie (Ci) americium-beryllium unit installed in 1968. The primary function of the neutron source is to provide neutrons for reactor startup. Insufficient neutrons would result in actuation of an operational interlock that would prohibit control rod withdrawal with the logarithmic channel nuclear instrument less than 2 counts per second as required by TS 3.2.3, Table 3. The licensee stated that a neutron source cladding failure would be detected during routine analysis of the RRR pool water for radioactivity as required by RRR TS 3.3, Specification e.

The NRC staff finds that the RRR neutron startup source is similar to other TRIGA reactor startup sources. On the basis of the information provided in the RRR SAR and discussed above, the NRC staff concludes that the RRR neutron startup source is appropriate for use in the RRR, and is therefore, acceptable to the NRC staff.

2.2.6 Core Support Structure

The RRR core support structure is described in the RRR SAR and in response to RAIs Nos. 11 and 14 (Ref. 4). The RRR core support structure is a square, all-welded aluminum-frame structure that rests on four legs that are held down by aluminum anchor bolts welded to the bottom of the aluminum tank in the RRR pool. Oversized bolt holes permit some horizontal adjustment. The reflector assembly rests on the reflector platform. Two aluminum channels welded to the bottom of the reflector container provide support. Four holes in the lower flanges of the channels are used to attach the reflector to the reflector platform with bolts and nuts.

The core components are contained between top and bottom aluminum grid plates. The plates have 127 total positions in 6 concentric rings around a central port (used for high-flux irradiations). Interstitial coolant passages in the bottom core plate provide coolant flow. The arrangement helps ensure a stable and reproducible core configuration. Penetrations in the core plates allow for sufficient coolant flow.

On the basis of its review of the information provided in the RRR SAR and described above, the NRC staff finds that the RRR reactor core components are typical of TRIGA reactors, will be capable of positioning and aligning the RRR fuel elements for all anticipated operating and accident conditions, and will provide adequate coolant flow to the fuel elements. On the basis of the information provided above, the NRC staff concludes that the RRR core support structure is acceptable

2.3 Reactor Tank or Pool

The RRR SAR, Chapter 4.1, and response to RAI No. 12 (Ref. 4), describes the RRR reactor tank. The RRR reactor core is located at the bottom of an oval-shaped aluminum tank that is 10 ft in diameter and 15 ft long with a 5-ft radius at each end. The tank is 25 ft deep and is bolted at the bottom to a 24-inch-thick poured concrete slab. The tank has a minimum wall thickness of 0.25 inches and is surrounded by approximately 2.5 ft of concrete. The tank is sealed by continuous welded joints; the integrity of the joints was verified by x-ray testing, pressure testing, dye-penetrate checking, and soap-bubble leak testing during construction. For corrosion protection, the outside of the tank is coated with a double layer of tar and felt. A 2-inch by 2-inch aluminum channel used for mounting the neutron detectors and underwater lights is welded around the top of the tank. The top of the tank is surrounded by a steel frame 11 ft wide and 16 ft long, which is fabricated with 10-inch structural-steel channels and recessed in the top of the shield structure. The tank is filled with demineralized water to a depth of 24.5 ft, providing approximately 20 ft of shielding above the top of the core.

The RRR is a natural convection water-cooled pool type reactor. The reactor pool is open to the atmosphere. The RRR core is cooled by natural circulation of the reactor tank water. The volume of the tank is approximately 25,000 gallons. There are no beam ports in the RRR. Based on the size and low power rating (250 kW) of the RRR, operation of the primary coolant system is not required as a safety system for the facility, but is used to maintain efficient reactor operation and water quality. The water in the reactor pool is used to moderate the reactor, to cool the fuel elements during reactor operation, and to shield against the 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, maintain control of potential of Hydrogen (pH), maintain optical clarity, and shield radiation generated in the core.

The following TS design features establish the basic requirements for the reactor coolant system:

2.3.1 TS 5.2 Reactor Coolant System

TS 5.2 states the following:

Specifications.

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

b. The pool water inlet and outlet pipes shall be equipped with siphon breaks not less than 5 meters above the upper core plate.

TS 5.2, Specification a, helps ensure proper cooling to the RRR core. Information provided in the RRR SAR confirms that the RRR core can be cooled by natural convection flow without the need for forced cooling. The NRC reviewed the operating history and finds that the design is in agreement with GA design assumptions, and concludes that natural convection is sufficient to provide adequate RRR cooling.

TS 5.2, Specification b, helps ensure that water cannot be drained by siphon inadvertently. This help ensures an adequate quantity of water for cooling the RRR core and provide radiation shielding.

On the basis of its review of the information provided in the RRR SAR and response to RAI No. 12 (Ref. 4), the NRC staff finds that the design of the coolant system will provide adequate cooling and radiological shielding for the RRR. The NRC staff also finds that RRR reactor tank including the constituents, materials, and components of the reactor tank are typical of other TRIGA reactors. TS 5.2, Specifications a and b, characterize important design features of the RRR which conform to the recommendations in NUREG–1537 and ANSI/ANS-15.1-2007. Furthermore, compliance with TS 5.2 will help ensure the RRR coolant system remains consistent with the assumptions and design bases in the RRR SAR. On the basis of the information provided above, the NRC staff concludes that the design of the reactor tank and the requirements of TS 5.2 are, therefore, acceptable.

2.4 Biological Shield

The RRR biological shield is described in the RRR SAR and the response to RAI No. 13 (Ref. 4). The RRR biological shield consists of the reactor pool water, the tank, and the surrounding concrete. The reactor sits at the bottom of the tank, which is 25 ft deep. Normally, at least 20 ft of water covers the core. The pool is surrounded by 2.5 ft of reinforced concrete embedded below floor grade. There are no accessible areas below floor grade.

As discussed in the RRR SAR Section 11.1.7, environmental monitoring is required to ensure compliance with 10 CFR Part 20 and the RRR TS. Installed monitoring systems include area radiation monitors and airborne contamination monitors. The facility has maintained a comprehensive environmental and facility monitoring program for approximately 35 years. The licensee stated that the program provides monitoring results demonstrating that the operation of the facility imposes an insignificant impact on local radiation levels and radiation exposure around the RRR facility. Annual reports from 2005 to 2011 were reviewed and the annual released were far below the allowable limits of 10 CFR Part 20. The corresponding IRs contain no contradictory findings.

Based on a review of the information provided in the RRR SAR and responses to RAI No. 13, the NRC staff finds that the RRR biological shield components are typical of TRIGA reactors and will limit radiation exposure from the reactor and reactor-related sources of radiation. On the basis of the information provided above, the NRC staff concludes that the RRR biological shield design is acceptable.

2.5 Nuclear Design

The reactor design bases, as described in the RRR 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

The SL is based on the fuel temperature, which, because of the strongly negative temperature coefficient of reactivity of the TRIGA fuel, contributes to the inherent safety of the TRIGA reactor. A limit on reactor power ensures operation within the RRR SAR design analysis as well as below the fuel temperature SL and pool water temperature limits.

2.5.1 Normal Operating Conditions

As previously discussed in Section 2.2 of this report, the RRR LCC consists of 64 fuel elements: 54 aluminum-clad fuel elements and 10 stainless-steel clad fuel elements. The fuel elements are arranged from the B-ring to the G-ring. The A-ring is used as a central irradiator.

The licensee used the MCNP computer modeling program to produce the design characteristics including the reactor power distributions and reactivity coefficients. The MCNP model solves the Boltzman transport equation using a Monte Carlo technique. It has been extensively benchmarked and is widely used in the RTR community for neutronic evaluations. The RRR LCC is described with the core loading diagram provided in the SAR (Ref. 5) using the geometry and core loading parameters obtained from the manufacturing drawings. The MCNP power distribution calculation is provided in the RRR RAI response (Ref. 4), Figure 9.

The NRC staff reviewed the licensee's use of MCNP for the RRR core analysis and concluded that the analysis in the RRR SAR and RAI response (Ref. 4) satisfied all TRIGA operational limits as described in NUREG–1537, and that the RRR normal operating conditions were bounded by the limits imposed by the RRR LCC. On the basis of the information provided above, the NRC staff concludes that the RRR LCC is, therefore, acceptable.

2.5.1.1 TS 3.1.1 Steady-State Operation

TS 3.1.1 states the following:

> Specification.
>
> The steady-state reactor power level shall not exceed 250 kW.

The RRR SAR RAI responses (Ref. 4 and Ref. 5) provide updated thermal hydraulic calculations and design analysis which provide the fuel temperature limits expected during steady-state operation. The accident analysis presented assumes a maximum reactor power of 285 kW which is greater than the licensed power level of 250 kW. This provides a level of conservatism to the results. The reactor power operational limit is maintained through operator observation of instrumentation at 230kW. An automatic scram is provided by TS 3.2.3 at 275 kW. The maximum calculated fuel temperature for the LCC is 264 degrees C at the steady-state operation of 250 kW in accordance with the TS 3.1.1. TS 3.1.1 helps ensure that the SL is maintained.

The NRC staff finds that TS 3.1.1 is appropriate and effective to help ensure that the licensee maintains operational limits with the RRR SAR design analysis. On the basis of the information provided above, the NRC staff concludes that TS 3.1.1 is, therefore, acceptable.

2.5.2 Reactor Core Physics Parameters

Calculational Methodology

The RRR core performance is evaluated by modeling the RRR core using the MCNP code. The calculated values for the purpose of licensing and demonstration of the adequacy of the applicable TSs were performed at the licensed power level of 250 kW. The NRC staff reviewed the modeling techniques used and finds that they were acceptable and appropriately implemented. The NRC staff also reviewed the level of accuracy used to model physical attributes and finds that the level of detail presented was acceptable. More specific review results are provided as follows:

Excess Reactivity, Shutdown Margin, and Control Rod Worth

The MCNP core model for the RRR was validated by comparing calculated excess reactivity and control rod worth with the corresponding measured values for these parameters in the RRR core. The SAR (Ref. 5), Section 6, summarizes calculated and measured control rod worths and excess reactivity.

The control rod worth comparisons indicated acceptable agreement in that the results vary from 10 percent to 30 percent. The main sources of measurement error are caused by the reading of control rod positions, power fluctuation during the rod drop, and the reading of the rod drop graph. The shutdown margin (SDM) was derived by assuming +$3.00 core excess reactivity (maximum allowed) and all rods in with the maximum worth rod stuck out of the core. The calculated excess reactivity and shutdown margin were within the TS requirements. Considering the measurement error noted above, and MCNP simulation confidence interval, the MCNP calculation results compared acceptably with the measurement data.

The NRC staff noted that the rod worth measurements are performed in accordance with RRR operating procedures. Following the procedure, the regulating rod is calibrated first and is withdrawn sequentially from full insertion (0 percent withdrawn) to full withdrawal (100 percent withdrawn). After the regulating rod is calibrated, the reactor is configured with the regulating rod and shim rod fully withdrawn and the safety rod inserted to maintain criticality. This process results in two control rods fully withdrawn and the third near mid-height. The safety and shim rods are then calibrated by pulling the safety rod, measuring the period, and then inserting the shim rod to reestablish criticality at the initial starting power. The result of this method is that the shim and safety rods curves are based on measurements taken on only the upper portions of the rods. The remainder of the rod worth curves is established by fitting a third-order polynomial to the collected data. Results are also customarily plotted to show rod worth as $0.00 when the rod is 100 percent withdrawn. The remainder of the rod worth curves indicates the amount of reactivity inserted by inserting a control rod to the indicated position. The NRC staff concludes that this method of rod worth measurement is acceptable.

Calculated and measured core parameters are provided in Table 2-4.

Table 2-4 RRR Comparisons of Measured and Calculated Core Parameters

Component	Calculated ($): MCNP	Measured ($): 8/13/2010	% Difference [C-M/average]
Excess reactivity ($)	+1.65	+1.54	6.8%
Safety control rod	-3.52	-3.10	12.7%
Shim control rod	-3.53	-3.05	14.6%
Regulating control rod	-1.06	-1.31	-21.1%
Sum of all rods	-8.11	-7.92	2.4%

The NRC staff finds that the measured values versus the calculated values of the RRR reactivity are typical of research reactor measurements and evidence that the overall acceptable agreement and validation of the MCNP code results are acceptable. On the basis of the results provided in Table 2-4, the NRC staff concludes that the core parameters, both measured and calculated are, therefore, acceptable.

Excess Reactivity

The purpose for monitoring excess reactivity is because it is a component of the SDM calculation, which is a basic safety requirement. In addition, the change in excess reactivity with burnup is expected to be predictable and consistent and this change may be reviewed over time to monitor for reactivity anomalies.

TS 3.1.3 Core Excess Reactivity

TS 3.1.3 states the following:

> Specification.
>
> The maximum available excess reactivity based on the reference core condition shall not exceed $3.00.

TS 3.1.3 establishes a limit on excess reactivity, allowing operational flexibility while limiting the reactivity available for reactivity addition accidents. The maximum excess reactivity helps establish a basis for ensuring that an adequate shutdown margin is available by control rod insertion.

Since RRR fuel has no burnable poisons, the excess reactivity of the core is reduced by reactor operation. The licensee calculated the RRR excess reactivity to be +$1.66 and measured +$1.65 for this parameter. The NRC staff reviewed the excess reactivity calculation and measurement, and concluded that the values provided were consistent with other TRIGA reactors, and were acceptable.

The NRC staff finds that RRR has selected the minimum excess reactivity in TS 3.1.3 that will allow the reactor to operate in accordance with the design features provided in the SAR, while allowing for the operational flexibility to conduct various experiments. On the basis of the information provided above, the NRC staff concludes that TS 3.1.3 is, therefore, acceptable.

Shutdown Margin

The guidance in NUREG-1537 provides that the SDM requirement is the amount by which the reactor must be subcritical following a scram or trip, with the strongest worth control rod fully removed from the core. The value often used by TRIGA research reactors, and used by this licensee, is -$0.50. To demonstrate that the SDM requirement is met, the licensee calculated the reactor shutdown reactivity. The estimated shutdown reactivity must be more negative than the SDM requirement under any and all operating conditions and appropriate accident scenarios. The guidance in NUREG–1537 also recommends that the licensee use the operating characteristics established in the LCC (e.g., control rod worths), to demonstrate that the SDM requirement (-$0.50) can be achieved under any operating conditions and all appropriate accident scenarios (see NUREG–1537, Section 4.5.1).

RRR has incorporated the SDM requirements into the TS as described below:

2.5.2.1 TS 3.1.2 Shutdown Margin

TS 3.1.2 states the following:

> Specification.
>
> The reactor shall not be operated unless the shutdown margin provided by control rods is greater than $0.50 with:
>
> a. Irradiation facilities and experiments in place and the total worth of all experiments in their most reactive state;
>
> b. The most reactive control rod fully withdrawn; and
>
> c. The reactor in the reference core condition.

TS 3.1.2, Specification a, helps ensure constraints on the core reactivity condition by considering the highest worth unsecured experiment to be in its most reactive state, to help ensure that the reactor remains subcritical, should an unsecured experiment move to its most reactive position.

TS 3.1.2, Specification b, helps ensure that the reactor can be shut down even if the most reactive control rod becomes stuck out of the reactor core.

TS 3.1.2, Specification c, helps ensure proper core reference conditions for deriving the SDM value. The reactivity state of a reactor can be affected by the fission product xenon, which is a neutron poison, and the temperature of the reactor. The purpose of defining a reference core condition is so that reactivity measurements can be adjusted to a fixed baseline. The reference core condition is the most limiting for determining the SDM.

TS 3.1.3 requires the maximum excess reactivity of the core to be $3.00 and the SDM requirement for the core to be at least -$0.50, assuming that the maximum worth control rod is stuck in the fully withdrawn position (the stuck control rod requirement). The SDM is highly dependent on the knowledge of the excess reactivity (ρ_X) and control rod worths. Surveillance requirements TS 4.2(d) provide determination of control rod worths, and TS 4.1, Specification b, provides determination of ρ_X.

Shutdown Margin—Confirmatory Analysis

The NRC staff performed a confirmatory analysis of shutdown margin using information provided in the RRR SAR which included both measured and calculated control rod worths. Table 2-5 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 (ARI-1) removed. In the case of the RRR, where the Shim1 control rod is the largest worth control rod in either measurements or calculations, this can be expressed as:

$$\rho_S = \rho_X - \rho_{SA} - \rho_R - \rho_E \leq -\$0.50$$

where the applicable component values and definitions are contained in Table 2-5.

Table 2-5 RRR Shutdown Margin Calculations

Calculation number (strongest rod withdrawn)	**Initial excess reactivity, ρ_X TS 3.1.3**	**Safety Rod (ρ_{SA})**	**Shim Rod (ρ_{SH})**	**Reg. Rod (ρ_R)**	**Exp. Worth (ρ_E) TS 3.6.1(b)**	**Shutdown reactivity (ρ_S)**	**SDM req. (ρ_{SDM})**
Calculated rod worths							
1. ARI-1	+$3.00	-$3.52	stuck out	-$1.06	+$1.00	-$0.58	-$0.50
Measured rod worths							
2. ARI-1	+$3.00	stuck out	-$3.05	-$1.31	+$1.00	-$0.36	-$0.50

Calculation 1—Calculated Rod Worths

In this calculation, the TS 3.1.3 value of the core excess reactivity is offset by insertion of all control rods except the maximum worth control rod. Calculated values of the control rod worths are used. As stated in the applicability of TS 3.1.2, the value (+$3.00) of the initial core reactivity consists of all components, including the reactivity of the unsecured experiments. The result is a shutdown reactivity that is more negative than the SDM requirement. This is a confirmation of the adequacy of the TS 3.1.3 requirements and is acceptable.

Calculation 2—Measured Rod Worths

In this calculation, the TS 3.1.3 value of the core excess reactivity is offset by insertion of all control rods except the maximum worth control rod. Measured values of the control rod worths are used. As stated in the applicability of TS 3.1.2, the value (+$3.00) of the initial core reactivity consists of all components, including the reactivity of the unsecured experiments. The result is a shutdown reactivity that is not more negative than the SDM requirement and thus would not be allowed by TS 3.1.2. However, in order to satisfy the SDM requirement, the licensee would have to reduce the excess reactivity (ρ_X) or the experimental reactivity (ρ_E) to in order to satisfy TS 3.1.2 requirement.

Based on the above discussion and calculations provided in Figure 2-5, the NRC staff finds that the licensee's values for measured and calculated excess reactivity, SDM, and control rod worths are acceptable. On the basis of the information provide above, the NRC staff concludes that the SDM requirement provided in TS 3.1.2 is, therefore, acceptable.

2.5.3 Reactivity Coefficients

Prompt Negative Fuel Temperature Coefficient

In the RRR SAR, the licensee states that a significant feature of a TRIGA reactor is the large, prompt, negative fuel temperature coefficient of reactivity (FTC), 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 fuel, clad, or moderator temperatures. 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 reactor operator 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. 15).

The FTC represents the change in reactivity per degree 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. RRR selected a constant value of -0.0103 $/K for use in its analysis.

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 2-5. In this model, a central rod region is used for stainless-steel cladded fuel elements; this region contains only fuel in aluminum-clad fuel elements. The physical dimensions of the model are taken from the RRR SAR.

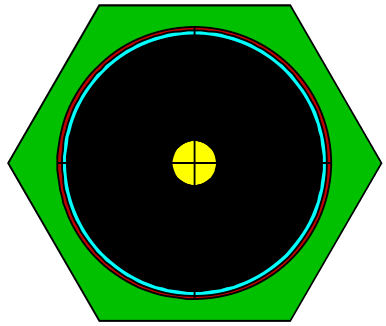

Figure 2-5 WIMS model of RRR fuel elements

The program WIMS-ANL (Ref. 19) is used to perform the confirmatory analysis. This program used a 69-group library specifically developed for research and test reactor (RTR) analysis. This library has cross-sections at a wide range of temperatures of 300 to 1,600 degrees Kelvin (K). 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 eigenvalue calculations was performed (e.g., for 150 degrees C, calculations are performed at 145 degrees C and 155 degrees C). Coefficients were calculated at each temperature of interest. Buckling values are selected to provide exactly critical conditions at 31 degrees C and were then used to model core average leakage in all of the cases. The results were converted into degrees K for presentation.

Reactivity 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.

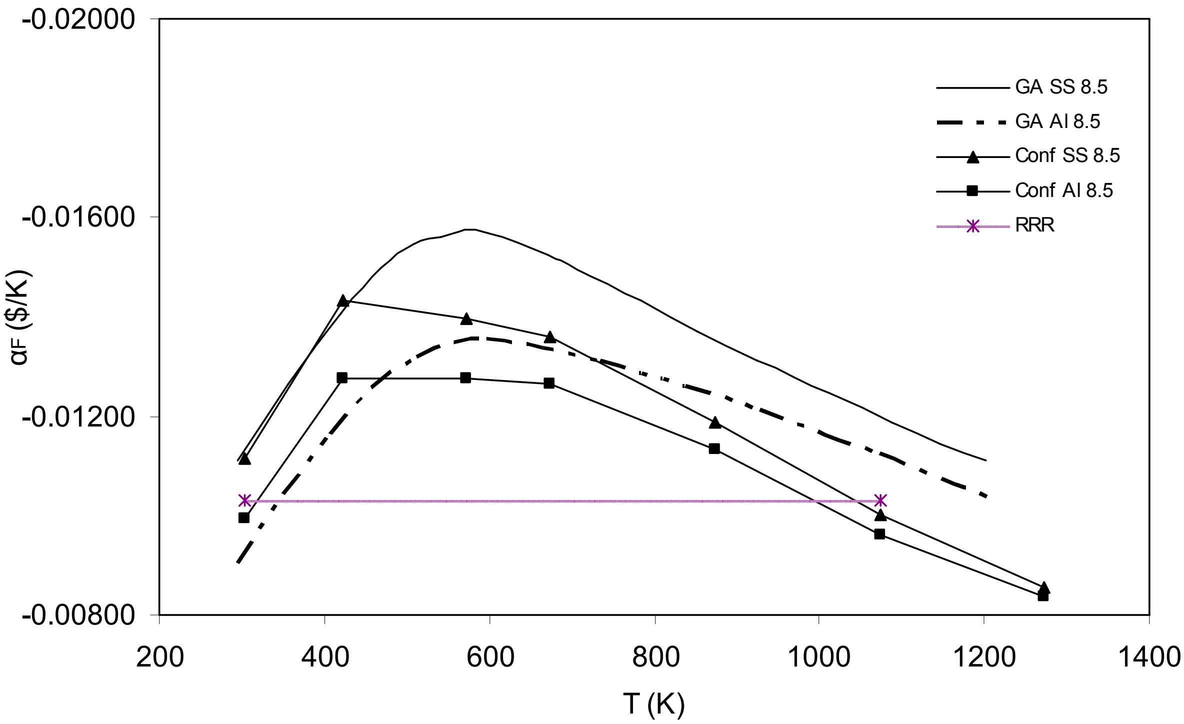

Figure 2-6 GA-7882, confirmatory, and RRR fuel temperature coefficients

The reference GA results, NRC staff confirmatory results, and RRR's chosen value of -0.0103 $/K, are displayed on Figure 2-6 above. The NRC staff finds that the FTC value the licensee selected, -0.0103 $/K, provides less negative reactivity feedback than is physically available in the reactor; therefore, it is conservative for the safety analyses results. On the basis of the information provided above, the NRC staff concludes that the RRR FTC is, therefore, acceptable.

2.5.4 Transient Analysis of an Uncontrolled Rod Withdrawal

In response to RAI No. 8 (Ref. 4), the licensee discusses the Oregon State point reactor kinetics model (PRKE) and states the conclusions of its analysis. The method outlined was compared with other methods used by RTRs for LRA review. The NRC staff reviewed the PRKE model and methods and finds that they are acceptable and appropriate for the analysis of reactivity insertion events in the RRR. This analysis is based on the following conservative assumptions as explained in the response to RAI No. 8 (Ref. 4):

1. the maximum worth rod (Shim, $3.53) is continuously withdrawn,
2. the withdrawal rate is $0.16/second,
3. the reactor scrams at a high power setpoint of 285 kW,
4. rod insertion takes 1 second following the initiation of the scram signal,
5. only the two lowest-worth rods are inserted during the scram (regulating and safety), and
6. the reactivity addition because of rod withdraw terminates at the same time that motion of the other two rods begins (i.e., the withdrawal of the rod ceases when the SCRAM signal initiates insertion of the other two rods).

The peak power attained is 3.9 MW and the reactor scram (trip) occurs 6.91 seconds following the initiation of the transient. The transient event occurs very fast such that the heat transfer conditions are essentially adiabatic and the maximum fuel temperature attained was calculated to be approximately 127 degrees C. The licensee concluded that the departure from nucleate boiling was not a concern during the most severe potential rod withdrawal accident scenario.

Uncontrolled Rod Withdrawal—Confirmatory Analysis

The NRC staff performed a confirmatory calculation using the TRACE model (Ref. 20) developed for the RRR. Section 2.6 contains a further discussion of this model. In this analysis, the maximum power density rod is modeled in order to determine the maximum fuel temperature resulting from the transient.

The confirmatory calculation used the RRR value of β_{eff} = 0.0075, with calculated rod worths of \$1.31 (regulating), \$3.05 (shim), and \$3.10 (safety). The maximum worth control rod (the safety) was withdrawn at the critical condition with the two remaining rods already fully withdrawn. The initial RRR power was 0.1 kW. The scram signal occurred at 285 kW (conservative assumption) and the rod continues to move out of the core momentarily. The calculation assumes a delay time between the initiation of the scram signal and the movement of the shim and regulating control rods of 0.5 seconds. Then the two remaining control rods are fully inserted. The calculation used the TS 3.2.1 scram time of 1 second and the withdrawal rate of \$0.16/s with the reactivity insertion worth of the two inserted control rods being linear over the insertion time.

The scram signal occurs at 6.5 seconds at a power of 285 kW. Power continues to rise until the delay time is exhausted. At this time the peak power of 1,103 kW is attained very briefly. Because nuclear processes occur much more rapidly than thermal-hydraulic processes, the peak fuel temperature, which begins the transient at ambient temperature (25 degrees C), peaks at 51 degrees C. The results are shown in Figure 2-7.

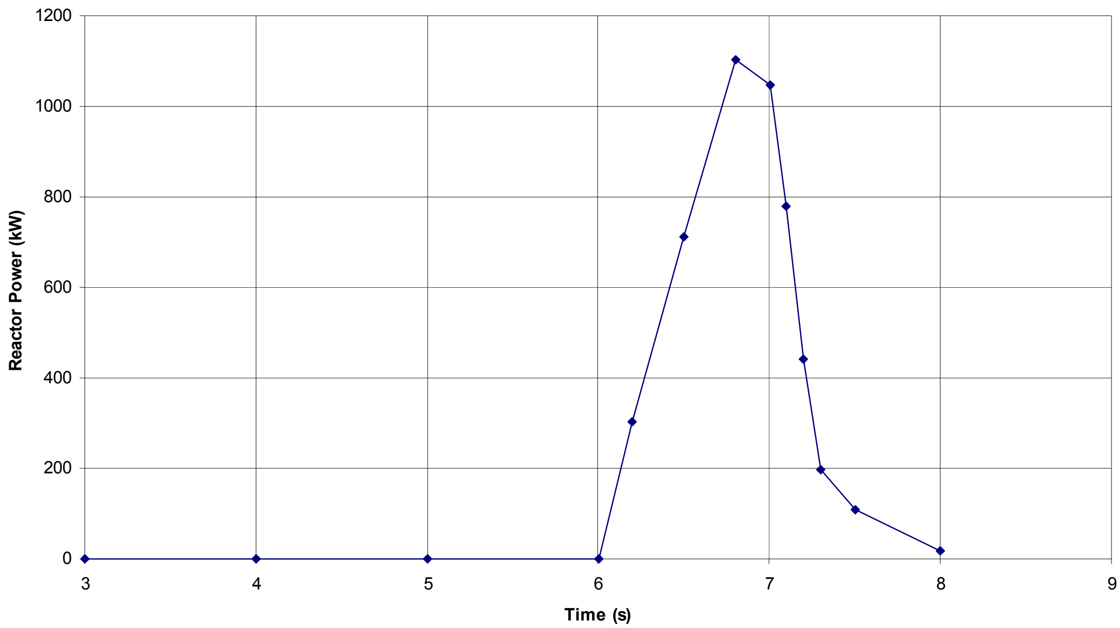

Figure 2-7 Uncontrolled rod withdrawal

The NRC staff finds that the peak power reached, for a very short time, does not pose a challenge to the fuel temperature SL or the DNBR because the time duration is extremely fast and the heat transfer so low that it cannot result in any appreciable temperature increase. Consequently, the confirmatory analysis substantiates the licensee's values provided in TS 3.2.1, Specification b, of 1 second for the control rod scram insertion time, and TS 3.2.1, Specification c, of $0.16/s for the control rod withdrawal rate limit (see Section 2.2.2 of this SER).

Based on the information provided above, the NRC staff finds that the licensee's results for the uncontrolled rod withdrawal accident scenario are acceptable. Furthermore, based on the results of the NRC staff's confirmatory analysis provided above, the NRC staff concludes that the results of the uncontrolled rod withdraw accident scenario are, therefore, acceptable.

2.5.5 Operating Limits

10 CFR 50.36(d)(1) requires reactors to specify SLs and limiting safety system settings (LSSSs). These regulations define safety limits as limits upon important process variables necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity. LSSSs for nuclear reactors are defined as settings for automatic protective devices related to those variables that have significant-safety functions. Whereas a TS 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, and 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 there is a buildup of excessive pressure between the fuel moderator and the cladding and if the fuel temperature then exceeds the SL. The presence of air, fission product gases, and H from the dissociation of the H and Zr in the fuel moderator causes such pressure. The fuel temperature and the ratio of H to Zr in the alloy determine the magnitude of this pressure.

The SL for the stainless-steel-cladded, high-hydride TRIGA fuel is based primarily on experimental evidence obtained during high-performance reactor tests on this fuel. 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 (Ref. 14).

The RRR TS establish a conservative value of 1,000 degrees C as the SL. This SL will prevent excessive stress on the cladding because of the H pressure caused by phase change of the U-ZrH fuel. Based on the theoretical and experimental evidence (Ref. 15), the NRC staff concludes that 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 SL TS 2.1 is provided in Section 2.2.2 of this report.

The licensee's TS include an LSSS to ensure that there is a considerable margin of safety before the safety limits specified above are reached. The LSSS is required for the operation of the reactor under 10 CFR 50.36, "Technical Specifications."

For each parameter on which the SAR establishes a safety limit, a protective channel should be identified that prevents the value of that parameter from exceeding the safety limit. The calculated setpoint for this protective action, providing the minimum acceptable safety margin considering process uncertainty, overall measurement uncertainty, and the transient phenomena of the process instrumentation, is defined as the LSSS. 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).

As the fuel temperature limit is established as a safety parameter in the RRR SAR, the NRC staff notes that the associated protective channel(s) are the percent power and linear power channels; the logarithmic channel is an additional monitoring channel and does not have a safety function and, therefore is not included for consideration here. The setpoint for these channels corresponds to 275 kW.

2.5.5.1 TS 2.2 Limiting Safety System Setting

TS 2.2 states the following:

> Specification.
>
> The limiting safety system setting shall be equal to or less than 300 kW as measured by a power measuring channel.

In the response to RAI No. 14 (Ref. 5), the licensee provided Table 2-6 below with shows the relationship between reactor core integrated power and maximum fuel temperature. At 300 kW, the maximum fuel temperature is 292 degrees C. This temperature is well below the TS 2.1 fuel temperature SL of 1,000 degrees C.

Core Power (kW)	Maximum Fuel Temperature (degrees C)
174	218
250	264
275	278
300	292
347	321
434	370
500	406

Table 2-6 Core Power vs. Maximum Fuel Temperature

Furthermore, the NRC staff finds that the percent power and linear power measuring channels provide a trip (scram) of the reactor at a power level of 275 kW, as specified in TS 3.2.3, Table 3. For this reason, the two redundant power measuring safety channels will protect the LSSS setpoint (300 kW). The scram setpoint of 275 kW is more conservative than the LSSS setpoint of 300 kW, which is consistent with the guidance in NUREG–1537. Based on the information provided above, the NRC staff concludes that the LSSS setpoint of 300 kW is acceptable.

2.5.5.2 TS 3.2.2 Reactor Power Measuring Channels

TS 3.2.2 states the following:

> Specifications.
>
> The reactor shall not be operated unless the reactor power measuring channels in Table 2 are operable.

Table 2: Power Measuring Channels[1]

Measuring Channel	Minimum Number Operable
Percent Power Channel	1
Linear Channel	1
Logarithmic Channel	1

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

The RRR SAR Section 7.2.3 describes the RRR safety control system as having three neutron detectors: a fission chamber, a compensated ion chamber, and an uncompensated ion chamber. The logarithmic channel uses the signal from the fission chamber; it provides no scam function. The linear channel uses the compensated ion chamber, which provides a scram function. The percent power channel uses the uncompensated ion chamber, which provides a scram function.

TS 3.2.2 states that three independent measuring channels must be operable when the RRR is operating. The NRC staff finds that the RRR TS 3.2.2 is consistent with the guidance in

NUREG–1537, Appendix 14.1, Section 3.2(8) by providing redundant power measurement capability to the RRR. On the basis of the information provided above, the NRC staff concludes that TS 3.2.2 is, therefore, acceptable.

2.5.5.3 TS 3.2.3 Reactor Safety Systems and Interlocks

TS 3.2.3 states the following:

> Specifications.
>
> The reactor shall not be operated unless the minimum number of safety channels described in Table 3 and interlocks described in Table 4 are operable.

Table 3: Minimum Reactor Safety Channels

Safety Channel	Function	Minimum Number
Percent Power	Scram at 275 kW or less	1
Linear Power	Scram at 275 kW or less	1
Loss of High Voltage	Scram	2
Console Manual Scram	Scram	1

Table 4: Minimum Interlocks

Interlock	Function	Minimum Number
Source Interlock	Prevent control rod withdrawal with neutron-induced signal less than 10^{-7}% of full power	1
Control Rod Drive Circuit	Prevent simultaneous manual withdrawal of two control rods	1

Power Level Scram

Both the percent power and linear power scram at 275 kW. As discussed above, these reactor safety channels are redundant, diverse, and protect the same setpoint, which is the LSSS. The setpoints are bounded by the RRR SAR neutronics and thermal-hydraulics analyses performed for the LCC (Ref. 5). Acceptable SAR results are provided for the DNBR, control rod worths, and radiation doses resulting from normal operation and accident analyses. The high-power trip setpoint of 285 kW was used in the analysis of reactivity insertion events and therefore bounds this setpoint. The NRC staff finds that these LCO setpoints are consistent with the guidance in NUREG–1537, Appendix 14.1, Section 3.2(4). On the basis of the information provided above, the NRC staff concludes that the TS 3.2.3 LCO scram setpoint of 275 kW is, therefore, acceptable.

Loss of High-Voltage Scram

The LCO setpoint for the automatic trip on loss of high voltage to the reactor console (less than 90 volts) is provided in the RRR SAR. This LCO helps ensure that the accuracy of reactor core measure 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 consistent with the guidance in NUREG–1537. On the basis of the information provided above, the NRC staff concludes that TS 3.2.3 LCO on the loss of high voltage is, therefore, acceptable.

Interlocks

The function of the source interlock is to prevent control rod withdrawal with neutron-induced signal less than 10^{-7}% of full power and control rod drive circuit interlock is to prevent simultaneous manual withdrawal of two control rods. The NRC staff finds that this circuit interlocks are typical for TRIGA facilities, are appropriate to RRR operation, have been properly considered in the RRR SAR, are supported by appropriate bases, and are consistent with the guidance in NUREG–1537. On the basis of the information provide above, the NRC staff concludes that the source and control rod drive circuit interlocks are, therefore, acceptable.

The NRC staff finds that the RRR nuclear design analysis described in the RRR SAR (Ref. 1 through Ref. 5) is typical of TRIGA reactors and is properly controlled and implemented in the applicable RRR TSs. The NRC staff finds that the LSSS and reactor safety channels in TSs 2.2 and 3.2.3, respectively, are based on acceptable analytical review and analysis by the licensee, confirmed by NRC staff calculations, and are sufficient to protect the SL in TS 2.1. On the basis of the information provided above, the NRC staff concludes that the RRR nuclear design and applicable TSs are therefore, acceptable.

2.6 Thermal-Hydraulic Design

The important parameter in the thermal-hydraulic design of a reactor is the critical heat flux (CHF), which describes the heat flux associated with the departure from nucleate boiling. The parameter of interest is the departure from nucleate boiling ratio, which is the ratio of the CHF to the maximum heat flux at full power. The guidance in NUREG–1537, Appendix 14.1, Section 2.1.2 recommends that this ratio should be greater than 2.0 for forced cooled systems; and this same value is often applied to natural convection system such as TRIGAs.

The licensee has presented a detailed analysis of the RRR DNBR using RELAP-3D as described in the RRR SAR (Ref. 4). The evaluation of the safety margin that exists during the operation of the RRR at the licensed power level is based on this analysis.

Figure 2-8 illustrates the RRR RELAP-3D model presented by the licensee. The model 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 for inlet pressure and temperature conditions to change with time. The cold leg is used 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 serves no physical purpose in the RRR but is simply a connector 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 RRR subchannel. Using the peaking factors from the neutronic analysis appropriately, this channel has been adjusted to represent the hot channel having the most conservative thermal-hydraulic parameters found in the RRR core.

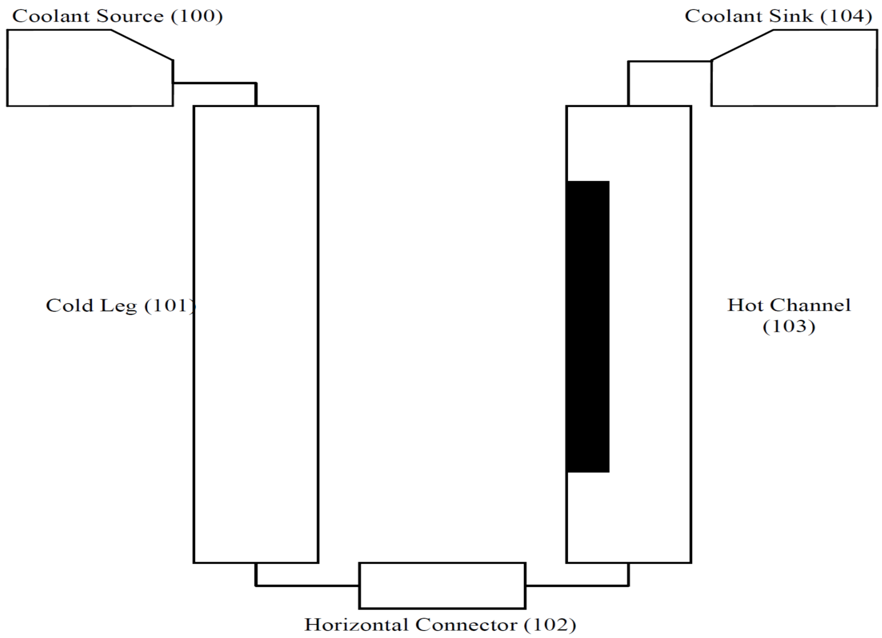

Figure 2-8 The RRR RELAP-3D model

RRR assumed that there was no cross-flow between adjacent channels which is conservative because it predicts higher values of temperature and lower margins to the DNBR. Publications cited in the RRR SAR state that the single channel model provides critical heat flux results that are within approximately 1.0 percent of those produced from two- and eight-channel models and the single-channel model produces the most conservative results.

In the licensee's thermal-hydraulic model, the hot channel in the core is cooled through natural circulation. The steady state thermal-hydraulic performance of the RRR is determined for the reactor operating at 250 kW with a water inlet temperature of 48.9 degrees C. The TS 3.3, Specification b limit for the pool temperature is 40 degrees C, so the analytic value provides conservatism in the DNBR calculation.

The maximum power fuel rod in a typical subchannel is described for the RRR under steady-state conditions. Figure 2-9 illustrates the RRR subchannel configuration. RELAP-3D used the Bernath correlation to determine the DNBR. The power in the hottest rod at which critical heat flux is predicted to occur was also calculated as was the maximum fuel temperature in the hottest rod.

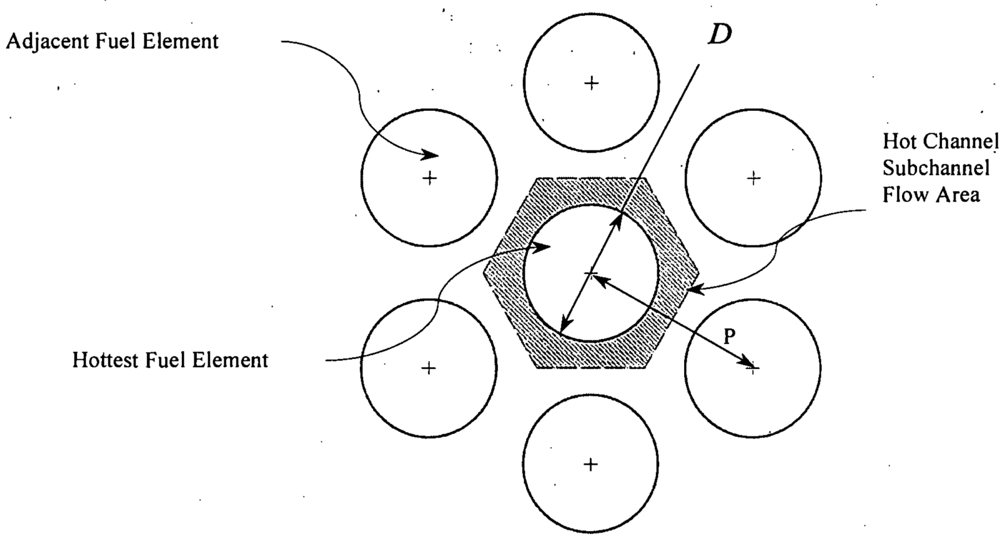

Figure 2-9 Schematic of the subchannel used for DNBR analysis

The RRR thermal-hydraulic analysis only considered stainless-steel fuel elements since the inner portion of the LCC only contained stainless-steel cladded elements. The fuel rod was represented in the hot channel by 24 discrete radial and axial nodes each. The mesh points within the fuel region that the model used correspond to 1 node for the central zirconium rod, 20 nodes of equal radial thickness for the fuel meat, 1 node for the fuel to clad gap, and 1 node for the clad.

Power-peaking factors for each core configuration were taken from the MCNP results provided in the neutronic analysis from the RRR SAR. The highest power rod for each configuration is determined by calculating the total power produced in each fuel element present in the configuration. After the highest power rod was determined, further analyses were performed to find the detailed axial and radial power shapes associated with that rod. The axial and radial power shapes were determined for 20 equally spaced nodes in both the axial and radial directions. The MCNP results were used to calculate three peaking factors:

- Hot channel fuel peak factor = (maximum fuel rod power)/(core average fuel rod power).
- Hot channel fuel axial peak factor = (maximum axial power in the hot rod)/(average axial power in the hot rod).
- Hot channel fuel radial peak factor = (maximum radial power in the hot rod)/(average radial power in the hot rod).

Table 2-7 below summarizes the peaking factor results.

Table 2-7 Hot Channel Fuel Power Summary

Hot Rod Location	Hot Channel Fuel Thermal Power [kW]	Hot Channel Fuel Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Axial Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Radial Peak Factor [P_{max}/P_{avg}]	Effective Peak Factor
B5	7.24	1.844	1.291	1.240	2.952

The average fuel element power was 3.91 kW (250 kW/64 fuel elements in the LCC). The ratio of the power in the maximum power fuel element to the average fuel element was 1.844. The MCNP analysis determined that the ratio of the peak power in the maximum power fuel element to the average power in that element was 1.291. Similarly, the ratio of the maximum fuel element power at the peak axial location to the average power at that location was 1.240.

The hottest fuel rod power, in the RRR LCC, was 7.20 kW (Ref. 5).

To evaluate DNBR for the RRR core, the licensee used Bernath's correlation because (1) it was traditionally used in research reactor SARs and (2) the correlation produces the most limiting CHF ratio values over all other correlations.

The licensee also provided an analysis demonstrating the relationship between fuel element power and peak fuel temperature using a RELAP-3D model. The results provided in the RRR SAR and reproduced in Figure 2-10 and Figure 2-11, demonstrate the expected linear relationship between fuel temperature, fuel rod thermal power, and DNBR.

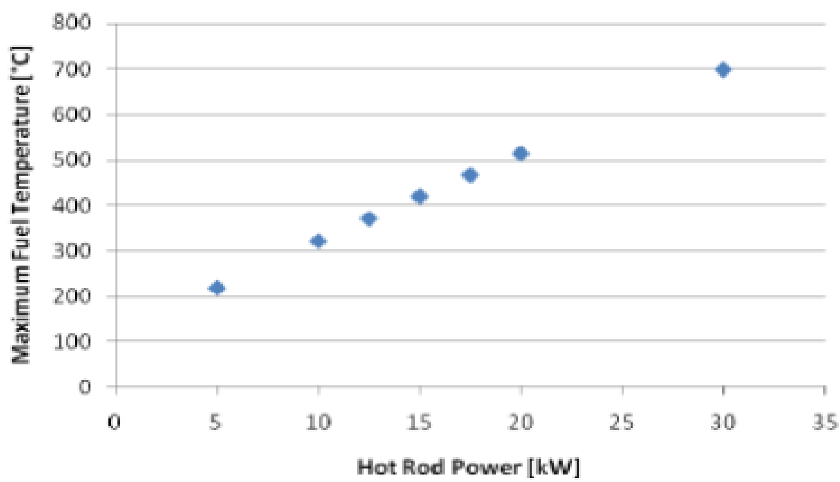

Figure 2-10 Maximum fuel temperature as a function of fuel element power

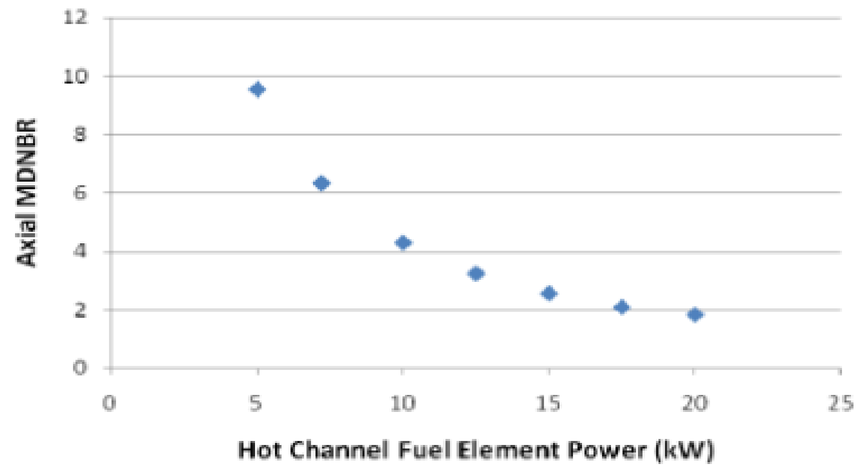

Figure 2-11 Hot channel MDNBR

Figure 2-10 and Figure 2-11 illustrate that the RRR can be operated with an acceptable DNBR and at a fuel temperature below the TS 2.1 SL value of 1,000 degrees C. The RRR LCC minimum DNBR was calculated to be 6.33 at the maximum fuel temperature of 264 degrees C at 250 kW steady-state operation. The guidance in NUREG–1537 recommends a minimum DNBR of 2; therefore, the RRR minimum DNBR of 6.33 is well above the minimum DNBR value. The RRR DNBR was consistent with the values reported for other TRIGA research reactors with similar operating power levels.

The NRC staff reviewed the assumptions used in the RRR thermal-hydraulic model and the calculated results for peaking factors, DNBR, and fuel rod temperature, and finds that the assumptions were acceptable and the calculated parameters are consistent with the values calculated and measured for similar research reactors. A confirmatory analysis performed by the NRC follows.

DNBR—Confirmatory Analysis

The NRC staff performed a confirmatory thermal-hydraulic analysis of the RRR. Table 2-7 shows data from the RRR SAR and the results of the thermal-hydraulics confirmatory analysis.

Table 2-8 Thermal-Hydraulic Model Parameters

Parameter Value	RRR	Confirmatory Analysis	% Difference (RRR vs. Confirmatory)
Flow area [m^2]	3.88×10^{-4}	3.2090×10^{-4}	20.9%
Fuel element pitch [m]	0.04054	0.04054	0.0%
Wetted perimeter [m]	0.117	0.117	0.0%
Hydraulic diameter [m]	1.301×10^{-2}	1.0906×10^{-2}	19.3%
Heated diameter [m]	3.724×10^{-2}	3.0099×10^{-2}	23.7%
Fuel element heated length [m]	0.381	0.381	0.0%
Fuel element surface area [m^2]	4.469×10^{-2}	4.4844×10^{-2}	-0.3%

The NRC staff confirmed the calculation of the RRR DNBR using the TRACE computer code (Ref. 20). The model developed is displayed in Figure 2-12.

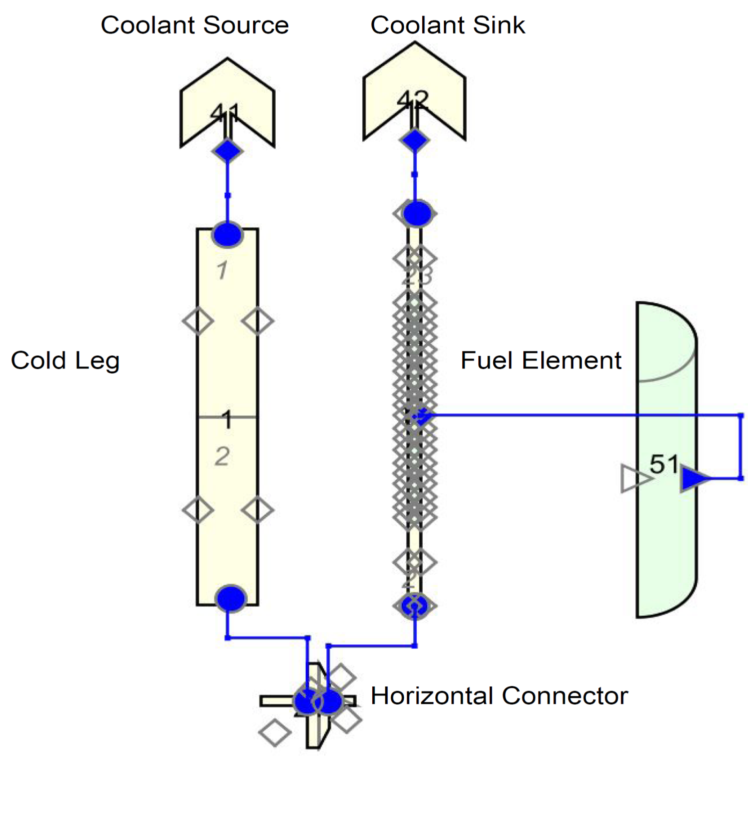

Figure 2-12 TRACE model of the RRR

The TRACE model used the corrected physical characteristics from the RRR SAR and the Bernath correlation. The results of the two models provided acceptable agreement. The

TRACE model DNBR was 6.08, while the RRR DNBR value was 6.33. Figure 2-13 illustrates the reactor analysis at various power levels to develop the natural circulation flow and heat transfer characteristics important to the results.

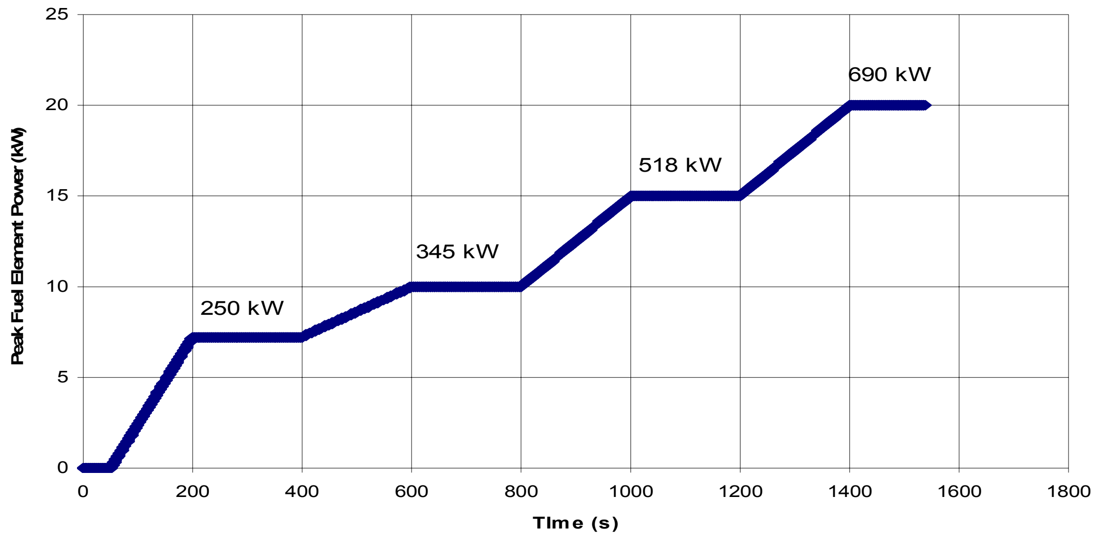

Figure 2-13 TRACE model power ramp

Figure 2-14 displays the TRACE calculated DNBR values as a function of reactor power. The calculations were performed using nominal inlet temperature (25 degrees C) and the TS limit on bulk pool temperature (40 degrees C) from TS 3.3, Specification b. The results illustrate acceptable agreement between the RRR thermal-hydraulic analyses using RELAP3-Dand the confirmatory calculation results by the NRC using TRACE.

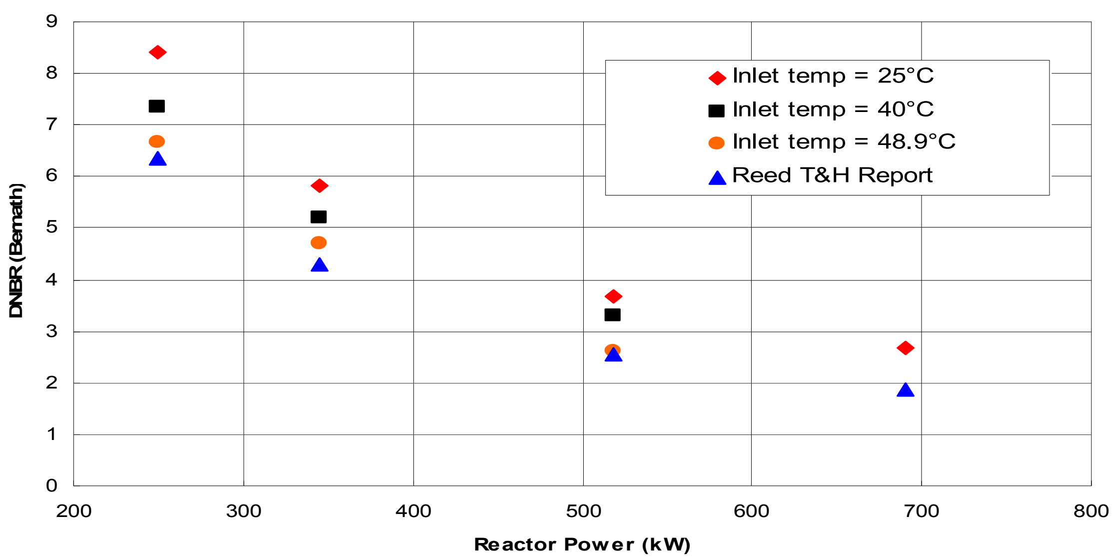

Figure 2-14 Comparison of confirmatory and RRR DNBR results

The NRC staff finds that the RRR thermal-hydraulic analysis for the RRR core adequately demonstrates that RRR can operate at the licensed power level of 250 kW with sufficient thermal-hydraulic safety margins. Furthermore, the NRC staff finds that its independent calculational results of the thermal-hydraulic analysis for the RRR provide a high level of confidence that RRR thermal-hydraulic match the actual core conditions at the licensed power level of 250 kW. Based on the information provided above, the NRC staff concludes that the RRR thermal-hydraulic analysis is, therefore, acceptable.

The licensee has provided TS pertaining to the conditions of the coolant system that are pertinent to safe and reliable operation of the RRR. They are discussed below.

2.6.1 TS 3.3 Reactor Primary Pool Water

TS 3.3 states the following:

Specifications.

a. The pool water level shall be greater than 5 meters above the upper core plate. The pool water level shall initiate an alarm signal if the pool level falls 10 cm below normal. The alarm indication shall be visible in the control room and outside the reactor facility.

b. The bulk pool water temperature shall be less than 40°C. The pool water temperature shall initiate an alarm if the pool temperature exceeds 40°C.

c. The conductivity of the pool water shall be less than 5.0 microsiemens/cm averaged over 1 month.

d. The pH of the pool water shall be between 5.0 and 7.5 averaged over 1 month.

e. The radioactivity activity of the pool water shall be less than the limits in 10 CFR 20 Appendix B, Table 3, for radioisotopes with half-lives greater than 24 hours.

TS 3.3, Specification a, helps ensure that at least 5 meters of water above the top of the core is maintained to provide adequate cooling and shielding. The TS is supported by information provided by the licensee in the responses to RAIs Nos. 12 and 22 (Ref. 4). As stated in the RAI responses, the minimum height of water of at least 5 meters provides sufficient water for effective cooling of the reactor and shielding to minimize radiation exposure to workers and the public.

TS 3.3, Specification b, helps ensure that the bulk water temperature limit is maintained to preserve the assumptions made in the DNBR analysis and to prevent the breakdown of resins important to water chemistry. The TS is supported by information provided by the licensee in RRR RAI No. 12 (Ref. 4).

TS 3.3, Specification c, follows the guidance provided in NUREG–1537, Section 5.4 and the information presented in RRR SAR Section 5.4. Many research reactor facilities have 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. Limiting this rate extends the longevity and integrity of the fuel cladding. It also helps ensure that the heat transfer between the clad and coolant will not degrade because of oxide buildup.

TS 3.3, Specification d, is supported by the information provided in the RRR SAR Section 5.4, and guidance provided in NUREG–1537, Section 5.4. The pH of tank pool water is kept between 5.0 and 7.5, ensuring the water is kept in an appropriate range. The NRC staff finds that a small rate of corrosion continuously occurs in a water-metal system. Limiting this rate extends the longevity and integrity of the fuel cladding. It also ensures that the heat transfer between the cladding and coolant will not degrade because of oxide buildup on the cladding

TS 3.3, Specification e, follows the guidance provided in NUREG–1537, Section 5.2, and provides oversight of the radiological conditions in the reactor coolant. Such monitoring will detect fuel failure before the continuous air monitor (CAM) or area radiation monitor (ARM) monitors.

The NRC staff finds that the requirements in TS 3.3, Specification a through e, follow the guidance provided in NUREG-1537. Based on the information provided above, the NRC staff concludes that TS 3.3 is therefore, acceptable.

The NRC staff finds that the RRR thermal-hydraulic design analysis is typical of TRIGA reactors and is properly controlled and implemented in the RRR TS 3.3. The NRC staff reviewed SAR Chapter 13.2.3 and finds that if the RRR reactor is operated in accordance with TS 3.3, the radiological consequences of a loss of primary coolant are acceptable. The licensee has the ability to detect and contain potential leakage from the tank. The information provided in the RRR SAR on thermal-hydraulic design was validated by confirmatory analysis and therefore is acceptable to the NRC staff. Based on the information presented above, the NRC staff concludes that the RRR thermal-hydraulic parameters and TS 3.3, Specifications a through e, are, therefore, acceptable.

2.7 Reactor Description Conclusions

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 RRR core without undue risk to public health and safety or the environment. The NRC staff 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 TS. The NRC staff concludes that the thermal-hydraulic analysis in the RRR SAR demonstrates that the RRR core results in acceptable safety margins for thermal-hydraulic conditions.

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 performed by the NRC staff. The NRC staff reviewed the analysis of the steady-state operation of the RRR core at a power level of 250 kW 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 RRR TS 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 TS. Therefore, on the basis of this review, NRC staff concludes that there is reasonable assurance that continued operation of the RRR, up to 250 kW, as limited by the TS, 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 RRR 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 as low as reasonably achievable (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 RRR. The licensee's historic performance in this area is acceptable, and the RRR SAR provides acceptable documentation of the licensee's management of this topic.

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 RRR can be categorized as airborne, liquid, and solids.

Airborne Radiation Sources

The RRR SAR, Section 11.1.1.1, and RAI responses (Ref.4) indicate that during normal operations of the RRR, the primary airborne sources of radiation are argon (Ar)-41 and nitrogen (N)-16. Ar-41 results from irradiation of the air in experimental facilities and dissolved air in the reactor pool water. The primary means of Ar-41 production is the rotating specimen rack. Other production sources include the pool water and the pneumatic irradiation system. N-16 is produced when oxygen in the pool water is irradiated by the reactor core. The NRC staff's review considered the licensee's calculations of the production and release of routine airborne radioactive effluents and the resultant doses to the RRR staff and members of the public.

The RRR SAR, Section 11.1.1.1. indicates that the major sources of Ar-41 that contribute to occupational radiation exposure is generated in, and released from, the reactor tank, the rotating rack, the rabbit, and the vertical beam. Based on the NRC staff review of RRR annual reports covering the last five years, the highest concentration of Ar-41 discharged while at full power was 3.0×10^{-7} microcuries per milliliter (μCi/ml) (equivalent to μCi per cubic centimeter (μCi/cm^3)). The licensee determined compliance with the derived air concentration (DAC) in the reactor bay by assuming a uniform mixing of Ar-41 in the reactor with the highest discharge concentration. This concentration was an order of magnitude less than the Ar-41 DAC found in Appendix B to 10 CFR Part 20, which is 3×10^{-6} μCi/cm^3 for a semi-infinite cloud.

The licensee determined compliance with the radiation dose limits to the maximally exposed member of the public by calculating the potential dose using the highest Ar-41 release occurring over the last 5 years, assuming a full year of continuous reactor operation, using the flow rate of the exhaust stack, and using a highly stable stability class with a low wind speed. Appendix B to 10 CFR Part 20 lists the allowable effluent concentration of Ar-41 as 10^{-8} μCi/cm^3, which results in 50 milli-roentgen equivalent man (mrem)/yr for continuous exposure. The licensee calculated the highest annual dose in an unrestricted area to be 8.1^5 mrem. Because N-16 has a very short half-life of approximately 7 seconds, exposure to the workers and public is negligible.

The licensee also evaluated the potential direct exposure from Ar-41 in the reactor bay to a person in the adjacent Psychology Building. The individual was represented as a point 9.1 meters from the building and 1 meter in the air. Two 1.24-cm concrete walls were modeled between the reactor bay and the individual and the entire bay is assumed to be filled with Ar-41 at a concentration of 3.0×10^{-7} µCi/ml. The calculated exposure rate is 3.17×10^{-5} mrem/hr.

The following is stated in the RRR TSs:

3.1.1.1 TS 3.5.2 Effluents

TS 3.5.2 states the following:

> Specifications.
>
> The annual average concentration of ^{41}Ar discharged into the unrestricted area shall not exceed 1×10^{-8} µCi/ml at the point of discharge.

Based on its review of the RRR SAR, the NRC staff finds that the production and control of the RRR routine airborne radiation sources and atmospheric effluent releases of Ar-41 and N-16 are within the limitations 10 CFR Part 20 requirements. The NRC staff finds that TS 3.5.2, and information that the licensee provided in the RRR SAR, indicate that during continued normal operation of the RRR, airborne radioactive releases will result in doses to the maximally exposed member of the public on the order of 1 mrem/yr or less. Based on the information provided above, the NRC staff concludes that normal operation of the RRR is within the limits of 10 CFR Part 20, and that TS 3.5.2, is therefore, acceptable.

Liquid Radiation Sources

The RRR SAR, Section 11.1.1.2 (Ref. 1) indicates that impurities in the primary coolant become activated by neutrons as they pass through the reactor core. Most of this material is captured in mechanical filtration or ion exchange in demineralizer resins and, therefore, dealt with as solid waste. However, small quantities and concentrations of nonroutine liquid radioactive waste are generated from decontamination or maintenance activities. The RRR SAR, Section 11.1.1.2, indicates that the RRR's policy does not allow the release of liquid radioactive waste as an effluent. The SAR also indicates that semiannual liquid scintillation counts of the primary coolant, secondary coolant, and environs have no detectable measure of any quantity of tritium in the water.

Based on a review of the information provided above, the NRC staff finds that liquid radioactive sources from continued normal operation of the RRR are properly controlled and will not pose a significant hazard to public health or safety, or the environment. The NRC staff concludes that liquid radioactive sources from normal operation of the RRR are, therefore, acceptable.

Solid Radiation Sources

The RRR SAR, Section 11.1.1.3, 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 anti-contamination clothing associated with reactor experiments, surveillance, or maintenance operations. The RRR SAR, Section 11.1.1.3 states that radioactive solid waste at the RRR historically has been small and consists mostly of consumable supplies such as absorbent materials or protective clothing. When possible, solid radioactive waste initially is segregated at the point of origin and screened based on the presence of detectable radioactivity. The NRC staff reviewed the RRR annual reports from 2005 through 2011 which reported that a total of twelve 55-gallon drums had been shipped as radioactive waste. Two years had no shipments (2005 and 2010) and 2 years had three drums shipped (2009 and 2011). The remaining years shipped two drums each year. Radioactive materials (liquids, solids, gaseous) generated for research and experiments under the Federal byproduct material of the reactor facility are transferred to the State of Oregon license before conducting research or experiments.

Based on a review of the information provided above, the NRC staff finds that solid radioactive sources from normal operation of the RRR are controlled and have not resulted in any significant exposures. Based on the information provided above, the NRC staff concludes that the control of solid radioactive sources at RRR is, therefore, acceptable.

3.1.2 Radiation Protection Program

10 CFR 20.1101(a), requires each licensee to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities. As indicated in the RRR SAR Section 11.1.2, certain aspects of the radiation protection program deal with radioactive materials regulated by the State of Oregon (an Agreement State) under license ORE-90010. The RRR Director is responsible for implementation of the radiation protection program and ensuring RRR compliance with 10 CFR Part 20. The Reactor Health Physicist oversees the implementation of the radiation protection program at the reactor. 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 implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR 20. The program shall use the guidelines of the ANSI/ANS 15.11-1993; R2004, “Radiation Protection at Research Reactor Facilities.”

The license states that RRR staff and Reed College Office of Environmental Health and Safety (EHS) personnel prepare the radiation protection plan to be compliant with NRC and State regulations, and to follow the guidelines in ANSI/ANS-15.11-1993 (R2004) (Ref. 21). The goal of the program is to limit radiation exposures and radioactivity releases to a level that is ALARA without restricting facility operations for education and research. Preparation, audit, and review of the radiation protection program are the responsibility of the RRR Director. The RRR Director and Reactor Health Physicist coordinate activities and program implementation with the Environmental Health and Safety office. The Reactor Review Committee (RRC) reviews the RRR Director’s activities and audits the program. Surveillance and record keeping are the responsibility of the Reactor Supervisor, who reports to the RRR Director. ALARA activities are incumbent on all radiation workers associated with the reactor facility. Substantive changes in the radiation protection program require approval of the RRC.

The EHS office and the RRR Director are responsible for the implementation of training for radiation protection. 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 staff is integrated with the training and requalification program. Specific training requirements of 10 CFR Parts 19 and 20, the RRR radiation protection plan, the EP, and a facility walkthrough are included in the training. The RRC reviews the activities of the Director and audits the program. Substantive changes to the radiation protection program require approval of the RRC.

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

3.1.3 ALARA Program

The RRR SAR, Section 11.1.3, describes the RRR ALARA program. RRR established a program designed to keep radiation exposures to personnel ALARA in compliance 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. An NRC licensed Senior Reactor Operator reviews all experiments.

As part of the licensee's commitment to ALARA, specific goals are established to 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 RRR Supervisor, the RRR Director or both, are responsible for planning and scheduling operations, experiments, and training. Contamination control procedures also are used to further minimize radiation exposures. The RRR applies the radiation exposure limits in 10 CFR Part 20 for occupational workers, members of the public, minors, and pregnant women. The ALARA program also defines and requires surveys, monitoring, radiation records, and personnel dosimetry.

The NRC staff reviewed the RRR ALARA program and finds that it complies with the regulations in 10 CFR 20.1101, is consistent with the guidance in ANSI/ANS-15.11-1993 (R2004), and is required in accordance with TS 6.3. Based on the information provided above, the NRC staff concludes that the RRR ALARA program is, therefore, acceptable.

3.1.4 Radiation Monitoring and Surveying

The RRR SAR, Section 11.1.4 provides an overview of RRR radiation monitoring and surveying program. The radiation monitoring program for the reactor is structured to ensure that all three categories of radiation sources—air, liquid, and solid—are detected and assessed in a timely manner. Area monitors in the reactor bay and in the gaseous effluent pathway are in continuous use. Contamination surveys with a Geiger counter or equivalent, and area dose rates with ion chambers, scintillation detectors, or energy-compensated Geiger counters are performed daily before reactor operation.

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.

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 3.5.1 Radiation Monitoring Systems and Effluents

TS 3.5.1 states the following:

Specifications.

The reactor shall not be operated unless the minimum number of radiation monitoring channels are operable as specified in the accompanying table:

Table 5: Minimum Radiation Monitoring Channels

Radiation Monitoring Channel	Minimum Number Operable
Radiation Area Monitor (RAM)[1]	1
Continuous Air Monitor (CAM)	1
Environmental Dosimeters	4

[1] When the RAM becomes inoperable, operations may continue only if portable instruments are substituted for the normally installed monitor within one hour of discovery for periods not to exceed one month.

RRR SAR, Table 11.10 lists the radiation monitoring equipment used in the radiation protection program. This includes three area radiation monitors and three air monitors. This equipment provides a comprehensive set of radiation survey instrumentation that covers, with sufficient ranges, the various types of radiation that may be encountered at the RRR. TS 3.5.1 helps ensure that at least one radiation air monitor (RAM) and one CAM are operable to support reactor operations and that the radiation monitoring systems alert the operator in the event alarm setpoints are exceeded during reactor operation.

The RRR facility director is responsible for calibration of the instruments on site using written procedures. Calibration is indicated through stickers on each instrument; the RRR staff maintains the records, which the RRC audits annually. Routine monitoring and surveillance is performed with the radiation monitoring equipment provided in the RRR SAR, Table 11.10. In addition, TS 3.5.1 requires the licensee to have a comprehensive set of area radiation monitors, air monitors for the stack, and a set of portable radiation survey instrumentation, with sufficient ranges, to detect the various types of radiation that may result from reactor operations. The licensee also has other specialized radiation monitoring equipment, such as gamma spectroscopy systems and a liquid scintillation spectrometer.

A CAM located in the reactor bay will actuate the ventilation system to isolation mode when alarmed. The CAM display is located on the reactor console for the operator's use. The CAM samples air drawn air from the reactor bay after it passes through a paper particulate filter. The

CAM will also alarm on high airborne activity. A similar unit samples air from the exhaust stack, through which all air from the facility passes; this unit may be used as a backup if the CAM fails.

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). Furthermore, based on the information described above, the NRC staff concludes 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 20.150(a) and (b), and TS 3.5.1 is, therefore, acceptable.

3.1.5 Radiation Exposure Control and Dosimetry

RRR SAR, Section 11.1.5 provides an overview of radiation exposure control and dosimetry program. Radiation exposure control depends on factors such as the facility's 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 estimates of annual doses at various locations.

The shielding for the RRR TRIGA reactor is similar to designs used successfully at many other research reactors. The principal design feature for control of radiation exposure during operation is the column of water surrounding the reactor, and the below grade elevation of the reactor tank. The RRR is designed so that radiation from the core area can be accessed through vertical ports for research and educational purposes. Radiation exposure is controlled by restricting access to areas of elevated radiation fields.

The regulations in 10 CFR 20.1502 requires monitoring of workers likely to receive an annual dose from sources external to the body, in excess of 10 percent of the limits described in 10 CFR 20.1501. 10 CFR 20.1502 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 RRR SAR, Table 11.11 provides a list of the average occupational exposures at the RRR for the last 6 years and indicates that all doses were below the regulatory limits. The licensee states that there have been no instances of any exposures in excess of 10 percent of the applicable limits. However, the RRR maintains a radiation dosimetry program and restricts access to areas of elevated radiation fields to help compliance and ensure personnel protection.

Personnel who enter the control room or the reactor room either hold the authorization for unescorted access, or do so under the direct supervision of an authorized escort. The RRR control room and reactor room are designated as restricted areas. This includes locked doors and access controls to prevent unauthorized entry. When the reactor is operating, the Senior Reactor Operator or the licensed operator in charge is responsible for controlling access to the control room and reactor room. Personnel who enter the reactor room have a record of accumulated dose measured by gamma dosimetry, using either a personnel dosimeter or a self-reading dosimeter. RRR staff evaluates the potential for personnel exposures before any work with radioactive materials begins to ensure that the correct dosimeters are issued. Internal dosimetry is evaluated by urinalysis for tritium and in vivo thyroid counting for radioiodine uptake.

The licensee collects and maintains records of occupational exposure information using the appropriate NRC forms. Records of self-reading dosimeters are kept in a log book maintained by the RRR staff as permanent records, as are measurement results of accidental releases to the environment. The environmental monitoring records from over 35 years of operation, quantify that the facility has had an insignificant effect on the local environment, with no accidental releases.

Based on its review of information provided in the RRR SAR, the NRC staff finds the facility's exposure and control program to be acceptable. The NRC staff finds that the RRR has the appropriate equipment and procedures for radiation exposure and control. The NRC staff finds that RRR personnel exposures are controlled through satisfactory radiation protection and ALARA programs and have had historically low radiation doses. Furthermore, based on the results of RRR annual reports from 2005–2011, annual radiological releases to the environment are far below NRC allowable limits. The NRC staff review of IRs from 2005 through 2011 found no contradictory findings. On this basis the information provided above, the NRC staff concludes that the licensee's control of personnel exposures and dosimetry is, therefore, acceptable.

3.1.6 Contamination Control

The RRR SAR, Section 11.1.6 describes the RRR's contamination control program which is comprised of written procedures for radioactive material handling, use of trained personnel, and implementation of a monitoring program designed to detect contamination in a timely manner. The RRR facility contains no areas with continuously removable contamination. Locations most likely to have radioactive contamination are the sample port at the rotary specimen rack and the sample-handling hood for receiving irradiated samples. Contamination control methods include training, procedures, and protective measures such as the use of removable absorbent paper pads with plastic backing are implemented as part of the contamination control program. Workers are required to wear protective gloves and other appropriate protective clothing when working in potentially contaminated areas. Workers are required to perform surveys prior to leaving a contaminated work area to ensure that no contamination remain present on hands, clothing, or shoes. If contamination is detected, personnel decontamination processes would be employed in accordance with RRR Standard Operating Procedures that involve the use of a mild soap and lukewarm water. Materials and tools are monitored for contamination before removal from contaminated areas or from restricted areas likely to be contaminated. All personnel leaving the reactor bay are surveyed for contamination. On a biweekly basis, routine radiation smears tests are performed for the reactor bay and control room. The NRC staff reviewed the RRR annual reports from 2005 through 2011 and IRs from 2005 through 2011 which indicated no contradictory findings.

Based on the information provided above and the NRC staff's review of the RRR 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 RRR facility.

3.1.7 Environmental Monitoring

The RRR SAR, Section 11.1.7, provides information on the facility's environmental monitoring program. Environmental monitoring is conducted at RRR to ensure compliance with Subpart F of 10 CFR Part 20 and the RRR TS. Installed monitoring systems include area radiation monitors and air monitors that have been managed and maintained in a comprehensive program. TS 3.5.1 requires one area radiation monitor; however, as noted in the RRR SAR, Table 11.10, the RRR maintains three monitors within the reactor bay. With the exception of Ar-41, which is discussed in Section 3.1.1 of this SER, there are no pathways for radioactive materials from the RRR to enter the unrestricted environment during normal operations. The RRR TS 3.5.1 requires one CAM in the reactor bay. The monitoring program includes two additional air monitoring systems to monitor stack effluents, which alarm in the reactor control room. Calibration of the air monitors is accomplished as required by the TS 4.5, and in accordance with facility procedures. Quarterly environmental monitoring is conducted by measuring both gamma-ray doses within the facility and exterior to the facility using fixed area dosimetry. Radiation surveys are conducted daily before operation at the RRR.

As required by 10 CFR 20.1501, contamination surveys are conducted to ensure compliance with regulations reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive material, and potential radiological hazards. The NRC staff reviewed of RRR annual reports and IRs from 2005 through 2011, and finds that the operation of the RRR has had an insignificant impact on the local environmental radiological level and radiation exposure in and around the facility.

Based on its review of the information described above, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the RRR 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 health and safety of the public and the environment. The RRR SAR, Section 11.2, provides an overview of the facility's radioactive waste management program.

3.2.1 Radioactive Waste Management Program

The objectives of the RRR radioactive waste management program are to minimize and properly handle, store, and dispose of waste. The NRC staff reviewed the RRR radioactive waste management program during a site visit conducted on November 10, 2009. The onsite review, and the review of RRR SAR, Section 11.2.1, confirmed that acceptable controls are in place to prevent uncontrolled personnel exposures from radioactive waste operations and provide the necessary accountability to prevent unauthorized release of radioactive waste. Based on the information provided above, the NRC staff concludes that the RRR radioactive waste management program is, therefore, acceptable.

3.2.2 Radioactive Waste Controls

The RRR SAR, Section 11.2.2 describes the radioactive waste controls at the RRR. RRR considers radioactive waste to be any item or substance that is no longer of any use to the facility and contains or is suspected of containing radioactivity above the established natural background radioactivity. The RRR SAR, Section 11.2.2 differentiates between radioactive waste and radioactive effluents, notably Ar-41. Waste volumes at the RRR historically have been small and of known characterization. When possible, radioactive waste is segregated at the point of origin from items not considered to be radioactive waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the projected future need for the materials involved. Consumable supplies, such as absorbent materials or protective clothing, are declared radioactive waste if radioactivity above background is found to be present.

When possible, radioactive waste is segregated at the point of origin from items not considered to be radioactive waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the projected future need for the materials involved. Oregon is an Agreement State; therefore, radioactive materials generated for research and experiments under the Federal byproduct material license of the reactor facility are transferred to the State of Oregon license for disposal. Although disposal of liquids to the sanitary sewer system is permitted in accordance with 10 CFR 20.2003, the RRR policy, as described below, does permit the release of liquid wastes.

Based on its review of the information in the RRR SAR, Sections 11.2.2 and 11.2.3, as described above, the NRC staff concludes that the RRR facility has adequate radioactive waste controls in place to monitor the radiation exposure from radioactive waste, perform required handling operations, and prepare proper documentation for transfer to offsite disposal.

3.2.3 Release of Radioactive Waste

The RRR SAR, Section 11.2.3 states that the RRR does not allow the release of radioactive waste into the environment. The licensee states that the only exception is gaseous radioactive effluents, notably Ar-41, which is regulated under 10 CFR Part 20 and the U.S. Environmental Protection Agency (EPA) discharge limits. Gaseous effluents are monitored using air monitors in the stack to ensure compliance with the regulatory limits (i.e., the allowable effluent concentration for Ar-41 is 10^{-8} μCi/cm^3). If contaminated liquids are produced, they are contained locally, added to an absorbent, and transferred to a solid radioactive waste barrel in preparation for transfer to a licensed burial facility. Solid waste is likewise routinely contained on site.

The NRC staff reviewed the information provided in the RRR SAR Section 11.2.3, and finds that the RRR 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

On the basis of the evaluation of the information presented in the RRR SAR, observations of the licensee's operations during a site visit, and results of the NRC inspection program, the NRC staff concludes the following regarding the RRR radiation protection program and waste management:

- The RRR 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 systems provided for the control of radiological effluents, when operated in accordance with the TS, 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 RRR 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 RRR 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 RRR, 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 acceptably identifies and describes potential radiation sources and controls them.

- 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. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas give reasonable assurance that doses to the RRR 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 RRR radiation protection program and waste management summary as described in the RRR SAR Chapter 11. The NRC staff finds that the RRR 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 RRR radiation protection and waste management programs will provide acceptable radiation protection to the workers, the public and the environment.

4. ACCIDENT ANALYSES

The RRR SAR provided accident analyses to demonstrate that the health and safety of the public and workers can be protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analyses presented in the RRR SAR provided the basis to establish the RRR TS described throughout this report. The accident analysis presented in this chapter ensured that no credible accident could lead to unacceptable radiological consequences to the RRR staff, the public, or the environment. Additionally, the licensee, consistent with the guidance in NUREG–1537, analyzed the consequences of the maximum hypothetical accident (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
- loss-of-coolant accident
- loss-of-coolant flow
- 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 RRR, the MHA is defined as the rupture of the cladding of one fuel element in air. The scenario assumes that such an accident occurs at full licensed power long enough for the inventories of all radionuclides in the scenario to be at their maximum concentration. The analysis assumes that, at the time of clad failure, the volatile fission products have accumulated in the gap and are released abruptly into the air with no radioactive decay; this includes the release of noble gases, halogens, and other volatile fission products. The licensee provided the MHA information in the RRR SAR and in response to RAI No. 36 (Ref. 4) which the NRC staff used it to evaluate the event.

Nuclide Inventory

For determining the radionuclide inventories, the licensee assumed the reactor had been operating continuously for 1 year at 250 kW. This resulted in all of the halogens and noble gases (except for krypton (Kr)-85) reaching their saturated activities. This assumption is extremely conservative because RRR seldom operates continuously at 250 kW for periods longer than 8 hours to 10 hours. The licensee calculated halogens and noble gases inventories based on a highest power fuel element of 7.2 kW in the LCC. The licensee used information on fission product yield to estimate the saturation inventories.

Based on the power history assumptions provided in the RRR SAR, the NRC staff determined that the inventories of isotopes dominant to the MHA dose calculations (halogens and noble gases) are at the saturation (maximum) concentration for continuous full power operation of the RRR with the exception of Kr-85, which is a long-lived isotope. However, the NRC staff noted that the Kr-85 contribution was small when compared with other more dominant contributors such as the iodine isotopes. The NRC staff performed an independent calculation using the saturation concentration of Kr-85 (at continuous full power operation) to confirm that the Kr-85 was a negligible contributor to the MHA dose because it only accounted for approximately 10^{-4} mrem in the MHA scenarios. The NRC staff reviewed the RRR analysis and performed confirmatory calculations using information on fission yields and information from NUREG/CR–2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," and confirmed that the licensee's approach was acceptable. On the basis of its review, the NRC staff concludes that the RRR inventory estimates are acceptable for the MHA dose calculations.

Release Fractions

The licensee calculated the releases of noble gases and halogens from fuel matrix to fuel gap using the GA-developed correlation for fission gas release (Ref. 22). The correlation calculates the release based on the average fuel temperature. The licensee used an average fuel temperature of 264 degrees C (corresponding to the LCC peak rod power of 7.24 kW) and calculated a release fraction of 1.51×10^{-5} to the fuel gap.

The licensee evaluated two scenarios following the postulated fuel cladding failure involving a release directly to the reactor room and a release into the reactor pool before release to the reactor room ambient air. For the first scenario where cladding failure occurs in the air, the licensee assumed that 25 percent of the halogens in the cladding gap will escape to the reactor room and eventually to the outside environment. The NRC staff finds this assumption consistent with the NRC staff position that assumes 50 percent of the gap inventory is released and 50 percent of halogens will remain airborne in the RRR reactor building. If the cladding fails in the water, only 5 percent of the halogens will escape the reactor pool water. Multiplying the 5 percent released from the reactor pool water by the fraction when the cladding fails in the air (25 percent) results in the fraction of halogens available to the reactor room and the outside environment as 1.25 percent. For both accident scenarios, 100 percent of noble gases in the fuel gap are available to the reactor room and outside environment. The total release fractions are summarized in Table 4-1. Based on the discussion above, the NRC staff finds the licensee's method acceptable.

Table 4-1 Total Release Fractions

Release Fractions	**Noble Gas**	**Halogens**
Release to air	1.5×10^{-5}	3.8×10^{-6}
Release to reactor pool	1.5×10^{-5}	1.9×10^{-7}

Atmospheric Dispersion Factor χ/Q

The licensee used the Gaussian plume diffusion model in the TID 24190, “Meteorology and Atomic Energy,” issued in July 1968 (available from the National Technical Information Service, Springfield, VA), to calculate nuclide concentration at selected downwind distances. The licensee assumes a stable atmospheric class F with a wind speed of 1 meter per second. The NRC staff used RG 1.145, Revision 1, to determine the atmospheric dispersion factors at select distances from the reactor building for comparisons and confirmatory analysis. The analysis confirmed that the method and data used in the atmospheric dispersion factor calculations are reasonable and conservative. Table 4-2 provides the comparisons of the licensee’s χ /Q values and those of a method using the guidance in RG 1.145.

Table 4-2 Comparison of χ/Q values for Ground Release

Distance (m)	RRR (s/m^3)	RG 1.145 (s/m^3)
100	3.46×10^{-2}	8.65×10^{-3}
200	1.14×10^{-2}	2.84×10^{-3}
300	5.26×10^{-3}	1.32×10^{-3}
400	3.03×10^{-3}	7.11×10^{-4}
500	2.08×10^{-3}	5.20×10^{-4}
600	1.52×10^{-3}	3.81×10^{-4}

Dose Calculations

The licensee calculated the occupational dose for an individual in the reactor room. Boundary conditions for these calculations included assuming failure of the hottest fuel element, incorporating the calculated release fractions, and assuming the reactor room has a volume of 300 cubic meters (m^3), which is an added conservatism because the cited room volume is at least 20 percent smaller than the actual volume. Other parameters used in the dose calculations include a breathing rate of 0.02 m^3/minute (m) or 3.33×10^{-4} m^3/second (s), consistent with the value given in Appendix B to 10 CFR Part 20; and a ventilation rate of 0.628 m^3/s. In addition, the licensee used dose conversion factors (DCF) for the inhalation and external exposure pathways from the DOE reports DOE/EH-0071, “Internal Dose Conversion Factors for Calculation of Dose to the Public,” issued July 1988 (Ref. 23), and DOE/EH-0070, “External Dose-Rate Conversion Factors for Calculation of Dose to the Public,” issued July 1988 (Ref. 24).

Based on the above considerations, the licensee estimated potential doses to the public at specified distances from the reactor and to individuals within the reactor facility. The NRC staff reviewed the licensee’s methods for computing the radiological dose both within and beyond the confines of the reactor facility in case of a fission product release. The licensee evaluated the following three scenarios as described in the RRR SAR and response to RAI No. 36 (Ref. 4):

Scenario A assumes the north wall of the reactor building has vanished. In this scenario, it takes 8.8 seconds for the radioactive materials in the reactor room air to be released to the outside atmosphere.

Scenario B assumes the gases are released to the outside atmosphere at a rate equal to the HVAC exhaust rate of 0.628 m^3/s. In this scenario, the licensee assumes the entire source to be exhausted to the outside atmosphere with one complete room-air change, which would take 478 seconds.

Scenario C assumes that the reactor room air leaks to the outside at a rate of 1.54×10^{-3} m^3/s, requiring 54.11 hours for one complete room-air change.

In these scenarios, the licensee assumed workers would be exposed to radiation in the reactor room for 2- and 5-minute time periods. In addition, the licensee determined the direct dose to an individual 9.1 meters from the outside of the reactor building.

MHA Dose—Confirmatory Analysis

The NRC staff performed confirmatory calculations to validate the adequacy of the licensee's total effective dose equivalent (TEDE) calculations. These calculations were performed using the assumptions, geometry, and source terms that the licensee provided. These calculations were used to confirm both intermediate estimate and final results that RRR provided.

For dose calculations, the licensee used DCFs from DOE/EH-0070 and DOE/EH-0071 (Refs. 23 and 24, respectively). The NRC staff's confirmatory analysis used DCFs from Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-502/1-88-020, issued September 1988 (Ref. 25); and FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA-402-R-93-081, issued September 1993 (Ref. 26).

Scenario A—In this scenario, the licensee assumed a sudden opening of the reactor bay north wall with an effective surface area of 34.1 m^2. This scenario assumed that a wind speed of 1 m/s would cause the air in the reactor room to exit to the environment at a rate of 34.1 m^3/s. Table 4-3 summarizes the total dose to an individual at select distances from the reactor room. This table provides both the licensee's calculated dose as well as the NRC staff's confirmatory calculated dose for the occupational and to the individuals at selected distances down wind. The table provides comparison of doses when the fuel clad failure occurs in air. Doses for failure in water are smaller.

Scenario B—In this scenario, the licensee assumes that the RRR reactor room remains intact and the HVAC system is operating. The licensee also assumes the entire reactor room air inventory would be exhausted to the environment with one complete air exchange, or about 478 seconds. The licensee calculates doses to the members of public assuming ground release. Table 4-4 summarizes the results providing both the licensee's and the NRC staff's confirmatory calculation results.

Scenario C—In this scenario, the reactor room leaks at a rate of 1.54×10^{-3} m^3/s through cracks in the reactor room wall. The licensee calculates doses to the members of public assuming ground release. Table 4-5 summarizes the dose results and provides both the licensee's and the NRC staff's confirmatory calculation results.

Table 4-3 MHA Occupational and Public Dose Estimates for Scenario A

Total Effective Dose Equivalent (mrem)			
Time (min) and Downwind distance (m)	**RRR Results**	**NRC Staff Confirmatory Calculational Results**	**Dose limit (mrem)**
Occupational Dose (mrem)			5000
2-min	0.51	0.49	
5-min	0.51	0.49	
Public Dose (mrem) at Downwind Distance			100
100 m	0.58	0.57	
200 m	0.18	0.19	
300 m	0.08	0.09	
400 m	0.05	0.05	
500 m	0.03	0.03	
600 m	0.02	0.02	

Table 4-4 MHA Occupational and Public Dose Estimates for Scenario B

Total Effective Dose Equivalent (mrem)			
Time (min) and Downwind distance (m)	**RRR Results**	**NRC Staff Confirmatory Calculational Results**	**Dose limit (mrem)**
Occupational Dose (mrem)			5000
2-min	6.04	5.85	
5-min	12.35	12.19	
Public Dose (mrem) at Downwind Distance			100
100 m	0.55	0.56	
200 m	0.18	0.18	
300 m	0.08	0.08	
400 m	0.05	0.05	
500 m	0.03	0.03	
600 m	0.02	0.02	

Table 4-5 MHA Occupational and Public Dose Estimates for Scenario C

Total Effective Dose Equivalent (mrem)			
Time (min) and Downwind distance (m)	**RRR Results**	**NRC Staff Confirmatory Calculational Results**	**Dose limit (mrem)**
Occupational Dose (mrem)			5000
2-min	6.82	6.61	
5-min	16.53	16.38	
Public Dose (mrem) at Downwind Distance			100
100 m	0.29	0.30	
200 m	0.09	0.10	
300 m	0.04	0.05	
400 m	0.03	0.03	
500 m	0.02	0.02	
600 m	0.01	0.01	

The non-occupational (public) dose calculations were performed at selected radial distances from the reactor in 100-meter increments and are not directionally dependent. The dose estimates can be considered as perimeters of concentric rings from the reactor (disregarding any shielding effects due to objects or obstacles that would result in a dose reduction from the preceding dose estimates). As demonstrated by the tables above, the estimated doses to the public are a small fraction of the limit in 10 CFR Part 20 for each scenario evaluated.

In addition, the licensee calculated the direct dose from the sources inside the reactor room to an individual 9.1 meters from the building in the adjacent psychology building. The licensee stated that its calculation considered the shielding effect provided by two 19.24 cm thick concrete walls between the individual in the psychology building and the reactor bay. The estimated dose rate was calculated to be 8.49×10^{-3} mR/hr at the beginning of the accident. The NRC staff performed an independent confirmatory calculation and obtained a dose rate of 4.13×10^{-3} mR/hr for a 38.48 cm thick concrete wall. These dose rates are consistent with the dose estimate the licensee provided and are expected to decrease with time due to decay of short half-life nuclides.

In the RRR SAR and in response to RAI No. 36 (Ref. 5), the licensee states that the reactor facility was connected to the psychology building by a single door, and that air must pass through four intervening spaces and a stairwell before reaching the connecting door. The licensee provided air leak rate measurements between the psychology building and the reactor facility through the connecting door with the following reactor facility ventilation modes: operating, isolation, and shutdown. For all reactor facility ventilation modes, the results clearly show that the air flow was a net flow into the reactor facility and out of the Psychology Building. Based on the information described above, the NRC staff concludes that the dose to an individual in the unrestricted area of the psychology building would be because of the direct transmission of the accumulated airborne radioactivity in the RRR facility, and not from an airborne transport through the four intervening spaces and stairwell. As stated above, the dose rate estimate is very small (8.49×10^{-3} mR/hr). The direct dose over time is also a small contributor, and the maximum offsite dose was calculated to be about 0.6 mrem.

MHA Dose Calculation Conclusions

The NRC staff reviewed the MHA analysis for all three scenarios, as well as the dose calculational results, and concluded that the licensee used appropriate assumptions and analytical techniques and that their conclusions were appropriate and acceptable. The independent confirmatory dose calculations performed by the NRC staff demonstrates 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 provided by the licensee. 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 RRR MHA doses are acceptable.

4.1.2 Insertion of Excess Reactivity

In the RRR SAR and in response to RAI No. 8 (Ref. 4), the licensee provided its method for analyzing the reactivity insertion event. This method was reviewed in Section 2.5.4 of this report. The licensee concluded that there was no credible means by which the control rods could be manipulated to promptly add reactivity without violating several conditions and procedures. The remaining credible option for the prompt insertion of positive reactivity was improper or unexpected movement of experiments. RRR analyzed this event in its response to RAI No. 42 (Ref. 4). As such, the licensee analyzed several combinations and determined that the limiting accident was the insertion of $2.00 positive reactivity through an experiment malfunction. TS 3.6.1(b) (see SER Section 2.1.3.1) helps ensure that this reactivity insertion limit is maintained. In this licensee's analysis, the RRR fuel temperature momentarily peaks at 1,016 degrees C, but remains below the guidance stated in NUREG-1537 of 1,150 degrees C.

Insertion of Excess Reactivity—Confirmatory Analysis

The maximum experiment reactivity of $2.00 from TS 3.6.1, Specification b, was used in the NRC staff confirmatory analysis. The values for the delayed neutron fraction and the prompt neutron lifetime were obtained from the RRR SAR. The initial reactor temperature is assumed to be 40 degrees C, consistent with the TS 3.3 limit. The initial reactor conditions were assumed to be critical at a power of 275 kW, thus assuming that the reactor was operating at greater than the licensed power but below the scram setpoint at the time of the accident. The fuel temperature coefficient used is 0.0009 ρ/°C, also from the RRR SAR. The method used was the Fuchs-Nordheim technique documented in GA-7882 (Ref. 13), and was the method GA typically used as the basis for TRIGA analysis of reactivity insertion events.

Table 4-6 Fuel Temperature Following a Step Reactivity Insertion

Parameter	RRR
Prompt neutron lifetime(s)	9×10^{-5}
Fuel temperature coefficient (ρ/°C)	7.72×10^{-4}
β_{eff}	7.5×10^{-3}
Initial reactor power (MW)	275
Initial fuel isothermal temperature (°C)	40
Amount of reactivity pulse ($)	2.00
Total peaking factor	1.844
Energy released during pulse (MW-s)	17.39
Peak power attained (MW)	343.78
Average final fuel temperature (°C)	393
Peak final fuel temperature (°C)	724

This confirmatory calculation indicates that an insertion of $2.00 in positive reactivity results in a peak fuel temperature of 724 degrees C, and does not exceed the TS SL. Based on the results of the licensee's and the NRC staff's calculations, the insertion of excess reactivity scenario provides acceptable fuel temperature results.

Based on the information provided below and the results of the NRC staff's confirmatory calculation, the NRC staff concludes that the results of the insertion of excess reactivity scenario are, therefore, acceptable.

4.1.3 Loss-of-Coolant Accident

The RRR SAR, Section 13.2.3, and response to RAI No. 37 (Ref. 4), provided the results of the licensee's loss-of-coolant accident (LOCA). The licensee considered such a LOCA extremely unlikely due to the design of the reactor pool structure, but calculated the maximum fuel temperature due to a sudden LOCA to be 150 degrees C. This fuel temperature was well below the TS SL; therefore, the LOCA is not expected to result in loss of cladding integrity.

The RRR SAR provided radiation dose rates from the loss of all shielding water. The licensee provided direct and scattered dose rates after extended 250 kW and 1.5 MW operations and a loss-of-all-shielding water event in the response to RAI No. 37 (Ref. 4). The latter results, at 1.5 MW power operation, were performed for Torrey Pines Mark F TRIGA reactor. The licensee used the ratio of RRR power level to the Torrey Pines data to estimate the potential direct and scattered radiation doses at RRR. The scattered dose was dependent on the shielding provided by the reactor building and structure. RRR SAR (Ref. 4), Table 12, provides the potential direct and scattered dose rate if all shielding water was removed. The NRC staff reviewed the licensee's LOCA results and agreed that the direct radiation doses would be proportioned to the ratio of reactor power between RRR and Torrey Pines. The NRC staff finds that the dose rates calculated for RRR were similar to other research reactors. The direct gamma dose rates from the unshielded core would require evacuation of the reactor room and the building, exclusion of the public from the vicinity of the facility boundary, and would ensure that the 10 CFR Part 20 dose guidelines to the workers, building occupants, and the public are satisfied. Therefore, the NRC staff concludes the LOCA estimated dose results were acceptable.

Based on the information provided above, the NRC staff finds that the licensee's LOCA analysis was performed with appropriate assumptions and analytical techniques. Furthermore, the NRC staff concludes that the results are acceptable.

4.1.4 Loss-of-Coolant Flow

The RRR SAR discusses the LOCA. Since the RRR uses natural convection cooling, the geometry and design of the reactor flow components make it highly unlikely that local flow blockages would occur. The RRR is located in a tank, surrounded by 25,000 gallons of water, and is cooled by natural convection flow. The RRR was designed to operate without any additional cooling capacity, such as an external heat exchanger. At 250 kW steady-state power, in a case of loss-of-coolant flow, the bulk pool water temperature would increase adiabatically at a rate of 0.037 degrees C/min. The slow rate of bulk water temperature increase would allow ample time for the RRR operators to identify the condition and evaluate and implement corrective actions.

Based on its review of the information in the RRR SAR, the NRC staff finds that the loss of the RRR ability to cool the primary coolant in the reactor pool would not result in any fuel-cladding failure or radiological consequences. The NRC staff noted that the RRR has numerous alarms (bulk water temperature, water level, water flow, and radiation monitors) available to signal the need for operator action to shut down the reactor. Given a complete loss in the ability to remove heat from the pool coolant (i.e., no coolant flow), with the reactor operating at full power (250 kW), it would take more than 40 hours for the water level to evaporate down to the top of the core. Numerous alarms would alert the RRR operators who could then easily provide additional coolant to the reactor pool from external sources.

The NRC staff reviewed the description of the RRR grid plates in RRR SAR, Section 4.2.3, which states the 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 finds that in the event of a possible blockage of a coolant channel created by a foreign object lodged in the grid plate, the open fuel element lattice would ensure sufficient continuing cooling of all fuel elements as a result of cross flow.

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 RRR SAR, Section 13.2.4, the licensee evaluated events that could cause a postulated accident involving the mishandling or malfunction of fuel. These included:

(1) a fuel-handling accident,
(2) the failure of the fuel cladding caused by a manufacturing defect or corrosion, and
(3) the overheating of a fuel element causing cladding failure.

In the unlikely event of such a failure in air, the event consequences would be bounded by the results of the MHA scenario, which have been discussed and accepted in SER Section 4.1.1 above. The NRC staff reviewed the licensee's postulated mishandling or malfunction of fuel accident scenario and concludes that the licensee's results are, therefore, acceptable.

4.1.6 Experiment Malfunction

In RRR SAR, Section 13.2.5, the licensee evaluated the scenario for a postulated experimental malfunction accident. Experiment malfunction can occur from three principal causes:

(1) unexpected reactivity,
(2) release of material from the specimen container, and
(3) detonation.

TS 3.6.1 prevents a step change in reactivity greater than $1.00 for any single unsecured experiment and $2.00 for all experiments. The analysis supplied in the RRR SAR Section 13.2.2 and evaluated in this SER Section 4.1.2, demonstrated the acceptability of this limitation. Therefore, the NRC staff concludes that experiment malfunction related to unexpected reactivity will not result in unacceptable consequences.

TS 3.6.2 limits the introduction of corrosive materials by requiring double encapsulation. This TS helps ensure that a highly unlikely double encapsulation failure is required to have a release of such material into the coolant system, and is consistent with the guidance in NUREG-1537. The NRC staff concludes that this TS helps ensure that an anticipated failure of a single capsule has acceptable consequences.

TS 3.6.2 establishes the requirement to limit the use of explosive material in RRR to 25 milligrams TNT equivalent and states that quantities less than 25 milligrams 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 this consistent with the guidance in RG 2.2 (Ref. 27) and NUREG–1537.

In an experiment malfunction leading to radiological release, the licensee states that TS 3.6.3 limits the quantity and type of material in the experiment so that in the case of experiment failure the airborne radioactivity in the reactor room or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR Part 20. A 100-percent release of this quantity into the reactor room air (TS 3.6.3, Specification a) is not expected to exceed the corresponding quantities listed in Table 3 of the RRR SAR and in response to RAI No. 36 (Ref. 4), which was evaluated as part of the MHA. As such, the dose results are bounded by the MHA analysis.

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

4.1.7 Loss of Normal Electrical Power

In RRR SAR, Section 13.2.6, the licensee evaluated the scenario for a postulated RRR accident involving the loss of normal electrical power. RRR does not require emergency backup electrical power to safely shutdown the reactor, to maintain the reactor in a shutdown condition indefinitely, and to maintain long term core cooling. The loss of normal electrical power will cause the reactor to shut down through loss of voltage to the control rod drive mechanism and a reactor scram, as required in TS 3.2.3. The licensee indicated that the postulated loss of normal electrical power will not result in any scenario that could cause the release of radioactive material. The licensee's staff has the ability to verify that the control elements are fully inserted by visual inspection at the reactor bridge. The loss of normal electrical power does not affect the radiation safety and alarm equipment in the reactor room because an emergency battery-powered backup unit is used. 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 has no impact on the reactor pool.

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

4.1.8 External Events

In RRR SAR, Sections 2.5, 3.4, and 13.2.7, the licensee describe the analysis of the potential impact to the RRR from external events. Floods, hurricanes, and tornadoes are not credible in the Portland area and, therefore, are not considered to pose a threat to the reactor. Seismic activity in the State of Oregon and adjacent areas are typically moderate with minor consequences. In Section 3.4 of the RRR SAR, the applicant states that there are no known faults near the RRR site. The construction of the RRR building and reactor pit are designed to resist the lateral forces of Zone 2, as specified in the Uniform Building Code when the reactor was installed. In an earthquake with significant severity, the consequences to the RRR facility are not expected to cause events more severe than the MHA. A severe earthquake accident may result in loss of electric power, which in turn results in a reactor trip and in a loss-of-coolant flow to the reactor tank. The consequence of a LOCA with the 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.

The NRC staff finds that severe storms, floods, and tornadoes were very unlikely for the area around the RRR site. The RRR building, reactor foundation, shielding structure, reactor tank, and core support structure were designed in accordance with Uniform Building Code Zone 2 requirements. Meeting these requirements helps ensure that the reactor can be safety shutdown following an earthquake likely to occur during the facility's lifetime. On this basis, the NRC staff concludes that the consequences of external events are bounded by the MHA analysis and are, therefore, acceptable.

4.1.9 Mishandling or Malfunction of Equipment

In RRR SAR, Section 13.2.8, the licensee evaluated the potential mishandling or malfunction of equipment. The licensee states that the RRR 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 would most likely result in reactor shutdown.

The NRC staff reviewed the licensee's postulated mishandling or malfunction of equipment accident scenario and finds that fuel damage is unlikely. Based on its review, the NRC staff concludes that the results of the licensee's analysis of the mishandling or malfunction of equipment results are, therefore, 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, 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.

- 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 $2.00 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 $2.00 by TS 3.6.1 and therefore does not pose a threat to fuel integrity. The licensee did not identify any additional accidents involving reactivity addition that are not bounded by the supplied analysis.

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

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

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

- The accident analysis confirms the acceptability of the licensed power of 250 kW, including the response to anticipated transients and accidents.

- The accident analysis confirms the acceptability of assumptions in excess reactivity limits ($2.00).

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

4.3 Accident Analyses Conclusions

The NRC staff reviewed the radiation source term and MHA calculation for the RRR. 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 RRR 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 having consequences not bounded by the MHA. The RRR design features and administrative restrictions found in the TS prevent the initiation of accidents and mitigate any consequences. Therefore, on the basis of this 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 RRR poses no undue risk to the facility staff, the environment, or the public.

5. TECHNICAL SPECIFICATIONS

In this section of the report, the NRC staff evaluated the licensee's technical specifications (TS). The RRR TS define specific features, characteristics, and conditions required for the safe operation of the RRR facility. NUREG–1537, Part 1, Chapter 14, Appendix 14.1, and ANSI/ANS-15.1-2007 provided guidance, including an accepted style, format, and content for RTR TS. The NRC staff also relied on the references provided in NUREG–1537 and the ISG to perform this review.

5.1 Technical Specification Definitions

The licensee proposed to add or modify the following definitions to be consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007 as follows:

Audit: A qualitative examination of records, procedures, or other documents after implementation from which appropriate recommendations are made.

Channel: The combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

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

Channel Check: 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: The introduction of a signal into the channel for verification that it is operable.

Control Rod: 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:

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.

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

Core Configuration: The core configuration includes the number, type, and arrangement of fuel elements, reflector elements, and control rods (Shim, Safety, Regulating) occupying the core grid.

Excess Reactivity: 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: Any operation, hardware, or target (excluding devices such as detectors or foils) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, beam port, or irradiation facility. Hardware rigidly secured to a core, shield, or tank structure so as to be a part of their design to carry out experiments is not normally considered an experiment. Specific experiments shall include:

> **Secured Experiment:** Any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means or by gravity. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.
>
> **Unsecured Experiment:** Any experiment or component of an experiment that does not meet the definition of a secured experiment.
>
> **Movable Experiment:** A movable experiment is one where it is intended that the entire experiment or part of the experiment may be moved in or near the core or into and out of the core while the reactor is operating.

Fuel Element: A single TRIGA® fuel element.

Irradiation Facilities: The central thimble, the rotating specimen rack, the pneumatic transfer system, sample holding dummy fuel elements, and any other in-pool irradiation facilities.

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

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

Operating: A system or component is operating when it is performing its intended function.

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

Reactor Facility: The physical area defined by the Reactor Bay, the Mechanical Equipment Room, the Control Room, the Hallway, the Loft, the Classroom, the Radiochemistry lab, the Counting room, the Break room, the Storeroom, the sump area, the stairway, and the Restroom.

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

Reactor Safety Systems: Those systems, including their associated input channels, that are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

Reactor Secured: The reactor is secured when either:

a. there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection; or

b. all of the following exist:

   1. the three control rods are fully inserted;

   2. the reactor is shut down;

   3. experiments or irradiation facilities in the core are being moved or serviced that have, on movement or servicing, a reactivity worth exceeding one dollar;

   4. 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; and

   5. the console key switch is in the “off” position and the key is removed from the console.

Reactor Shut Down: The reactor is shut down if it is subcritical by at least \$1.00 in the reference core condition with the reactivity worth of all installed experiments included.

Reference Core Condition: The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<\$0.30).

Review: An examination of records, procedures, or other documents prior to implementation from which appropriate recommendations are made.

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

Scram Time: The elapsed time between the initiation of a scram signal to the time the slowest scrammable control rod reaches its fully inserted position.

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

Substantive Changes: Changes in the original intent or safety significance of an action or event.

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

Quintennial: interval not to exceed 6 years.

Biennial: interval not to exceed 30 months.

Annual: interval not to exceed 15 months.

Semiannual: interval not to exceed 7.5 months.

Quarterly: interval not to exceed 4 months.

Monthly: interval not to exceed 6 weeks.

Weekly: interval not to exceed 10 days.

Unscheduled Shutdown: 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 RRR TS definitions follow the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The RRR TS are, therefore, acceptable to the NRC staff.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 TS 2.1 Safety Limits: Fuel Temperature

See the evaluation in Section 2.2.2 of this report.

5.2.2 TS 2.2 Limiting Safety System Settings

See the evaluation in Section 2.5.5 of this report.

5.3 Limiting Conditions for Operation

5.3.1 TS 3.1 Reactor Core Parameters

5.3.1.1 TS 3.1.1 Steady-State Operation

See the evaluation in Section 2.5.1 of this report.

5.3.1.2 TS 3.1.2 Shutdown Margin

See the evaluation in Section 2.5.2 of this report.

5.3.1.3 TS 3.1.3 Core Excess Reactivity

See the evaluation in Section 2.5.2 of this report.

5.3.1.4 TS 3.1.4 Fuel Parameters

See the evaluation in Section 2.2.2 of this report.

5.3.2 TS 3.2 Reactor Control and Safety System

5.3.2.1 TS 3.2.1 Control Rods

See the evaluation in Section 2.2.3 of this report.

5.3.2.2 TS 3.2.2 Reactor Power Measuring Channels

See the evaluation in Section 2.5.5 of this report.

5.3.2.3 TS 3.2.3 Reactor Safety Systems and Interlocks

See the evaluation in Section 2.5.5 of this report.

5.3.3 TS 3.3 Reactor Primary Pool Water

See the evaluation in Section 2.6 of this report.

5.3.4 TS 3.4 Ventilation System

TS 3.4 states the following:

> Specifications.
>
> The reactor shall not be operated nor irradiated fuel moved unless the facility ventilation system is operable in one of the following operational modes:
>
> a. **Normal mode:** The exhaust, supply and control room fans are operating. The reactor bay pressure is maintained negative with respect to the control room.
>
> b. **Isolation mode:** Isolation mode is initiated by high radiation readings on the continuous air monitor. The exhaust and control room fans are operating. The reactor bay pressure is maintained negative with respect to the control room and all exhaust is diverted through an HEPA filter.

The basis for TS 3.4 provides the RRR staff the ability to operate the reactor while in isolation mode so as to minimize radioactive particulate releases during a period when the staff could be determining the location of a leaking fuel element. The NRC staff determined that this was an appropriate precaution for the leaking fuel scenario. The TS allows the HVAC system to be in both the normal mode or the isolated mode and still permits the RRR staff to operate the reactor or to move fuel. This does not change the normal operating mode, which is that the system will change to isolation mode upon receiving alarms as explained in the RRR SAR Section 1.3.5.2. The NRC staff concludes that TS 3.4, Specifications a and b, are acceptable.

5.3.5 TS 3.5 Radiation Monitoring Systems and Effluents

5.3.5.1 TS 3.5.1 Radiation Monitoring Systems

See the evaluation in Section 3.1.4 of this report.

5.3.5.2 TS 3.5.2 Effluents

See the evaluation in Section 3.1.1 of this report.

5.3.6 TS 3.6 Limitations on Experiments

5.3.6.1 TS 3.6.1 Reactivity Limits

See the evaluation in Section 2.1.3 of this report.

5.3.6.2 TS 3.6.2 Materials

See the evaluation in Section 2.1.3 of this report.

5.3.6.3 TS 3.6.3 Experiment Failures and Malfunctions

See the evaluation in Section 2.1.3 of this report.

5.4 Surveillance Requirements

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.

5.4.0 TS 4.0 General

TS 4.0 states the following:

> Specifications.
>
> a. Surveillance requirements may be deferred during reactor shutdown (except TS 4.3 a, d, and e); however, if deferred, they shall be completed prior to reactor operation unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor operation. Scheduled surveillance that cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.
>
> b. Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool, the pool coolant system, the rod drive mechanism radiation monitors, or the reactor safety systems 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.

TS 4.0 helps ensure that the quality of systems and components will be maintained to their original design and fabrication specifications. TS 4.0, described above, follows the guidance in NUREG–1537, Appendix 14.1, Section 4.0. Accordingly, the NRC staff finds that TS 4.0, Specifications a and b, provide appropriate RRR surveillance practices, are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007 and, therefore, acceptable to the NRC staff.

5.4.1 TS 4.1 Reactor Core Parameters

TS 4.1 states the following:

> Specifications.
>
> a. The shutdown margin shall be determined annually, following changes in the fuel or control rods, and following any other significant change (>$0.25) from the reference core.
>
> b. The core excess reactivity shall be determined annually, following changes in the fuel or control rods, and following any other significant change (>$0.25) from the reference core.
>
> c. Forty percent of the fuel elements in the reactor core shall be inspected visually for damage or deterioration biennially such that each fuel element is inspected quintennially.

TS 4.1, Specification a, helps ensure the determination of SDM as required to support TS 3.1.2.

TS 4.1, Specification b, helps ensure the determination of core excess reactivity as required to support TS 3.1.3.

TS 4.1, Specifications a and b, establish the requirement for ensuring that changes to the core are subject to limitation of the LCC defined in the RRR SAR and RAI responses.

TS 4.1, Specification c, addresses inspection of the fuel. The guidance in NUREG-1537 requests that this be accomplished as 20 percent per year over 5 years.

The NRC staff reviewed TS 4.1, Specifications a through c, for reactor core components. The NRC staff finds that TS 4.1 is consistent with NUREG–1537 and ANSI/ANS-15.1-2007 and is, therefore, acceptable to the NRC staff.

5.4.2 TS 4.2 Reactor Control and Safety Systems

TS 4.2 states the following:

> Specifications.
>
> a. The control rod drives shall be visually inspected for damage or deterioration annually.
>
> b. The poison sections of the control rods shall be visually inspected for damage or deterioration biennially.
>
> c. The control rod scram time shall be measured annually.
>
> d. The total reactivity worth and reactivity addition rate of each control rod shall be measured annually or following any significant change (>$0.25) from a reference core.
>
> e. A channel check of each of the reactor power measuring channels in TS 3.2.2, Table 2, shall be performed prior to each operation of the reactor.
>
> f. A channel calibration of the Linear and Percent Power Channels in TS 3.2.2, Table 2 shall be performed annually.
>
> g. A channel test of each item in TS 3.2.3, Tables 3 and 4, shall be performed annually.

TS 4.2, Specifications a and b, help ensure the operability of the control rods as required to support TS 3.2.1, Specification a.

TS 4.2, Specification c, helps ensure the acceptability of the control rod scram time as required to support TS 3.2.1, Specification b.

TS 4.2, Specification d, helps ensure the acceptability of the control rod worths to support SDM determination per TS 3.1.2 and to maintain the limits imposed by analysis on the control rod withdrawal rate per TS 3.2.1, Specification c.

TS 4.2, Specification e, helps ensure the operability of the linear power and percent power channels as required to support TS 3.2.2 channel operability.

TS 4.2, Specification f, helps ensure the accuracy of the linear and percent power channels as required to support TS 3.2.2.

TS 4.2, Specification g, helps ensure the operability of the control rod movement as required to support TS 3.2.1, Specification c.

The NRC staff reviewed TS 4.2, Specifications a through g, for reactor control and safety systems. The NRC staff finds that TS 4.2 is consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable the NRC staff.

5.4.3 TS 4.3 Reactor Primary Pool Water

TS 4.3 states the following:

Specifications.

a. A channel check of the reactor pool water level shall be performed monthly.

b. A channel check of the reactor pool water temperature and level monitors shall be performed prior to each day's operation or prior to each operation extending more than one day.

c. A channel calibration of the reactor pool water level and temperature monitors shall be performed annually.

d. The reactor pool water conductivity and pH shall be measured monthly.

e. The reactor pool water radioactivity shall be measured quarterly.

TS 4.3, Specification a, helps to ensure the availability of the reactor tank water level monitor as required to support TS 3.3, Specification a.

TS 4.3, Specification b, helps ensure that the reactor tank water temperature and level are acceptable prior to reactor operation as required to support TS 3.3, Specifications a and b.

TS 4.3, Specification c, helps ensure the accuracy of reactor tank water level and temperature monitors as required to support TS 3.3, Specifications a and b.

TS 4.3, Specification d, helps ensure the operability of the reactor pool water conductivity and pH levels as required to support TS 3.3, Specifications c and d.

TS 4.3, Specification e, helps ensure the activity of the reactor pool water is maintained as required to support TS 3.3, Specification e.

The NRC staff reviewed TS 4.3, Specifications a through e, for the reactor primary pool water system. The NRC staff finds that TS 4.3 is consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007 and, therefore, is acceptable to the NRC staff.

5.4.4 TS 4.4 Ventilation System

TS 4.4 states the following:

> Specifications.
>
> a. A channel check of the reactor bay ventilation system, to verify that it is operating, shall be performed prior to each day's operation or prior to each operation extending more than one day.
>
> b. A channel test of the reactor bay ventilation system's Isolation mode, as described in TS 3.4 b., shall be performed quarterly.

TS 4.4 helps ensure that the RRR ventilation system is operational as described in the SAR and satisfies the analysis assumptions of the SAR accident analysis and TS 3.4. The NRC staff reviewed TS 4.4 for ventilation system and finds that TS 4.4 helps ensure the assumptions in the RRR SAR and is consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.4 is, therefore, acceptable.

5.4.5 TS 4.5 Radiation Monitoring System

TS 4.5 states the following:

> Specifications.
>
> a. For the RAM and CAM listed in TS 3.5.1, Table 5:
>
>   1. A channel check shall be performed prior to each day's operation or prior to each operation extending more than one day;
>
>   2. A channel test shall be performed quarterly; and
>
>   3. A channel calibration shall be performed annually.
>
> b. Fixed-area dosimetry shall be exchanged and evaluated quarterly.
>
> c. Effluent concentration shall be evaluated annually.

TS 4.5, Specification a, parts 1 through 3, helps ensure the availability, operability, and accuracy of the RAM and CAM radiation monitoring equipment to support TS 3.5.1.

TS 4.5, Specification b, helps ensure that fixed-area dosimetry is available and evaluated as required to support TS 3.5.1.

TS 4.5, Specification c, helps ensure that the effluent concentration is available and evaluated as required to support TS 3.5.2.

The NRC reviewed TS 4.5, Specifications a thru c, and finds that the surveillance requirements, are appropriate and the surveillance intervals are consistent with the frequencies recommended in NUREG-1537 and ANSI/ANS 15.1-2007. On this basis, the NRC staff concludes that TS 4.5 is, therefore, acceptable.

5.4.6 TS 4.6 Experimental Limits

TS 4.6 states the following:

> Specifications.
>
> a. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before the reactor is operated with the experiment.
>
> b. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with Section 3.6 of these TS by the Reactor Operations Committee in accord with Section 6.5 of these TS and the procedures that are established for this purpose.

TS 4.6, Specification a, helps ensure that the reactivity worth of an experiment is determined prior to use in RRR as required to support TS 3.6.1.

TS 4.6, Specification b, helps ensure that experiments are not inserted into the reactor unless a valid safety analysis has been performed and reviewed by the Reactor Operations Committee as required to support TS 6.5.

The staff reviewed TS 4.6, Specifications a and b, surveillance requirements for controlling experiments. The NRC staff finds that TS 5.6 is consistent with the guidance in NUREG–1537 and ANSI/ANS 15.1-2007. On this basis, the NRC staff concludes that TS 4.6 is, therefore, is acceptable.

5.5 Design Features

The RRR TS design features are described and evaluated by the NRC as follows:

5.5.1 TS 5.1 Site and Facility Description

TS 5.1 states the following:

> Specifications.
>
> a. The site boundary is that boundary extending 250 feet in every direction from the center of the reactor core.
>
> b. The restricted area is that area inside the reactor facility. The unrestricted area is that area outside the reactor facility.

The NRC staff finds that TS 5.1, Specifications a and b, provide important features of the physical design of the facility used to house RRR and define the boundaries of the facility being licensed. These specifications support the accident analysis fundamental to acceptably meeting 10 CFR Part 20 requirements, define the operational and site-area boundaries for the facilities, and are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 5.1 is, therefore, acceptable.

5.5.2 TS 5.2 Reactor Coolant System

See the evaluation in Section 2.3 of this report.

5.5.3 TS 5.3 Reactor Core and Fuel

5.5.3.1 TS 5.3.1 Reactor Core

See the evaluation in Section 2.2 of this report.

5.5.3.2 TS 5.3.2 Control Rods

See the evaluation in Section 2.2.3 of this report.

5.5.3.3 TS 5.3.3 Reactor Fuel

See the evaluation in Section 2.2.2 of this report.

5.5.4 TS 5.4 Ventilation System

TS 5.4 states the following:

> Specifications.
>
> a. The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the reactor bay is approximately 300 cubic meters.
>
> b. The reactor shall be equipped with a ventilation system designed to filter and exhaust air or other gases from the reactor and release them from a stack 3.6 meters from the ground.
>
> c. The ventilation system shall be equipped with inlet dampers that can be closed from the control room. Closing the inlet dampers changes the ventilation system to isolation mode.

The NRC staff finds that TS 5.4, Specifications a through c, provide important design features of the RRR ventilation system. These specifications support the SAR accident analysis and are fundamental to acceptably meeting the dose analysis to satisfy 10 CFR Part 20 requirements. The NRC staff finds that TS 5.4 is consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 5.4 is, therefore, acceptable.

5.5.5 TS 5.5 Fuel Storage

TS 5.5 states the following:

> Specifications.
>
> a. All fuel elements or fueled devices shall be rigidly supported during storage in a safe geometry (k_{eff} less than 0.8 under all conditions of moderation).
>
> b. Irradiated fuel elements shall be stored in the reactor pool in an array that will permit natural convection cooling by water.

The NRC staff finds that TS 5.5, Specification a, limits the k_{eff} value to 0.8, which is less than the recommended value provided in NUREG–1537 and ANSI/ANS-15.1-2007. Response to RAI No. 24 (Ref. 5) describes a comprehensive analysis of the fuel element criticality for the in-tank storage racks, which demonstrates that the fuel cannot exceed the subcritical value cited in TS 5.5, Specification a, under normal or accident conditions. On this basis, the NRC staff concludes that fuel cooling is assured and acceptable.

TS 5.4, Specification b, provides the basic design requirement to help ensure adequate cooling by natural convection cooling, either by water or air, of stored irradiated fuel rods and fueled devices. The licensee provided in the RRR SAR, an analysis by Foushee (Ref. 28) that provides a comprehensive analysis of fuel element criticality for the in-tank storage racks mounted on the inside surface of the tank at an elevation similar to the core. Detailed analysis shows that the fuel stored as such cannot violate the subcritical value cited in Specification a under normal or accident conditions. Since the fuel is located on a spacing greater than spacing in the core and the elevation is the same as the core, cooling of the fuel is ensured.

The NRC staff finds that TS 5.4, Specifications a and b, for fuel storage, are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 5.4 is, therefore, acceptable.

5.6 Administrative Controls

TS 6.0, Administrative Controls, provides requirements for the conduct of operations for RRR. The administrative controls presented in TS 6.0 include responsibilities, facility organization, staff qualifications, training, the safety committee, operational review and audits, procedures, required actions, and 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 TS are based on these standards. In some cases, the wording that the licensee proposed was not identical to that given in NUREG–1537 and ANSI/ANS-15.1-2007. However, this review considered these cases and determines 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, TS, and federal regulations.

The NRC staff finds that the organization described in TS 6.1 is consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG–1537. On this basis, the NRC staff concludes that TS 6.1 is, therefore, acceptable.

5.6.1.1 TS 6.1.1 Structure

TS 6.1.1 states the following:

> The reactor administration shall be as shown in Figure 1. The Levels refer to ANSI/ANS-15.4-1988; R1999.

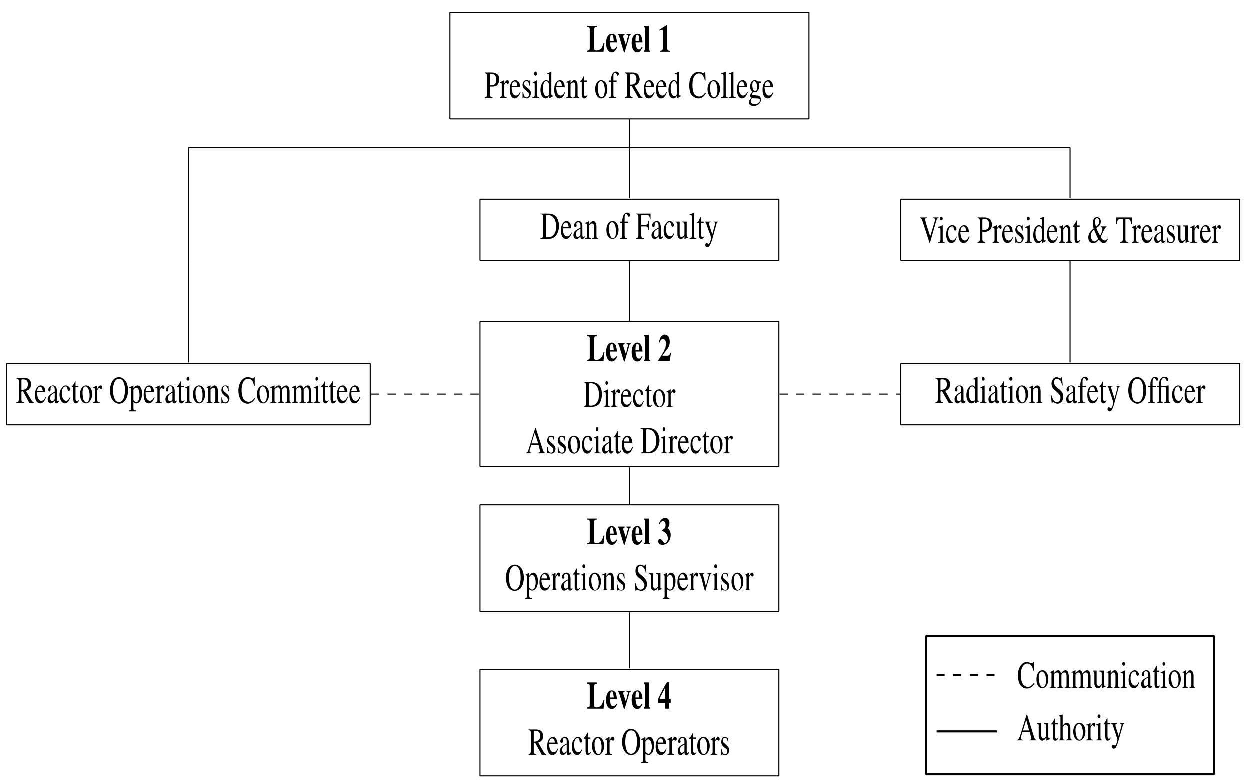

Figure 1 Administrative Structure

The NRC staff finds that the organizational structure described in TS 6.1.1 is consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG–1537. On this basis, the NRC staff concludes that TS 6.1.1 is, therefore, acceptable.

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. President (Level 1): The President of Reed College is responsible for the facility license and represents Reed College.

b. Director and Associate Director (Level 2): The Director reports to the President of Reed College via the Dean of the Faculty and is accountable for ensuring that all regulatory requirements, including implementation, are in accordance with all requirements of the NRC and the Code of Federal Regulations. The Associate Director reports to the Director and is responsible for guidance, oversight, and technical support of reactor operations.

c. Operations Supervisor (Level 3): The Operations Supervisor reports to the Associate Director and Director and is responsible for directing the activities of the reactor staff and for the day-to-day operation and maintenance of the reactor.

d. Reactor Operators and Senior Reactor Operators (Level 4): The Reactor Operators (RO) and Senior Reactor Operators (SRO) report to the Operations Supervisor, Associate Director, and the Director, and are primarily involved in the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor-related equipment.

e. Radiation Safety Officer: The Radiation Safety Officer reports to the President of Reed College via the Vice President and Treasurer and is responsible for directing health physics activities including implementation of the radiation safety program. The Radiation Safety Officer shall communicate with the Reactor Director regarding health physics issues.

The NRC staff finds that the organizational responsibilities delineated in TS 6.1.2 are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.1.2 is, therefore, acceptable.

5.6.1.3 TS 6.1.3 Staffing

TS 6.1.3 states the following:

a. The minimum staffing when the reactor is operating shall be:

   1. A licensed reactor operator in the control room;

   2. A second person present in the reactor facility able to scram the reactor and summon help;

   3. If neither of these two individuals is an SRO, a designated SRO shall be readily available on call. “Readily available on call” means an individual who:

a) has been specifically designated and the designation known to the operator on duty,

b) can be contacted quickly by the operator on duty, and

c) is capable of getting to the reactor facility within 15 minutes.

b. 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:

1. Reactor Director;

2. Reactor Associate Director;

3. Operations Supervisor;

4. Radiation Safety Officer; and

5. At least one other person who is a licensed SRO.

c. Events which require the presence of an SRO in the facility shall include:

1. Initial start-up and approach to power of the day or following significant changes (> $0.25) to the core;

2. All fuel or control rod relocations in the reactor core;

3. Maintenance on any reactor safety system;

4. Recovery from unscheduled reactor scram or significant power reduction; and

5. Relocation of any in-core experiment or irradiation facility with a reactivity worth greater than one dollar.

TS 6.1.3, Specification a, describes the minimum staffing necessary to safely operate the RRR. 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 b, describes the organization of the facility and the requirement for establishing formal responsibilities and authorities for the operating staff.

TS 6.1.3, Specification c, requires a senior reactor operator 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 significant 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 in accordance with the requirements of 10 CFR 50.54(k) and 10 CFR 50.54(m) and consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.1.3 is, therefore, acceptable.

5.6.1.4 TS 6.1.4 Selection and Training Personnel

TS 6.1.4 states the following:

> The selection, training, and requalification of 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 established the 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. 29), as guidance for selecting and training personnel. The NRC staff finds that TS 6.1.4 is consistent with the guidance provided in NUREG–1537. On this basis, the NRC staff concludes that TS 6.1.4 is, therefore, acceptable.

5.6.2 TS 6.2 Review and Audit

TS 6.2 states the following:

> The Reactor Operations Committee (ROC) shall have primary responsibility for review and audit of the safety aspects of reactor facility operations, and to assure that the facility is operated in a manner consistent with public safety and within the conditions specified in the facility license. Minutes, findings, or reports of the ROC shall be presented to the President (Level 1) and the Director (Level 2) within ninety days of completion.

The function of the ROC, as outlined in TS 6.2, is consistent with the guidelines of NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.2 is, therefore, acceptable.

5.6.2.1 TS 6.2.1 ROC Composition and Qualifications

TS 6.2.1 states the following:

> The ROC shall have at a minimum 3 members, at least two of whom are knowledgeable in fields that relate to physics and nuclear safety. The Dean of the Faculty, the Reactor Director, and the campus Radiation Safety Officer shall be voting members. Additional voting members shall be added at the President's discretion.

The composition and qualifications for the ROC conform to the recommendations of NUREG-1537 and ANSI/ANS-15.1-2007, Section 6.2.1. On this basis, the NRC staff concludes that TS 6.2.1 is, therefore, acceptable.

5.6.2.2 TS 6.2.2 ROC Rules

TS 6.2.2 states the following:

> The operation of the ROC shall be in accordance with written procedures including provisions for:
>
> a. Meeting frequency: not less than once per calendar year.
>
> b. Quorums: a group consisting of at least half of the voting members, of which the operating staff (i.e., the director and anyone who reports to that person) does not constitute a majority.
>
> c. Use of subcommittees.
>
> d. Review, approval, and dissemination of minutes.

TS 6.2.2 establishes the ROC meeting frequency, rules, and the committee charter. The NRC staff finds that TS 6.2.2 is consistent with the guidance in NUREG–1537 and ANSI/ANS–15.1–2007. On this basis, the NRC staff concludes that TS 6.2.2 is, therefore, is acceptable.

TS 6.2.3 ROC Review Function

TS 6.2.3 states the following:

> The responsibilities of the ROC, or designated subcommittee thereof, include, but are not limited to, the following:
>
> a. Review changes made under 10 CFR 50.59;
>
> b. Review new procedures and substantive changes to existing procedures;
>
> c. Review proposed changes to the TS or license;
>
> d. Review violations of TS, license, or violations of internal procedures or instructions having safety significance;
>
> e. Review operating abnormalities having safety significance;
>
> f. Review events from reports required in Section 6.6.1 and 6.7.2 of these TS;
>
> g. Review and approve new experiments under Section 6.5 of these TS; and
>
> h. Review audit reports.

TS 6.2.3 establishes the ROC review functions. The NRC staff finds TS 6.2.3 consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.2.3 is, therefore, acceptable.

5.6.2.3 TS 6.2.4 ROC Audit Function

TS 6.2.4 states the following:

> The ROC, or a subcommittee thereof, shall audit reactor operations at least annually. The annual audit shall include at least the following:
>
> a. Facility operations for conformance to these TS and applicable license conditions;
>
> b. The requalification program for the operating staff;
>
> c. The results of action taken to correct deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety; and
>
> d. The Emergency Plan and implementing procedures.

TS 6.2.4 establishes the ROC audit functions. The NRC staff finds TS 6.2.4 consistent with NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.2.4 is, therefore, acceptable.

5.6.3 TS 6.3 Radiation Safety

See the evaluation in Section 3.1.2 of this report.

5.6.4 TS 6.4 Procedures

TS 6.4 states the following:

> Written operating procedures shall be adequate to ensure the safe operation of the reactor, but shall not preclude the use of independent judgment and action if the situation requires. Operating procedures shall be in effect for the following:
>
> a. Startup, operation, and shutdown of the reactor;
>
> b. Fuel loading, unloading, and movement within the reactor;
>
> c. Maintenance of major components of systems that could have an effect on reactor safety;
>
> d. Surveillance checks, calibrations, and inspections required by the TS or those that have an effect on reactor safety;
>
> e. Radiation protection;
>
> f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;

g. Implementation of required plans, such as the Emergency and Security Plans; and

h. Use, receipt, and transfer of byproduct material held under the reactor license.

Substantive changes to the above procedures shall be made only after review and approval by the ROC. Non-substantive changes shall be reviewed and approved prior to implementation by the Director or Associate Director.

Temporary deviations from the procedures may be made by the responsible SRO when the procedure contains errors or in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported by the next working day to the Director or Associate Director.

TS 6.4, Specifications a through h, establish operational procedures for the RRR. The NRC staff finds that TS 6.4 is consistent with the guidance of ANSI/ANS-15.1-2007 and NUREG-1537. On this basis, the NRC staff concludes that TS 6.4 is, therefore, acceptable.

5.6.5 TS 6.5 Experiment Review and Approval

TS 6.5 states the following:

a. Approved experiments shall be carried out in accordance with established and approved procedures.

b. All new experiments or classes of experiments shall be reviewed and approved by the ROC.

c. Substantive changes to previously approved experiments shall be made only after review by the ROC and approval in writing by the Director or Associate Director.

d. Minor changes that do not significantly alter the experiment may be approved by the Operations Supervisor, Associate Director, or Director.

TS 6.5, Specifications a through d, require review and approval of different types of experiments before being performed at the RRR and specify the extent of the analysis submitted for review. TS 6.5 helps ensure acceptable management control over experiments. The NRC staff finds that TS 6.5 is consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.5 is, therefore, 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 is exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

b. An immediate notification of the occurrence shall be made to the Director, the Chair of the ROC, the NRC, and the President of Reed College.

c. A report shall be prepared and reviewed by the ROC. The report shall describe the following:

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

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

    3. Corrective action to be taken to prevent recurrence.

TS 6.6.1, Specifications a through c, require the facility to shut down in the event that a safety limit is exceeded. The facility may not resume operation without authorization from the NRC. The violation must also 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.3 specifies the corrective actions to be taken to prevent recurrence. The NRC staff finds that the actions the licensee proposes are consistent with the guidance of ANSI/ANS-15.1-2007 and NUREG–1537. On this basis, the NRC staff concludes that TS 6.6.1 is, therefore, acceptable.

5.6.6.2 TS 6.6.2 Action To Be Taken in the Event of an Occurrence of the Type Identified in 6.7.2 Other Than a Safety Limit Violation

TS 6.6.2 states the following:

> For all events that are required by regulations or TS to be reported to the NRC within 24 hours under Section 6.7.2, except a safety limit violation, the following actions shall be taken:
>
> a. The reactor shall be shut down and the Director or Associate Director and ROC chair notified;
>
> b. Operations shall not resume unless authorized by the Director or Associate Director;
>
> c. The ROC shall review the occurrence at or before their next scheduled meeting; and
>
> d. A report shall be submitted to the NRC in accordance with TS 6.7.2.

TS 6.6.2 requires the facility to shut down in case 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 within 24 hours 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 of ANSI/ANS-15.1-2007 and NUREG–1537. On this basis, the NRC staff concludes that TS 6.6.2 is, therefore, acceptable.

5.6.7 TS 6.7 Reports

TS 6.7.1 Annual Operating Report

TS 6.7.1 states the following:

> An annual report shall be created and submitted by the Director to the NRC by November 1 of each year consisting of:
>
> a. A brief summary of operating experience including the energy produced by the reactor;
>
> b. The number of unscheduled shutdowns, including reasons therefor;
>
> c. A tabulation of major preventative and corrective maintenance operations having safety significance;
>
> d. 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;
>
> e. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to 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 percent of the concentration allowed or recommended, a statement to this effect is sufficient;
>
> f. A summarized result of environmental surveys performed outside the facility; and
>
> g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed.

The NRC staff finds that TS 6.7.1, Specifications a through g, annual operating report requirements, are consistent with guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.7.1 is, therefore, acceptable.

5.6.7.1 TS 6.7.2 Special Reports

TS 6.7.2 states the following:

> In addition to the requirements of applicable regulations, and in no way substituting therefore, the Director shall report to the NRC as follows:
>
> a. A report not later than the following working day by telephone and confirmed in writing by facsimile to the NRC Operations Center, to be followed by a written report

that describes the circumstances of the event within 14 days to the NRC Document Control Desk of any of the following:

1. Violation of the safety limit;

2. Release of radioactivity from the site above allowed limits;

3. Operation with actual safety system settings from required systems less conservative than the limiting safety system setting;

4. Operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in Sec. 3;

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

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

7. Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable; or

8. An observed inadequacy in the implementation of 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.

b. A report within 30 days in writing to the NRC Document Control Desk of:

   1. Permanent changes in the facility organization involving Level 1–2 personnel; or

   2. Significant changes in the transient or accident analyses as described in the Safety Analysis Report.

The NRC staff finds that TS 6.7.2, Specifications a and b, special report requirements, are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.7.2 is, therefore, acceptable.

5.6.8 TS 6.8 Records

5.6.8.1 TS 6.8.1 Records To Be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years

TS 6.8.1 states the following:

a. Normal reactor operation;
b. Principal maintenance activities;
c. Reportable occurrences;
d. Surveillance activities required by the TS;

e. Reactor facility radiation and contamination surveys;
f. Experiments performed with the reactor;
g. Fuel inventories, receipts, and shipments;
h. Approved changes to the operating procedures; and
i. ROC meetings and audit reports.

The NRC staff finds TS 6.8.1, Specifications a through i, record requirements, are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.8.1 is, therefore, acceptable.

5.6.8.2 TS 6.8.2 Records To Be Retained for the Duration of a Requalification Cycle

TS 6.8.2 states the following:

> Records of retraining and requalification of licensed reactor operators and senior reactor operators shall be retained at all times the individual is employed or until the certification is renewed. For the purpose of this technical specification, a certification is an NRCissued operator license.

The NRC staff finds TS 6.8.2, records retention requirements, consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.8.2 is, therefore, acceptable.

5.6.8.3 TS 6.8.3 Records To Be Retained for the Lifetime of the Reactor Facility

TS 6.8.3 states the following:

a. Gaseous and liquid radioactive effluents released to the environs;
b. Offsite environmental monitoring surveys;
c. Radiation exposures for all personnel monitored;
d. Drawings of the reactor facility; and
e. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

The NRC staff finds TS 6.8.3, Specifications a through e, lifetime record retention requirements, are consistent with the guidance in NUREG–1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.8.3 is, therefore, acceptable.

5.7 Technical Specifications Conclusions

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

- To satisfy the requirements of 10 CFR 50.36(a), RRR provided proposed TS with the application for license renewal. As required by the regulations, the proposed TS included their appropriate summary bases.

- The RRR 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 license will include the TS. To satisfy the requirements of 10 CFR 50.36(b), RRR provided TS derived from analyses in the RRR SAR.

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

- The TS acceptably implement the recommendations of NUREG–1537, Part 1, and ANSI/ANS-15.1-2007 by using definitions that are acceptable.

- The TS contain LCOs 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 TS contain design features that satisfy the requirements of 10 CFR 50.36(c)(4).

- The TS contain administrative controls that satisfy the requirements of 10 CFR 50.36(c)(5). The RRR'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 RRR TS are acceptable and concludes that normal operation of the RRR within the limits of the TS 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 RRR TS provide reasonable assurance that the facility will be operated as analyzed in the RRR SAR, and that adherence to the TS will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 4, "Accident Analysis," of this SER.

6. CONCLUSIONS

On the basis of its evaluation of the application, as discussed in the previous chapters of this SER, the following conclusions are in order:

- The application for license renewal dated August 29, 2007, as supplemented on January 26, July 30, and October 12, 2010, and May 20, August 3, and December 12, 2011, and January 27, and March 26, 2012, 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 the 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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