Safety Evaluation Report

Renewal of the Facility Operating License for the Armed Forces Radiobiology Research Institute TRIGA Reactor

License No. R-84
Docket No. 50-170

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report (SER) 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 filed by the Armed Forces Radiobiology Research Institute (AFRRI, the licensee) for a 20-year renewal of the Facility Operating License No. R-84 in order to continue to operate the AFRRI Training, Research, Isotopes, General Atomics (TRIGA) reactor. In its safety review, the NRC staff considered information submitted by the licensee, its past operating history recorded in the licensee's annual operating reports to the NRC, inspection reports prepared by NRC staff and firsthand observations. On the basis of its safety and environmental review, the NRC staff concludes that the licensee 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, ACRONYMS, AND SYMBOLS

$	dollar (of reactivity)
% Δk/k	excess reactivity in percent
10 CFR	Title 10 of the *Code of Federal Regulations*
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
AFRRI	Armed Forces Radiobiology Research Institute
ALARA	as low as reasonably achievable
Am-Be	americium-beryllium
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar	argon
C	Celsius
cm	centimeters
CAM	continuous air monitor
Ci	curie
DAC	derived air concentration
DIF3D	Diffusion Theory 3 Dimensional
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
EP	emergency plan
F	Fahrenheit
ft	feet
FFCR	fuel-follower control rod
FR	*Federal Register*
FY	fiscal year
GA	General Atomics
in	inches
I	iodine
IRs	Insepctions reports
IFE	instrumented fuel element

ISG	interim staff guidance
kW	kilowatt
kWh	Kilowatt-hour
kWt	kilowatt thermal
LCO	limiting condition of operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LSSS	limiting safety system setting
μs	microsecond
MCNP	Monte Carlo N-Particle Transport
MCNPX	Monte Carlo N-Particle eXtended
MHA	maximum hypothetical accident
μmhos/cm	micromhos per centimeter
m	meters
mg	milligram
mrem	millirem
MW	megawatt
MWh	megawatt-hour
MWt	megawatt thermal
n/cm^2-s	newton per cubic centimeter per second
N	nitrogen
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
pCi/ml	picocurie per milliliter
PDR	Public Document Room
pH	hydrogen-ion concentration
PSP	physical security plan
RAI	request for additional information
RAM	remote area monitor
RELAP5	Reactor Excursion and Leak Analysis Program 5
RFD	reactor facility director
RG	regulatory guide
RRFSS	Reactor and Radiation Facilities Safety Subcommittee
RSO	radiation safety officer

RWP	rod withdrawal prevent
rem	roentgen equivalent man
SL	safety limit
SAR	safety analysis report
SER	safety evaluation report
SGM	stack gas monitor
SNM	special nuclear material
SOI	statement of intent
SRM	staff requirements memorandum
SRO	senior reactor operator
SS	stainless-steel
TEDE	total effective dose equivalent
TRIGA	Training, Research, Isotopes, General Atomics
TS	technical specification
U-Zr	uranium-zirconium
$U\text{-}ZrH_x$	uranium-zirconium hydride
$U\text{-}Zr_x$	uranium-zirconium
W	watt
w%	weight-percent

1. INTRODUCTION

1.1 Overview

By letter dated June 24, 2004 (Ref. 1), as supplemented by letters dated March 4 (Ref. 2), August 13 (Ref. 3), September 27 (Ref. 4), October 21 (Ref. 5), and December 15, 2010 (Ref. 6); February 7 (Ref. 7), June 20 (Ref. 8), September 6 (Ref. 9), October 20 (Ref. 10), and November 28, 2011 (Refs. 11 and 80); January 17 (Ref. 12), April 20 (Ref. 13), and September 21, 2012 (Ref. 14); June 28 (Ref. 15) and August 27, 2013 (Ref. 16); December 4, 2014 (Ref. 17); March 30, 2015 (Ref. 18); and February 9 (Ref. 19), February 29 (Ref. 20), August 5 (Ref. 21), September 12 (Refs. 76 and 77) , September 21 (Ref. 78), September 26 (Ref. 79), September 27 (Ref. 83), September 30 (Ref. 84), and November 16, 2016 (Ref. 36), the Armed Forces Radiobiology Research Institute (AFRRI, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for a 20-year renewal of the Class 104c Facility Operating License No. R-84, Docket No. 50-170, for the AFRRI Training, Research, Isotopes, General Atomics (TRIGA) reactor. A Notice of Opportunity for Hearing was published in the *Federal Register* (FR) on November 15, 2012 (77 FR 68155).

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.51(a) states, in part, that "[e]ach license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from date of issuance." AFRRI holds the Facility Operating License No. R-84 originally issued on June 26, 1962, by the Atomic Energy Commission. The NRC reissued the AFRRI facility operating license on August 1, 1984 (License Amendment No. 18) (Ref. 22). By letter dated February 28, 2000 (Ref. 23), AFRRI asked for an extension of the expiration date from November 8, 2000, to August 1, 2004, to make the license term 20 years from the issuance date of the previous license renewal. License Amendment No. 23, dated September 5, 2000 (Ref. 24), authorized the requested extension for the license to expire on August 1, 2004. A renewal would authorize continued operation of the AFRRI TRIGA reactor facility for an additional 20 years from date of issuance. Because AFRRI filed the request for license renewal in a timely manner, and until the NRC staff completes action on the renewal request, the licensee is permitted to continue operation of the AFRRI facility under the terms and conditions of the existing license in accordance with 10 CFR 2.109, "Effect of timely renewal application."

The NRC staff conducted its review for renewing the AFRRI operating license based on information in the license renewal application and in supporting supplements and licensee responses to NRC staff requests for additional information (RAIs). Specifically, the renewal application included the safety analysis report (SAR), the environmental report, technical specifications (TSs), the reactor emergency plan (EP), the reactor physical security plan (PSP), the reactor operator requalification program, financial qualifications, decommissioning information, and a statement describing proposed changes to the EP, PSP, and the requalification plan. As part of the review, the NRC staff also reviewed AFRRI's annual operating reports of the facility operation for the years 2008 through 2015 (Ref. 25) and NRC inspection reports (IRs) for the years 2008 through 2016 (Ref. 26). The NRC staff issued RAIs in letters dated June 23 (Ref. 27), July 19 (Ref. 28), and September 13, 2010 (Ref. 29); March 24, 2011 (Ref. 30); May 3, 2013 (Ref. 31); May 16, 2014 (Ref. 32); November 2, 2015 (Ref. 33); and June 17 (Ref. 34), and September 15, 2016 (Ref. 81). The NRC staff conducted site visits at the facility to observe facility conditions.

The licensee responded to the NRC staff's RAIs by letters dated March 4 (Ref. 2), August 13 (Ref. 3), September 27 (Ref. 4), October 21 (Ref. 5), and December 15, 2010 (Ref. 6); February 7 (Ref. 7), June 20 (Ref. 8), September 6 (Ref. 9), October 20 (Ref. 10), and November 28, 2011 (Ref. 11); January 17 (Ref. 12), April 20 (Ref. 13), and September 21, 2012 (Ref. 14); June 28 (Ref. 15) and August 27, 2013 (Ref. 16); December 4, 2014 (Ref. 17); March 30, 2015 (Ref. 18); and February 9 (Ref. 19), February 29 (Ref. 20), August 5, (Ref. 21), September 12 (Refs. 76 and 77) , September 21 (Ref. 78), September 26 (Ref. 79), September 27 (Ref. 83), and September 30, 2016 (Ref. 84).

With the exception of the PSP, and portions of the SAR, RAI responses, and EP that contain security-related information, material pertaining to this review may be examined or copied, for a fee, at the NRC's Public Document Room (PDR), Room 01-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. The NRC maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Publicly available documents related to this license renewal may be accessed online through the NRC's Public Library, ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if you experience problems accessing the documents in ADAMS, contact the NRC PDR staff by telephone at 1-800-397-4209 or 301-415-4737, or send an e-mail to the PDR at PDR.Resource@nrc.gov. The PSP and material containing security-related information are protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements," and 10 CFR 2.390(d). Since portions of the SAR, RAI responses, and the EP contain security-related information and are protected from public disclosure, redacted versions are provided to the public in ADAMS.

Chapter 7, "References," of this safety evaluation report contains the dates and associated ADAMS accession numbers of the licensee's renewal application and related supplements.

In conducting its safety review, the NRC staff evaluated the facility against the requirements in the regulations, including 10 CFR Part 20, "Standards for Protection against Radiation," 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," and 10 CFR Part 73, "Physical Protection of Plants and Materials;" the recommendations of applicable regulatory guides (RG); and relevant accepted industry standards, such as those of the American National Standards Institute/American Nuclear Society (ANSI/ANS)-15 series. The NRC staff also considered the recommendations in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," issued February 1996, and Part 2, "Standard Review Plan and Acceptance Criteria," issued February 1996 (Ref. 39). For research reactors, the NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20.

In SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," dated October 24, 2008 (Ref. 40), the NRC staff described plans to streamline the review of license renewal applications for research and test reactors. The Commission issued astaff requirements memorandum (SRM) for SECY-08-0161, "Staff Requirements—SECY-08-0161 – Review of Research and Test Reactor License Renewal Applications," dated March 26, 2009 (Ref. 41). The SRM 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 directs the NRC staff to implement a graded approach whose scope is commensurate with the risk posed by each facility. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 to SECY-08-0161. In the alternative safety review approach, used in this SER, the NRC staff considered the results of past NRC staff reviews. A basic requirement, as contained in the SRM, is that licensees must be in compliance with applicable regulatory requirements.

The NRC staff developed RTR Interim Staff Guidance (ISG) ISG-2009-001, "Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors," (Ref. 42), to assist in the review of license renewal applications. The streamlined review process is a graded approach based on licensed power levels. Under the streamlined review process, the facilities are divided into two tiers. Facilities with a licensed power level of 2 megawatt thermal (MWt) and greater undergo a full review using NUREG-1537. Facilities with a licensed power of less than 2 MWt undergo a focused review that centers on the most safety-significant aspects of the renewal application and relies on past NRC reviews for certain safety findings. The NRC issued a draft of the ISG for public comment and considered public comments in its development of the final ISG.

The NRC staff conducted its review of the license renewal application for the AFRRI research reactor using the guidance in the final ISG, dated October 15, 2009 (Ref. 42). Since the licensed power level of the AFRRI research reactor is less than 2 MWt, the NRC staff performed a focused review of the licensee's license renewal application. Specifically, the NRC focused on reactor design and operation, accident analysis, TSs, radiation protection, waste management programs, financial requirements, environmental assessment, physical security, emergency plans, operator requalification, and changes to the facility made after submission of the application.

As part of its review of the license renewal application, the NRC staff reviewed the PSP, "Armed Forces Radiobiology Research Institute Reactor Facility Physical Security Plan," submitted with the license renewal application. The NRC staff issued RAIs to the licensee in a letter dated February 23, 2015 (Ref. 43), and the licensee responded by letters dated June 25 (Ref. 44) and November 4, 2015 (Ref. 45), including a revised AFRRI reactor facility PSP. The NRC staff reviewed the revised AFRRI reactor facility PSP, found that it meets the applicable regulations, and, based on that finding, concludes that the revised AFRRI reactor facility PSP, dated November 4, 2015, is acceptable. The licensee maintains the program to provide for the physical protection of the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73. The licensee can make changes to the PSP in accordance with 10 CFR 50.54(p), as long as those changes do not decrease the effectiveness of the plan. In addition, the NRC staff routinely inspects the licensee's compliance with the requirements of the P and the applicable regulations. The NRC staff's review of the AFRRI reactor facility for the past several years identified no violations of the security plan requirements.

The NRC staff reviewed the "AFRRI Reactor Facility Emergency Plan" (Ref. 1), submitted with the license renewal application, using the following regulations and guidance:

- 10 CFR Part 50.34(b)(6)(v)
- RG 2.6, "Emergency Planning for Research and Test Reactors," Revision 1, issued March 1983 (Ref. 46)
- ANSI/ANS-15.16-2015, "Emergency Planning for Research Reactors" (Ref. 47)

- NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," issued October 1983 (Ref. 48)

- NRC Information Notice 97-34, "Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20" (Ref. 49)

- NRC Information Notice 92-79, "Non-Power Reactor Emergency Event Response," dated December 1, 1992 (Ref. 50)

The NRC staff's review determined that the licensee maintains an EP in accordance with the requirements in 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 continue to be prepared to assess and respond to emergency events. The NRC staff routinely inspects the licensee's compliance with the requirements of the emergency plan, and no violations have been identified for the past several years. Based on its review, the NRC staff concludes that the EP complies with the regulations and meets applicable guidance.

The NRC staff reviewed the AFRRI Reactor Facility Reactor Operator Requalification Program, submitted as part of the license renewal application (Ref. 1), and amended on December 4, 2014 (Ref. 17), and September 12, 2016 (Ref. 77), in response to an NRC staff request for additional information dated May 16, 2014 (Ref. 32). The updated requalification plan eliminates the exemption from operator requalification exams for the Reactor Facility Director, bringing the plan into compliance with 10 CFR Part 55, "Operators' Licenses." The NRC staff finds that the updated AFRRI Reactor Facility Reactor Operator Requalification Program dated September 12, 2016, meets the applicable requirements of 10 CFR Part 55 and is consistent with guidance contained in industry standard ANSI/ANS-15.4-2016, "Selection and Training of Personnel for Research Reactors" (Ref. 51). Based on the above, the NRC staff concludes that the AFRRI Reactor Facility Reactor Operator Requalification Program is acceptable.

The NRC staff separately evaluated the environmental impacts of the renewal of the license for the AFRRI reactor in accordance with 10 CFR Part 51. The NRC staff published an Environmental Assessment and Finding of No Significant Impact in the *Federal Register* on November 25, 2016 (81 FR 85268), (Ref. 82), which concluded that renewal of the AFRRI operating license will not have a significant effect on the quality of the human environment.

The purpose of this SER is to summarize the findings resulting from the safety review of the AFRRI TRIGA reactor and to delineate the technical details considered in evaluating the radiological safety aspects of continued operation. The SER provides the basis for renewing the license for operation of the AFRRI TRIGA reactor at steady-state thermal power levels up to and including 1.1 MWt and short-duration power pulses with reactivity insertions up to $3.50 (2.45 percent delta k/k (% Δk/k, excess reactivity in percent)).

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1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff's evaluation considered the information submitted by the licensee, including past operating history recorded in the licensee's annual operating reports to the NRC and IRs prepared by the NRC staff. On the basis of this evaluation and resolution of the principal issues reviewed for the AFRRI TRIGA reactor, the NRC staff concludes the following:

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

- The facility will continue to be useful in the conduct of research and development activities.

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA), emphasizing those accidents that could lead to a loss of integrity of fuel element cladding and a release of fission products. The licensee performed analyses, using conservative assumptions, of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses to the facility staff and members of the public, would not exceed 10 CFR Part 20 dose limits.

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

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

- The licensee's TSs, which provide limits controlling operation of the facility, provide reasonable 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 AFRRI license renewal SAR (Ref. 1), as supplemented, and the TSs will continue to help ensure that no significant degradation of safety-related equipment will occur.

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

- The licensee maintains a PSP for the facility and its SNM in accordance with the requirements of 10 CFR Part 73, which reasonably ensures that the licensee will continue to provide the physical protection of the facility and its SNM.

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

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

On the basis of these findings, the NRC staff concludes that there is reasonable assurance that AFRRI can continue to operate its TRIGA reactor in accordance with the Atomic Energy Act of 1954, as amended (AEA); NRC regulations; and Renewed Facility Operating License No. R-84 without endangering public health and safety, 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 of the Facility

The reactor is located on the grounds of the Naval Support Activity Bethesda military installation, Bethesda, Montgomery County, MD. AFRRI is a triservice military organization under the Uniformed Services University of the Health Sciences. Its mission is to conduct scientific research in the field of radiobiology and related radiation research in support of the U.S. Department of Defense (DOD). The licensee is licensed to operate the reactor at up to 1.1 MWt steady-state thermal power with pulsing capability using reactivity insertions up to $3.50 (2.45 % $\Delta k/k$).

The AFRRI TRIGA Mark F is a heterogeneous pool-type reactor cooled by the natural convection of light water. The reactor is moderated primarily by zirconium hydride and, to a lesser extent, by the cooling water and is reflected by pool water and a graphite moderator as end plugs of each fuel element. The core and control system are suspended on a bridge mounted on rails that allows the core to be moved across the pool to face exposure rooms on both sides of the tank.

The current reactor core contains 85 fuel elements that contain uranium enriched to less than 20 percent with a homogeneously mixed zirconium-hydride moderator. The reactor is controlled by four stainless steel-clad borated graphite control rods that are suspended from electromagnets, which are part of the control system on the bridge.

Three of the four control rods are moved in and out of the reactor core by individual mechanical drives. The forth is a transient control rod with a combination of pneumatic-electromechanical drive that may be used either as a control rod or as a transient rod to generate a neutron pulse in pulse-mode operation. The three standard control rods have fuel-followers, whereas the transient rod has a void-follower.

Ionization chambers are installed above the core to sense neutron and gamma-ray levels. The control room in which the control console is located is adjacent to the reactor room to allow the operator to observe the reactor room, reactor pool, and the top structures of the reactor through a large window. The control console contains the typical instrumentation and control equipment used at non-power reactors. The reactor is located within a reinforced concrete biological shield structure.

1.4 Shared Facilities and Equipment

The AFRRI reactor building is attached to a laboratory and support building that is used primarily for radiation biology and biomedical research. The complex shares common utilities, such as water, sewage, electricity, and natural gas, supplied by the Naval Support Activity Bethesda military installation. The reactor building has a dedicated ventilation system that exhausts air through an elevated stack on the roof and that can be isolated independently from other parts of the building complex.

1.5 Comparison with Similar Facilities

In Section 1.3.1 of the SAR, (Ref. 1) the licensee stated that a large number of TRIGA reactors built by GA, similar to the AFRRI TRIGA, have been operating for 30 years. This operating experience has demonstrated the inherent safety of the family of TRIGA reactors. TRIGA reactors exist in a variety of configurations and capabilities (Ref. 52). The reactor core is similar to most of the TRIGA reactor installations. The reactor operates at a power level similar to seven other NRC-licensed TRIGA research reactors (The Pennsylvania State University, Oregon State University, Texas A & M University, U.S. Geological Survey, University of Texas, University of Wisconsin and, Washington State University); however, the AFRRI reactor has a unique pool design that supports the use of two large exposure rooms for radiobiology research. The instruments and controls used in the AFRRI reactor facility are similar in principle to most NRC-licensed non-power reactors.

1.6 Summary of Operations

The AFRRI reactor facility is used to conduct radiobiology and related research and a range of irradiation services for the Department of Defense. As described in the 2015 annual operating report, that since 1967, the total energy produced by the core is 1,153,063.6 kilowatts, and the total number of pulses greater than $2.00 performed is 4,219. Total annual usage has varied over time depending on the schedule of the research programs that use the facility. According to AFRRI's annual operating reports for the years 2008 through 2015 (Ref. 25), the energy generated by the reactor core each year was 0 kilowatt-hour (kWh) in 2015, 4,496.6 kWh in 2014, 9,427.3 kWh in 2013, 6,021.6 kWh in 2012, 17,812.8 kWh in 2011, 34,206.7 kWh in 2010, 65,141.4 kWh in 2009, and 6,571.4 kWh in 2008.

Since 1967, the reactor has operated at a rate of about 26 MW-hours per year mostly in steady-state power operation. The reactor has been pulsed on average about 300 times per year, with most pulses corresponding to reactivity insertions of $2.00 or less without a fuel element failure (Ref. 1). Expectations for the upcoming license renewal period are to at least maintain or improve the current usage rate achieved to date.

This review considered AFRRI annual operating reports for the years 2008 through 2015 (Ref. 25), and the NRC IRs for the years 2008 through 2016 (Ref. 26). The annual operating report summaries did not indicate any significant degradation of fuel element integrity, control rod operability issues, or radiological exposure concerns. The fuel temperature and scram circuits required for operation are calibrated routinely. The NRC staff's review of the AFRRI IRs did not identify any violations.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982, 42 U.S.C.§10222(b)(1)(B), specifies that the NRC may require, as a precondition to issuing or renewing a facility operating license for a research or test reactor, that the licensee enter 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 (Ref. 53), R.L. Morgan of DOE informed H. Denton of the NRC that DOE had determined that universities and other Government agencies operating non-power reactors have entered into contracts with the DOE providing that DOE retains title to the fuel and that DOE is obligated to take the spent fuel and high-level waste for storage or reprocessing. The AFRRI reactor is a Federally owned and operated non-power reactor. The current SNM Interagency Agreement is No. 1037, Amendment 0003, valid from September 30, 2015, until September 29, 2018. This agreement provides that DOE will accept the fuel at cessation of operation. DOE renews these contracts prior to their expiration to ensure that the contract remains valid. By entering into such a contract and abiding by the contract with DOE, AFRRI has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

On June 26, 1962, the U.S. Atomic Energy Commission issued an operating license to AFRRI for a TRIGA reactor located on the grounds of the current Naval Support Activity Bethesda military installation. Facility Operating License No. R-84 initially authorized the AFRRI TRIGA reactor to operate at steady-state power levels up to 100 kilowatts thermal (kWt). The AFRRI TRIGA reactor was authorized for routine operation in three modes—(1) Mode I, up to a steady state of 100 kWt, (2) Mode II, between 100 kWt and 1 MWt for periods not to exceed 10 minutes for a maximum of 1 MWt-hour per day (square wave), and (3) Mode III, pulsed, with step reactivity insertions up to 1.9 % $\Delta k/k$. After operating the facility for several years, AFRRI (1) substituted an improved version of fuel for the original core, (2) modified some of the original control instrumentation, (3) increased the reactor primary cooling capacity, and (4) applied for a license amendment for authorization to operate the reactor at steady-state power levels up to and including 1 MWt. The NRC issued License Amendment No. 13 for that power level on August 29, 1968 (Ref. 54).

By letter dated August 14, 1970, AFRRI requested an extension of the expiration date of its initial operating license, which was due to expire on November 8, 1970. On November 4, 1970, the NRC staff amended Facility Operating License No. R-84 by extending the expiration date to November 8, 1980. AFRRI made a timely application dated October 3, 1980, requesting 20 years of continued reactor operations at a steady-state power level of 1 MWt. The NRC staff reissued the AFRRI facility license on August 1, 1984 (License Amendment No. 18) (Ref. 22), authorizing steady-state operation up to 1 MWt and step reactivity insertions up to 2.8% $\Delta k/k$ with an expiration date of November 8, 2000. License Amendment No. 19, dated July 23, 1990 (Ref. 56), approved the installation of a microprocessor-based instrument and control system. The NRC issued License Amendment No. 21 on October 8, 1991, authorizing the installation of fuel-follower control rods and increasing the steady-state power level up to 1.1 MWt (Ref. 55). By letter dated February 28, 2000 (Ref. 23), AFRRI asked for an extension of the expiration date from November 8, 2000, to August 1, 2004, to make the license term 20 years from the issuance date of the previous license renewal. License Amendment No. 23, dated September 5, 2000 (Ref. 24), authorized the requested extension years and specified a license expiration date of August 1, 2004.

The NRC staff's review of modifications made since 2008 without license amendment indicates that these modifications have been technological upgrades to instrumentation or minor changes to the existing design that either enhanced capabilities or improved reactor operations. All of the modifications were subject to evaluation under 10 CFR 50.59, "Changes, test and experiments," to ensure that there was no prior NRC approval required or impact on the safety of the AFRRI TRIGA reactor. The NRC staff also reviewed the licensee's annual operating reports for the years 2008 through 2015 (Ref. 25), and NRC IRs for the years 2008 through 2016 (Ref. 26) that documented these changes. The results of these reviews indicated that the changes were performed, as required, in accordance with the requirements of 10 CFR 50.59. AFRRI did not request any changes to its facility as part of this license renewal application.

1.9 Financial Considerations

1.9.1 Financial Ability To Operate the Facility

The regulation, 10 CFR 50.33(f) states:

> 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 in this chapter, the activities for which the permit or license is sought.

AFRRI does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Furthermore, 10 CFR 50.33(f)(2), states "[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 AFRRI must meet the financial qualifications requirements under 10 CFR 50.33(f) and that AFRRI is subject to a full financial qualifications review. AFRRI must demonstrate that it possesses or that it has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the renewed facility operating license. Pursuant to 10 CFR 50.33(f)(2), AFRRI 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 must indicate the sources of funds to cover those costs. This is consistent with the guidance provided in NUREG-1537 as it pertains to financial qualifications assurance.

By letter dated June 24, 2004 (Ref. 1), AFRRI submitted an initial license renewal application to the NRC to renew Amended Facility Operating License No. R-84 for its research reactor. In supplements to the application dated August 13, 2010 (Ref. 3), June 28, 2013 (Ref. 15), and February 9, 2016 (Ref. 19), AFRRI submitted its projected operating costs in 2013 for the AFRRI TRIGA reactor for each fiscal year (FY) 2013 through 2018. The projected operating costs for the AFRRI TRIGA reactor are estimated to range from $1.3 million in FY 2017 to $1.4 million in FY 2018. On February 9, 2016, the licensee stated that the projected operating costs have not changed for the period FY 2016 through FY 2021. According to AFRRI, its primary source of funding to cover the AFRRI TRIGA reactor operating costs will come from the congressional appropriations process, as further delegated within the DOD. AFRRI expects that this funding source will continue for FY 2016 through FY 2021. The NRC staff reviewed AFRRI's estimated operating costs and projected source of funds to cover those costs and finds them to be reasonable.

Based on its review, the NRC staff finds that AFRRI has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs for the period of the renewed facility operating license and has met the acceptance criteria on financial assurance for operations under NUREG-1537. Accordingly, the NRC staff finds that AFRRI has met the financial qualifications requirements in 10 CFR 50.33(f) and that it is financially qualified to engage in the proposed AFRRI TRIGA reactor activities.

AFRRI TRIGA reactor is currently licensed as a facility that is useful in research and development under Section 104.c of the AEA, 42 U.S.C. § 2234(c). The regulation, 10 CFR 50.21(c), provides for issuance of a license to a facility which is useful in the conduct of research development activities if no more than 50 percent of the annual cost of owning and operating the facility is devoted to production of materials, products, or the sale of services, other than research and development or education or training. SAR Section 1.2 states that the AFRRI TRIGA reactor facility is used to conduct research to support the U.S. Department of Defense. The research associated with the reactor typically involves radiobiolgical and related matters. Because 10 CFR 50.21(c) requires that the majority of AFRRI TRIGA reactor operating costs be funding by non-commerical uses and AFRRI, an entilty of the U.S. Department of Defense, is primarily funded by the U.S. Government, the NRC staff concludes that the AFRRI TRIGA reactor can be renewed as a Section 104.c license.

1.9.2 Financial Ability To Decommission the Facility

The NRC staff has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to ensure the adequate protection of public health and safety. The regulations in 10 CFR 50.33(k) require that an application for an operating license for a production or utilization facility must contain information to demonstrate how the applicant will provide reasonable assurance that funds will be available to decommission the facility.

Under 10 CFR 50.75(d), each non-power applicant for, or holder of, an operating license shall submit a decommissioning report that contains a cost estimate for decommissioning the facility, an indication of the funding methods that the applicant will use to ensure funding for decommissioning, and a description of the means to adjust the cost estimate and associated funding level periodically over the life of the facility. The regulation at 10 CFR 50.75(e)(1) specifies the acceptable methods for providing financial assurance for decommissioning. The staff used guidance in NUREG-1537 to complete its review of the AFRRI license renewal as it pertains to financial assurance for decommissioning.

In a letter dated, June 28, 2013 (Ref. 15), AFRRI updated and confirmed its decommissioning cost estimate to $14.831 million in 2011 dollars. The cost estimate summarized costs by labor, radioactive waste disposal, energy, and a 25-percent contingency factor. The licensee stated that the decommissioning cost estimates have not changed since the previous submittals. According to AFRRI, its updated decommissioning cost estimate for the TRIGA reactor was developed using a 1990 decommissioning cost study of the AFRRI TRIGA reactor as a basis. This study compared the costs of decommissioning a similar reactor facility, as described by Pacific Northwest Laboratory in NUREG/CR-1756, “Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors,” issued March 1982 (Ref. 57). The licensee stated that it will update its decommissioning cost estimate by (1) reviewing the assumptions for decommissioning, waste disposal, and shipment, (2) revalidating the tasks required for decommissioning, including the amount of labor and skill sets required, (3) updating the estimates with current and projected rates for energy, civilian labor, overhead, and direct costs, and (4) applying inflation using published U.S. Bureau of Labor Statistics consumer price index rates. The NRC staff has reviewed the information submitted by AFRRI on the decommissioning of the reactor facility and finds that the decommissioning approach and cost estimates submitted are reasonable. In the letters dated June 28, 2013, and February 9, 2016 (Ref. 19), AFRRI stated that the basis for the cost estimate had not changed since 2010 and that there were no changes to the means of adjusting the cost estimate and associated funding level periodically over the life of the facility.

AFRRI also stated that it will update its estimate at 5-year intervals using methodology proposed in 10 CFR 50.75(c)(2), based on factors in the most recent version of NUREG-1307, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities," Revision 15, issued January 2013 (Ref. 58), and the U.S. Bureau of Labor Statistics for labor, energy, and waste burial.

AFRRI has elected to use a statement of intent (SOI) to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv), for a Federal, State, or local government licensee. The SOI must contain or reference a cost estimate for decommissioning and must indicate that funds for decommissioning will be obtained when necessary.

By letters dated August 13, 2010 (Ref. 3), and February 9, 2016 (Ref. 19), AFRRI provided an SOI, stating, in part, that "funding will be sought from the [U.S.] Department of Defense in accordance with established programming and budgeting procedures." The decommissioning cost estimate is $14.831 million for the DECON option. In letters dated June 28, 2013 (Ref. 15), and February 9, 2016, AFRRI confirmed that there were no changes to the SOI and that decommissioning funding obligations of the AFRRI facility remain backed by the full faith and credit of the U.S. Government.

To support the SOI and AFRRI's qualifications to use an SOI, the application stated that AFRRI, an entity of the Uniformed Services University of the Health Sciences in the DOD, is part of the Federal Government and has included documentation that corroborates this statement. The application, as supplemented, also provides information supporting AFRRI's representation that the decommissioning funding obligations of AFRRI are backed by the full faith and credit of the U.S. Government. AFRRI also provided documentation verifying that William T. Bester, Brigadier General, U.S. Army (retired), and Acting President of the Uniformed Services University of the Health Sciences (the signatory of the SOI), is authorized to execute contracts on behalf of AFRRI.

The NRC staff reviewed AFRRI's information on decommissioning funding assurance and finds that (1) AFRRI is a Federal Government licensee under 10 CFR 50.75(e)(1)(iv), (2) the SOI is acceptable, (3) the decommissioning cost estimate and the costs for the decommissioning option are reasonable, and (4) AFRRI's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable. Therefore, AFRRI meets the regulatory requirements and the guidance provided in NUREG-1537. The NRC staff notes that any adjustment of the decommissioning cost estimate must incorporate, among other things, changes in costs resulting from the availability of disposal facilities and that AFRRI has an obligation under 10 CFR 50.9, "Completeness and accuracy of information," to update any changes in the projected cost, including changes in costs resulting from increased disposal options.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d of the AEA of 1954, as amended, 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 regulations in 10 CFR 50.38, "Ineligibility of certain applicants," contain language to implement this prohibition.

In addressing foreign ownership, control, and domination, the NRC staff considered guidance in the "Final Standard Review Plan on Foreign Ownership, Control and Domination," issued

June 1999 to determine whether the applicant is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC published this Standard Review Plan in the *FR* on September 28, 1999 (64 FR 52355) (Ref. 59).

According to the application, AFRRI is a Federal Government entity, within the DOD, and 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 AFRRI currently has an indemnity agreement with the Commission which continues in effect until all radioactive material has been removed from the site and the license is terminated. Therefore, the licensee will continue to be a party to the present indemnity agreement following issuance of the renewed facility operating license. Under 10 CFR 140.71, "Scope," AFRRI, as a Federal Government licensee, is not required to furnish nuclear liability insurance. The Commission will indemnify AFRRI for any claims that arise from a nuclear incident under the Price-Anderson Act and Section 170 of the AEA, as amended, and in accordance with the provisions under its indemnity agreement in 10 CFR 140.94, "Appendix D—Form of indemnity agreement with Federal agencies," up to $500 million. In addition, the licensee is not required to purchase property insurance under 10 CFR 50.54(w). The NRC staff concludes that AFRRI maintains an acceptable indemnity agreement which satisfies applicable requirements.

1.9.5 Financial Considerations Conclusions

As described above, 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 AFRRI TRIGA reactor and, when necessary, to shut down the facility and carry out decommissioning activities. In addition, the NRC staff concludes that there are no foreign ownership, control or domination issues or insurance issues that would preclude the issuance of a renewed facility operating license.

1.10 Facility Operating License Possession Limits and License Changes

The NRC staff reviewed the current AFRRI TRIGA reactor facility license and finds that the license contains conditions that control the receipt, possession, and use of byproduct material and SNM in accordance with 10 CFR Parts 30 and 70. However, the format and organization used in some license conditions (LCs) were not consistent with that in recently issued licenses. The NRC staff reformatted and reorganized the LCs to make them easier to read and understand.

The renewal of the Facility Operating License No. R-84 for the AFRRI TRIGA reactor authorizes the receipt, possession, use, but not the separation of 100 grams total of special nuclear and byproduct materials in connection with the operation of the facility and such material as may be produced by such operation. This change is consistent with the licensee's request in letter dated August 5, 2016. SNM consists of such material as the U-235 in the reactor fuel, SNM in the form of detectors, fission plates, foils, solutions, and SNM produced by operation of the reactor. Byproduct material consists of a 3-Ci sealed americium-beryllium neutron startup source and such material as may be produced by the operation of the reactor, including activation products produced by operation of the reactor in the fuel, experiments, and reactor structure. As is current practice, LCs were added by the NRC staff and reviewed by the licensee (Ref. 36). These added LCs were added to prevent the separation of SNM and to clarify the byproduct material possession requirements to allow the separation of only byproduct material produced in non-fueled experiments. The restricted area is defined in TS 5.1 and SAR Section 1.3.1, and all activities performed within this area fall under the jurisdiction of the TRIGA reactor license. The NRC inspection program has shown that the licensee has procedures and equipment to safely and securely handle licensed material within the restricted area. Based on its review as discussed above and the acceptable results of the NRC inspection program, the NRC staff concludes that the licensee has procedures and equipment in place to safely receive, possess, and use the materials authorized by the facility operating license.

In its responses to RAI dated August 5, 2016, the licensee states that it currently holds Uranium-235 and plutonium in the form of detectors, fission plates and foils. Much of the current inventory is aging and will need to be replaced. AFRRI will require no more than 100 grams in the foreseeable future. The NRC staff reviewed the license's responses and finds that the a total of 100 grams possession limit requested by the licensee is reasonable for facility operation and is, therefore, acceptable. No change was requested for licensed authorization for receiving, possessing, and using, in connection with the operation of the facility, 5.0 kilograms of source material. The NRC staff concludes that the changes to the license are reasonable and that the licensee's procedures and equipment are adequate to handle licensed material within the restricted area.

2. REACTOR DESCRIPTION

2.1 Summary Description

The Armed Forces Radiobiology Research Institute (AFRRI) has a Training, Research, Isotopes, General Atomics (TRIGA) Mark F reactor designed and manufactured by General Atomics (GA) that achieved criticality in 1962. It is a light-water, tank-type reactor that can operate in the steady-state mode with thermal power up to 1.1 megawatts thermal (MWt) or pulse mode with a step reactivity insertion up to \$3.50 (2.45 percent delta k/k (% Δk/k, excess reactivity in percent)), resulting in an average maximum pulse of less than 35 MWt-seconds (integrated power). The reactor core is cooled through the natural convection of the pool water. The reactor uses standard design stainless-steel clad GA fuel elements.

The AFRRI TRIGA reactor has a horizontally movable core that allows the use of a variety of experimental facilities, including two separate exposure rooms and a pneumatic transfer system. Extensive experience gained from similar designs used throughout the world has demonstrated the safety of TRIGA reactors. The TRIGA fuel is characterized by high fission product retention and the ability to withstand water quenching at temperatures as high as 1,150 degrees Celsius (°C) (2,102 degrees Fahrenheit (°F)). The safety of the fuel arises from the strongly prompt negative temperature coefficient characteristic of uranium-zirconium (U-Zr_x) fuel-moderator elements. As the fuel temperature rises, this coefficient immediately compensates for reactivity insertions. The AFRRI fuel temperature safety limit (SL) (TS 2.1) is specified not to exceed 1,000 °C (1,832 °F) under any conditions of operation. To ensure that this SL is not exceeded, the limiting safety system setting (LSSS) (TS 2.2) is established for steady-state or pulse operation for fuel temperature to be less than or equal to 600 °C, (1,112 °F) as measured in the instrumented fuel element (IFE), which is located in specific locations in the core. (Section 2.5.3 of this safety evaluation report (SER) discusses the SL and LSSS).

A series of GA and NRC reports discuss features of this type of reactor, such as reactor kinetic behavior (GA-7882, “Kinetic Behavior of TRIGA Reactors,” dated March 31, 1967 (Ref. 60)), fission product retention (NUREG-1282, “Safety Evaluation Report of High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors,” issued August 1987 (Ref. 61), and GA-4314, E-117-833, “The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel,” issued February 1980 (Ref. 62)), and accident analysis (NUREG/CR-2387, “Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors,” issued April 1982 (Ref. 63)).

2.2 Reactor Core

Table 2-1 presents the current AFRRI TRIGA reactor core configuration. The AFRRI TRIGA reactor cylindrical core can contain up to 87 standard TRIGA stainless-steel-clad fuel elements. The current core contains 85 fuel elements, including two IFEs. The AFRRI TRIGA reactor safety analysis report (SAR) states that, as fuel is consumed, the dry tube and water hole will be replaced with standard fuel elements.

Table 2-1 AFRRI Reactor Core Configuration

Core Element	Core 85-3 Configuration
Standard fuel elements	83
Instrumented fuel elements	2
Fuel-follower control rods (FFCRs)	3
Void-follower transient rod	1
Dry tube	1
Water hole	1
Total	91

The uranium-zirconium hydride (U-ZrH_x) fuel-matrix material is U-$ZrH_{1.7}$ enriched to less than 20-percent uranium-235 (U-235). The ratio of 1.7 is acceptable because AFRRI's SL is 1,000 °C (1,832 °F), and this accords with NUREG-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors" (Ref. 63) and NUREG-1282 (Ref. 61) guidance. The core, as allowed by the TS, contains four aluminum or stainless-steel-clad borated graphite control rods with air-, fuel-, or aluminum-followers; a startup source; and a guide tube. The core moderator comprises both water and the zirconium hydride in the fuel. The reactor core is reflected by graphite end plugs located in both ends of the fuel elements and by water at the periphery.

The cylindrical fuel elements and control rods are positioned in the core in five concentric rings surrounding the centrally located transient control rod, as shown in Figure 2-1. The fuel elements and the control rod guide tubes are positioned in the core by the top and bottom grid plates. The grid plates are attached to a cylindrical shroud that surrounds the core. The shim, safety, and regulating rods constitute the three standard control rods and are in positions D1, D7, and D13, whereas the transient rod is in position A1.

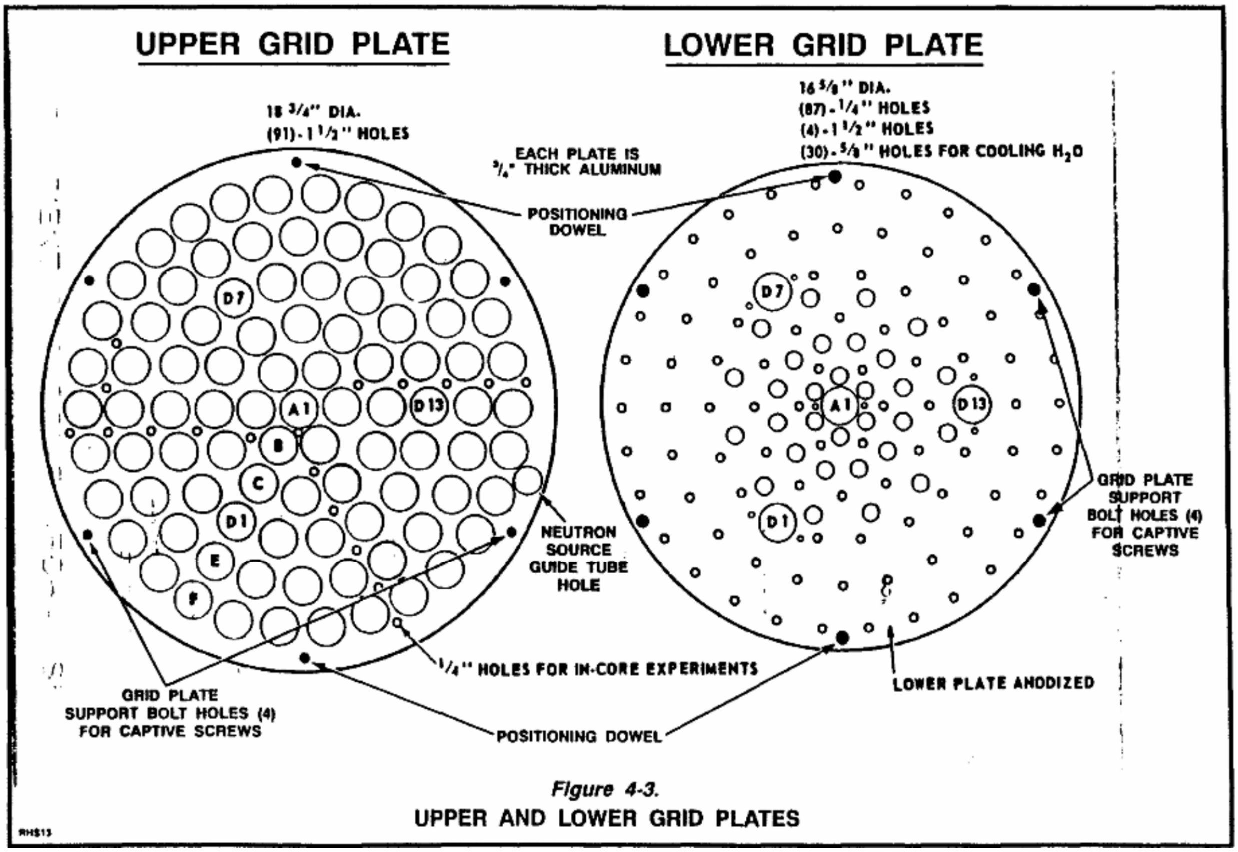


Figure 2-1 AFRRI reactor core configuration

TS 1.0, "Definitions," defines the reactor core components as follows:

(…)

1.9. CORE GRID POSITION

The core grid position refers to the location of a fuel element, control rod, or experiment in the grid plate. It is specified by a letter indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

(…)

1.14. FUEL ELEMENT

A fuel element is a single TRIGA fuel rod or the fuel portion of a fuel follower control rod (FFCR).

(…)

1.17. INSTRUMENTED FUEL ELEMENT

An instrumented fuel element is a fuel element in which one or more thermocouples have been embedded for the purpose of measuring fuel temperatures.

(…)

1.41. STANDARD CONTROL ROD

A standard control rod is a control rod having electromechanical drive and scram capabilities. It is withdrawn by an electromagnet/armature system.

(…)

1.44. TRANSIENT ROD

The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It is activated by applying compressed air to a piston.

The licensee stated in SAR Section 7.3.1, "Control Rod Drives," that the standard control rods, which include the regulating control rod drive and two shim control rod drives, are rack-and-pinion linear actuators. The regulating rod drive uses a stepper motor that can operate at variable speeds, and the shim rod drives use a stepper motor that operates at a single speed. The transient control rod is also called the pulse control rod and is operated by a pneumatic electric drive.

The NRC staff reviewed the proposed definitions above and finds that they are standard definitions used in research reactor TSs and are consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," and Part 2, "Standard Review Plan and Acceptance Criteria," (Ref. 39) and American National Standards Institute/American Nuclear Society, (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors." On this basis, the NRC staff concludes that the definitions above are acceptable. SER Section 2.2.2 contains the NRC staff's review of the control rods.

TS 5.2.2, "Reactor Core," states the following:

Specifications

a. The reactor core shall consist of TRIGA reactor fuel elements in a close packed array with a minimum of two thermocouple instrumented TRIGA reactor fuel elements.

b. There shall be four single core positions occupied by the three standard control rods and transient rod, a neutron startup source with holder, and positions for possible in-core experiments.

c. The core shall be cooled by natural convection water flow.

d. In-core experiments shall not replace B ring, C ring, and/or D ring fuel elements within the reactor core.

TS 5.2.2, Specification a, describes the approved reactor core configuration, which consists of standard TRIGA fuel elements, including two IFEs geometrically arranged in a close-packed array to control water holes in the reactor core. The control of water holes is necessary to control power peaking in fuel elements. TS 5.2.2, Specification a, helps ensure that the fuel elements used in the operating core configuration remain consistent with the assumptions used in the safety analysis. TS 5.2.2, Specification b, establishes the arrangements for the control and transient rods and additional position for a neutron startup source, and for in-core experiments. It helps ensure that the physical arrangement of fuel limits empty fuel locations and thereby controls power peaking in fuel elements. TS 5.2.2, Specification c, establishes the cooling requirements for the reactor core that are consistent with the licensee's thermal-hydraulic analysis. TS 5.2.2, Specification d, places a restriction on in-core experiments located at the highest power density locations in the core to prevent reactivity excursions and ensure safe power-peaking limits in adjacent fuel element positions are not exceeded. The NRC staff reviewed TS 5.2.2, Specifications a through d and finds them acceptable because they are consistent with the guidance in NUREG-1537, and the licensee's safety analysis.

Currently AFRRI has a TS 5.2.2, Specification e, which states "Fuel elements indicating an elongation greater than 0.100 inch, a lateral bending greater than 0.0625 inch, or significant visible damage shall be considered damaged, and shall not be used in the reactor core." Because this specification is a limiting condition for operation as defined by 10 CFR 50.36(c)(2)(i), given that it specifies "the lowest functional capability or performance levels of equipment required for safe operation of the facility," it was moved to TS 3.7. In the proposed TS 3.7, the term "transverse bend" replaces the previous term, "lateral bend." Lateral is defined as "to the side; of or pertaining to the side" and in mechanics has a similar meaning to transverse, which is defined as "situated or lying across; side to side, relative to some defined 'forward' direction." Transverse bending is a more accurate description of forces causing sagitta on fuel elements. NUREG-1537 guidance states that, for stainless-steel-clad UrZrH1.65 TRIGA fuel, the sagitta (or sag) shall not exceed 0.0635 inches (in) (0.159 centimeters (cm)) over the original length of the cladding in a circular gird core arrangement (Ref. 39).

TS 3.7, "Fuel Parameters," states the following:

<u>Specification</u>

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

    a. The transverse bend exceeds 0.0625 inches over the length of the cladding;

    b. The length exceeds its original length by 0.100 inches;

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

    d. Visual inspection identifies bulges, gross pitting, or corrosion.

2. The burnup of uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration.

TS 3.7, Specifications 1, a through d, and 2, help ensure that the reactor is only operated with fuel that has an effective cladding barrier to the release of any potential fission products. TS 3.7, Specifications a and b, help to ensure that elongation and lateral bending of fuel elements are limited and are based on criteria determined through analysis by GA, the reactor designer. Fuel growth and deformation can occur during normal operations, as described in NUREG-1537 (Ref. 39); GA-4314, E-117-833 (Ref. 62); and Simnad's 1981 article, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel" (Ref. 65). TS 3.7, Specifications c and d, help establish inspection requirements to detect gross failure or visual deterioration of the fuel. Damage preventing the removal of a fuel element from the cylindrical grid plate would be detected by the periodic measurement and inspection under TS 3.7, Specifications a through d. TS 3.7, Specification e, is based on NUREG-1537. The TRIGA fuel qualification in TS 3.7, Specification e, limits the burnup of the fuel to 50 percent of the original gram amount of uranium-235. The licensee's calculations showed that, by 80 megawatt thermal (MWt) days of usage, excess reactivity will have diminished such that the AFRRI reactor core will not be capable of maintaining power, and in order to continue operation will need to refuel. The licensee stated in a letter dated February 9, 2016 (Ref. 19), that if the reactor were to continue to be operated at its current utilization it would take 2,140 years for the core average burnup to reach 50 percent. The reactor core will run out of excess reactivity before the core reaches an average burnup of 50 percent of the original weight of U-235. The typical power-peaking factor for a TRIGA core is 2 or less. Considering this additional factor of 2, sufficient margin still remains to the 50-percent burnup limit for the fuel element operating at the hottest point in the core.

Based on its review, the NRC staff finds that TS 5.2.2, Specifications a through d, and TS 3.7, Specifications 1, a through d, and Specification 2, specify the AFRRI core design features and help ensure that core loading conforms and is limited to the analysis in the AFRRI SAR (Ref. 2). TS 5.2.2 and 3.7 help ensure that (1) excessive power densities will not result from any allowed core loading, (2) core components are adequate to control the power level of the AFRRI TRIGA reactor, (3) core components do not create unnecessary radioactivity, and (4) the reactor core is adequately cooled. The NRC staff also finds that TSs 5.2.2 and 3.7 are consistent with the guidance in NUREG-1537, Section 4.5.1, which recommends that the licensee identify the highest power density of any possible core arrangement. Based on the information above, the NRC staff concludes that TS 5.2.2, Specifications a through d, and TS 3.7, Specifications 1 and 2, are acceptable.

TS 4.6, "Reactor Fuel Elements," describes the surveillance requirements for the fuel as follows:

Specification

Fuel elements shall be inspected visually for damage or deterioration and measured for length and bend in accordance with the following:

a. Before being placed in the core for the first time or following long-term storage;

b. Every two years, not to exceed 30 months, or at intervals not to exceed 500 pulses of insertion greater than $2.00, whichever comes first, for fuel elements in the B, C, and D rings;

c. Every four years (not to exceed 54 months), or at intervals not to exceed 500 pulses of insertion greater than $2.00, whichever comes first, for fuel elements in the E and F rings; and

d. If damage, deterioration, or unacceptable length and bend measurements are found in one or more fuel elements, all fuel elements in the core shall be inspected for damage or deterioration and measured for length and bend.

TS 4.6 provides the surveillance for TS 3.7, "Fuel Parameters," which specifies that fuel elements indicating an elongation greater than 0.100 in (0.254 cm), a transverse bending greater than 0.0625 in (0.159 cm) or significant visible damage shall be considered damaged, and shall not be used in the reactor core. TS 4.6 helps ensure that the licensee will perform regular inspection of fuel elements for damage or deterioration. TS 4.6, Specification b, describes the surveillance of the fuel elements, including inspecting length and bend biannually for fuel elements in the B, C, and D rings, and Specification c, covers inspection of the fuel elements in the E and F rings every 4 years. TS 4.6, Specifications b and c, require the licensee to inspect any fuel element that has been operated for 500 pulses of insertion greater than $2.00. TS 4.6, Specification d, states that if damage, deterioration, or unacceptable length and bend measurements are found in one or more fuel elements, then all fuel elements in the core shall be inspected for damage or deterioration and measured for length and bend. During a 2-year cycle, the licensee must inspect all fuel elements in the B, C, and D rings, and during a 4-year cycle, all fuel elements in the core must be inspected.

The NRC staff reviewed the surveillance frequencies and concluded that they are consistent with NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The intervals are sufficient to ensure that the facility maintains fuel element integrity and can detect any deterioration in cladding integrity. Since TS 3.7, Specification 1, states that "The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements," the licensee must remove fuel elements with visible damage or deterioration or length or bow measurements in excess of specifications from the core, in order to operate. TS 4.6 specifies that if a single element fails to pass an inspection, it creates a trigger event for the inspection of all fuel elements. Inspection of all fuel elements provides a high degree of confidence that the only undamaged fuel will be used. If a fuel element cannot be removed from the core due to damage and/or bulging, the fuel element will not pass inspection. The NRC staff finds that the inspections of the fuel elements required by TS 4.6 require adequate oversight of the physical condition of the fuel and are acceptable.

TS 4.6 helps ensure that the AFRRI reactor fuel elements are not operated in a damaged condition, which could result in the potential loss of the cladding integrity and release of radioactive gases. The NRC staff finds that TS 4.6 is typical of TRIGA reactors and consistent with the guidance provided in NUREG-1537, Appendix 14.1 (Ref. 39). The NRC staff's review of AFRRI annual operating reports for the years 2008 through 2015 (Ref. 25) did not identify any reported fuel failures, damaged fuel, or fuel with out-of-tolerance conditions. The NRC staff finds that TS 4.6 helps ensure that the quality of the AFRRI TRIGA reactor fuel is maintained. Therefore, based on the information above, the NRC staff concludes that TS 4.6 is acceptable.

TS 3.1.3, "Reactivity Limitations," defines the reactivity limitations for the core as follows:

Specifications

a. The reactor shall not be operated with the maximum available excess reactivity greater than $5.00 (3.5% Δk/k).

b. The shutdown margin provided by the remaining control rods with the most reactive control rod in the most reactive position shall be greater than $0.50 (0.35% Δk/k) with the reactor in the reference core condition, all irradiation facilities and experiments in place, and the total worth of all non-secured experiments in their most reactive state.

TS 3.1.3, Specification a, helps establish an appropriate limit on the AFRRI core excess reactivity that allows 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 the control rod design. In its March 4, 2010, letter (Ref. 2), the licensee stated that the measurement of the core excess reactivity for the current core is $4.37. The licensee calculated an excess reactivity of $4.89 for a middle-of-life core using the Monte Carlo N-Particle eXtended (MCNPX) computer code, which agreed well with the measured value. MCNPX is a code that research reactors use and that has been affirmed by confimatory calculations performed by the NRC staff. The MCNPX computer code predicts an excess reactivity of $2.99 for the end-of-life core. The NRC staff finds that these values are less than the TS limit (+$5.00) and provide an acceptable measure of agreement between calculated and measured values. The NRC staff concludes that the limit of $5.00 maximum excess reactivity, along with the maximum available control rod worths (discussed later in this section), ensure that an adequate shutdown margin is available under all operational conditions and that, therefore, TS 3.1.3, Specification a, is acceptable.

TS 3.1.3, Specification b, helps ensure that the reactor can be shutdown from any operating condition with the highest worth control rod stuck out of the core and all non-secured experiments in their most reactive position. The AFRRI reactor core is suspended from a movable carriage and can be positioned at numerous locations within the reactor tank. According to the SAR (Ref. 2), Section 4.4, "Support Structure," the four-wheeled carriage travels on two tracks that span the reactor tank and is used to move the reactor core laterally from one operating position within the tank to another. Reflector conditions change as the core is repositioned. AFRRI stated in the resubmitted Chapter 4 of the SAR (Ref. 2), dated March 4, 2010, and in its response to a request for additional information (RAI) dated September 27, 2010 (Ref. 4), that the AFRRI staff measures the core excess reactivity each day on which a control rod withdrawal is planned. During annual reactor shutdown and maintenance, control rod reactivity curves are plotted. The control rod reactivity is recorded as a function of rod position at three lateral core positions-north, center, and south. The measured reactivity identified in Table 4-9 in the SAR applies to the central position, which is the reactor core position with the highest reactivity. Repositioning the core to any other location results in lower excess reactivity. The core configuration of AFRRI reactor is, in part, based on meeting TS 3.1.3, Specification a, excess reactivity limit of $5.00 or less in any of the core positions. Because the central core position is the most reactive position, the licensee must ensure that the TS 3.1.3, Specification a, limit for excess reactivity is met when the core in the central position.

The licensee used the MCNPX code to compute the control rod worths and the shutdown margin. The combined control rod worth (not including the transient rod) is \$9.55, and the shutdown margin is \$4.66 at the middle of life and \$6.12 at the end of life without accounting for experiments. The MCNPX calculation used the most reactive rod in the most reactive position for the shutdown margin calculation. The available shutdown margin allows flexibility in designing experiments and ensures that the TS 3.1.3, Specification b, requirement of a minimum shutdown margin of \$0.50 is met and that the reactor can be shut down from any operating condition.

When the licensee removes a control rod for maintenance, it uses a procedure in the reactor annual maintenance checklist to control the core configuration and maintain subcriticality by at least \$0.50. A sufficient number of fuel elements are removed so that if the most reactive control rod and a second control rod were removed simultaneously, the core would still be subcritical by at least \$0.50, helping to ensure adequate shutdown margin.

TS 1.0, “Definitions,” defines the reference core condition as follows:

> 1.34. REFERENCE CORE CONDITION
>
> The reference core condition is when the core is at ambient temperature and the reactivity worth of xenon is negligible (<\$0.01).

Given that the reactivity required to satisfy the AFRRI reactor shutdown margin (TS 3.1.3, “Reactivity Limitations,” Specification b) is \$0.50 in the reference core condition, the definition of reference core condition is consistent with the guidance in NUREG-1537. Based on the information above, the NRC staff concludes that the licensee’s definition is acceptable and that the minimum shutdown margin of \$0.50 provides adequate shutdown margin for the AFRRI TRIGA reactor. The NRC staff finds that TS 3.1.3, Specifications a and b, are consistent with the guidance in Section 3.1 of Appendix 14.1 to NUREG-1537 (Ref. 39); therefore, the NRC staff concludes that TS 3.1.3, Specification b, is acceptable.

TS 4.1, “Reactor Core Parameters,” presents the corresponding surveillance requirements for the shutdown margin and the excess reactivity as follows:

> Specifications
>
> a. The reactivity worth of each standard control rod/transient rod and the shutdown margin shall be determined annually, not to exceed 15 months, or following any significant (>\$0.25) changes to core configuration (excluding in-core experiments).
>
> (…)
>
> c. The core excess reactivity shall be measured each day of operation involving the movement of control rods, or prior to each continuous operation exceeding more than a day, and following any significant (>\$0.25) core configuration changes. At a minimum, excess reactivity shall be measured annually, not to exceed 15 months. The measurement is also a complete channel test of the linear power channel and log power channel.
>
> (…)

TS 4.1, Specification a, requires control rod worth to be determined annually and after significant core configuration changes. Accurate control rod worths are used to determine other reactivity-based limits, such as excess reactivity, shutdown margin, and the reactivity worth of experiment. Specification a, also requires that the licensee determine the shutdown margin annually. This determination is done to ensure that the required shutdown margin is available. TRIGA operating experience shows that annual rod worth reactivity measurement is adequate to ensure no significant changes to the shutdown margin. SER Section 2.2.2 further discusses this issue.

TS 4.1, Specification c, requires that the licensee measure core excess reactivity at the beginning of each day of operation involving the movement of control rods, or before each continuous operation exceeding more than 1 day, and following any significant (>$0.25) core configuration changes. Based on the surveillance requirements for measuring core excess reactivity, the NRC staff concludes that this surveillance will help ensure that the core excess reactivity is within TS limits.

TS 4.1, Specifications a and c, control the important aspects of the design and the basic overall characteristics of AFRRI's TRIGA reactor core that are defined in more detail in other portions of the TSs. TS 5.2.2, TS 3.7, and TS 3.1.3 specify the normal operating conditions of the reactor core and include limits on the allowable core configurations, shutdown margin, and excess reactivity. TS 4.1, Specifications a and c, specify the corresponding surveillance. The NRC staff finds TS 5.2.2, TS 3.7, TS 3.1.3, and TS 4.1, Specifications a and c, consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff finds that the analysis presented in the SAR (Refs. 1 and 2) provides a basis for TS 5.2.2, TS 3.7, TS 3.1.3, and TS 4.1, Specifications a and c, and shows that normal operation within the constraints of these TSs will not lead to the release of fission products from the fuel. Based on these considerations, the NRC staff concludes that the licensee has adequately analyzed the reactor core for the expected normal operation during the period of the renewed facility operating license. The NRC staff further concludes that TS 5.2.2, TS 3.7, TS 3.1.3, TS 4.1, Specifications a and c, and TS 4.6 provide reasonable assurance that normal operation of AFRRI's TRIGA reactor core will not pose a significant risk to the public health and safety or the environment, and are therefore acceptable.

2.2.1 Reactor Fuel

The development and use of U-ZrH fuels for TRIGA reactors began in 1957. GA's research and development program has demonstrated that the fuel used in TRIGA reactors can be operated safely in both the steady-state and pulse modes. More than 6,000 fuel elements of seven distinct types have been fabricated for use in 60 TRIGA reactors worldwide. More than 25,000 pulses have been safely performed domestically and abroad with TRIGA fuel elements. U-ZrH fuel has exhibited unique safety features, including a large prompt negative temperature coefficient of reactivity, high fission product retention ability, chemical stability when quenched from high temperatures in water, and dimensional stability over large ranges of temperatures.

AFRRI uses standard TRIGA stainless-steel-clad cylindrical fuel elements in which the zirconium hydride moderator is homogeneously mixed with the uranium fuel. The active part of each element consists of three cylindrical slugs of U-ZrH containing 8.5 weight-percent (w%) uranium with less than 20-percent U-235 enrichment. The hydrogen-to-zirconium atom ratio is about 1.7 to 1. Graphite end plugs are located in the fuel element above and below the fuel section and serve as neutron reflectors. The fuel element cladding is a Type 304 stainless-steel tube sealed at the top and bottom with stainless-steel end fittings. A solid zirconium rod

0.225 in (0.572 cm) in diameter is centered in the fuel region of each fuel element to provide structural support.

In the AFRRI SAR, Section 4.2.1, “Reactor Fuel,” of March 4, 2010 (Ref. 2), the licensee stated that samarium is used in each fuel element to minimize reactivity changes from fuel burnup and fission product buildup. The samarium is mixed with aluminum to form thin wafers. Samarium is a burnable neutron poison. As the reactor operates, the samarium is consumed, which results in a positive reactivity addition that helps to offset the negative reactivity addition from burning the uranium in the fuel. Samarium allows a longer fuel life without unacceptable reactivity effects by allowing more uranium to be initially put in the fuel. The wafers are located between the fuel-moderator section and the end plugs.

TS 5.2.1, “Reactor Fuel,” defines the individual design characteristics of the fuel elements as follows:

> <u>Specifications</u>
>
> The individual non-irradiated TRIGA fuel elements shall have the following characteristics:
>
> a. Uranium content: Maximum of 9.0 weight percent enriched to less than 20% uranium-235. In the fuel follower, the maximum uranium content shall be 12.0 weight percent enriched to less than 20% uranium-235.
>
> b. Hydrogen-to-zirconium atom ratio (in the ZrH_x): Nominal 1.7 H atoms to 1.0 Zr atoms with a range between 1.6 and 1.7.
>
> c. Cladding: 304 stainless steel, nominal 0.020 inches thick.
>
> d. Any burnable poison used for the specific purpose of compensating for fuel burnup or long-term reactivity adjustments shall be an integral part of the manufactured fuel elements.

TS 5.2.1, Specifications a through c, control the important aspects of the fuel design, including the uranium content, enrichment, hydrogen-to-zirconium ratio limits, and the clad material and thickness for standard TRIGA fuel elements. The effect of a maximum uranium content of 9 w% U-235 for the standard TRIGA elements was analyzed for a reactor with similar fuel to the AFRRI TRIGA reactor (Washington State University) and was found to be about 6 percent greater than the design value of 8.5 w% U-235. Such an increase in loading would result in an increase in the power density of 6 percent. The maximum hydrogen-to-zirconium ratio of 1.7 could cause the maximum stress in the fuel element clad to be higher than a factor of 2 greater than the value resulting from a hydrogen-to-zirconium ratio of 1.6. However, this increase in the clad stress would not exceed the rupture strength of the clad, according to an analysis performed by Washington State University (Ref. 66). Based on the information above, the NRC concludes that TS 5.2.1, Specifications a through c, are acceptable.

TS 5.2.1, Specification d, states that any burnable poison content must be an integral part of the fuel element. This process could lead to a variation of burnable poison content for a single fuel element of about 1 to 2 percent over the content used in the analysis, increasing the local power density by about 2 percent (Ref. 66). The NRC staff finds TS 5.2.1, Specification d, acceptable because the AFRRI safety analysis, SAR Section 4.2.1, has adequately demonstrated that the fuel design has sufficient safety margins at its licensed power level of operation in regard to uncertainties in the burnable poison distribution.

Fuel growth and deformation can occur during normal operations, as described in NUREG-1537 (Ref. 39) and GA-4314 (Ref. 62). Damage mechanisms include fission recoils and fission gases, both of which are strongly influenced by thermal gradients. Swelling of the fuel is dependent on the amount of time the fuel spends over a temperature threshold of about 750 °C (1,382 °F). At a 1.0 MWt steady-state power level, the calculated peak fuel temperature in the hot rod is 440.7 °C (825.26 °F) (Ref. 1). At the licensed 1.1 MWt maximum power level, the peak fuel temperature is less than 500 °C (932 °F), and swelling would be minimal, if present at all (Ref. 14). Although fuel temperatures could go above 750 °C (1,382 °F) during pulsing, the time at temperature is short enough that pulsing would not cause fuel swelling by these mechanisms. The NRC staff reviewed the data provided by the licensee and concludes that the data provides reasonable assurance that fuel swelling by the above mechanism is precluded.

Two IFEs are used to monitor fuel temperature. These fuel elements have three chromel-alumel thermocouples embedded axially in the fuel moderator section. The sensing tips are located halfway between the outer radius and the vertical centerline on the fuel section and 1 inch above and below the horizontal center. The IFE allows the licensee to directly measure the temperature of the fuel. With the exception of the thermocouples, the IFEs are identical to the standard fuel elements.

An important parameter used to ensure fuel integrity is the fuel temperature. TS 3.2.1, “Reactor Control System” (discussed in Section 2.2, “Control Rods,” of this SER), and TS 3.2.2, “Reactor Safety System” (discussed in Section 2.5.1.2, “Pulse Mode Operation,” of this SER), require two fuel temperature measuring channels and two fuel temperature safety channels. To help ensure that the fuel temperatures are properly monitored, TS 4.2.3, “Fuel Temperature,” defines the surveillance requirements of the fuel temperature measuring channels and fuel temperature safety channels as follows.

TS 4.2.3 states the following:

<u>Specifications</u>

a. A channel check of the fuel temperature scrams shall be made each day that the reactor is to be operated.

b. A channel calibration of the fuel temperature measuring channels shall be made annually, not to exceed 15 months.

c. A weekly channel test shall be performed on fuel temperature measuring channels, whenever operations are planned.

d. If a reactor scram caused by high fuel element temperature occurs, an evaluation shall be conducted to determine whether the fuel element temperature exceeded the safety limit.

TS 4.2.3, Specification a, requires the licensee to perform daily channel checks of fuel temperature scrams.

TS 4.2.3, Specifications b and c, require periodic calibrations and tests of the fuel temperature measuring channels.

TS 4.2.3, Specification d, requires the licensee to determine whether the temperature actually exceeded the SL in the event that a high-temperature scram occurs.

The NRC staff finds that these surveillances of fuel element parameters and fuel element temperature measuring and safety channels and their intervals are consistent with the guidance in NUREG-1537, Appendix 14.1 (Ref. 39), and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff finds that these surveillance frequencies will ensure performance and operability of the fuel elements, including the fuel element temperature measuring systems or components.
The licensee has described the fuel elements in detail (constituents, materials, components, and fabrication), in its SAR Section 4.2.1 “Reactor Fuel,” (Ref. 2) and has discussed the design limits along with technological and safety-related bases for all applicable limits. The design limits are identified for use in applicable design bases to support the TS. The design and development program for the TRIGA fuel offers reasonable assurance that the fuel will function safely for the period of the license renewal without adversely affecting the public health and safety.

The NRC staff reviewed the constituents, materials, and components of the fuel elements stated in the SAR, as supplemented. Based on its review, the NRC staff finds that the licensee adequately described the fuel elements used in the AFRRI reactor, including the design limits, and the technological and safety-related bases for these limits. The NRC staff concludes that compliance with TS 5.2.1, and TS 4.2.3 limits will ensure uniform characteristics and compliance with design bases and safety-related requirements. Based on the information above, the NRC staff concludes that TS 5.2.1, and TS 4.2.3 are acceptable.

2.2.2 Control Rods

Reactor power at AFRRI is controlled by three standard control rods, which include a shim rod, a safety rod, and a regulating rod (as discussed in SAR Section 4.6.2, “Standard Control Rod Drives”) and one transient control rod, (SAR Section 4.6.3, “Transient Control Rod”) (Ref. 2). TS 5.2.2, “Reactor Core,” Specification b, specifies the core configuration and arrangements of the control rods.

Each standard control rod consists of a sealed stainless-steel tube with an air-filled upper section. The middle segment contains boron as the neutron absorber. The lower part can contain a 12-w% U-ZrH fuel mixture with a solid zirconium rod in the center, an air follower, or a solid aluminum follower. The definitions section of the TS defines “control rod” as a device fabricated from neutron-absorbing material or fuel, or both, that is used to establish neutron flux changes and to compensate for routine reactivity losses. Scrammable control rods can be quickly uncoupled from their drive units to rapidly shut down the reactor if needed. All control rods have a design-limited maximum travel. The licensee projected that the lifetime of the control rods will extend beyond the period of renewal and has no plans to replace or store depleted control rods.

The shim, safety, and regulating control rods are driven within the core by rack and pinion electromechanical drive mechanisms and are positioned within the core by guide tubes, as discussed in SAR Section 4.6.2, “Standard Control Rod Drives” (Ref. 2) and SAR Section 7.3.1, “Control Rod Drives” (Ref. 78). Rod position within the core is displayed on the reactor console. An electromagnet is secured to the bottom of the draw tube to which the rack is mounted. The magnet is moved up or down in response to rotation of the pinion shaft. The control rod is attached to the armature by a long connecting rod. When the magnet is energized, the armature is magnetically coupled to the draw tube. Deenergizing the magnet causes the rod to drop. A dash pot is incorporated into the armature section to decelerate the rod near the bottom following a scram. Limit switches sense when the magnet is fully withdrawn, the magnet is fully down, and the armature (and thereby the rod) is fully down. A 10-turn potentiometer is coupled to the pinion shaft to provide for rod position indication. The pinion shafts are chain-and-sprocket coupled to a direct current (DC) stepper motor. A series of micro switches provide full-in and full-out indications for the control rods. The reactor interlock system prevents the simultaneous manual withdrawal of two or more standard control rods during steady-state operation and prevents the withdrawal of any standard control rod during pulse operation.

The SAR Section 7.3.2, "Servo System" (Ref. 78), provides a description of the servo system. In the Automatic and Square-Wave modes of operation, the regulating rod is controlled by the servo system to control reactor power based on input signals from a power channel, reactor period signal, and the power demand control. The regulating rod drive uses a stepper motor that operates at variable speeds when operated by the servo system, however limited by its maximum speed in the manual mode. The shim and safety rod drives use the same type of stepping motor as the regulating rod drive but operate at a single speed.

In Automatic mode, the reactor power is compared against the power demand setting to obtain power error. The period signal is monitored by the controller to limit the reactor period to a minimum of +8 seconds when power is being increased. To reduce hunting of the regulating rod, a deadband is incorporated in the system. The power error signal is used by the derived air concentration computer to determine which direction (if any) the regulating rod needs to move to correct the power error. The regulating rod speed is variable and it will move slowly for small errors and it will move fast for large errors. The regulating rod speed cannot exceed the travel speed that is used in manual control. The variable speed ability of the servo system reduces power overshoot during transients.

Square Wave mode operation requires that the reactor must first be configured in Steady-State mode. The reactor power is raised to some nominal low power (less than 1,000 watts) with the air to the transient rod off. The transient rod cylinder is then raised to the position corresponding to the desired reactivity insertion and then the square wave mode switch is depressed to change the console mode from Steady-State to Square Wave and the transient rod fire button pressed. Reactor power will increase to the desired power level and then switch to the Automatic mode to maintain, a constant power level.

The fourth control rod is the transient rod, sometime called the pulse rod, is described in SAR Section 4.6.3, "Transient Control Rod," (Ref. 2). The transient rod consists of a sealed aluminum tube that is slightly larger than the standard control rod. A portion of the rod is filled with a boron compound. The lower portion of the rod contains either a machined solid aluminum follower, a combined poison and aluminum follower, or an air follower. The transient rod is operated by a pneumatic/electric drive. A connecting rod couples the transient rod to a piston rod assembly. The piston resides within an externally threaded cylinder. A ball screw nut acts on these external threads to raise or lower the cylinder. Rotation of the ball screw nut is accomplished by a worm gear coupled to a motor. A potentiometer is gear-driven by the worm gear shaft to provide rod position indication. A hydraulic shock absorber is incorporated into the top of the cylinder. Air from a compressor is connected to a normally closed port of a three-way air solenoid valve. The common port is connected to the transient control rod drive cylinder below the piston. The normally open port is vented. When the air solenoid valve is energized, air pressure is placed on the bottom of the piston, causing the piston to be brought in contact with the shock absorber. The resulting reactivity insertion is dependent on the position of the cylinder before applying air. With air applied, energizing the motor in the up or down direction will cause the cylinder, piston, and control rod to move up or down as a unit acting as a steady-state mode control rod. Scram of the transient rod is accomplished by deenergizing the air solenoid valve. This vents the air pressure under the piston and results in the control rod dropping. Limit switches provide for sensing cylinder up, cylinder down, and rod down. A bracket extends over the top of the cylinder. A switch on the bracket opens a contact in the up circuitry when the shock absorber assembly contacts it. The bracket itself is substantial enough to stall the motor should the switch contact fail to open. The electromechanical portion of the drive, which consists of an electric motor, a ball-nut drive assembly, and an externally threaded air cylinder, allows the transient rod to operate like a standard control rod when it is not being used for pulse operation.

TS 5.2.3, "Control Rods," contains the design specifications for the standard control rods as follows:

Specifications

a. The standard control rods shall have scram capability, contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. These rods may have an aluminum, air, or fuel follower. If fuel followed, the fuel region will conform to Technical Specification 5.2.1.

b. The transient control rod shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless-steel cladding. This rod may incorporate an aluminum, poison, or air follower.

The AFRRI reactor control rods are standard design for the TRIGA Mark F reactor and have long operating histories. The design specifications give the requirements for the standard and transient rods. The objective is to ensure that control rods are fabricated to reliably perform their intended control and safety function.

TS 5.2.3, Specifications a and b, help ensure that the design specifications and requirements for the standard and transient rods are maintained as discussed in the SAR, Section 4.10, "Reactor Control Components." The NRC staff finds that the material characteristics in TS 5.2.3 will help ensure that the important aspects of the design of the control rods are maintained, and will help ensure that the control rods will perform their safety function. TS 5.2.3 describes the important aspects of the design of the standard control and transient rods to ensure they will perform their safety function. Based on the information above, the NRC staff concludes that TS 5.2.3, Specifications a, and b, are acceptable.

TS 3.2.1, Specifications b and c, define requirements to ensure that the control rods are available and will promptly shut down the reactor upon a scram signal.

TS 3.2.1, “Reactor Control System,” states the following:

> Specifications
>
> (…)
>
> b. The reactor shall not be operated unless the four control rod drives are operable except:
>
>   a. the reactor may be operated at a power level no greater than 250kW with no more than one control rod drive inoperable with the associated control rod drive fully inserted.
>
> c. The time from scram initiation to the full insertion of any control rod from a full up position shall be less than 1 second.

TS 3.2.1, Specifications b and c, help ensure that, during normal operation of the AFRRI reactor, either all control rods are operable except that the reactor may be operated at a power level no greater than 250 kilowatt thermal (kWt) with no more than one control rod drive inoperable with the associated control rod drive fully inserted.

The licensee provided supplemental information on September 30, 2016 (Ref. 84), concerning reactor operation limited to 250 kWt with one control rod fully inserted in AFRRI core 85-3. The license conservatively assumed that the inserted control rod suppressed all power production in 30 percent (26 elements) of the 87 fuel elements in the core. The licensee’s analysis results in all of the power being produced in the remaining 61 elements with an average power density of 4.1 kWt per element which is less than the GA thermal-hydraulic limit of approximately 21 kWt per element. Significant safety margin exists between the average power density and the limiting power density to conclude that the peak power density would need to exceed five times that of the average power density to exceed the GA limiting power density. The NRC staff notes that only an unrealistic, significant change to the flux distribution in the core, due to the inserted control rod, could impact the power monitoring channels. Such a change would require that the reactor would have be operating above its licensed power limit when the power monitoring channels were indicating 250 kWt. Therefore, the NRC staff concludes that operating the reactor at a power level of up to 250 kWt with one control rod fully inserted is acceptable.

The NRC staff also finds that the time required for the control rods to be fully inserted from the instant that a scram signal is initiated is rapid enough to prevent fuel damage. Adherence to TS 3.2.1, Specifications b and c, help ensure that the reactor will be promptly shut down when a scram signal is initiated. The 1-second value is an analytical assumption in the reactivity insertion events described in Section 6.2.2, “Reactor Power Transients,” of the SAR (Ref. 1). For the range of transients anticipated for the AFRRI TRIGA reactor, the specified 1-second scram time is adequate to ensure the safety of the reactor because it protects the integrity of the fuel.

The NRC staff finds that the requirements in TS 3.2.1, Specifications b and c, support the basic design requirements described in the SAR, as amended, to prevent reactor fuel damage. Therefore, the NRC staff concludes that TS 3.2.1, Specifications b and c, are acceptable.

TS 4.2.1 give the surveillance requirements for the control and transient rods.

TS 4.2.1, "Reactor Control Systems," states the following:

> Specifications
>
> a. The standard control rods/transient rod shall be visually inspected for damage and deterioration annually, not to exceed 15 months.
>
> (…)
>
> c. On each day that pulse mode operation of the reactor is planned, the transient rod system is channel tested to verify that the system is operable. Semiannually, not to exceed 7.5 months, the transient rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary.

TS 4.2.1, Specification a, requires the licensee to visually inspects the control rods for deterioration annually (in intervals not to exceed 15 months). Visual inspections are made for indications of corrosion or wear characteristics, which, along with the functional checks, will help ensure acceptable performance. The NRC staff finds that the surveillance interval specified in TS 4.2.1, Specification a, is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64) and, therefore, is acceptable.

TS 4.2.1, Specification c, requires the licensee to perform a channel test of the transient rod system to verify the system is operable, on each day that pulse mode operation is planned. Semiannually (periods not to exceed 7.5 months), the licensee must inspect, clean, and lubricate the transient rod drive cylinder and the associated air supply system. These inspections and maintenance activities help ensure that the necessary quality of the control systems and components is maintained. The NRC staff finds that the surveillance interval specified in TS 4.2.1, Specification c, is consistent with the guidance in NUREG-1537 (Ref. 39) and ANS/ANSI-15.1-2007 (Ref. 64) and, therefore, is acceptable.

TS 4.2.1, "Reactor Control Systems," Specification b, gives the requirements for the control rod drop times of all rods as follows:

> Specification
>
> (…)
>
> b. The control rod drop times of all rods shall be measured semiannually, not to exceed 7.5 months. After work is done on any rod or its rod drive mechanical components, the drop time of that particular rod shall be verified.
>
> (…)

TS 4.2.1, Specification b, requires that the licensee verifies the control rod drop times semiannually (at intervals not to exceed 7.5 months). This helps ensure that the licensee confirms both the scram time assumptions used in the safety analysis and the TS 3.2.1, Specification c, requirement that scram time for the control rods must be less than 1 second. Specification b also establishes the requirements to confirm by testing, so that the minimum required scram time is maintained after work is done on a control rod or its drive mechanism.

The NRC staff finds that the surveillance interval specified in TS 4.2.1, Specification b, is consistent with the guidance in NUREG-1537 (Ref. 39) and ANS/ANSI-15.1-2007 (Ref. 64) and, therefore, is acceptable.

Based on a review of the information provided by the licensee, the NRC staff concludes that the control rods conform to the applicable design bases described in the SAR, Section 4.10, “Reactor Control Components,” and can shut down and maintain the reactor shutdown under normal, abnormal and accident conditions. The TSs require that the scram features will perform as required during the renewal period to ensure fuel integrity and to protect public health and safety. The design and functional description of the transient rod system help ensure that pulses will be reproducible and will be limited to values that maintain fuel integrity. The control rod design for the AFRRI TRIGA reactor includes reactivity worths that can control the excess reactivity planned for the AFRRI reactor, including the assurance of an acceptable shutdown reactivity and margin.

The licensee has described appropriate design limits, limiting conditions for operation, and surveillance requirements for the control rods. TS 5.2.3, TS 3.2.1, TS 4.1, and TS 4.2.1 require that the control rods will promptly shut down the reactor upon a scram signal. These TSs specify surveillance intervals to help ensure the operability of the control rods and require verification that all control rods meet the scram time requirement. Measurement of the scram time not only checks the scram system electronics but also indicates the capability of the control rods to perform properly. The NRC staff finds that these intervals for control rod inspection, scram time determination, and control rod maintenance are sufficient to help ensure operability and that they are consistent with the surveillance intervals recommended by NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Based on this finding, the NRC staff concludes that the requirements related to the AFRRI reactor control rods, TS 5.2.3, TS 3.2.1, TS 4.1, and TS 4.2.1 are acceptable for ensuring the performance of the control rods.

2.2.3 Neutron Moderator and Reflector

As described in the SAR, Section 4.2, “Reactor Tank,” the predominant moderator of the AFRRI reactor core is the zirconium hydride incorporated into the low-enriched uranium fuel elements. The pool water between the fuel elements also serves as a moderator. The core is reflected on the top and bottom by graphite end plugs in each fuel element and by the water at the periphery. The AFRRI reactor core can be positioned at numerous horizontal locations within the reactor tank, thus creating a variety of radial reflector combinations. The radial reflector materials comprise water, aluminum, and lead. The reactor core is located under approximately 16 feet (ft) (4.9 meters (m)) of water. TS 3.2.2 states, in part, that the maximum setpoint for the pool water level is 14 ft (4.27 m) from the top of the core. This is equivalent to a drop of 6 inches from normal pool water level. A drop of approximately 6 in (15.2 cm) of the pool water will result in a reactor scram per TS 3.2.2. As discussed in SAR Section 4.3, “Reactor Tank or Pool” (Ref. 2), the pool water level alarm ensures that a loss of biological shielding (pool water) would result in a reactor shutdown.

According to SAR, Section 4.9, Fuel Elements,” the axial reflector, both above and below the core, consists of the graphite end plugs in the fuel elements and the pool water above and below the core region. The graphite reflectors are known to maintain their structural integrity but are susceptible to graphite expansion with sufficient neutron exposure. Such growth may distort the graphite and cause the clad to deform. The licensee performs routine visual inspections according to periodicity specified in TS 4.6, “Reactor Fuel Elements,” that include the cladding enveloping the graphite plugs (reflectors) to detect significant structural changes. Visual inspections are required to ensure that no major degradations have occurred. The NRC staff concludes that the license renewal SAR, as supplemented, provides assurance that the moderator and reflectors will function safely in the AFRRI reactor core during the renewal period.

2.2.4 Neutron Startup Source

The AFRRI reactor uses a 3 curie americium-beryllium (Am-Be) neutron startup source. The primary function of the neutron source is to show that the instrumentation is functioning properly by providing sufficient counts on a neutron-monitoring channel during startup of the reactor. It is composed of pellets of an americium oxide and beryllium mixture in a 1-to-10 weight ratio; these pellets are doubly encapsulated in a Type 304L stainless-steel source holder. The licensee stated that the current neutron source will be used indefinitely.

The neutron source is inserted into the AFRRI reactor core inside a 0.875 in (2.2 cm) diameter tubular source guide tube. A hole in the upper grid plate, near the edge of the grid array, receives the guide tube. When bolted to the upper grid plate, the guide tube projects downward alongside the fuel array. When in position, the source material is located at the horizontal centerline of the core. A top adapter, which is pinned to the source holder, has a hole through which a handling wire is attached. This configuration enables the source to be removed from the guide tube for tests or storage.

When not in use, the neutron source is stored in the reactor tank. A neutron source clad failure would be detected during routine measurement of the radioactivity of the reactor pool during analysis of pool water, as required by TS 4.3, Specification c. The NRC staff finds the surveillance requirements specified in TS 4.3, Specification c, acceptable for limiting the radioactivity content of the pool water, reducing personnel exposure, and detecting potential damage to the source cladding.

The design of the neutron source used at AFRRI is similar to the designs used reliably in other TRIGA reactors. The Am-Be source will not degrade in a radiation environment during reactor operation. The use of this source has not resulted in any personnel overexposure, and the licensee does not anticipate any such exposure through its continued use during the period of license renewal. The location of the source in the core allows it to be cooled by convection of the pool water. The source strength produces an acceptable count rate on reactor startup instrumentation and allows for a monitored reactor startup under all operating conditions.

As discussed in SER Chapter 5, TS 3.2.2 requires the control and safety system to have an interlock that prevents rod withdrawal when the neutron count is less than 0.5 counts per second.

The NRC staff reviewed the information on the neutron startup source in the license renewal SAR, as supplemented. Based on its review, the NRC staff concludes that the neutron startup source is adequate to allow controlled reactor startup and, therefore, it is acceptable.

2.2.5 Core Support Structure

The AFRRI reactor core can be positioned at numerous horizontal locations within the reactor tank. A four-wheeled carriage, which travels on two tracks that span the reactor tank, moves the reactor core laterally. In addition to supporting the core, the carriage also supports the control rod drives, the nitrogen-16 diffuser system, and various electronic devices.

The core support structure consists of an aluminum cylinder and an aluminum adapter. The cylinder connects the adapter to the core carriage. A vertical slot extends the full height of the cylinder. This slot allows for access to the inside of the support structure. As a result, the slot enables the installation and removal of core components without raising them above the pool water level.

The reactor core is enclosed in an aluminum shroud that is attached to the bottom of the core support adapter. The upper and lower grid plates are bolted to the top and bottom of the core shroud. The fueled region of the core is located above the bottom of the reactor tank.

An upper and lower grid plate holds the fuel elements, control rods, and other in-core assemblies in place. The grid plates are attached to a cylindrical shroud that surrounds the core. The upper grid plate contains 91 holes in concentric rings. Four of the holes receive the control rod guide tubes, while the remaining 87 holes hold the fuel elements and the in-core experiment tube. Eighteen smaller holes are used for small in-core experiments and dosimetry.

The lower grid plate is gold-anodized to reduce wear and to aid light reflection in the core. In addition to 30 flow holes, the lower plate contains a similar hole pattern as that in the upper grid plate. Both grid plates are constructed from aluminum plates and are bolted to the shroud by four bolts. Two positioning dowels ensure correct positioning of the plates.

The core system at AFRRI allows the reactor core to be moved within the reactor tank. Upper and lower grid plates accurately position and align the fuel elements. The core support also provides guides and supports for the control rod drives, nuclear detectors, and in-core experiments. The design is open to allow sufficient coolant flow for the fuel elements. The structure is composed of aluminum components that are resistant to radiation damage and corrosion.

The licensee visually inspects the carriage structure to observe structural degradations. The licensee indicated that the carriage structure maintains its structural integrity and that visual inspections during reactor core changes are sufficient to recognize structural degradations. The NRC staff reviewed the licensee’s information and finds that the licensee adequately analyzed its core support structure and that required inspections will adequately detect structural degradations. The NRC staff concludes that the reactor carriage is adequate to perform its function during the renewal period.

2.3 Reactor Pool

The AFRRI reactor tank is a cloverleaf-shaped configuration, approximately 19.5 ft (5.9 m) deep with a distance across the tank lobes of approximately 13 ft (4 m). The cloverleaf projections allow the core to face the exposure rooms over a 210-degree arc in the horizontal plane. The tank bottom thickness is 0.5 in (1.27 cm).

The tank is constructed of aluminum and embedded in concrete. A Phenoline coating was applied between the tank and the concrete. The core is shielded radially by the reactor tank water and by ordinary concrete (with the exception of the two exposure rooms). The vertical shielding comprises approximately 16 ft (4.9 m) of water above the core and ordinary concrete. All concrete was placed in accordance with prescribed building standards.

The normal water level in the reactor pool is approximately 14.5 ft (4.57 m) above the top of the reactor core. The licensee stated in the application that the reactor tank water level is continuously monitored by a float-activated switch located on the east side of the pool with a liquid crystal display readout on the auxiliary console in the reactor control room. In addition, a water level drop below 6 in (15.24 cm) from normal level activates an audible and visual alarm on the reactor console and on the annunciator panel.

The tank is illuminated by lights that are suspended from the top of the tank walls. The primary coolant pump takes a suction at about 4 ft (122 cm) below the pool surface and returns that coolant to the pool at about 8 ft (244 cm) below the pool surface. Small holes are drilled in the primary coolant suction and return lines about 4 in (10.16 cm) below the pool surface to act as a siphon break preventing water from being siphoned out of the pool and uncovering the core in the unlikely event of a primary coolant line break outside of the pool.

The licensee also stated that it continually monitors the reactor pool for leakage. The operator is required to measure pool water level on a daily and weekly basis. The minimum detectable loss of water is 0.1 in (0.254 cm). Historical records show that the reactor pool level can vary by as much as 0.4 in (1.016 cm) due to evaporation over a 3-day period. The manual gravity-fed makeup water system compensates for small amounts of leakage. Any further drop in water level activates a low-water alarm. Leakage pathways through pipes, fittings, beam ports, and the pool wall would result in leakage into observable areas and would be discovered in a 24-to 72-hour time period; potential leakage would not result in an unmonitored and uncontrolled release outside of the facility.

Because the licensee continuously monitors the pool water level and logs the frequency and quantity of makeup water, detection of pool water leakage would occur. The requirements for pool makeup water to replace water lost to evaporation are well known to the AFRRI staff, and the licensee would investigate off-normal changes. Based on the analysis presented above, the NRC staff concludes that that the licensee has adequate measures in place to maintain the water level above the core.

The licensee stated that, in the event of a leak in the primary cooling system, drains are provided for AFRRI's radioactive waste water system. The facility would retain water in holdup tanks until the leakage water is sampled before its release to the environment. During nonduty hours, AFRRI's guard force has standing orders to notify the reactor on-call person and safety and facilities staff if they detect any water leaks or if they notice puddles in the reactor areas. If a leakage is not observable and if water subsequently leaks to the environment, the periodic monitoring of the pool water ensures that the radioactivity level of the released water would be very low and within the effluent release limits in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation."

The licensee stated that, to ensure reactor pool integrity it performs conductivity measurements monthly of the primary water system and that it tests primary pool water activity monthly on a multiple-channel analyzer to detect any activation or fission products. The conductivity measurements have been demonstrated as an effective method to detect contamination of the pool water. TS conductivity limits and surveillance requirements help ensure that the pool water quality is maintained and any degradation is promptly identified and corrected.

The licensee also stated that visual inspection is performed to further assess the material condition of the fuel, reactor internals and the pool walls. TS 3.7, "Fuel Parameters," and companion surveillance requirements include requirements for the routine inspection of the fuel. Additionally, an U.S. Navy dive team performed an underwater corrosion exam of the pool in

April 2001, and no corrosion was detected. During a site visit, the NRC staff found no visible signs of pool degradation or any evidence of external leakage.

In its RAI response dated August 5, 2016 (Ref. 21), AFRRI stated that abnormally high levels of activity in the water, due to activated contaminates, activated corrosion or ware products, or fission products, would be detected by the myriad of radiation detection equipment in the reactor room including, remote area monitors, continuous air particulate monitors, and a stack gas monitor. Because the water is never directly discharged to the environment, it does not pose a risk to either reactor staff, the public, or the environment. AFRRI Health Physics Department personnel collect and analyze primary and secondary reactor water monthly. The licensee also performs an annual chemical water analysis. Any isotopic analysis result or direct survey that exceeds an Action Level would be brought to the attention of the Reactor Facility Director (RFD) within 24 hours of the posting of results.

The following action levels are compared to the results of primary water and secondary cooling water analysis results:

Reactor Pool Water Action Levels

Type Analysis*	Action Limit 1 (pCi/ml)	Action Limit 2 (pCi/ml)
Gross Alpha	0.1	1
Gross Beta	0.1	1
Activation Products	0.1	1
Fission Products	*	*

* The presence of any fission product shall be brought to the immediate attention of the RSO and RFD. Fission products include Kr-85, Sr-89, Sr-90, Sr-91, Sr-92, Y-90, Y-91, Zr-95, Nb-95, Nb-95m, Nb-97, Mo-99, Ru-103, Ru-106, Rh-103, Rh-106, Ag-111, Sn-125, Sb-125, Sb-127, Te-127, Te-127m, Te-129, Te-129m, Te-132, I-131, I-132, I-133, I-134, I-135, Xe-131, Xe-133, Xe-135, Cs-136, Cs-137, Cs-138, Ba-137, Ba-140, La-140, Ce-141, Ce-143, Ce-144, Pr-143, Pr-144, Nd-147, and Pm-147.

Secondary Cooling Pool Water Action Levels

Type Analysis*	Action Limit 1 (pCi/ml)	Action Limit 2 (pCi/ml)
Gross Alpha	0.1	0.5
Gross Beta	0.1	0.5
Gamma Emitters	*	*

* The presence of any gamma emitter other than naturally occurring radionuclides shall be brought to the immediate attention of the RFD.

The NRC staff finds that the chemical sampling, isotopic analysis, and periodic visual examinations adequately monitor the quality of pool water, the fuel, reactor internals, and the pool walls to help ensure that corrosion mechanisms do not result in pool wall failure; that corrosion and wear products do not foul heat transfer surfaces; and that coolant activity remains in compliance with as low as reasonably achievable requirements and remains below the 10 CFR Part 20 public and occupational exposure limits during normal operation, transient conditions and in the unlikely event of an accident.

TS 3.3 and TS 4.3 specify the reactor pool water quality requirements.

TS. 3.3, “Coolant Systems,” states the following:

Specifications

a. The reactor shall not be operated if the bulk water temperature exceeds 60°C;

b. The reactor shall not be operated if periodic measurements taken IAW TS 4.3 show conductivity of the bulk water greater than 5 micromhos/cm; and

c. Both audible and visual alarms shall be provided to alert the AFRRI security guards and other personnel to any drop in reactor pool water level greater than 6 inches.

d. The reactor shall not be operated if the measurement required by TS 4.3 shows concentrations of radionuclides above the values in 10CFR part 20 appendix B table 2 are found in the primary coolant until the source of the activity is determined and appropriate corrective actions are taken.

TS 3.3 Specification a, requires that the pool water temperature is consistent with the assumptions used in the thermal hydraulic calculations for the limiting hot rod, and assumes a maximum pool water temperature of 60 °C (140 °F). See Section 2.6, “Thermal Hydraulic Design,” of this SER for additional discussion of the thermal hydraulic analysis. The AFRRI thermal hydraulic analysis shows that full power operations are safe with a pool water temperature of 60 °C (140 °F). At a power level of 1MWt, the fuel temperature reaches approximately 400 °C (752 °F). TRIGA fuel has been tested to be safe beyond 1,000 °C (1,852 °F). The overarching objective is to protect the integrity of the fuel. Additionally, TS 3.3 Specification a, also helps ensure proper performance of the water purification system. Limiting pool water temperature to a maximum of 60 °C (140 °F) maintains the effectiveness of the ion exchange resins which in turn, helps control the level of activated contaminants in the pool, a radiological hazard, and helps inhibit corrosion of the fuel cladding, the coolant system piping, reactor internals, and the reactor pool walls. The NRC staff finds that this TS provides assurance that safe operations are maintained.

TS 3.3, Specifications b, requires that the conductivity of the bulk water is maintained at an acceptable level to minimize the corrosion of reactor components and the radiation dose resulting from the activation of contaminates in the reactor coolant. The licensee stated in the SAR, as supplemented, 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. The conductivity limit is a longstanding value for research reactors, which has been shown to be effective in controlling corrosion in aluminum and stainless steel (SS) systems. This limit helps to ensure that the heat transfer between the cladding and coolant will not be degraded due to oxide buildup on the cladding. The limit also helps to maintain the concentration of activation products in the coolant to acceptably low levels so that activation products do not pose a significant radiological hazard. TS 3.3, Specification b, helps ensure that the radioactive content of the primary coolant water remains low in the event of any pool or primary coolant leakage. The NRC staff reviewed TS 3.3, Specification b, and finds that the conductivity limit of 5 micromhos per centimeter (μmhos/cm) is consistent with the guidance in NUREG-1537.

TS 3.3, Specification c, requires both audible and visual alarm signals if the pool water level drops more than 6 in (15.24 cm) The alarms are monitored in the reactor control room and by security guards stationed in the building. Procedures are in place for response to the alarms by both the security guards and the reactor on-call staff. Because the licensee continuously monitors the pool water level and logs the frequency and quantity of makeup water, detection of pool water leakage would occur. The requirements for pool makeup water to replace water lost to evaporation are well known to the reactor staff, and the licensee would investigate off-normal changes.

TS 3.3, Specification d, helps ensure early detection of potentially abnormal pool water activity. The limiting condition of operation (LCO) prohibits reactor operation if radionuclide concentrations of the pool water exceeds the 10 CFR Part 20, Appendix B, Table 2, "Effluent Concentrations," until the source of the activity is identified and appropriate corrective actions are taken. TS 3.3, Specification d, was proposed to be added by the licensee in the September 21, 2016, RAI response (Ref. 78), as an LCO for reactor radioactivity. The corresponding surveillance is discussed below.

TS 3.3, "Coolant Systems," Specifications a through d, covering the coolant temperature, level, conductivity, and activity limit, are consistent with other TRIGA reactors and with the guidance in NUREG-1537. These specifications give the required control to prevent operation of the reactor unless the water level and quality are within specified limits. Therefore, the NRC staff concludes that TS 3.3, Specifications a through d, are acceptable.

TS 4.3, "Coolant Systems," states the following:

> <u>Specifications</u>
>
> a. The pool water temperature, as measured near the input to the water purification system, shall be measured daily, whenever operations are planned.
>
> b. The conductivity of the bulk water shall be measured monthly, not to exceed 6 weeks.
>
> c. The reactor coolant shall be measured for radioactivity quarterly, not to exceed 4 months.
>
> d. The audible and visual reactor pool level alarms shall be tested quarterly, not to exceed 4 months.

TS 4.3 applies to the surveillance requirement for monitoring the pool water and the water-conditioning system. The objective is to ensure the reactor operation is limited to the maximum temperature assumed in the thermal hydraulic analysis and the integrity of the ion exchange resin is maintained thereby maintaining the purity of the reactor pool water, controlling possible radiation hazards from activated impurities in the pool water, and limiting the potential corrosion of fuel cladding and other components in contact with pool water.

TS 4.3, Specification a, requires the licensee to measure the pool water temperature daily at or near the inlet of the purification system. In the February 9, 2016 (Ref. 19), RAI responses, AFRRI stated that the AFRRI staff, as part of procedures, performs calibration for thermocouple and resistance tempature detector [RTD] probes in accordance with the TRIGA TRAKER manufacturer's requirements. The NRC staff concludes that TS 4.3, Specification a, is acceptable.

TS 4.3, Specification b, requires the licensee to measure the conductivity of the pool water monthly to provide timely information of possible changes in the pool water chemistry. In the February 9, 2016 (Ref. 19), RAI responses, AFRRI stated that monthly measurements are sufficient to control conductivity as confirmed by 50 years of acceptable conductivity measurements. Conductivity values over that period were stable at or about 1.2 μmhos/cm and did not vary with reactor usage. NUREG-1537, Appendix 14.1, (Ref. 39) "Coolant Systems," Item b, "Conductivity and pH," provides guidance that monthly conductivity and pH measurements are permitted if the reactor is shutdown for long periods of time or if otherwise justified. Given that conductivity measurements have been monitored over a long period; have shown no variability; have continuously met TS conductivity limits; and have not resulted in observations of corrosion of fuel cladding or reactor component corrosion during visual inspections, the NRC staff finds that monthly conductivity measurements are sufficient to provide adequate assurance that fuel clad corrosion rate will remain slow and neutron activation of dissolved materials will remain within TS limits. The NRC staff concludes that TS 4.3, Specification b, is acceptable.

TS 4.3, Specification c, requires the licensee to measure the radioactivity in the pool water quarterly to detect, in a timely manner, a potential introduction of material into the pool water which has been activated by the neutron flux or the presence of fission products potentially indicative of a fuel cladding failure. AFRRI health physics personnel have historically tested the reactor pool water monthly for radioactivity comparing results to established alert levels provided in AFRRI health physics procedures. TS 4.3, Specification c is the surveillance TS for TS 3.3, Specification d, which requires that the reactor not be operated if measurements required by TS 4.3, Specification c, the LCO that limits reactor radioactivity. Because TS 4.3, Specification c is consistent with NUREG-1537 guidance and the SAR, the NRC staff concludes that TS 4.3, Specification c is acceptable.

TS 4.3, Specification d, provides for the quarterly testing of the pool water level alarm. TS 3.3, Specification d, helps ensure that the reactor is operated only with an operable alarm system. The NRC staff concludes that TS 4.3, Specification d, is acceptable.

TS 3.2.2, Table 2, "Minimum Reactor Safety Systems Scrams," states that the pool water level channel maximum setpoint is 14 ft (4.27 m) from the top of the core, which is equivalent to a drop of approximately 6 in (152 mm) of the pool water. The corresponding surveillance, TS 4.2.2, Specification f, is evaluated below.

TS 4.2.2, "Reactor Safety Systems," states the following:

> <u>Specifications</u>
>
> (…)
>
> f. The low pool water scram shall be tested weekly not to exceed 10 days whenever operations are planned.
>
> (…)

TS 4.2.2, Specification f, requires that the low pool water scram channel required by TS 3.2.2, Table 2, be tested weekly whenever reactor operations are planned. This surveillance requirement and interval is consistent with guidance in with ANSI/ANS-15.1-2007 (Ref. 64). Therefore, the NRC staff concludes that TS 4.2.2, Specification f, is acceptable.

The AFRRI reactor tank has withstood all mechanical loads and stresses from operation without any loss of coolant or other indications that would impact safe reactor operation or the reactor's ability to safely shut down. A review of the design and materials of the reactor tank provides reasonable assurance that it can continue to perform as designed for the period of the license renewal. The penetrations and attachments to the tank to allow for water cooling, purification, and makeup are designed to help ensure safe operation. Design of penetrations below the water level include analyses of potential malfunctions, including loss of coolant. The combination of the aluminum reactor tank and the maintenance of water chemistry will minimize the potential for corrosion. The design of the reactor tank helps to ensure sufficient radiation shielding to protect operating personnel and other facility components.

The NRC staff reviewed the information in the AFRRI license renewal SAR, as supplemented, with regard to pool water level and quality (Refs. 1 and 2). The NRC staff finds that the water level instrumentation and the water quality program are adequate for ensuring that the water level exceeds 14 feet above the core and that the water quality is maintained. In addition, the AFRRI reactor staff monitors the pool water level and would investigate leakage. On the basis of its review the NRC staff concludes that that the risk of a significant release to the environment from pool leakage is acceptably low.

2.4 Biological Shield

The AFRRI reactor core is shielded in the radial direction by the reactor pool water and by concrete, except for the intrusions into the two exposure rooms. Vertical shielding consists of 14 to 15 feet of water above the reactor core and about 1.5 feet of water as well as concrete below it. The concrete separates the reactor tank from the underlying subsoil of the reactor building.

Two lead shielding doors are located in the reactor pool, thus dividing it into two equal sections. When fully closed, the shield doors allow access to one exposure room without significant radiation exposures while experiments are taking place in the other exposure room. The interlocking doors are constructed of aluminum plates and 8-inch aluminum z-sections. Each watertight door is filled with lead shot and transformer oil to fill the gaps between the lead shot. The interlocking end pieces (z-sections) of the shield doors are stepped to prevent radiation streaming. A low-friction thrust bearing supports each door on the bottom.

The shield doors may be rotated 90 degrees to allow the core support carriage to move past. A fractional horsepower drive motor provides the motive force. Limit switches are used to indicate the positions of the doors (fully open or closed). These limit switches are part of the facility interlock system that prevents movement of the core support carriage if the doors are not fully open and that prevents the control rod drives from energizing unless the doors are fully opened or closed. The reactor console indicates the shield door position, and a camera gives the operator in the control room an overhead view of the reactor tank.

TS 3.2.3, "Facility Interlock System," specifies the interlocks and the prevention of reactor operation or movement to prevent inadvertent exposures as follows:

Specifications

Facility interlocks shall be provided so that:

a. The reactor cannot be operated unless the shield doors within the reactor pool are either fully opened or fully closed;

b. The reactor cannot be operated unless the exposure room plug door adjacent to the reactor core position is fully closed and the lead shield doors are fully closed; or if the lead shield doors are fully opened, both exposure rooms plug doors must be fully closed; and

c. The lead shield doors cannot be opened to allow movement into the exposure room projection unless a warning horn has sounded in that exposure room, or unless two licensed reactor operators have visually inspected the room to ensure that no personnel remain in the room prior to securing the plug door.

TS 3.2.3, Specifications a through c, establish the requirements for facility interlocks that give the required protection to prevent inadvertent operation of the reactor unless the shield doors and plug doors are correctly positioned and a determination has been made that personnel have been evacuated from the exposure rooms. The NRC staff finds that these interlocks are consistent with the guidance in NUREG-1537 and prevent the operation and movement of the reactor core into an area until assurance is provided that inadvertent exposure will be prevented. The NRC staff concludes that the interlock requirements specified in TS 3.2.3, Specifications a through c, are acceptable for preventing and limiting inadvertent radiological exposure to the operating staff.

TS 4.2.4, "Facility Interlock System," specifies the surveillance requirements for the facility interlock system as follows:

Specifications

Functional checks shall be made annually, not to exceed 15 months, to ensure the following:

a. With the lead shield doors open, neither exposure room plug door can be electrically opened.

b. The core dolly cannot be moved into region 2 with the lead shield doors closed.

c. The lead shield doors cannot be opened to allow movement Into the exposure room projection unless a warning horn has sounded in that exposure room, or unless two licensed reactor operators have visually inspected the room to ensure that no personnel remain in the room prior to securing the plug door.

TS 4.2.4 specifies the surveillance requirements that ensure the integrity of the facility interlock system. The performance of these surveillance procedures in regard to the functionality and operability of the interlock system is adequate to help ensure its proper operation. The NRC staff concludes that TS 4.2.4, Specifications a through c, are acceptable.

The information in Section 4.13, “Facility Interlock System,” of the SAR (Ref. 2) shows that the shield doors are designed to limit exposures from the reactor so that persons do not exceed the limits in 10 CFR Part 20. Based the information above, the NRC staff finds that the biological shield design is acceptable to minimize radiation streaming from the core as well as the exposure rooms.

The NRC inspection program routinely reviews the licensee’s radiation protection program and independently measures radiation levels in the facility. The NRC staff concludes that the application, as amended, and the results of NRC inspections provide reasonable assurance that the AFRRI biological shield and the shield doors will bound exposures from the reactor and reactor-related sources of radiation to the limits in 10 CFR Part 20.

2.5 Nuclear Design

Section 2.5 of this SER addresses issues associated with nuclear design. The information discussed establishes the design bases for other chapters, especially the safety analyses.

2.5.1 Normal Operating Conditions

The AFRRI reactor operates at a steady-state power level of 1.0 MWt and is licensed for 1.1 MWt. The AFRRI reactor presently contains 85 fuel elements, as described in Section 2.2.1 of this SER. They are placed in five concentric rings labeled B through F. The transient control rod is located at position A, a single position in the center of the fueled rings. The B ring contains six fuel elements. There are three control rods located in the D ring. The F ring contains 30 fuel elements. The core can be positioned within the tank to maximize exposures in either of two exposure rooms located on the north and south sides of the reactor tank.

The following definitions delineate the operational state of the reactor and are used, in part, to determine the applicability of other TS to the facility.

TS 1.0 defines the operational states of the reactor as follows:

(…)

1.27. PULSE MODE

Operation in the pulse mode shall mean that the reactor is intentionally placed on a prompt critical excursion by making a step insertion of reactivity above critical with the transient rod. The reactor may be pulsed from a critical or subcritical state.

(…)

1.29. REACTOR OPERATING

The reactor is operating whenever it is not secured or shutdown.

(…)

1.32. REACTOR SECURED

The reactor is secured when:

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

b. All of the following conditions exist:
   1. All control rods are fully inserted into the core;

   2. The console key switch is in the off position and the key is removed;

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

   4. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding $1.00.

1.33. REACTOR SHUTDOWN

The reactor is shutdown when it is subcritical by at least $1.00 of reactivity in the reference core condition with the reactivity worth of all installed experiments included.

1.42. STEADY STATE MODE

Operation in the steady state mode shall mean operation of the reactor either by manual operation of the control rods or by automatic operation of one or more control rods at power levels not exceeding 1.1 MW. Square wave mode is a subset of the steady state mode of operation.

(…)

These definitions describe the operational states of the reactor. They are standard definitions used in research reactor TSs and are consistent with the definitions in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff finds these definitions acceptable.

2.5.1.1 2.5.1.1 Steady-State Operation

TS 3.1.1 specifies that the licensed reactor steady-state power level should not exceed 1.1 MWt; however, the AFRRI reactor normally operates at a steady-state power of 1.0 MWt.

TS 3.1.1, “Steady State Operation,” specifies the steady-state operating power level as follows:

Specification

The reactor steady state power level shall not exceed 1.1 MW.

TS 3.1.1 helps ensure that the licensee operates the AFRRI reactor with a maximum steady-state thermal power level consistent with the analysis in the SAR. The AFRRI accident analysis results show that, at all power levels up to 1.1 MW, the steady-state maximum fuel temperature in the hottest element is less than 440.7 °C (825.3 °F), which is substantially below the fuel temperature SL of 1,000 °C (1,832 °F).

TS 3.1.1 specifies a steady-state power limit to help ensure that natural convection of the pool water provides adequate cooling for the fuel. (SER Section 2.6 discusses the thermal-hydraulic analysis.) Consistent with the licensee's accident analysis, Table 2 in TS 3.2.2 indicates that a reactor scram would occur at a power level of 1.1 MWt to ensure that the safety limit is not reached, and no fuel element reaches the LSSS temperature of 600 ^{o}C . Based on the discussion above, the NRC staff finds that the requirement in TS 3.1.1 that the reactor power should not exceed 1.1 MW during steady-state operation provides an acceptable margin of safety for operation. On the basis of its review of the information provided above, the NRC staff concludes that TS 3.1.1 is acceptable.

TS 3.2.2 specifies the minimum number of reactor safety system channels that shall be operable for proper operation consistent with the SAR Section 4.11, "Reactor Instrumentation." Additionally, the specification states that the reactor shall not be operated unless the safety systems described in tables 2 and 3 are operable for the specific mode of operation.

TS 4.2.2, "Reactor Safety Systems," Specifications a through e and Specification g, help ensure that the licensee maintains reactor safety system channels as follows (TS 4.2.2, Specification f, is discussed in SER Section 2.3):

<u>Specifications</u>

a. A channel test of the scram function of the high-flux safety channels shall be made each day that reactor operations are planned.

b. A channel test of each of each of the reactor safety system channels for the intended mode of operation shall be performed weekly, whenever operations are planned.

c. Channel calibration shall be made of the NP, NPP, NM1000, NLW, NMP or any other console instrumentation designated to provide direct power level information to the operator, annually not to exceed 15 months.

d. A thermal power calibration shall be completed annually not to exceed 15 months.

e. The emergency stop scram shall be tested annually, not to exceed 15 months.

(…)

g. The console manual scram button shall be tested weekly not to exceed 10 days whenever operations are planned.

As stated in the licensee's RAI response dated August 5, 2016 (Ref. 21), TS 4.2.2, Specification c, is designed to cover power-level monitoring channels, which are defined in the definitions as channels that are intended to provide real-time power-level readings to the operator. In SAR Section 7.1 (Ref. 67), the NM1000 is the digital neutron log power channel. In SAR Section 7.2.1.1 (Ref. 67), AFRRI describes the purpose and function of the NP, NPP, and

NM1000. Reactor power is measured by three separate detectors—a fission chamber serving the operational channel and either ion chambers or fission chambers serving the safety channels. The signal from the fission chamber is used by the NM1000 to provide wide-range log power from 10^{-8} percent to 100 percent reactor power and period indication from -30 seconds to +3 seconds. One ion chamber or fission chamber is connected to the NP-1000 safety channel. A second ion chamber or fission chamber is used by the NPP-1000 percent power and pulsing channel. Both the NP-1000 and NPP-1000 provide indication of linear reactor power from 0 percent to 120 percent steady-state reactor power, and the NPP-1000 also provides indication of reactor power for pulsing operations. The NM-1000 log display gives a continuous indication from 10^{-8} percent to 100 percent of full power for the console display, analog bar graph display, and the console chart recorder. The NP-1000 safety channel provides a linear power signal to the console display and analog bar graph display. These displays are scaled at 0 to 120 percent of full power. A bistable circuit provides scram and alarm functions if the high-power setpoint is exceeded. The detector input to the NP-1000 safety channel is disabled during pulse mode operations.

The definition for safety channel is a high-flux safety channel with scram capability. TS 4.2.2 provides for periodic checks, tests, and calibrations of high-flux channels, reactor safety system channels, and power-level monitoring channels.

Table 2 in TS 3.2.2 specifies the reactor safety system scram functions required for steady-state and pulse operation, including two power safety scrams in steady-state mode. TS 4.2.2, Specification a, helps ensure that the high-flux safety channels, listed in Table 2 of TS 3.2.2, are channel tested each day that the reactor is operated.

TS 4.2.2, Specification b, requires that the licensee perform a channel test for each reactor high-flux safety channels weekly for the intended mode of operation. The licensee states that a channel test of the two high-flux safety channels and a test of interlocks before reactor startup (see Section 5.3.2.2 of this SER), when combined with the channel calibrations required by TS 4.2.2, provide assurance that the high-flux percent power-level measuring channels are providing accurate power-level indications.

TS 4.2.2, Specification c, requires the licensee to calibrate the reactor power-level monitoring channels annually and helps ensure that the power-level information is accurately provided to the operator and that the reactor is operated within the authorized power levels.

TS 4.2.2, Specification d, requires the licensee to perform thermal power calibration (calorimetric) to confirm the calibrations required by TS 4.2.2, Specification c. The licensee stated in the Section 4.11.3, “High Flux Safety Channels One and Two,” of the SAR (Ref. 2) that an annual calibration helps ensure that the power-level measuring channels are accurately indicating the reactor power level.

TS 4.2.2, Specification e, helps ensure the licensee tests the emergency stop scram, which is designed to prevent and cease reactor operations for the exposure rooms and console.

TS 4.2.2, Specification g, helps ensure that the licensee tests the console manual scram button when reactor operations are planned.

The surveillance intervals are sufficient to maintain the accuracy of the safety channels. A channel test of the two power-level safety channels and a test of interlocks before reactor startup, when combined with the channel calibrations, provide assurance that the power-level measuring channels are providing accurate power-level indications that help prevent the reactor power level from exceeding 1.1 MWt.

The NRC staff reviewed TS 4.2.2, Specifications a through e, and g, and finds that these surveillance requirements and their intervals are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64) and with the requirements and intervals used at similar research reactors. TS 4.2.2, Specification f, is discussed in Section 2.3 of this SER. These specifications provide assurance that component degradation and failure will be detected in a timely manner and that operability and performance will be as required. On the basis of its review of the information provided above, the NRC staff concludes that TS 4.2.2, Specifications a through e and Specification g, are acceptable.

2.5.1.2 <u>2.5.1.2 Pulse Mode Operation</u>

The licensee stated that the AFRRI TRIGA reactor is designed to be pulsed from a low to a high power level by the rapid insertion of reactivity. In this mode of operation, the maximum reactivity insertion is limited to that which will limit the peak fuel temperature to 830 °C (1,526 °F). In addition, the facility may not initiate the pulse from a core power in excess of 1 kWt. A required interlock, TS 3.2.2, Table 2, “Minimum Reactor Safety Systems Scrams,” prevents the transient rod from firing, prevents pulsing from a power level greater than 1 kWt. In Section 4.14, “Scram Logic Circuitry,” of the SAR (Ref. 2), the licensee stated that the pulse timer causes a reactor scram in pulse mode at the time set on the timer (not less than 15 seconds) or an automatic software scram timeout at 15 seconds. Table 3, “Minimum Reactor Safety System Interlocks,” requires an interlock to prevent pulsing when the reactor power level is 1 kWt or above. The footnote to TS 3.2.2, Table 3, states that the reactor safety system interlocks shall be tested daily whenever operations involving these functions are planned. SAR Section 7.3.3, “Interlocks” (Ref. 67), and Section 4.11, “Reactor Instrumentation” (Ref. 2), describe the interlocks. Tables 2 and 3 of TS 3.2.2 are discussed in Chapters 2 and 5 of this SER. The AFRRI’s pulse mode operation is similar to other TRIGA reactors with pulse capability.

TS 3.1.2 applies to the peak fuel temperatures in the reactor that result from a rapid insertion of reactivity to help ensure that fuel element damage does not occur.

TS 3.1.2, “Pulse Mode Operation,” specifies the limitation on pulse mode operation as follows:

> <u>Specification</u>
>
> The maximum step insertion of reactivity shall be \$3.50 (2.45% $\Delta k/k$) in pulse mode.

TS 3.1.2 helps ensure the maximum allowable reactivity addition for pulsing, so that the reactor may be safely pulsed without fuel damage. GA performed many tests with step insertions up to \$5.00 before any fuel damage became apparent; therefore, the TS limit of \$3.50 is well within the safety envelope established by GA.

2.5.2 Reactor Core Physics Parameters

The peak fuel temperature limitation and its associated maximum reactivity insertion for pulsing will help ensure that the reactor can be safely pulsed without fuel damage. The large prompt negative temperature coefficient of reactivity of the U-Zr_x fuel moderator supplies a basis for safe operation of the reactor in the pulse mode. Pulse capability is limited to help ensure that the fuel temperature stays below the SL of 1,000 °C (1,832 °F) for standard fuel, as discussed in this SER Section 2.5.3, and that the peak fuel temperature remains below 830 °C (1,526 °F). NUREG-1537 and GA recommend a fuel temperature limit of 830 °C (1,526 °F) during pulsing to ensure that no fuel damage occurs because of internal pressure caused by hydrogen migration (Ref. 69). The limit of 830 °C (1,526 °F) was established following minor fuel damage experienced at the Texas A&M University TRIGA reactor. Calculations using the BLOOST code (Ref. 70) show that the maximum step insertion of $3.50 leads to a maximum fuel element temperature of less than 830 °C (1,526 °F). The NRC staff finds that licensee's calculations of the peak fuel temperature limit and the associated maximum reactivity addition limit for pulsing help ensure that the reactor can be safely pulsed without fuel damage and, therefore, concludes that these pulse reactivity addition limits in the TSs are acceptable. Reactor Core Physics Parameters

The safe operation of a TRIGA reactor during normal operations is accomplished by the control rods and is monitored accurately by the core power level (neutron) detectors. The licensee stated that an important safety feature of a TRIGA reactor is the reactor core's inherent large prompt negative temperature coefficient of reactivity resulting from an intrinsic molecular characteristic of the U-ZrH_x matrix at elevated temperatures. The prompt negative temperature coefficient results principally from the neutron-hardening properties of the fuel matrix at elevated temperatures, which increases the leakage of neutrons from the fuel-bearing material into the water moderator material that absorbs them preferentially. This reactivity decrease is a prompt effect because the fuel and zirconium hydride are mixed homogeneously. Therefore, the zirconium hydride temperature rises essentially simultaneously with the fuel temperature, which is directly related to reactor power. An additional contribution to the prompt negative temperature coefficient is the Doppler broadening of U-238 resonances at high temperatures, which increases nonproductive neutron capture in these resonances.

As a result of the large prompt negative temperature coefficient, the fuel matrix will rapidly and automatically compensate for a step insertion of reactivity that results in an increasing fuel temperature. This compensation can terminate the resulting power excursion without any dependence on the electronic or mechanical reactor safety systems or on the actions of the reactor operator. In addition, the fuel matrix can rapidly compensate for changes of reactivity that result in a change in fuel temperature during steady-state operation, thus limiting the reactor steady-state power level (Ref. 62). This inherent characteristic of the U-ZrH_x fuel has been the basis for designing TRIGA reactors with a pulsing capability as a normal licensed mode of operation.

Table 2-2 lists the major reactor core physics parameters computed for the AFRRI TRIGA reactor.

Table 2-2 Reactor Core Physics Parameters

Parameter	Beginning of Life	End of Life
Peak thermal neutron flux	2.45×10^{13} n/cm^2-s	
Shutdown margin	\$2.69	\$6.12
Prompt neutron lifetime	41.4 μs	39.5 μs
Void coefficient	-0.080% Δk/k per 1% void	
Prompt negative temperature coefficient *	-1.10×10^{-4} Δk/k/°C at ~700 °C	-1.13×10^{-4} Δk/k/°C at ~700 °C
Effective delayed neutron fraction	0.0075	0.0068

* prompt negative temperature coefficients are quoted from Tables 4-12 and 4-13 in the AFRRI SAR.

The main reactor core physics parameter discussed in Section 4.16.3, “Negative Temperature Coefficients of Reactivity,” of the SAR (Ref. 2) is the very large prompt negative temperature coefficient of reactivity. The prompt negative temperature coefficient of reactivity was measured to be approximately -0.011% Δk/k/°C at a temperature of approximately 700 °C (1,292 °F), which is a value typical of other TRIGA cores. Therefore, the fuel will compensate for any large (greater than \$1.00) step insertion of excess reactivity, thus causing the transient to end.
The thermal neutron flux for the AFRRI reactor 85-3 core (85 fuel elements and three control rods) operating at 1.1 MWt is approximately 2.45×10^{13} neutrons per centimeter squared per second, which is consistent with other 1-MWt research reactors. The calculated neutron lifetime was about 41 microseconds at beginning of life. The best estimate value for the effective delayed neutron fraction for the 85-3 core was found to be *ßeff* = 0.0075 at beginning of life. The licensee stated that the calculated void coefficient was -0.08% Δk/k per % water void. With the exception of shutdown margin, Table 2-2 indicates no significant changes to the calculated reactor parameters with burnup. The increase in shutdown margin with burnup is consistent with TRIGA behavior as the uranium in the core is used up. The reactor core parameter values are similar to those of other TRIGA reactors. The licensee discussed the core physics parameters and the methods used to determine them. The NRC finds this information, and the licensee’s analysis, to be consistent with the methodologies in the guidance in NUREG-1537.

TS 4.1, “Reactor Core Parameters,” Specification d, establishes the surveillance requirement for the power coefficient as follows:

> <u>Specifications</u>
>
> (…)
>
> d. The power coefficient of reactivity at 100 kW and 1 MW shall be measured annually, not to exceed 15 months.

TS 4.1, Specification d, establishes requirements for surveillance of the power coefficient to help ensure that reactivity insertions are consistent with the required power level.

The NRC staff reviewed the licensee’s analyses, as discussed above, and finds that the licensee considered appropriate core physics parameters and has established an adequate surveillance TS that helps ensure reactivity insertions are consistent with the required power level and the safety analysis. On this basis, the NRC staff concludes that the values of the core physics parameters and TS 4.1, Specification d, are acceptable.

2.5.3 Operating Limits

The regulation in 10 CFR 50.36(d)(1) requires reactors to specify SL and LSSSs. The regulation in 10 CFR 50.36(c)(1) defines SLs as limits on important process variables that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. In addition, 10 CFR 50.36(c)(1) defines LSSSs for nuclear reactors as settings for automatic protective devices related to those variables that have significant safety functions. If a licensee specifies an LSSS for a variable on which it has placed a SL, it must choose the setting such that automatic protective action will correct the abnormal situation before the facility exceeds a SL.

TS 2.1, “Safety Limit: Fuel Element Temperature,” states the following:

> <u>Specification</u>
>
> The maximum temperature in a TRIGA fuel element shall not exceed 1,000°C under any mode of operation.

TS 2.1 helps ensure that the maximum fuel temperature for standard TRIGA fuel is limited to prevent possible damage to the fuel.

The licensee stated in Section 4.11.4, “Fuel Temperature Safety Channels,” of the SAR (Ref. 2) that a key parameter for a TRIGA reactor is the fuel element temperature. This SL specification can be measured directly using an IFE. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding if the fuel temperature exceeds the SL. The pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of the hydrogen and zirconium in the moderator. The magnitude of this pressure is determined by fuel moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The licensee also stated in Section 4.14, “Scram Logic Circuitry,” of the SAR (Ref. 2) that the SL for the standard TRIGA fuel element is based on data that includes the large amount of experimental evidence obtained during high-performance reactor tests on this fuel. The SL for the reactor fuel is based on data indicating that the stress in the cladding, resulting from the hydrogen pressure of the disassociation of zirconium hydride will remain below the ultimate yield stress (clad integrity failure), provided that the temperature of the fuel does not exceed 1,150 °C (2,102 °F), and the fuel cladding is below 500 °C (932 °F). (Ref. 62). Based on analysis and experiments performed by GA, a suggested conservative SL is 1,150 °C (1,150 °F). The temperature SL for the AFRRI reactor standard TRIGA fuel under any mode of operation is set at 1,000 °C (1,832 °F). The NRC staff evaluated the properties and performance of standard TRIGA low-enriched uranium (LEU) fuel in NUREG-1282 (Ref. 61) and approved the fuel for use with the provision that case-by-case analysis discusses individual reactor operating conditions when using the fuel. The AFRRI reactor SL is consistent with the fuel temperature found acceptable in NUREG-1282 (Ref. 61).

An additional consideration is the need to provide adequate cooling relative to the maximum heat flux to prevent departure from nucleate boiling (DNB) and the resulting rapid increase in clad temperature, which will lead to failure of the clad (SER Section 2.6). A power-level limit is calculated that ensures that the fuel temperature SL will not be exceeded and that film boiling will not occur. The design-bases analysis has shown that operation at a thermal power level of 1.1 MWt (the power level at which a reactor scram occurs) across a broad range of core and coolant inlet temperatures with natural convection flow will not lead to film boiling.

The NRC staff reviewed the licensee's analysis of the fuel SL, as discussed above, and finds that TS 2.1 helps ensure that the maximum fuel element temperature SL for the AFRRI reactor LEU fuel is consistent with the SLs used for other TRIGA reactor fuel elements (supported by research conducted by GA) and has been previously approved by the NRC staff (e.g., Texas Engineering Experiment Station/Texas A&M University Nuclear Science Center Reactor, 1.0 MWt SL is 1150 °C (2100 °F) (Ref. 69). Based on the information above, the NRC staff finds the fuel element temperature SL acceptable.

The licensee stated in Section 4.14, "Scram Logic Circuitry," of the SAR (Ref. 2) that the LSSS is the measured IFE temperature that, if exceeded, will initiate a reactor scram to prevent exceeding the fuel element temperature SL. For the AFRRI TRIGA reactor, the LSSS is set equal to, or less than, 600 °C (1,112 °F), as measured at the IFE at specific locations in the core. The location of the IFE is important for ensuring that the hottest fuel location in the reactor core is protected from excessive temperature. A relationship between the measured temperature in the IFE and the actual temperature at the fuel hot spot in the core has been determined to show that the setting of 600 °C (1,112 °F) protects the SL at the hottest point in the core. The IFE contains three axial thermocouples that measure the fuel temperature in a vertical distribution.

TS 2.2, "Limiting Safety System Setting for Fuel Temperature," specifies the LSSS for fuel temperature as follows:

> Specification
>
> The limiting safety system setting shall be equal to or less than 600°C, as measured in the instrumented fuel elements. There shall be two fuel temperature safety channels. One channel shall utilize an instrumented fuel element in the B ring, and the second channel shall utilize an instrumented fuel element in the C ring.

TS 2.2 specifies the acceptable locations for the IFE and the temperature limit as measured by any one of the three thermocouples in the IFE. One of the IFEs must be located in the B ring, which contains fuel elements with the highest power density in the core. However, the location of the IFE may not be the same as the location of the hottest fuel element in the reactor core. The other IFE must be located in the C ring, which is the location of the next highest power density fuel elements. This configuration helps ensure that the IFE temperature measurements are close to the hot rod temperatures. Calculations performed by the licensee assume that one of the IFEs is located in the lowest power density position in either the B or C ring. The calculations show that an indicated temperature of 600 °C (1,112 °F) on the fuel temperature measuring channel corresponds to a peak fuel temperature of 611 or 628 °C (1,162.4 °F) with the IFE in the B or C ring, respectively (Ref. 14). The allowed locations are chosen based on calculations performed for the AFRRI reactor core to ensure that the hottest location is protected not only against exceeding the SL, but also against the DNB. The NRC staff reviewed the calculation methodologies and results and finds them reasonable and acceptable.

The NRC staff finds that the LSSS provides a safety margin of about 400 °C (752 °F) to the fuel element SL of 1,000 °C (1,832 °F), which is a significant margin between true and measured temperatures that result from thermocouple uncertainties and core location variations. The licensee has shown that, for both the hottest and the coldest thermocouples, an IFE located in the above core positions would protect against the fuel temperature SL and would limit the steady-state temperature in the fuel to less than 628 °C (1,162.4 °F). The NRC staff concludes that the LSSS of 600 °C (1,112 °F) is sufficient to protect the fuel temperature SL and is consistent with NUREG-1537 guidance on the SL and LSSS; therefore, the LSSS is acceptable. The NRC staff finds that TS 2.2 helps ensure that an LSSS is established to protect the fuel cladding integrity and prevent the release of fission products. The LSSS value in TS 2.2 is supported by the SAR, as supplemented, and provides a substantial margin of safety. Based on the information above, the NRC staff concludes that TS 2.2 is acceptable.

2.5.3.1 2.5.3.1 Pulse Operation

The licensee indicates in the license renewal SAR, as supplemented, that the LSSS is applicable not only in steady-state operation but also in pulse mode. However, the temperature channel will not limit the peak power generated during a pulse because the response time of the temperature channel is relatively slow as compared to the width (time) of a pulse. The temperature scram would limit the total amount of energy generated in a pulse by cutting off the tail of the energy transient in the event that the fuel temperature limit was met or exceeded. Thus, the fuel temperature scram provides an additional degree of safety in the pulse mode of operation by limiting the total amount of energy generated in a pulse. In addition, a pulse timer will also initiate a reactor scram within 15 seconds after the initiation of the pulse, thus further reducing the power level and the total generated energy during the pulse.

Pulse capability is limited to help ensure that the peak fuel temperature in pulse operating mode remains below 830 °C (1,529.6 °F). GA recommends the fuel temperature limit of 830 °C (1,529.6 °F) during pulsing to ensure that no fuel damage occurs. TS 3.1.2 limits the maximum allowable reactivity insertion to help ensure that the facility does not exceed the 830 °C (1,529.6 °F) temperature limit. (Section 2.5.1 of this SER further discusses the pulsing limit.)

The NRC staff reviewed the licensee’s analyses, as discussed above. The NRC staff finds that the SL and LSSS for the AFRRI reactor are based on acceptable analytical and experimental investigations and are consistent with those approved by the NRC staff and used at other TRIGA reactors. On this basis, the NRC staff concludes that the LSSS of 600 °C (1,112 °F) and the accompanying conditions for IFE placement in the core are sufficient to prevent the fuel temperature from exceeding the SL and, therefore, are acceptable.

2.6 Thermal-Hydraulic Design

2.6.1 Thermal-Hydraulic Design

The guidance in NUREG-1537 instructs the licensee to perform a thermal-hydraulic analysis to demonstrate that the LSSS setpoints (and LCO setpoints, if necessary) help ensure fuel integrity under all conceivable operating conditions. Because the AFRRI TRIGA is a pulse reactor, the licensee must perform thermal-hydraulic analysis for the steady-state and pulse modes of operation of the reactor and also present the corresponding LSSS setpoints for both modes of operation. The licensee must demonstrate that during normal and pulse operation, the maximum fuel temperature remains below the SL using suitably conservative analysis and assumptions. Another important parameter in thermal-hydraulic design is the critical heat flux, which describes the heat flux associated with the DNB. The parameter of interest is the DNB ratio (DNBR), which is the ratio of the critical heat flux at the calculated conditions to the maximum heat flux generated by the highest power fuel element at the LSSS or LCO power. NUREG-1537, Section 4.6 (Ref. 39) acceptance criteria suggest that this ratio should be no less than 2.0. The SL fuel temperature must also be satisfied.

The licensee presented an analysis of the AFRRI TRIGA DNBR using the Reactor Excursion and Leak Analysis Program 5 (RELAP5)-3D (Ref. 2). The methodology and computer codes used for this analysis have been compared to actual measurements and consistently produce conservative results. The NRC staff considers the use of RELAP5 with the Bernath correlation to determine the critical heat flux (and thus departure from nucleate boiling) to provide a best estimate analysis of DNBR. The licensee's thermal-hydraulic design report describes the basic features of the licensee-supplied model used in its analysis. Figure 2-2, reproduced from that report, illustrates the basic features of the model, which includes a hot channel with its associated coolant and heat structure (the hot fuel element). There is also an average channel that represents the remaining fuel, fuel followers, and the associated coolant inside the reactor core. The channels are connected to a riser section that directs the coolant flow away from the top of reactor core into the reactor tank and then into the downcomer structure. The returning flow is directed into the reactor core near the bottom of the reactor tank.

The model assumes that there is no crossflow between adjacent channels. This assumption is conservative because higher values of temperature and lower margins to the DNBR are typically predicted when crossflow between adjacent channels is ignored. RELAP5-3D is used to calculate the steady-state natural circulation flow through the water channels adjacent to the fuel elements. The code also calculates all radial heat transfer from the fuel element into its associated coolant channel. Accordingly, the code calculates the cladding and fuel axial and radial temperature distributions as well as the axial temperature distribution in each coolant channel.

The RELAP5 model used by the licensee is based on a circular arrangement of fuel elements. The model also includes a representation for the shroud, which extends 5 feet (1.5 meters) from the core top grid plate. The water exiting the top of the core shroud enters the reactor pool water. The pressure at the top of the reactor core is sum of atmospheric pressure at top of the pool plus the hydrostatic pressure associated with the water column above the core at this elevation. The core height plus the adapter height provide the buoyancy head for natural circulation of water inside the reactor tank and the core region.

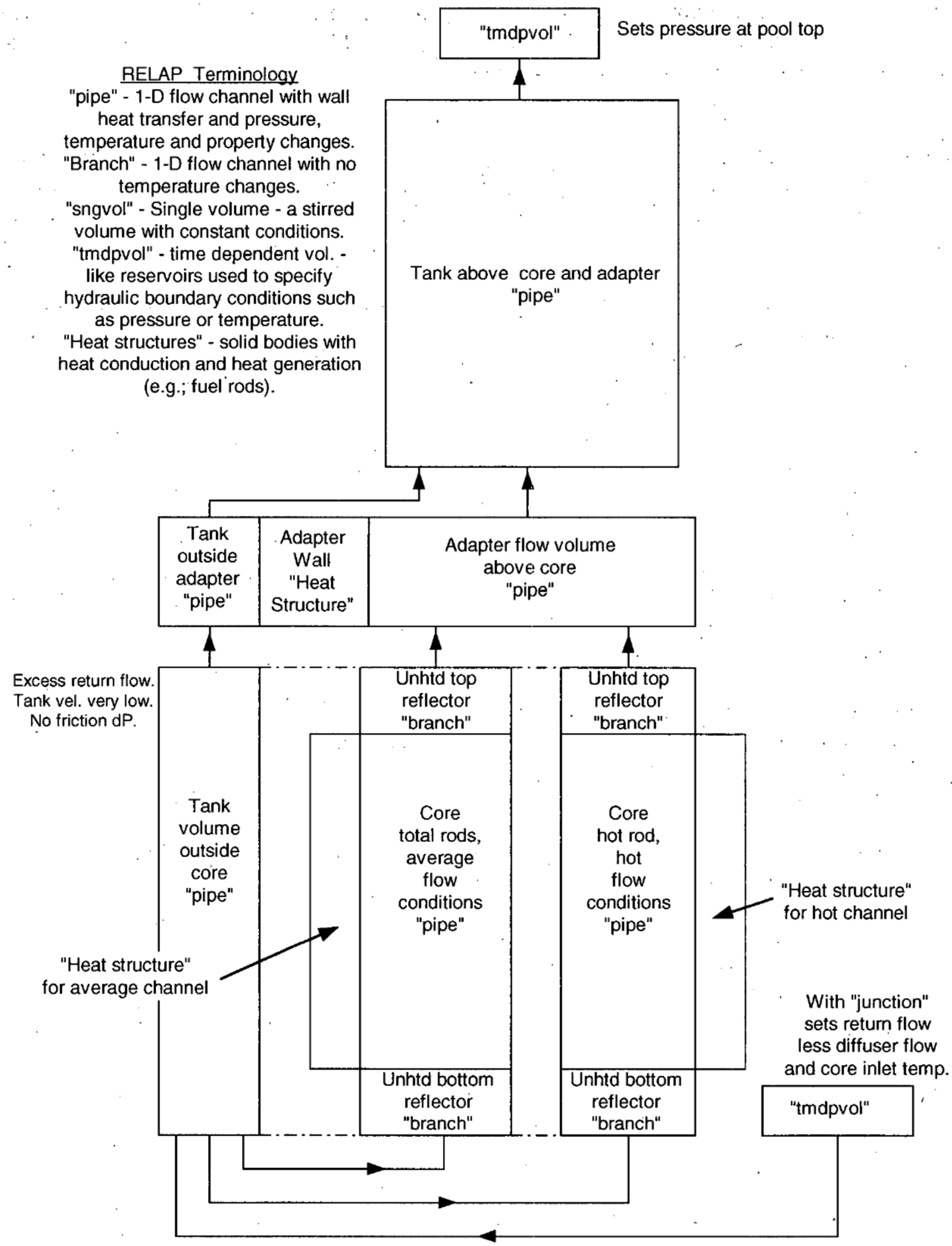


Figure 2-2 The AFRRI TRIGA RELAP5 model

The licensee used the MCNPX code to characterize the fuel element power distribution for the limiting core condition. In this model, 85 fuel elements and three FFCRs were loaded in what is termed the 85-3 core. Figure shows the fuel element location layout. Figure 2-4, revised from the MCPNX output, confirms the fuel loading for the 85-3 core.

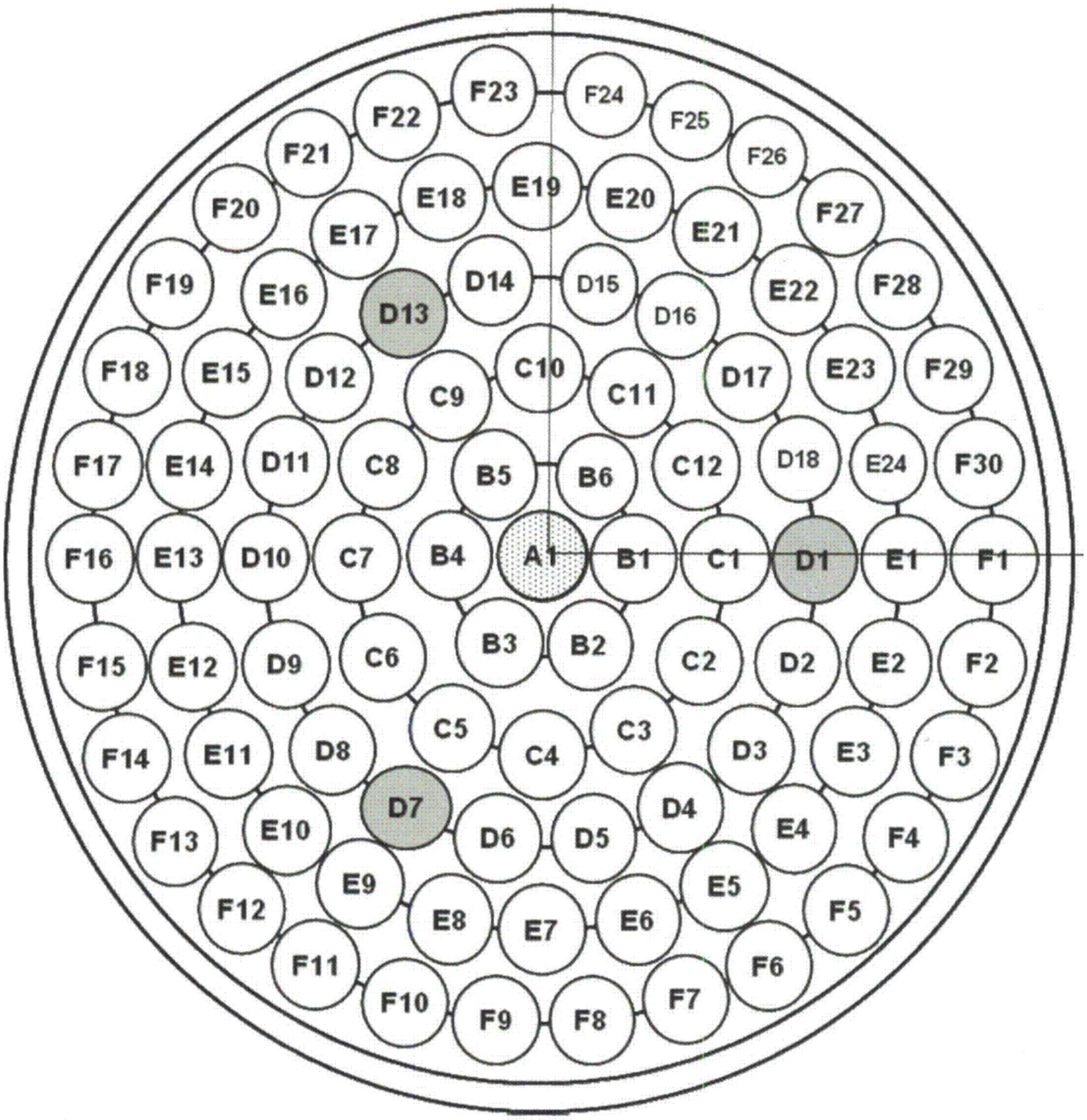


Figure 2-3 AFRRI fuel element location map

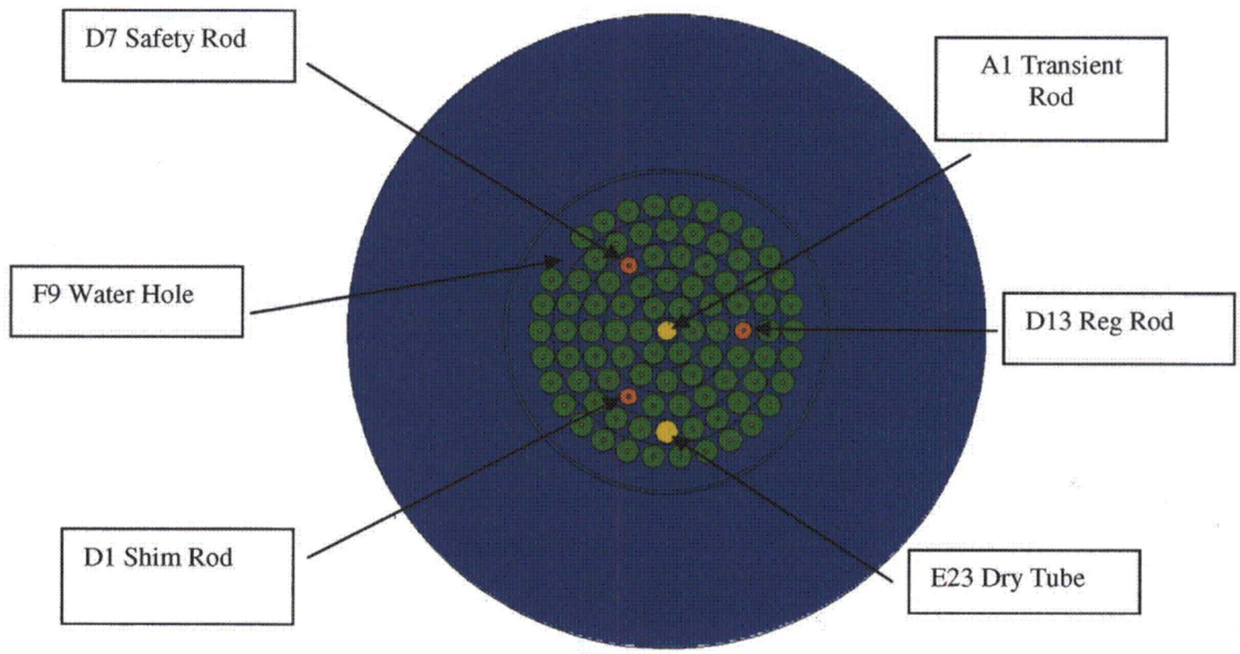


Figure 2-4 The 85-3 core loading

The licensee calculated the hot channel geometry as follows (see Figure 2-4):

> The tri-cusp area between the core center rod, A-1, and two adjacent B ring rods forms the minimum flow area in the core and includes the maximum power rod. The three rods that form this tri-cusp flow area constitute one-half of a rod. Additionally, the heated perimeter for this flow area represents only one-third of a rod because the central location A-1 is not a fuel rod. Customarily, this area is doubled to represent the flow area and hydraulic diameter for a whole rod. The flow area then represents two-thirds of a heated rod. Input to RELAP5 for the "maximum rod" is two-thirds of a rod surface area and two-thirds of the hot rod power.

Although it is true that the central thimble provides no power for the cell described, this is the limiting channel to consider for the DNBR analysis (Figure 2-5). Scaling the flow area and power to a full fuel element shows that this channel is more limiting than the channels with more fuel elements. This is further discussed below in Section 2.6.1 of this SER under “Confirmatory Analysis.”

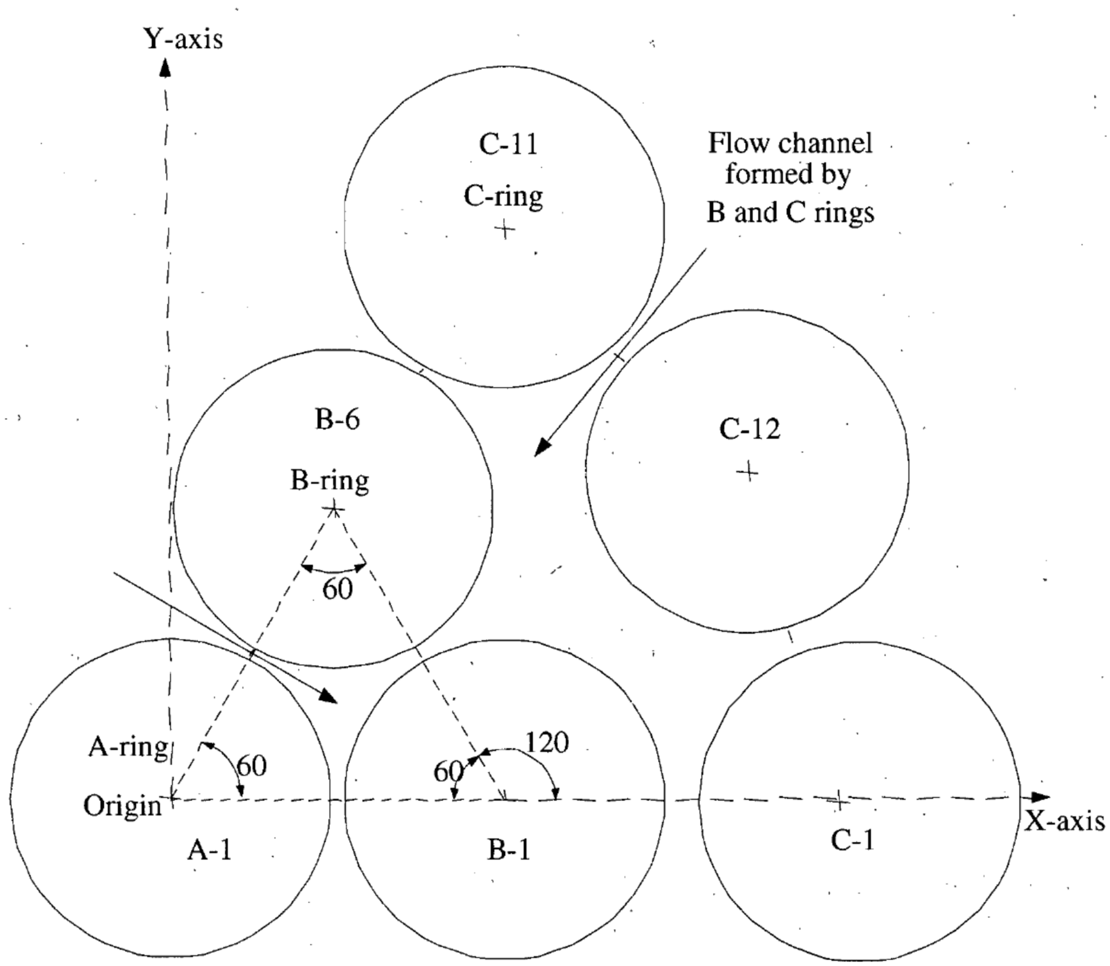


Figure 2-5 AFRRI hot coolant channel model

The RELAP5 model of both the average and hot fuel elements contains the heated fuel and the unheated sections below and above the heated section. These unheated sections represent the reflectors and the fuel element end fittings. A single inlet flow loss coefficient represents the flow losses associated with the bottom grid plate and the bottom fuel element fittings, which is estimated by calculating the individual contraction and expansion losses from the tank through the lower core distribution plate into the reactor core. The sum of these losses is then converted to the single inlet loss coefficient based on the flow area. A similar calculation is performed for the exit loss coefficient that represents the hydraulic losses due to flow expansions and contractions from the top of the fuel element, through the grid plate and into the adapter flow region.

The applicant stated that the fuel element is 1.475 in (3.75 cm) in diameter, and the fuel meat is 15 in (53 cm) long. The fuel element is represented in the heat structure by 25 discrete axial nodes and 20 discrete radial nodes. The NRC staff notes that this radial dimension is different by 0.002 in (0.1 cm) from the AFRRI drawings which show this dimension to be 1.473 in (3.74 cm). Information obtained from GA indicates that it should be 1.478 in (3.76 cm). This is discussed more fully under DNBR - Confirmatory Analysis below.

Power peaking factors for each core configuration are analyzed using MCNPX. The nuclear cross-sections were based on Evaluated Nuclear Data File (ENDF)/B VII data included in the MCNPX data libraries. The highest power fuel element for each configuration is determined by calculating the total power produced in each fuel element present in the configuration. After the highest power fuel element is determined, further analyses are performed to find the detailed axial and radial power shapes associated with that fuel element. The axial power shapes are applied to the fuel element nodes.

To evaluate the DNBR for the AFRRI TRIGA core, the licensee used the Bernath critical heat flux correlation. The NRC staff finds that the analysis performed by the licensee used qualified calculation methods and conservative or justifiable assumptions, and that this analysis is typically used for TRIGA reactor and is therefore acceptable.

2.6.2 Steady-State Operations

The licensee carried out the steady-state thermal-hydraulic analysis of the AFRRI TRIGA reactor using the AFRRI circular core configuration as the basis. The AFRRI TRIGA reactor is located in a nonpressurized tank that contains 15,000 gallons (57,000 liters) of light, demineralized water. The analysis assumed the reactor power to be the maximum allowable, 1.1 MWt, and the reactor inlet temperature to be 48 °C (118 °F). It also assumed that the core is cooled by the natural circulation of the water through the core and that the circulation disperses the heat to the rest of the water in the tank. The column of water surrounding the core establishes the driving force for the natural convection flow rate.

The licensee used the RELAP5 computer code—a widely used code for power and research reactors—to calculate the coolant and fuel element temperatures in the AFRRI reactor. The code also computes the radial thermal fluxes through the fuel element to the adjacent vertical water coolant channel in natural circulation flow. The code provides the clad, fuel, and zirconium rod temperatures and determines the axial distribution of water temperature in the coolant channels.

The RELAP5 model for the AFRRI TRIGA reactor core contains two separate fuel elements and their corresponding flow channels. An “average fuel element” represents the average of the 88 fuel elements in the entire core (85 standard fuel elements and three FFCRs), and a “maximum fuel element” represents the hottest fuel element in the core. The RELAP5 model calculates the “average fuel element” flow area as the total flow area divided by the number of fuel elements in the core. The wetted perimeter and the heat source are taken as those for a single element multiplied by the number of fuel elements. For the “hot fuel element,” those in the B ring of the reactor core, the RELAP5 model accounts for the fact that the central core location contains the nonheated transient control rod. The channel geometry used is the one with the smallest fuel element-to-element pitch (distance between two fuel element centers) and the one with the smallest flow area. The model does not assume any heat transfer or crossflow between the two representative fuel elements. The licensee neglected crossflows based on previous work as quoted in the SAR, which shows that neglecting crossflow conservatively decreases the DNBR. The licensee used the Bernath correlation to predict the critical heat flux. The Bernath critical heat flux correlation is the more conservative of any of the correlations that are applicable to TRIGA cores because the its predictions are closer to the departure from nucleate boiling ratio than other correlations. The NRC staff finds that the licensee’s approach is consistent with guidance in NUREG-1537 and accepts the correlation for use in TRIGA reactor thermal-hydraulic analysis.

The core model includes the geometry of the aluminum shroud above the top grid plate. The core height plus the adapter height provide the buoyancy head for natural circulation flow through the core. The hydraulic model of flow through the core uses standard engineering practice for inlet, contraction, and expansion losses and for friction losses. Input to RELAP5 included radial and axial heat source distributions within the fuel. The licensee used the Diffusion Theory 3 Dimensional (DIF3D) code to compute the fuel element power factors. The DIF3D code is widely used to perform neutronic calculations, including the prediction of power distributions in research and power reactors.

The licensee performed RELAP5 thermal-hydraulic calculations for the core at the maximum power of 1.1 MWt with a core water inlet temperature of 48 °C (118 °F). Table 2-3 summarizes the licensee-calculated results.

Table 2-3 AFRRI TRIGA Thermal Results Summary for a 1.1-MW Core

Parameter	Initial Core
Axial peaking factor—average element	1.316
Axial peaking factor—hot element	1.343
Hot element power factor	1.560
Inlet coolant temperature	48 °C (118 °F)
Coolant saturation temperature at core inlet	110.3 °C (230.5 °F)
Exit coolant temperature—average element	67.11 °C (152.8 °F)
Exit coolant temperature—hot element	82.51 °C (180.5 °F)
Average temperature in pool above core	60.2 °C (140.4 °F)
Coolant mass flow	13.60 kg/s (107,900 lb/h)
Average flow velocity	29.48 cm/s (0.967 ft/s)
Core average fuel temperature	247.1 °C (476.7 °F)
Peak fuel temperature in average fuel element	360.0 °C (679.9 °F)
Maximum wall temperature in hot element	149.2 °C (300.6 °F)
Peak fuel temperature in hot fuel element	440.7 °C (825.3 °F)
Average heat flux	27.87 W/cm^2 (88,362 BTU/h-ft^2)
Maximum heat flux in hot element	58.40 W/cm^2 (185,125 BTU/h-ft^2)
Reactor Power at Minimum DNBR of 1.0	1.99 MWt

Section 4.7.1 of the SAR (Ref. 2) states that the DNBR is 1.0 at a power level of 1.99 MWt. In addition, the licensee used the RELAP5 code to predict the steady-state performance of the fuel elements at the maximum allowable operating power level of 1.1 MWt. The RELAP5 calculations predict a peak fuel temperature in the core of 440.7 °C (825 °F) at 1.0 MWt and less than 500 °C (932 °F) at 1.1 MWt, which leaves a large margin to the LSSSs of 600 °C (1,112 °F) and a larger margin to the SL of 1,000 °C (1,832 °F).

The NRC staff finds that the analysis performed by the licensee used acceptable calculation methods and conservative or justifiable assumptions. The NRC staff also finds that the results of the thermal hydraulic calculations yield conservative values, and are reasonable and acceptable. Therefore, the NRC staff concludes that the thermal-hydraulic analysis in the AFRRI license renewal SAR, as supplemented, demonstrates that the AFRRI 85-3 core results in acceptable safety margins with regard to thermal-hydraulic conditions.

2.6.3 NRC Staff Analysis of Limiting Conditions

The NRC confirmatory calculations used both the TRACE and RELAP5/MOD3.3 computer codes to represent the AFRRI reactor's thermal-hydraulic behavior using components that represent the major features, such as the reactor core, the downcomer, and the core adapter region that acts as a chimney above the core and enhances natural circulation. This analyses use a one-dimensional noninteracting representation of a hot and an average channel (i.e., N-1 channel) inside the core based on RELAP5 and TRACE "PIPE" components (hereafter referred to as the "pipe model"), as shown in Figure 2-6.

In the models (both TRACE and RELAP5) used for the NRC analyses, the reactor core is represented by two noninteracting (except at the inlet and exit to the reactor core) PIPE components and their associated heat structures, in which a hot channel and its associated heat structure represent the fuel element and coolant channels of the highest power, while the N-1 channel and its associated heat structure represent the remaining fuel elements. These two PIPE components are represented with the conservative assumption that neglects crossflow effects between the two core channels. The model uses inlet and outlet flow loss coefficients consistent with what the licensee used.

Each heat structure for the fuel element is represented by 20 axial nodes and 24 radial nodes. Each PIPE component is represented by 24 axial cells in which cells 3–22 are connected to the corresponding heat structure cells representing the active fuel region. The remaining cells represent the unheated axial reflector portions of the fuel element.

One difference in the NRC model from the licensee model is that the flow holes connecting the downcomer to the adapter are modeled. These flow holes allow cold water from the downcomer region to enter into the adapter chimney flow, which reduces the buoyancy head of the natural circulation flow system.

(Top of Figure 2-6)

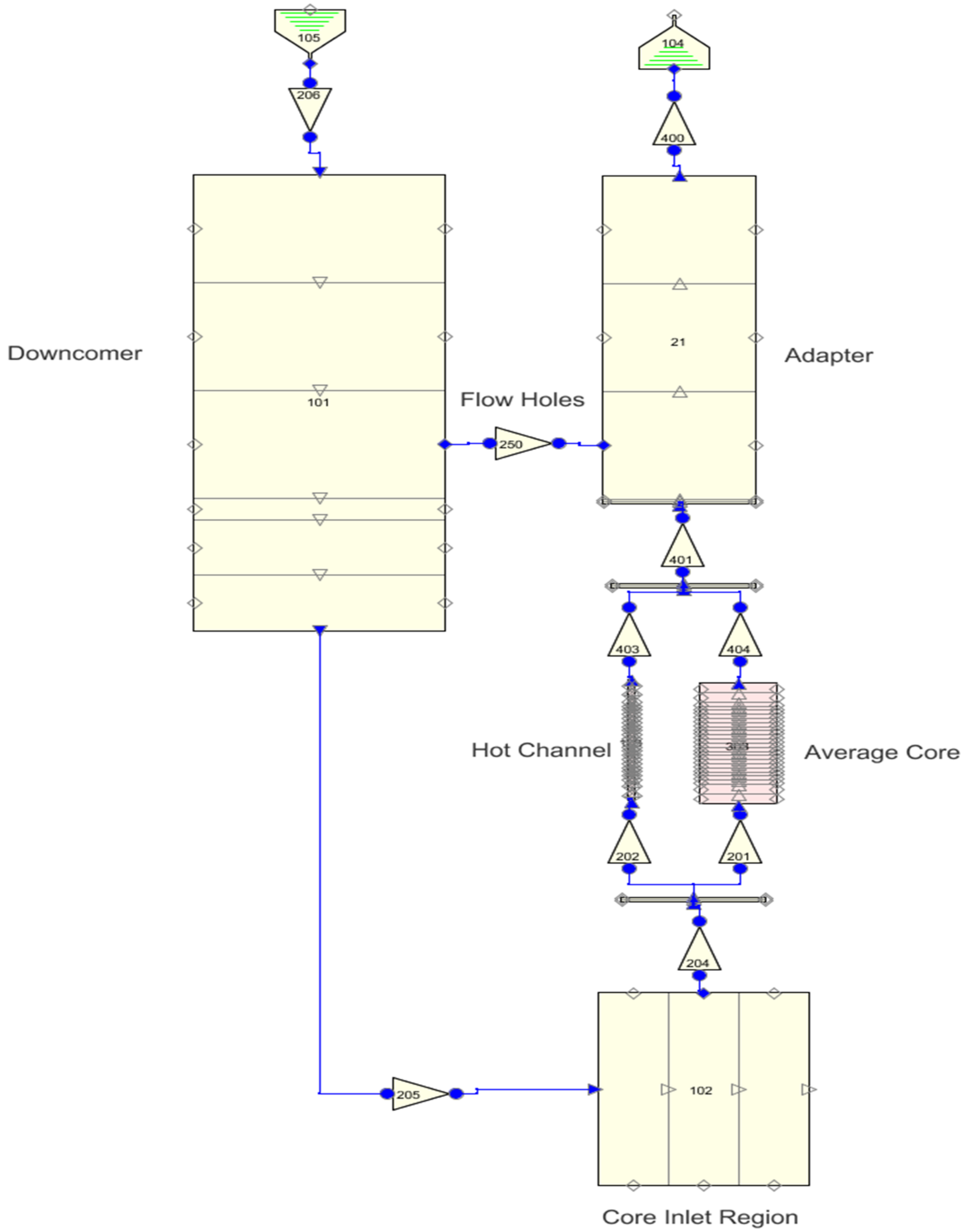


Figure 2-6 AFRRI models used by the NRC for confirmatory TRACE and RELAP5 analysis

Because the AFRRI reactor hot channel is somewhat irregular, it can be represented using six triangles (see Figure 2-7). Each of these triangles creates a unique thermal-hydraulic region contributing to the hot fuel element. By using the coordinates from the AFRRI reactor core plate and the correct geometry of the fuel pin and guide thimbles, it is possible to accurately calculate the flow area of each triangle. Three pairs of identical triangles are then determined. Each triangle then represents half of the flow area for a full fuel element. The doubled (full fuel element) flow areas are 3.051, 4.60, and 7.05 cm^2.

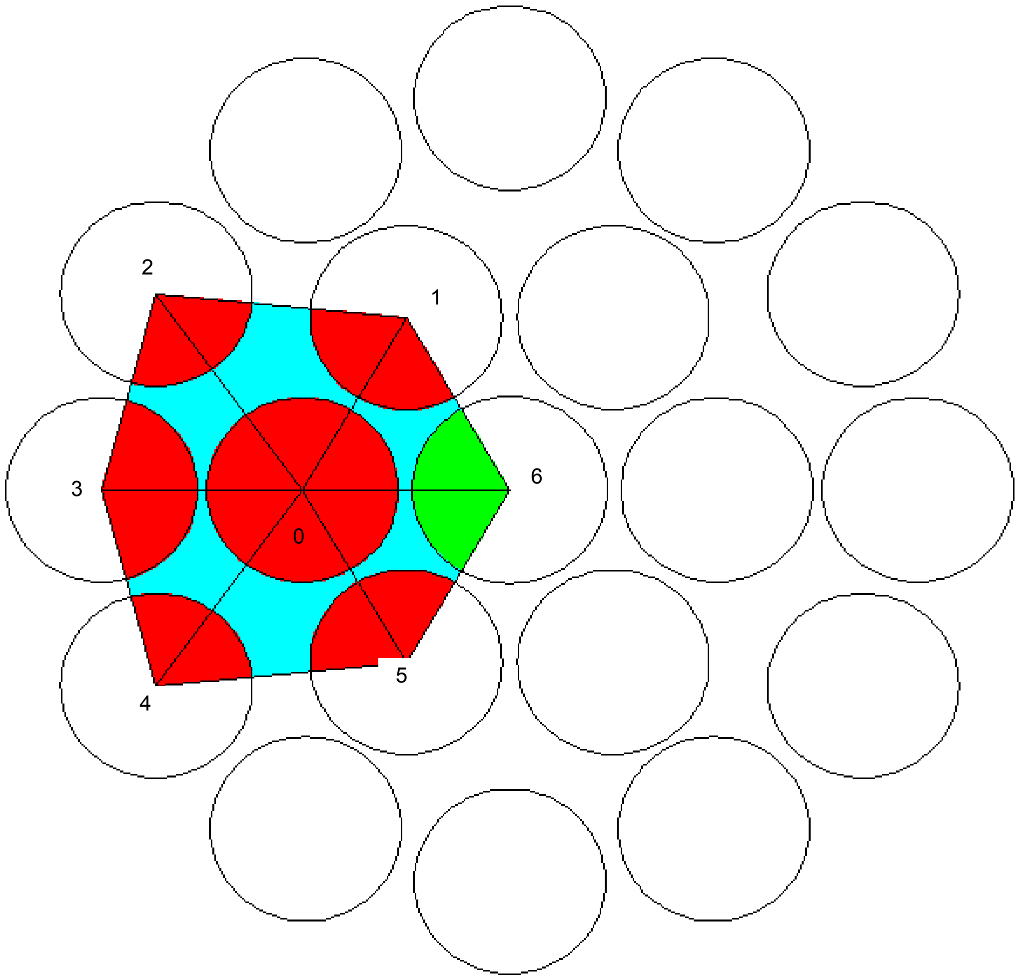


Figure 2-7 Fuel element diagram for rings A–C drawn to scale

Figure 2-7 shows that this forms six triangles that have the associated flow areas listed in Table 2-4.

Table 2-4 Candidate Triangles for DNBR Analysis

Triangle	Triangle Flow Area * 2 (cm^2)
0-1-2	7.05
0-2-3	4.60
0-3-4	4.60
0-4-5	7.05
0-5-6	3.051
0-6-1	3.051

The triangles having heated fuel elements on each the vertices are the 0-2-3 and the 0-3-4 triangles. The hydraulic diameter associated with the representation in Figure 2-7 is 1.56 cm. The hot fuel element power of 19.6 kWt is used with a constant inlet temperature of 60 °C (140 °F) (i.e., the limiting value).

Figure 2-8 is a radial model of the TRIGA fuel element. Table 2-5 lists the accepted fuel radial dimensions for the Stainless Steel TRIGA fuel element illustrated in Figure 2-8.

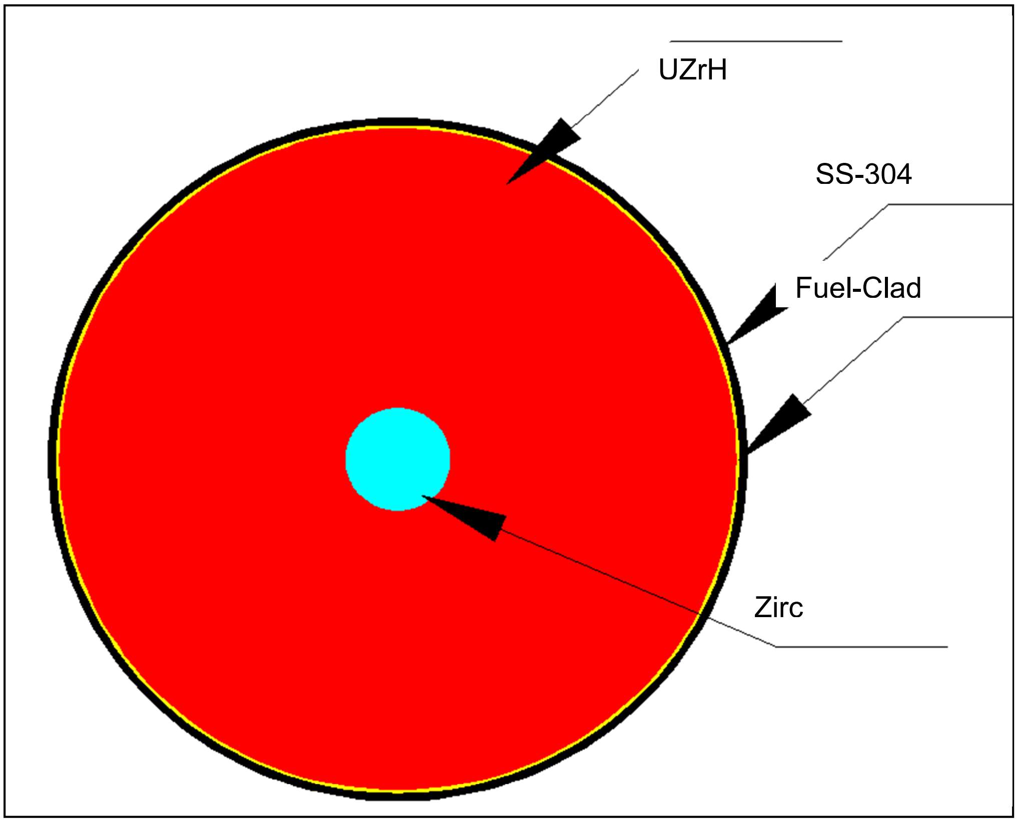


Figure 2-8 TRACE fuel element radial model

Table 2-5 AFRRI Fuel Element Dimensions

Radial Dimensions		
Region	**Inner Diameter (Inches)**	**Outer Diameter (inches)**
Zirc Rod	0.000	0.225
UZrH Fuel	0.225	1.435
Fuel-Clad Gap	1.435	1.438
SS-304 Clad	1.438	1.478
Axial Dimensions		
Region	**Bottom (inches)**	**Top (inches)**
Upper Graphite Slug	18.720	21.280
Upper Fuel Slug	13.720	18.720
Middle Fuel Slug	8.720	13.720
Lower Fuel Slug	3.720	8.720
Lower Graphite Slug	0.000	3.720
Length including end fittings	0.000	29.680

The NRC staff used the TRACE and RELAP5 pipe models described above, starting with a reactor power of 0 kWt where the calculated flow through the core is confirmed to be zero. Subsequently, the reactor power is ramped at steps up to the TS 3.1.1 setpoint of 1.1 MWt and then terminates at 2 MWt. Each ramp leads to a series of power plateaus, each of which allows sufficient time for the code calculations to reach steady-state conditions. The code calculates the thermal-hydraulic conditions at each power plateau to determine if it exceeds the DNBR limit of 2 as set by the NRC guidance in NUREG-1537 (Ref. 39).

Table 2-6 summarizes the confirmatory calculations using the TRACE and RELAP5 (MOD3.3) computer codes. Both calculations assume a maximum allowable inlet temperature of 60 °C (140 °F). The predicted DNBR remains above 2 for power levels up to the TS 3.1.1 limit based on both the TRACE- and RELAP5-calculated results. The NRC staff RELAP5 and TRACE results show that the calculated hot channel flow is stable up to the power level of 2.0 MWt, where the DNBR is approximately 1.13 in the RELAP5 calculation and 1.3 in the TRACE calculation. The confirmatory TRACE and RELAP5 calculations demonstrate that at the AFRRI reactor's licensed conditions, the TS 3.1.1 conditions are satisfactory.

Table 2-6 Confirmatory DNBR Results

Computer Code	Hot Fuel Element Power (kWt)	Inlet Temperature (°C)	Confirmatory DNBR at 1.1 MWt	Peak Fuel Temperature at 1.1 MWt (°C)
TRACE	19.6	60	2.20	490.75
RELAP5/MOD3.3	19.6	60	2.21	437.02

The NRC staff finds that the confirmatory analysis demonstrates that the TS 3.1.1 setpoint of 1.1 MWt helps to ensure that the AFRRI reactor, when operated in accordance with the TS, will exhibit thermal-hydraulic conditions that are acceptably bounded by the NUREG-1537 (Ref. 39) regulatory guidance and the established SL for fuel temperature at all steady-state conditions.

2.6.4 NRC Staff Confirmatory Calculations

The NRC staff also performed confirmatory calculations to review the licensee-supplied DNBR analysis in further detail. To accomplish this, the staff compared the results of additional TRACE and RELAP5 cases to estimates provided as part of the AFRRI submittal also using the RELAP5 computer code. Table 2-7 summarizes the results of the comparison calculation. The primary difference between these cases and the confirmatory analysis above is that the licensee used an inlet temperature of 48 °C (118 °F). The NRC staff replicated this assumption in the analysis below while observing that it is not limiting. It does, however, serve the purpose of demonstrating the adequacy of the licensee's model.

In general, the results of the NRC staff's confirmatory calculations using the two codes RELAP5 and TRACE are consistent with the results of the licensee's analysis, with differences noted in the calculated mass flow rate through the channels after the onset of subcooled boiling. There are also differences in fuel temperature that are caused by differences in the gap conductivity. However, after taking that observation into account, the NRC staff finds that there is a reasonable agreement between the licensee's RELAP5 results and those of the confirmatory models in term of the predicted fuel temperature and DNBR.

Table 2-7 Licensee and Confirmatory Analysis at 1.1 MW

Parameter	RELAP5[a] (Licensee)	RELAP[b] (Conf.)	TRACE[c] (Conf.)
Axial peaking factor—average element	1.316	1.346	1.346
Axial peaking factor—hot element	1.343	1.346	1.346
Hot element power factor	1.56	1.56	1.56
Inlet coolant temperature (°C)	48	48	48
Exit coolant temperature—average element (°C)	67.1	66.4	68.0
Exit coolant temperature-hot element (˚C)	82.5	83.4	84.5
Average temperature in pool above core (˚C)	60.2	61.8	62.4
Total coolant mass flow rate for all fuel elements (kg/s)	13.6	14.1	13.0
Average flow velocity for all fuel elements (cm/s)	29.48	30.2	27.8
Peak fuel temperature in average fuel element (˚C)	360.0	322.0	358.15
Peak fuel temperature in hot fuel element (˚C)	440.7	437.0	491.8
Maximum clad surface temperature in hot element (˚C)	149.2	134.1	129.7

Parameter	RELAP5a (Licensee)	RELAPb (Conf.)	TRACEc
Average heat flux for all fuel elements (W/cm^2)	27.87	27.64	27.87
Maximum heat flux in hot element (W/cm^2)	58.4	58.3	59.1
Minimum DNBR ratio	1.99[d]	2.44	2.41

[a] The parameters in this table are those selected by and published in the AFRRI analysis (Ref. 2).

[b] The confirmatory RELAP5 model uses the same inlet coolant temperature, core unheated length, distance from top of the core to the pool surface, channel flow areas, hydraulic diameters, fuel element diameter including radial dimensions, use of Bernath correlation, and inlet/exit loss coefficients for grid plate as those for the AFRRI RELAP5 model.

[c] TRACE model input conditions are also the same as those of the confirmatory RELAP 5 model.

[d] The SAR table (Ref. 2) giving this value states that the DNBR is for a 1 MWt condition. However, all other elements in that same table are clearly stated to be at 1.1 MWt. The NRC staff assumes that this is a typographical error and the actual conditions for the quoted DNBR are for 1.1 MWt.

In summary, the NRC staff finds that the licensee analysis submitted in SAR, Chapter 4, is adequate for the characterization of important parameters and for the confirmation of the acceptability of SLs and parameters that correspond to regulatory guidance (e.g., the recommended acceptance criteria for DNBR to be no less than 2.0 in NUREG-1537). The results of the AFRRI analysis satisfy these criteria. The NRC staff concludes that the thermal-hydraulic analysis in the AFRRI SAR, as supplemented, demonstrates that the AFRRI reactor LEU core has acceptable safety margins for thermal-hydraulic conditions during steady-state operations.

The licensee also used RELAP5 to calculate the power level at which the DNBR approaches 1—a cladding dryout condition. The licensee stated that this occurs at a power of 1.99 MWt. NRC staff was able to confirm these results with RELAP5. The NRC staff used both RELAP5 and TRACE to calculate the DNBR at the licensed power level and found the results consistent with the licensee's analysis.

An important parameter for pulse operation of the AFRRI TRIGA reactor is the peak fuel temperature. The TS limits the fuel temperature to a maximum of 830 °C (1,529.6 °F) based on early experience with TRIGA fuel, which demonstrated that fuel damage could occur as a result of hydrogen gas accumulation and redistribution in the hydride fuel if the reactor is pulsed after an extended period of operation at 1 MWt (Ref. 69). Because of this physical feature of the fuel, the pulse maximum reactivity insertion is limited to $3.50, and the allowable initial power level at the instant of pulse initiation is set below or at 1 kWt, as discussed below.

The basic objective of the analysis is to demonstrate safe operation of the reactor under pulsing conditions. The maximum allowable pulse insertion of reactivity should lead to a fuel temperature in the maximum power fuel element that is less than the acceptable temperature limit for pulse operations.

The designer of the reactor, GA, recently reviewed the temperature limit data for TRIGA fuel for steady-state and pulse operation of TRIGA reactors (Ref. 69). In this report, GA provides its temperature limit recommendations for all TRIGA fuels for steady-state and pulse operations. Based on a reconsideration of all the available data and on the Texas A&M TRIGA reactor fuel damage experience, GA continues to recommend temperature SLs in the range of 1,025 °C (1,877 °F) to 1,150 °C (2,102 °F). However, based on the analysis of the Texas A&M TRIGA reactor fuel damage experience under pulse operations, GA recommends additional conservatism for pulse operation and recommends a peak fuel temperature of 830 °C (1,529.6 °F) for the pulse operation of all TRIGA fuels.

The licensee used the GA BLOOST computer code to compute the performance of a TRIGA reactor core in a step input of reactivity. The licensee provided a description and validation results of the BLOOST code (Refs. 13 and 70). The code performs combined reactor kinetics and heat transfer calculations using a point kinetics model that analyzes reactor core power and fuel temperature reactivity transients with a variable-temperature fuel heat capacity model. The fuel element is reasonably considered adiabatic for the short duration of the neutronic pulse. The BLOOST model can predict the core average fuel temperature response, or it can use power peaking factors to model the highest power density locations in the reactor core. The BLOOST code is used to predict the transient response of the fuel to reactivity insertions up to $4.00.

Table 2-8, as presented by the licensee, summarizes the results of its analysis of reactivity insertions ranging from $1.50 to $4.00.

Table 2-8 BLOOST Results for the Various Pulses at Beginning of Life

Parameter	$1.50	$1.60	$2.00	$3.00	$4.00
Peak pulsed power (MW)	109	176	605	2,609	4424
Energy release (Megajoule)	6.1	8.1	15.3	31.5	43.1
Peak fuel temperature (°C)	185	243	382	662	831
IFE temperature (°C)	151	189	308	530	666

Section 4.5.9, "Pulse Results," of the SAR, states that the calculated results indicate that a limit of $3.50 reactivity pulse gives adequate protection because at that pulse limit, the reactor core fuel temperature is maintained below 830 °C (1,529.6 °F). TS 3.1.2 specifies the maximum step insertion of reactivity ($3.50 (2.45% Δk/k)). The NRC staff concludes that the TS 3.1.2 limit of $3.50 provides acceptable safety margins for limiting the maximum fuel temperature below 830 °C (1,529.6 °F). The analysis was done with qualified calculation methods and conservative or justifiable assumptions. Because the BLOOST code has been shown to conservatively predict the pulse power and energy deposition in the fuel, its use for these calculations is acceptable.

AFRRI established an administrative procedure and physical design features to prevent the initiation of a pulse from power levels exceeding 1 kWt. The accidental pulsing of the transient rod requires the failure of the 1 kWt interlock that prevents air from being applied to the transient rod piston for a reactor power level above 1 kWt, and the failure of the operator to follow written procedures.

The NRC staff reviewed the licensee's analyses in the SAR as described above. The NRC staff finds that the licensee's analysis used qualified, acceptable calculation methods, conservative and justifiable assumptions, and produced results that demonstrate that the safety limit is not exceeded. The NRC staff concludes that the thermal-hydraulic analysis in the SAR, as supplemented, demonstrates that the operation of the AFRRI TRIGA reactor results in acceptable safety margins for thermal-hydraulic conditions during pulse operations as limited by the TS without exceeding operational or SLs on the fuel temperature.

2.7 Conclusions

Based on the above considerations, the NRC staff finds that the licensee has supplied adequate information and analysis to demonstrate the technical ability to configure and operate the AFRRI TRIGA reactor core without endangering public health and safety. The NRC staff's review of the facility has included a study of its design and installation, its controls and safety instrumentation, its operating procedures, and its operational limitations as identified in the TSs. The NRC staff finds that the thermal-hydraulic analysis in the SAR, as supplemented, demonstrates that the reactor can be operated in accordance with the TSs with acceptable safety margins for thermal-hydraulic conditions. Although the AFRRI TRIGA reactor is typically operated at a power of 1.0 MWt, the reactor can be safely operated to at least 1.1 MWt and maintain adequate critical heat flux margins.

The NRC staff also finds that the licensee's analyses used qualified calculation methods and conservative or justifiable assumptions. The NRC staff reviewed and confirmed that the analysis of the pulse operation of the AFRRI TRIGA reactor and finds that, with pulse sizes up to the administrative limit of $3.50, the maximum core fuel temperature will remain below the 830 °C (1,529.6 °F) limit set by the known mechanical and thermal properties of the fuel. The design features of the reactor are similar to those typical of research reactors of the TRIGA type operating in many countries worldwide. The NRC staff finds that the AFRRI TSs for the reactor design, reactor core components, reactivity limits, and related surveillance requirements are consistent with the analysis, meet 10 CFR 50.36 requirements, and provide assurance that the reactor will be operated safely. On the basis of its review and findings, the NRC staff concludes that there is reasonable assurance that the licensee's activities can be conducted safely for the AFRRI reactor, as limited by the TS, for the period of the requested license renewal.

3. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

Activities involving radiation at the Armed Forces Radiobiology Research Institute (AFRRI) Training, Research, Isotopes, General Atomics (TRIGA) reactor are controlled under a radiation protection program that must meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1101, "Radiation protection programs." The regulations in 10 CFR 20.1101 state, in part, that each licensee shall develop, document, and implement a radiation protection program and shall use, to the extent practical, procedures and engineering 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). In accordance with 10 CFR 20.1101(c), the licensee shall periodically (at least annually) review the radiation protection program content and implementation to ensure continued compliance with the requirements of 10 CFR Part 20, "Standards for Protection against Radiation."

The U.S. Nuclear Regulatory Commission (NRC) inspection program routinely reviews radiation protection and radioactive waste management at the AFRRI TRIGA reactor facility. The licensee's historical performance in these areas as documented in NRC inspection reports (IRs) and the annual operating reports of the AFRRI reactor facility, and the safety analysis report (SAR), as supplemented, provide documentation that measures are in place to minimize radiation exposure to AFRRI staff and the public and to provide adequate protection against operational releases of radioactivity to the environment.

3.1.1 Radiation Sources

The NRC staff reviewed the descriptions of potential radiation sources in each physical form (airborne, liquid, or solid) presented in the SAR, as supplemented, including the inventories and location of the sources. The review of radiation sources included identification of potential radiation hazards, and verification that the hazards were accurately depicted and comprehensively identified.

3.1.1.1 Airborne Radiation Sources

In the SAR (Ref. 1), as supplemented by its June 28, 2013, response (Ref. 15) to the NRC staff's request for additional information (RAI), the licensee stated that during normal reactor operations, argon-41 (Ar-41) and nitrogen-16 (N-16) are the primary airborne radioisotopes produced. N-16 is produced when oxygen in the pool water that is circulated through the reactor core is irradiated by neutrons from the reactor (Ref. 1). As the N-16 rises to the reactor pool surface and is evolved from the pool surface, it decays with a half-life of 7.13 seconds, emitting gamma radiation. As discussed in the SAR (Ref. 1), radiation measurements indicate that for steady-state operations at 1.0 megawatts thermal (MWt), the dose rate is 200 millirem (mrem) per hour at the reactor pool surface, and 14 mrem per hour at the boundary of the chained area around the reactor pool. (The reactor is typically operated at 1.0 MWt. However, the licensed power of the reactor is 1.1 MWt. The licensee did not provide measurements of radiation dose from N-16 at 1.1 MWt, but given that N-16 production is proportional to reactor power, the NRC staff note that dose rates from N-16 at 1.1 MWt would be approximately 10 percent greater than dose rates from N-16 at 1.0 MWt.) Therefore, in order to limit occupational exposures from N-16, access inside the chained area is limited during operations above 0.1 MWt. Because the amount of time that it takes to exhaust the N-16 from the reactor room to the stack is long in comparison to the half-life of N-16, the radiation from N-16 at the release point from the stack is negligible.

Ar-41 is by far the most significant airborne radiation source at the AFRRI TRIGA reactor facility. Ar-41 production results primarily from the neutron irradiation of Ar-40, a natural component of air. Ar-41 is generated in air-filled tubes associated with experiments or reactor instrumentation and is also generated in the reactor pool water, which contains dissolved air. This Ar-41 generation occurs regardless of the reactor core location within the pool. When the core is positioned next to either of the two exposure rooms at the facility, or near the pneumatic transfer system, additional Ar-41 is generated in the air within those rooms or that system (Refs. 1 and 17). In its December 4, 2014, RAI response (Ref. 17), the licensee provided measurements of Ar-41 production for reactor operation at various pool locations showing that the Ar-41 production rate can range from 0.0024 millicuries per kilowatt-hour (kWh) of reactor operation when the core is positioned at mid-pool (away from the exposure rooms and pneumatic transfer system) to 1.89 millicuries per kWh when the core is positioned next to exposure room No. 2. These Ar-41 production measurements were determined based on stack gas monitor (SGM) readings taken following sustained reactor operation for a long enough period of time that Ar-41 production saturation had occurred.

The reactor room has a ventilation system that reduces the concentration of Ar-41 in the room by removing Ar-41 through the building exhaust system, thus minimizing the dose to workers from Ar-41 (see Technical Specification (TS) 3.4, “Ventilation System”). The nominal ventilation exhaust flow rate from the reactor room is 9.64E7 milliliters per minute (Ref. 17).

In responses to RAIs (Refs. 17 and 21), the licensee provided an estimate of Ar-41 concentration in the reactor room and discussed doses to AFRRI workers in the reactor room from exposure to this Ar-41. The licensee stated that, although Ar-41 is produced at higher levels when the core is positioned away from mid-pool, the areas where the additional argon is produced (the exposure rooms) are not accessible to personnel, and the additional argon is exhausted out of the stack and does not diffuse to the reactor room where workers could potentially be exposed to it. Therefore, the Ar-41 concentration in the reactor room does not increase when the core is positioned away from mid-pool. As stated above, for operation with the core at mid-pool, Ar-41 is produced at 0.0024 millicuries per kWh. Although the reactor is typically operated at a maximum of 1.0 MWt, the licensed power is 1.1 MWt. The licensee used the 0.0024 millicuries per kWh production rate to calculate an Ar-41 production rate of 0.74 microcuries per second for mid-pool operation at 1.1 MWt. Using the 0.74 microcuries per second production rate and the nominal reactor room exhaust flow rate, the licensee calculated a reactor room Ar-41 concentration of 4.6E-7 microcuries per milliliter, which is 15.3 percent of the derived air concentration limiting value of 3E-6 microcuries per milliliter established in Table 1 of Appendix B, “Annual Limits on Intakes (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20. If it is assumed that the reactor is operated at 1.1 MWt continuously throughout the year, and that an AFRRI worker remains in the reactor room for the entirety of 2,000 working hours during that year, the dose to that worker from exposure to Ar-41 would be 765 mrem. These are conservative assumptions, because although there are no time restrictions on reactor operation in the license, the reactor is typically operated much less frequently and at lower power (as discussed later in this section, the reactor was operated less than 20 megawatt-hours (MWh) annually for most of the years 2008 through 2015), and because the presence of workers in the reactor room during operation is an infrequent occurrence. The 765 mrem dose is well below the 5,000 mrem limit established in 10 CFR 20.1201, “Occupational dose limits for adults.”

The licensee also provided, in responses to RAIs (Refs. 14, 17, and 21), an estimate of the annual dose from Ar-41 to a maximally exposed member of the public located at the nearest residence to the facility (i.e., the nearest location that is continually occupied by members of the public, or where members of the public are assumed to be present for the entire year). The nearest residence to the AFRRI reactor facility stack is the Fisher House, located 91 meters (m) (299 feet (ft)) from the stack. Although members of the public may be located in other areas, such as roads, sidewalks, parking lots, and patios that are less than 91 m (299 ft) from the stack, these areas would not be continually occupied (i.e., the areas are typically only occupied for brief periods of time, and no individual member of the public would be expected to spend more than a small fraction of the year in these locations). Since the nearest residence is continually occupied, but locations closer to the stack are not continually occupied, annual public exposure from routine Ar-41 effluents is greatest at the nearest residence. The licensee used the COMPLY v1.6 code for its analysis of the dose at the nearest residence. The analysis is based on the assumption that 313.5 curies of Ar-41 is released from the facility stack (13 m (43 ft) above ground level) over a 1-year period. (313.5 curies [Ci] is the annual Ar-41 release limit in TS 3.5.2, "Effluents: Argon-41 Discharge Limit," Specification b. The licensee chose this quantity of Ar-41 as a limiting value in the TSs to bound doses in the unrestricted environment. Depending on the configuration of the reactor in the pool and the amount of time that the reactor is operated, reactor operations could be limited by the 313.5 Ci limit on Ar-41 release, as noted in the discussion of TS 3.5.2, Specification b, below.) The licensee calculated that the dose from Ar-41 at the nearest residence would be 9.9 mrem for a member of the public located at the nearest residence for the entire year. This is well below the 100 mrem per year public dose limit established in 10 CFR 20.1301, "Dose limits for individual members of the public," and is also below the 10 mrem ALARA constraint established in 10 CFR 20.1101(d).

Also in responses to RAIs (Refs. 14 and 21), the licensee provided an additional analysis of the theoretical dose to a member of the public located closer to the facility (10 ft (3 m) from the AFRRI exterior wall) and exposed to direct radiation shine from an Ar-41 plume passing overhead following its release from the stack. The licensee performed this analysis using the MicroSkyshine code. Assuming 313.5 Ci of Ar-41 is released from the facility stack during 1 year, with a uniform rate of release, the dose rate from the overhead plume at 10 ft (3 m) from the AFRRI exterior wall (downwind and directly under the plume, in a position maximally exposed to the shine from the plume) would be 0.045 mrem per hour. Using the conservative assumptions of a 1/20 occupancy factor for a person at this location (i.e., an individual member of the public is assumed to be present in this location for 5 percent of the total hours in a year), and that the wind blows precisely in the direction of this location relative to the stack 50 percent of the year (i.e., greater than the conservative default assumption of 25 percent used in the COMPLY code), the annual dose from the overhead Ar-41 plume to a person at this location would also be 9.9 mrem per year, below the 10 mrem ALARA constraint of 10 CFR 20.1101(d).

The NRC staff performed a confirmatory analysis of the dose from Ar-41 to a person continually located at the nearest residence. The NRC staff's analysis assumed that a total of 313.5 Ci of Ar-41 is released from the AFRRI reactor facility stack (13 m (43 ft) above ground level (Ref. 17)) over the course of 1 year. The analysis used the Pasquill-Gifford method, neglected building wake effects, and considered a range of meteorological conditions (i.e., various unstable and stable atmospheric stability classes, designated as Pasquill A through Pasquill F). The analysis conservatively assumed an average annual wind speed (for all atmospheric stability classes) of 2 m per second (6.6 ft per second), and that the wind blows in the direction of the receptor 25 percent of the time (the conservative default assumption used in the COMPLY code). The NRC staff's analysis determined that for the nearest residence, the highest dose occurs during Pasquill B, or moderately unstable, atmospheric stability conditions.

Conservatively assuming that the worst-case Pasquill B conditions occur the entire year, the NRC staff calculated an annual dose at the nearest residence of 13.0 mrem. This calculated dose is comparable (with some variance, which is expected given the different models and assumptions used) to the licensee's results, and is also well below the 10 CFR 20.1301 annual public dose limit of 100 mrem. The NRC staff's calculated dose is higher than the 10 mrem constraint of 10 CFR 20.1101(d). However, the 10 mrem dose is not a regulatory limit, but a constraint used to demonstrate that public doses are ALARA. If AFRRI exceeded this dose constraint, it would be required to report the exceedance to NRC and promptly take appropriate corrective action to help ensure against recurrence.

The NRC staff's confirmatory analysis also calculated the doses from Ar-41 to persons continually located at other locations closer to, or farther away from, the facility stack compared to the nearest residence. The NRC staff calculated that the worst-case doses (conservatively assuming that one worst-case atmospheric stability class occurs for the entire year) for locations beyond the nearest residence were all below the 13.0 mrem calculated for the nearest residence. For locations closer than the nearest residence, the highest dose was 15.5 mrem for a person at a location 50 m (164 ft) from the stack, assuming Pasquill A, or extremely unstable, atmospheric stability conditions occur the entire year. The dose at this location is greater than the dose for a person at the nearest residence. However, the location 50 m (164 ft) from the stack is not a continually occupied location, and the difference in the doses for the two locations is small. Therefore the NRC staff finds that the use of the nearest residence to represent the location of the maximally exposed member of the public from Ar-41 is reasonable, conservative, and acceptable.

As reported in the licensee's annual operating reports for the years 2008 through 2015 (Ref. 25), historical annual Ar-41 releases have been well below 313.5 Ci. The total activity of Ar-41 discharged was 0 Ci in 2015, 0.47 Ci in 2014, 4.55 Ci in 2013, 2.36 Ci in 2012, 6.21 Ci in 2011, 1.22 Ci in 2010, 49.92 Ci in 2009, and 2.63 Ci in 2008. Therefore, review of the AFRRI annual operating reports demonstrates that the Ar-41 released from the reactor facility during normal operations does not present a significant exposure hazard.

TS 3.5.2, "Effluents: Argon-41 Discharge Limit," states the following:

> <u>Specifications</u>
>
> (…)
>
> b. If calculations, which shall be performed at least quarterly but not to exceed 20 MWh of operation, indicate that argon-41 release in excess of 313.5 curies to the unrestricted environment could be reached during the year as a result of normal reactor operations, reactor operations that generate and release significant quantities of argon-41 shall be curtailed for the remainder of the year as needed to ensure adherence with the 10 mrem constraint.

TS 4.5.2, "Effluents," presents surveillance for effluent releases as follows:

> <u>Specifications</u>
>
> (…)

c. A gaseous effluent release report shall be generated quarterly or every 20 MW hours of reactor operations (whichever comes first) to ensure radioactive effluents will not exceed the annual dose limits to the public.

TS 3.5.2, Specification b, requires that if calculations show that the AFRRI reactor facility could release in excess of 313.5 Ci of Ar-41 to the unrestricted environment in 1 year, reactor operations must be curtailed as needed to ensure than the public dose from Ar-41 effluents does not exceed 10 mrem per year. In its August 5, 2016, RAI response (Ref. 21), the licensee stated that these calculations are performed for each period's operation using actual Ar-41 production measurements collected during historical reactor operations. As discussed above, the licensee calculated that an annual Ar-41 release of 313.5 Ci corresponds to a maximum annual public dose from Ar-41 of 9.9 mrem, well below the 100 mrem public dose limit in 10 CFR 20.1301 and below the 10 mrem ALARA constraint of 10 CFR 20.1101(d). Therefore, TS 3.5.2, Specification b, helps ensure that the reactor will be operated such that the discharge of Ar-41 from the facility will not cause the facility to exceed the ALARA constraint of 10 CFR 20.1101(d) or the public dose limit of 10 CFR 20.1301.

TS 4.5.2, Specification c, requires that reports containing the calculations required by TS 3.5.2, Specification b, be generated quarterly or every 20 MWh of reactor operation, whichever comes first. Therefore, TS 4.5.2, Specification c, imposes a surveillance requirement that helps ensure that the licensee is in compliance with TS 3.5.2, Specification b. Because Ar-41 effluent releases are analyzed at an interval not to exceed 20 MWh of operation or quarterly (whichever comes first), the maximum dose to the public between effluent analyses is 1.2 mrem. This is because the highest rate at which Ar-41 could be produced and discharged from the stack is 1.89 millicuries per kWh (for operation with the reactor core adjacent to an exposure room) (Ref. 17), and the limit of 313.5 Ci would be reached in 165.9 MWhs of operation. Therefore, the 20 MWh calculation interval is conservative because it helps ensure reactor operations would be curtailed if Ar-41 doses approach 10 CFR Part 20 limits. As reported in the licensee's annual operating reports for the years 2008 through 2015 (Ref. 25), the reactor has been operated less than 20 MWhs per year for the majority of the years 2008 through 2015. Total reactor operation in 2015 was 0 MWhs; in 2014, 4.50 MWhs; in 2013, 9.43 MWhs; in 2012, 6.02 MWhs; in 2011, 17.81 MWhs; in 2010, 34.21 MWhs; in 2009, 65.14 MWhs; and in 2008, 6.57 MWhs.

The NRC staff reviewed the information and analyses in the SAR, as supplemented, related to airborne radiation sources. For the analyses of doses from Ar-41, the NRC staff reviewed the licensee's calculation methodologies and assumptions, and determined that they were conservative and consistent with accepted industry practices. The NRC staff also performed confirmatory calculations of the dose from Ar-41 to a member of the public at the nearest residence. Based on the information and analyses that demonstrate that the AFRRI routine gaseous effluent releases are within the limits in 10 CFR Part 20, as well as the NRC staff's review of historical Ar-41 releases at the AFRRI reactor as described in AFRRI annual operating reports, the NRC staff concludes that AFRRI's production and control of airborne radiation sources are acceptable. The NRC staff further finds that adherence to TS 3.5.2, Specification b, and TS 4.5.2, Specification c, provides reasonable assurance that, during the continued normal operation of the AFRRI TRIGA reactor, the airborne radioactive releases will be in compliance with 10 CFR Part 20 and will not pose a significant risk to public health and safety or the environment. Therefore, based on the information above, the NRC staff concludes that TS 3.5.2, Specification b, and TS 4.5.2, Specification c, are acceptable.

3.1.1.2 Liquid Radiation Sources

The reactor coolant water, including water in the pool and piping, is a liquid radiation source at the AFRRI reactor. As discussed in Section 3.1.1.1 of this safety evaluation report (SER), the coolant contains dissolved N-16 and Ar-41, although most exposure from N-16 and Ar-41 occurs after they have been evolved from the coolant. Reactor coolant also contains impurities, such as corrosion products, that become activated by neutrons as they pass through the reactor core. Because the piping carries pool water that has been circulated through the reactor core, activated impurities produced during normal operation may be capable of causing exposure to personnel near the piping, especially when the reactor is operating at high power. The AFRRI reactor has a primary water purification system that removes impurities from the reactor water and helps to minimize any dose from activated impurities in the coolant (Ref. 1). The implementation of the ALARA policy, which is discussed in Section 3.1.3 of this SER, also helps minimize the potential exposure to operating staff. Additionally, TS 3.3, Specification b, limits the conductivity of the reactor coolant; TS 4.3, Specification b, requires monthly surveillance of the reactor coolant conductivity; and TS 4.3, Specification c, requires the licensee to sample pool water for radioactivity on a quarterly basis (see SER Section 2.3). These TSs help ensure (1) that the level of activation products in the reactor coolant is minimized and (2) detection of potential fission product leakage from the reactor fuel or leakage from sealed sources.

In its August 5, 2016, RAI response (Ref. 21), the licensee stated that no liquid radioactive waste is analyzed, processed, or disposed of under the 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Facility Operating License No. R-84 for the AFRRI reactor. Any liquid radioactive waste from the reactor is transferred to the 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," AFRRI "O2" byproduct license (NRC Materials License No. 19-08330-02). These transfers are internal to the AFRRI facility and no shipment of radioactive material is involved. Such transfers may not exceed the byproduct material possession limits in the "O2" license. The "O2" byproduct license requires that all waste handling and disposal conducted under the "O2" license be in accordance with NRC regulations. In accordance with 10 CFR 30.51 and 10 CFR 30.52, records must be kept of all wastes disposed of under the "O2" license, and these records are subject to inspection by the NRC.

Depending on its source and level of activity, any liquid waste from the reactor is sent to one of two liquid radioactive waste subsystems (hot or warm), where it is comingled with the liquid radioactive waste from the entire AFRRI complex. At this point, the waste is measured, diluted as necessary, and released to sanitary sewerage in accordance with 10 CFR 20.2003, "Disposal by release into sanitary sewerage." The licensee stated that procedures are used to help ensure that handling of liquid radioactive waste is conducted without endangering the health and safety of AFRRI personnel or the general public (Ref. 1).

The NRC staff reviewed the information above and finds that the licensee has implemented procedures and controls to help ensure that doses from liquid radioactive sources associated with the continued normal operation of the AFRRI reactor facility are small. The NRC staff also finds that the licensee controls access to the liquid sources, and that their disposal will be in compliance with 10 CFR Part 20. Therefore, the NRC staff concludes that these sources do not present a significant hazard to the public, AFRRI personnel, or the environment.

3.1.1.3 Solid Radiation Sources

The fission products contained in the reactor fuel constitute the most significant solid radiation source at the AFRRI TRIGA reactor facility. The reactor pool water and concrete shielding around the pool provide protection for personnel from this source of radiation dose. Nonfuel solid radiation sources include sealed sources in the reactor pool, activated reactor components, ion-exchange resins and other filter media from the pool water demineralizer, and irradiated samples. The sources that may be in the pool include an antimony-beryllium startup source that is doubly encapsulated in stainless steel (Ref. 1). SER Section 2.2.4 discusses this startup source. As discussed in SER Section 2.3, TS 3.3, Specification b, limits the conductivity of the reactor coolant; TS 4.3, Specification b, requires monthly surveillance of the reactor coolant conductivity; and TS 4.3, Specification c, requires the licensee to sample pool water for radioactivity on a quarterly basis. These TSs help ensure the detection of any leak of radioactive material from fuel or other nonfuel solid radiation sources into the reactor pool. TS 6.4 requires new experiments at AFRRI, including those involving irradiation of samples, to be reviewed for radiological safety and approved by the Reactor Facility Director, Health Physics Department, and Reactor and Radiation Facilities Safety Subcommittee (RRFSS) before they are performed. The AFRRI radiation protection program controls exposure from all solid radiation sources discussed above.

Radioactive waste is another solid radiation source at the AFRRI reactor. Solid radioactive waste generation at the AFRRI reactor facility generally consists of routine laboratory wastes such as glassware, plastics, paper, disposable gloves, vials, and biological waste. The Safety and Health Department at AFRRI packages all solid wastes, and the radiological waste facility stores them in preparation for shipment to an NRC-authorized disposal facility (Ref. 1). NRC staff review of the AFRRI reactor facility annual operating reports for the years 2008 through 2015 (Ref. 25) showed that, in those years, all solid radioactive waste generated at the reactor facility was transferred to the AFRRI “O2” 10 CFR Part 30 byproduct license (see SER Section 3.1.1.2), and no solid radioactive waste was disposed of under Facility Operating License No. R-84 for the AFRRI reactor. The transfers to the “O2” byproduct license may not exceed the byproduct material possession limits in the “O2” license. The “O2” byproduct license requires that all waste handling and disposal conducted under the “O2” license be in accordance with NRC regulations. In accordance with 10 CFR 30.51 and 10 CFR 30.52, records must be kept of all wastes disposed of under the “O2” license, and these records are subject to inspection by the NRC.

The NRC staff reviewed the information on solid radiation sources above and finds that the licensee has implemented procedures and controls to help ensure that doses from solid radioactive sources at the AFRRI reactor facility are small. The NRC staff also finds that the licensee controls access to the solid sources, and that their disposal will be in compliance with NRC regulations. Therefore, the NRC staff concludes that the control of solid radioactive sources at the AFRRI TRIGA reactor is acceptable.

3.1.2 Radiation Protection Program

The regulation at 10 CFR 20.1101(a) requires each licensee to develop, document, and implement a radiation protection program. As specified in TS 6.1 (see SER Section 5.6.1), the licensee for the AFRRI reactor, and the individual with ultimate responsibility for the license, is the AFRRI Facility Director. The Safety and Health Department at AFRRI is responsible for maintaining a comprehensive radiation protection program, which encompasses all sources of radiation at AFRRI including those at the reactor (Ref. 1). The Safety and Health Department’s

Radiation Safety Officer (RSO) is the individual responsible for radiation protection at AFRRI, including at the AFRRI reactor facility, and reports directly to the AFRRI Facility Director as specified in TS 6.1. Cooperation and coordination occur between the RSO and the AFRRI Reactor Facility Director, who also reports directly to the AFRRI Facility Director. The AFRRI RSO provides onsite advice about personnel and radiological safety and supplies technical assistance and review in the area of radiation protection.

TS 6.1.2 (see SER Section 5.6.1) specifies, in part, that AFRRI's RSO implement a radiation protection program that complies with 10 CFR Part 20. In its October 20, 2011, RAI response (Ref. 10), AFRRI stated that this radiation protection program would be conducted in accordance with the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.11, "Radiation Protection at Research Reactor Facilities" (Ref. 71).

As discussed in the SAR, as supplemented, as observed by the NRC staff during site visits, and in accordance with the TSs and the guidance in ANSI/ANS-15.11 (Ref. 71), the AFRRI reactor's radiation protection program performs the following functions:

- implementing and maintaining the ALARA program
- establishing administrative and internal exposure limits, and reviewing and investigating any doses that exceed these limits
- establishing guidelines for receiving, monitoring, handling, transporting, and testing radioactive materials
- establishing a procedures and record system for surveys and monitoring
- establishing requirements and responsibilities for personnel dosimetry
- ensuring management commitment and worker responsibility
- ensuring qualification of personnel and adequacy of resources
- ensuring adequacy of the authority of personnel responsible for radiation safety
- training new staff and supplying continuing education for all personnel
- ensuring that radiological design is an integral aspect of facility and experiment design
- ensuring that radiological planning is an integral aspect of operations planning
- conducting performance reviews of designs and operations
- maintaining and analyzing personnel exposure records
- conducting periodic assessment and trend analysis of the radiological environment
- conducting periodic assessment and audits of the radiation protection program

- conducting surveillance activities (including testing, calibration, and quality assurance) on equipment related to radiation protection

- issuing appropriate personnel monitoring devices and any required protective clothing

- maintaining the adequate supply of, and quality assurance for, protective equipment

- ensuring compliance with all applicable radiation protection requirements and regulations

The radiation protection program is implemented using written standard operating procedures, in accordance with TS 6.3, Specification c (see SER Section 5.6.3).

As observed by NRC staff during site visits, and in accordance with the guidance in ANSI/ANS-15.11 (Ref. 71), the general areas of radiation protection training given to personnel who are permitted unescorted access to the AFRRI reactor restricted area include (1) access control rules, (2) radiological safety principles, policies, and procedures, (3) dosimetry requirements, (4) monitoring instruments, (5) protective equipment, (6) procedures for handling and storing radioactive material, (7) posting and labelling requirements for radiation areas or radioactive materials, (8) ALARA principles and exposure limits, (9) risks and hazards of radiation exposure, (10) emergency procedures, and (11) responsibility of individuals to report unsafe conditions related to radiological safety.

The licensee performs radiation surveys of the facility on a regular basis (see Section 3.1.4 of this SER). The licensee monitors all liquid and gaseous effluents for compliance with the limits in 10 CFR Part 20 (see Sections 3.1.1 and 3.1.4 of this SER) and conducts environmental radiation monitoring of areas outside the facility (see Section 3.1.7 of this SER). As required by TS 6.7 (see SER Section 5.6), records relating to personnel exposures, radioactive effluents, and environmental monitoring must be kept for the lifetime of the facility.

The regulation in 10 CFR 20.1101(c) requires that licensees shall periodically (at least annually) review the radiation protection program content and implementation. In accordance with the guidance in ANSI/ANS-15.11 (Ref. 71), the radiation protection is reviewed annually to assure that the program requirements are met. TS 6.2.5, Specification a (see SER Section 5.6), requires that an annual audit of the conformance of facility operation (including radiation protection activities) to the TSs and license be performed under the cognizance of the RRFSS. The NRC inspection program also routinely reviews the radiation protection program at the AFRRI facility, and the results of these reviews are documented in IRs. The NRC staff reviewed the IRs for the years 2008 through 2016 (Ref. 26) and found no significant radiation safety issues.

The NRC staff reviewed the information above and finds that the AFRRI reactor facility radiation protection program complies with 10 CFR 20.1101, paragraphs (a) and (c); is implemented in an acceptable manner; and provides reasonable assurance that for all facility activities, the AFRRI staff, the public, and the environment will be protected from unacceptable radiation exposures. Therefore, the NRC staff concludes that the radiation protection program is acceptable.

3.1.3 ALARA Program

The regulation stated in 10 CFR 20.1101(b) requires licensees to use procedures and engineering controls to achieve occupational doses and doses to members of the public that are

ALARA. To comply with the regulation stated in 10 CFR 20.1101(b), the AFRRI RSO implements a policy that all operations are to be planned and conducted in a manner that keeps all exposures ALARA (Ref. 10). The AFRRI ALARA program is a part of the broad radiation protection program at AFRRI, which is based on the guidance in ANSI/ANS-15.11 (Ref. 71). The licensee applies the program through written procedures and guidelines, in accordance with TS 6.3, Specification c (see SER Section 5.6.3). AFRRI implements controls that limit access and personnel exposure in the facility, and also reviews all proposed experiments and operational procedures to help ensure that radiation exposures to personnel will be minimized. The AFRRI Safety and Health Department staff use procedures that help ensure handling and disposal of radioactive waste is conducted safely (Ref. 1).

TSs 3.5.2 and 4.5.2 (see SER Section 3.1.7), require AFRRI to conduct an environmental monitoring program. This program helps ensure that the facility will operate in a manner that minimizes radiation exposure to the public, and that the radiation exposure to the public resulting from operation of the reactor is maintained ALARA.

TS 6.2.5, Specification g (see SER Section 5.6), requires that an annual audit of the ALARA program be performed under the cognizance of the RRFSS. The NRC inspection program also routinely reviews the ALARA program. The NRC staff reviewed the IRs for the years 2008 through 2016 (Ref. 26) and found that the ALARA program was implemented sufficiently and the licensee met regulatory requirements related to radiation doses.

The NRC staff reviewed the information above and finds that the ALARA program implemented at the AFRRI reactor facility complies with 10 CFR 20.1101. Therefore, based on its review, the NRC staff concludes that the AFRRI ALARA program will limit occupational doses and doses to members of the public so that they are ALARA, and that the program is acceptable.

3.1.4 Radiation Monitoring and Surveying

The regulations stated in 10 CFR 20.1501, “General” paragraph (a) require each licensee to make, or cause to be made, surveys of areas, including the subsurface, that—

(1) May be necessary for the licensee to comply with the regulations in this part; and

(2) Are reasonable under the circumstances to evaluate- -

- (i) The magnitude and extent of radiation levels; and
- (ii) Concentrations or quantities of radioactivity; and
- (iii) The potential radiological hazards of the radiation levels and residual radioactivity detected.

The regulation stated in 10 CFR 20.1501(c) requires the licensee 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.

The licensee has a comprehensive set of radiation monitors that can (1) detect the various types of radiation and radioactive material that may be encountered at the facility and (2) give information to operating personnel on any impending or existing danger from radiation. AFRRI maintains these monitors as a means of ensuring compliance with the radiation limits established in 10 CFR Part 20. These instruments include radiation area monitors (RAMs), an SGM, and continuous air monitors (CAMs) (Ref. 67).

Four RAMs are located in various areas of the reactor building where potential radiation hazards may exist due to reactor operation; these RAMs use scintillation detectors to measure gamma radiation. TS 3.5.1, "Monitoring System," which is discussed later in this section, imposes requirements for these four RAMs. Two of the RAMs are located in the reactor room on the reactor deck, and one additional RAM is located adjacent to each of the two exposure rooms (Ref. 67).

Two CAMs (primary and backup) are located in the reactor room. The primary CAM is located approximately 36 inches (91.4 centimeters) above the reactor pool inside the core carriage, and the backup CAM is located near the warm drain along the west side of the reactor tank. These CAMs are of primary importance for ensuring safe operation of the reactor because they help detect any potential fission product release. TS 3.5.1, which is discussed later in this section, imposes requirements for reactor room CAMs (one CAM shall be operable to sample the air above the reactor pool when the reactor is not secured). Three additional CAMs, which have no TS requirements, monitor the exposure rooms and preparation areas. The five CAMs provide continuous air sampling and monitoring of radioactive airborne particulate matter by using an air pump to draw air into a shielded filter assembly. A Geiger-Mueller detector measures any radioactive particulate matter that is trapped on the filter (Ref. 67). As required by TS 3.5.1, Specification c, alarm of the reactor room CAM above the reactor pool shall initiate closure of the ventilation system dampers, restricting air leakage from the reactor room.

The SGM is part of the stack monitoring system, which includes both the SGM and the stack flow monitoring system. The SGM is a sodium iodide scintillation radiation detector system that samples exhaust air from the reactor stack. The exhaust air is passed through a filter to remove particulates before being analyzed. The stack flow monitoring system measures the average flow rate of the air exhausted through the reactor stack. Collectively, the stack monitoring system provides data about the radioactive effluents discharged through the reactor stack (Ref. 67).

In its August 5, 2016, RAI response (Ref. 21), the licensee stated that alarm setpoints for the RAMs, CAMs, and SGM are determined based on background radiation levels during operations, ALARA considerations, regulatory and internal administrative dose limits, and best practice. For example, the reactor room RAMs are typically set to alarm at 10 mrem per hour, and the CAMs are typically set to alarm at 40,000 counts per minute.

TS 3.5.1, "Monitoring System," states the following:

> <u>Specifications</u>
>
> The reactor shall be secured unless the following radiation monitoring systems are operable:
>
> a. Radiation Area Monitoring System:
>
>   i. 2 RAMS on the reactor Deck (Room 3160) are operable
>
>   ii. If operating in an exposure room (ER1 or ER2) the RAM adjacent to the exposure room in use shall be operable
>
> b. Stack Gas Monitor: The stack gas monitor (SGM) shall sample and measure the gaseous effluent in the exhaust system;

c. Continuous Air Particulate Monitor: The continuous air particulate monitor (CAM) shall sample the air above the reactor pool. This unit shall be sensitive to radioactive particulate matter. Alarm of this unit shall initiate closure of the ventilation system dampers, restricting air leakage from the reactor room; and

d. Table 4 specifies the alarm and readout system for the above monitors.

Table 4. Locations of Radiation Monitoring Systems

Sampling Location	Location(s) of readouts Audible alarms and visual Indicators
RAM	
Reactor Room (2 required)	Reactor and Control Rooms
Exp. Room 1 Area	Prep Area and Control Room
Exp. Room 2 Area	Prep Area and Control Room
SGM	
Reactor Exhaust	Reactor and Control Rooms
CAM	
Reactor Room	Reactor and Control Rooms

TS 3.5.1 imposes requirements that the two RAMs in the reactor room on the reactor deck, the RAMs adjacent to the exposure rooms (if the exposure rooms are in use), the CAM above the reactor pool, and the SGM be operable whenever the reactor is not secured. The objective of TS 3.5.1 is to help ensure that adequate radiation monitoring channels will be available to the operator to ensure safety whenever the reactor is in an unsecured condition (i.e., the reactor is not shut down, or other activities are in progress such as work in the core, fuel movement, etc.) that could potentially result in elevated radiation levels or an airborne release of radioactivity. By requiring both local and remote (control room) audible alarms and visual indicators, TS 3.5.1, Specification d, helps ensure that the radiation monitoring systems will inform operating staff of any impending or existing danger from radiation. This helps ensure that staff will have sufficient time to evacuate the facility and take the necessary steps to minimize the spread of radioactivity to the surroundings. When the CAM alarms, the automatic closure of the ventilation system dampers restricts air leakage from the reactor room, which also helps to prevent any release of radioactivity to the outside environment.

The NRC staff reviewed TS 3.5.1 and the information above regarding the radiation monitoring system at the AFRRI reactor facility. The NRC staff finds that the licensee has, and TS 3.5.1 requires, adequate instruments to provide reasonable assurance that radiation and airborne radioactive material will be properly measured and evaluated, helping to ensure compliance with 10 CFR Part 20. The NRC staff also finds that TS 3.5.1 is consistent with the guidance in ANS/ANSI-15.1-2007 (Ref. 64), which recommends that research reactors have TSs specifying minimum numbers of radiation monitors. Additionally, the NRC staff finds that TS 3.5.1 is consistent with 10 CFR 50.36(c)(2), which requires, in part, that licensees have TSs on limiting conditions for operation on installed instrumentation such as radiation monitors that are necessary to detect a radiological release. Therefore, based on the information above, the NRC staff concludes that TS 3.5.1 is acceptable.

TS 4.4, “Ventilation System,” (see Section 5.4.3 of this SER) has a surveillance requirement to ensure operability of the ventilation system dampers that close when the CAM alarms.

TS 4.5.1, “Monitoring System,” specifies the surveillance requirements for the radiation monitoring system as follows:

> Specification
>
> The radiation area monitoring, continuous air particulate monitoring, and stack gas monitoring systems shall be channel tested quarterly, not to exceed 4 months. A channel check of these systems shall be performed daily to verify operability when operations are planned. These systems shall be calibrated annually, not to exceed 15 months.

The licensee’s basis for TS 4.5.1 states that experience has shown that quarterly verification of radiation monitoring system setpoints, in conjunction with quarterly channel testing of the radiation monitoring systems, is adequate. The NRC staff reviewed TS 4.5.1 and the information above and finds that the surveillance requirements in TS 4.5.1 are consistent with the guidance in ANSI/ANS-15.1-2007, “The Development of Technical Specifications for Research Reactors,” issued in 2007 (Ref. 64), which recommends that the operability of radiation monitoring systems be tested monthly to quarterly, and that the systems be calibrated annually to biennially. In addition, the daily channel checks of the radiation monitoring systems during periods when reactor operations are planned will also help to ensure system reliability. The NRC staff also finds that TS 4.5.1 is consistent with 10 CFR 50.36(c)(3), which requires that licensees have surveillance requirements for limiting conditions for operation. Based on the information above, the NRC staff finds that the surveillance requirements in TS 4.5.1 are sufficient to help ensure that the radiation monitoring systems will perform their intended functions, and concludes that TS 4.5.1 is acceptable.

In addition to using the radiation monitoring system, the licensee also conducts radiation and contamination surveys at the reactor in accordance with the guidance in ANSI/ANS-15.11 (Ref. 71), and as observed by the NRC staff during site visits. The health physics staff at the AFRRI reactor facility routinely perform radiation and contamination surveys. The selection, use, control, testing, and calibration of the radiation monitoring and surveying equipment are in accordance with applicable national standards, guidance, and regulations. The equipment is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility, and the surveys are at appropriate frequencies to ensure compliance with the requirements in 10 CFR Part 20 and the facility ALARA program under all operating conditions. In accordance with the guidance in ANSI/ANS-15.11 (Ref. 71), and as observed by the NRC staff during site visits, the licensee has a personnel monitoring program that requires individuals who could be exposed to radiation to be equipped with dosimetry to help ensure that the dose limits in 10 CFR Part 20 will not be exceeded. Section 3.1.5 of this SER discusses radiation exposure control and dosimetry at the AFRRI reactor facility.

An environmental monitoring program that includes perimeter monitoring and environmental sampling is also conducted by AFRRI staff, primarily to measure environmental doses from radionuclides released in reactor effluents. TS 3.5.2, Specification a and TS 4.5.2, Specifications a and b, impose environmental monitoring program requirements. Section 3.1.7 of this SER discusses TS 3.5.2, Specification a, and TS 4.5.2, Specifications a and b, and the AFRRI environmental monitoring program, and concludes that the environmental monitoring program and associated TSs are acceptable.

Based on its review of the information above, the NRC staff finds that the licensee's radiation monitoring and surveying, including the equipment used and the surveillances performed on that equipment, are adequate for detecting the types and intensities of radiation likely to be encountered within the facility and will help ensure compliance with 10 CFR 20.1501(a) and 10 CFR 20.1501(c). Therefore, based on the information above, the NRC staff concludes that the radiation monitoring and surveying at the AFRRI reactor facility is acceptable.

3.1.5 Radiation Exposure Control and Dosimetry

The AFRRI reactor pool water and the concrete pool structure provide shielding of radiation from the reactor core, limiting dose rates in the reactor room (Ref. 2). As discussed in Section 3.1.1.1 of this SER, during operations at 1.0 MWt, the dose rate above the reactor pool is about 200 mrem per hour, and the dose rate immediately outside the chained-off area around the reactor pool is about 14 mrem per hour. However, these dose rates are predominantly due to N-16 evolved from the pool, as the radiation from the core is shielded. Because of the short half-life of N-16, Ar-41, rather than N-16, is the most significant airborne radiation source during routine operation of the reactor. The facility ventilation system maintains Ar-41 in the reactor room at levels that ensure the occupational dose limits in 10 CFR 20.1201 will not be exceeded. The ventilation system also maintains the reactor room at negative pressure with respect to its surroundings, helping to minimize any release of airborne radioactive material to the surrounding environment (Ref. 1).

The licensee uses contamination control procedures to help control radiation exposures by limiting the spread of radioactive contamination. Contamination control at the AFRRI reactor facility is discussed in Section 3.1.6 of this SER.

The licensee has a radiation monitoring system to monitor dose rates, concentrations of airborne radioactive material, and radioactive effluents at the facility, and also uses portable survey meters to periodically measure radiation and contamination levels, as discussed in Section 3.1.4 of this SER. As observed by the NRC staff during site visits, these radiation and contamination levels are posted as required in accordance with 10 CFR Part 20.

The regulations in 10 CFR 20.1502, "Conditions requiring individual monitoring of external and internal occupational dose," require monitoring of workers likely to receive, in 1 year from sources external to the body, a dose in excess of 10 percent of the occupational dose limits in 10 CFR 20.1201. To comply with 10 CFR 20.1502, the licensee has a personnel monitoring program in accordance with ANSI/ANS-15.11 (Ref. 71) that requires individuals who could be exposed to radiation to be equipped with dosimeters. All dosimeters for AFRRI radiation workers are analyzed approximately quarterly and flagged if they exceed AFRRI's internal administrative dose limits, which are set below 10 CFR Part 20 limits to help keep doses ALARA. The licensee stated that this personnel monitoring program helps ensure that the dose limits in 10 CFR Part 20 will not be exceeded (Ref. 14).

As specified in TS 6.7.3, the licensee is required to maintain personnel exposure records, records of radioactive effluents released to the environment, and environmental monitoring survey results for the life of the AFRRI reactor facility. The NRC staff reviewed the licensee's annual operating reports for the years 2008 through 2015 (Ref. 25) and noted that no exposures to reactor staff or visitors greater than 25 percent of 10 CFR Part 20 dose limits were reported. The NRC staff also reviewed the NRC IRs for the years 2008 through 2016 (Ref. 26) and noted no significant issues related to radiation exposure control and dosimetry.

The NRC staff reviewed the licensee's exposure control and dosimetry program and finds that personnel exposures at the AFRRI reactor facility are satisfactorily controlled though the AFRRI radiation protection and ALARA programs. Therefore, based on the information above, the NRC staff concludes that the radiation exposure control and dosimetry at the AFRRI reactor facility are acceptable.

3.1.6 Contamination Control

In accordance with the procedures implemented as part of the radiation protection program, which is conducted in conformance with ANSI/ANS-15.11 (Ref. 71), the licensee performs contamination surveys as necessary, depending on the frequency of the use or handling of radioactive material. As discussed in Section 3.1.2 of this SER, workers are trained on access control procedures, radiological safety principles, and working with radioactive material, which helps ensure that personnel conduct themselves such that they limit the spread of radioactive material when entering and exiting an area that contains such material. The NRC staff reviewed the licensee's annual operating reports for the years 2008 through 2015 (Ref. 25) and the NRC IRs for the years 2008 through 2016 (Ref. 26). This review showed no radiation exposures greater than 25 percent of 10 CFR Part 20 dose limits and no significant issues related to contamination control at the AFRRI reactor facility, indicating that adequate controls exist to prevent the spread of radiological contamination within the facility. Based on its review of the information above and on AFRRI's history of satisfactory contamination control, the NRC staff concludes that adequate controls exist to prevent the spread of contamination within the facility.

3.1.7 Environmental Monitoring

TS 3.5.2, "Effluents: Argon-41 Discharge Limit," presents requirements for environmental monitoring at the AFRRI reactor facility as follows:

> Specifications
>
> a. An environmental radiation monitoring program shall be maintained to determine the effects of the facility on the environs; and
>
> (…)

TS 4.5.2, "Effluents," presents surveillance requirements related to environmental monitoring as follows:

> Specifications
>
> a. The unrestricted area outside of AFRRI shall be monitored by dosimeters that shall be analyzed quarterly, not to exceed 4 months.
>
> b. Samples of soil, vegetation, and water in the vicinity of the reactor shall be collected and tested for radioactivity quarterly, not to exceed 4 months.
>
> (…)

TS 3.5.2, Specification a, requires periodic measurement of environmental radiation exposure from radionuclides produced by the AFRRI reactor facility, through the implementation of an environmental radiation monitoring program. TS 4.5.2, Specifications a and b, require the licensee to analyze environmental dosimeters and perform environmental sampling on a quarterly basis. The basis that the licensee provided for TS 4.5.2 states that experience has shown that quarterly environmental monitoring is sufficient to detect and quantify any release of radioactive material from research reactors. The licensee stated that the environmental monitoring program uses a reporting method approved by the NRC and the U.S. Environmental Protection Agency (Ref. 67).

The NRC staff noted that the surveillance intervals in TS 4.5.2, Specifications a and b, are consistent with the guidance in ANSI/ANS-15.1-2007 (Ref. 64), which recommends that environmental dosimeter analysis and environmental sampling be performed quarterly to annually.

As required by TS 6.7.3 (see Section 5.6 of this SER), the licensee must maintain records of offsite environmental monitoring surveys for the life of the facility. TS 6.6, "Operating Reports," Specification a, item 8 (see Section 5.6 of this SER), requires that a description of the results of environmental radiological surveys performed outside the facility be included in the AFRRI reactor facility's annual operating report. The NRC staff reviewed the licensee's annual operating reports for the years 2008 through 2015 (Ref. 25) and noted that all environmental sampling results were indistinguishable from normal environmental background activity levels. The NRC staff also reviewed the NRC IRs for the years 2008 through 2016 (Ref. 26) and noted the environmental dosimetry program was being conducted appropriately to evaluate impacts to the public and that no significant issues with the program were reported.

The NRC staff reviewed the information above about environmental monitoring at the AFRRI reactor facility, including annual operating reports and NRC IRs, which indicated that the operation of the AFRRI reactor facility had not adversely affected the environment. The NRC staff also reviewed TS 3.5.2, Specification a, and TS 4.5.2, Specifications a and b. The NRC staff finds that the required environmental monitoring program, as imposed by TS 3.5.2, Specification a, and TS 4.5.2, Specifications a and b, is sufficient to properly assess, and provide an early indication of, any environmental impact caused by the reactor facility operation. Therefore, based on the information above, the NRC staff concludes that the environmental monitoring program at the AFRRI reactor facility, and TS 3.5.2, Specification a, and TS 4.5.2, Specifications a and b, are acceptable.

3.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to help ensure that radioactive waste materials are identified, assessed, controlled, and disposed of in accordance with all applicable regulations, and in a manner that will protect AFRRI staff, the public, and the environment. The SAR, as supplemented, discusses the waste management program at the AFRRI reactor facility.

3.2.1 Radioactive Waste Management Program

The handling of radioactive waste at the AFRRI reactor facility is conducted under the supervision of the AFRRI Safety and Health Department. Waste handling and disposal is conducted using procedures to ensure compliance with NRC regulations, so that the health and safety of AFRRI personnel and the public will not be endangered (Ref. 1). Reactor operations generate radioactive waste in gaseous, solid, and liquid form. Gaseous effluents, primarily

Ar-41, are diluted and discharged through the facility stack while being monitored by the SGM, as discussed in SER Sections 3.1.1.1 and 3.1.4.

Solid radioactive waste generation at the AFRRI reactor facility generally consists of routine laboratory wastes such as glassware, plastics, paper, disposable gloves, vials, and biological waste (Ref. 1). As discussed in SER Section 3.1.1.3, solid waste generated at the AFRRI reactor facility is typically transferred to the AFRRI "O2" 10 CFR Part 30 byproduct license, and is not disposed of under Facility Operating License No. R-84 for the AFRRI reactor. The AFRRI Safety and Health Department packages all solid wastes, and the AFRRI radiological waste facility stores them in preparation for shipment to an NRC-authorized disposal facility (Ref. 1).

The regulations stated in 10 CFR 20.2003 allow liquid radioactive waste to be discharged into sanitary sewerage if certain conditions are met. The regulations at 10 CFR 20.2003(a) state, in part, that a licensee may discharge licensed material into sanitary sewerage if the material is readily soluble in water, and the quantity of licensed or other radioactive material that the licensee releases into the sewer in 1 month divided by the average monthly volume of water released into the sewer by the licensee does not exceed the concentration listed in 10 CFR Part 20, Appendix B, Table 3. The licensee stated that any liquid radioactive waste from the reactor is transferred to the AFRRI "O2" 10 CFR Part 30 byproduct materials license (see SER Section 3.1.1.2), and depending on its source and level of activity, is sent to one of two liquid radioactive waste subsystems (hot or warm), where it is comingled with the liquid radioactive waste from the entire AFRRI complex. At this point, the waste is measured, diluted as necessary, and released to sanitary sewerage in accordance with 10 CFR 20.2003(a) (Ref. 21).

TS 6.6, "Operating Reports," Specification a, specifies the content of the annual facility report to the NRC. TS 6.6, Specification a, item 7, specifies content pertaining to the reporting of radioactive effluents and waste as follows:

> A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the Licensee as determined at or prior to the point of such release or discharge. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient.
>
> a. <u>Liquid Waste</u> (summarized on a monthly basis)
>
> i. Radioactivity discharged during the reporting period:
>
> Total radioactivity released (in curies);
>
> Concentration limits used and isotopic composition for fission and activation products
>
> Total radioactivity of each nuclide released during the reporting period and, based on representative isotopic analysis, average concentration at point of release during the reporting period;
>
> ii. Total volume of effluent water (including diluents) during periods of release;
>
> b. <u>Gaseous Waste</u> (summarized on a quarterly basis)

Radioactivity discharged during the reporting period for:

Argon-41;

Particulates with half-lives greater than eight days;

c. Solid Waste (summarized on a quarterly basis)

Total cubic feet and combined activity in curies of materials in solid form disposed of under license R-84;

TS 6.6, Specification a, item 7.a, provides annual reporting requirements related to liquid waste releases. Because 10 CFR 20.2003(a) requires, in part, that the quantity of licensed or other radioactive material that the licensee releases into the sewer in 1 month divided by the average monthly volume of water released into the sewer by the licensee does not exceed the concentration listed in 10 CFR Part 20, Appendix B, Table 3, liquid waste is summarized on a monthly basis to help ensure compliance with 10 CFR 20.2003(a) (Ref. 21). The annual report includes (1) total radioactivity released in liquid waste, (2) the concentration limits used to show compliance with 10 CFR 20.2003(a) and the isotopic compositions used to determine the concentration limits, if liquid waste containing a mixtures of fission and activation product radionuclides is released, (3) total radioactivity and average concentration of each radionuclide released, and (4) total volume of liquid waste effluent (including any water added to dilute the effluent). TS 6.6, Specification a, item 7.b, specifies reporting requirements related to gaseous waste discharges, and requires annual reporting of Ar-41 and particulate (with half-lives greater than 8 days) discharges (summarized on a quarterly basis). TS 6.6, Specification a, item 7.c, specifies reporting requirements related to solid waste, and requires annual reporting of the total volume and combined activity of any solid radioactive materials that are disposed of (summarized on a quarterly basis).

As discussed above, liquid and solid radioactive wastes from the AFRRI reactor facility are typically not disposed of under NRC Facility Operating License No. R-84 for the reactor but are transferred to the AFRRI “O2” NRC byproduct license (see SER Sections 3.1.1.2 and 3.1.1.3) and disposed of under that license. TS 6.6, Specification a, item 7, associated with the reactor Facility Operating License No. R-84, does not require reporting of materials disposed of under any license other than the reactor Facility Operating License No. R-84. However, TS 6.6, Specification a, item 7, would require reporting of any liquid or solid radioactive materials if these were disposed of under reactor Facility Operating License No. R-84.

The transfers to the “O2” byproduct license are internal to the AFRRI facility, and there is no shipping of radioactive materials involved. In accordance with the regulations in 10 CFR 30.51 and 10 CFR 30.52, byproduct material licensees must maintain records of all radioactive waste disposed of under the 10 CFR Part 30 byproduct license, and these records are subject to inspection by the NRC. The “O2” byproduct license requires that all waste handling and disposal conducted under the “O2” license be in accordance with NRC regulations.

The NRC staff reviewed the annual reporting requirements related to radioactive waste in TS 6.6, Specification a, item 7, and finds that they (1) require an adequate summary of radioactive effluent and waste releases, (2) are consistent with the guidance provided in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 1, “Format and Content,” issued February 1996 (Ref. 39), and ANSI/ANS-15.1-2007 (Ref. 64), (3) help ensure that any liquid waste discharges to sanitary sewerage are in compliance with 10 CFR 20.2003(a), and (4) help ensure that any gaseous waste discharges are in compliance with applicable regulations. Therefore, based on the information above, the NRC staff concludes that TS 6.6, Specification a, item 7, is acceptable.

The NRC staff reviewed the licensee's radioactive waste release practices as described in the SAR, as supplemented. The NRC staff also reviewed the annual operating reports for the years 2008 through 2015 (Ref. 25) and NRC IRs for the years 2008 through 2016 (Ref. 26) for the facility and noted that all waste was released in compliance with regulations and that there were no significant issues related to radioactive waste handling or releases. The NRC staff finds that the licensee's practices demonstrate reasonable assurance that radiological releases from the facility will not exceed applicable regulatory limits nor will they pose unacceptable radiation risk to the environment or the public. The NRC staff also finds that the licensee has adequate controls in place to prevent uncontrolled personnel exposures from radioactive waste operations and to provide the necessary accountability to prevent any potential unauthorized release of radioactive waste. Therefore, based on the information above, the NRC staff concludes that the radioactive waste management program at the AFRRI reactor facility is acceptable.

3.2.2 Radioactive Waste Packaging and Labeling

As discussed in Section 3.2.1 of this SER, liquid and gaseous radioactive wastes from the AFRRI reactor facility are diluted and discharged to sanitary sewerage and through the facility stack, respectively, in compliance with the regulations in 10 CFR Part 20. Because these wastes are not packaged or shipped for offsite disposal, radioactive waste packaging and labelling is not applicable to these wastes.

As discussed in SER Sections 3.1.1.3 and 3.2.1, solid waste generated at the AFRRI reactor facility is typically transferred to the AFRRI "O2" byproduct license, and is not disposed of under Facility Operating License No. R-84 for the AFRRI reactor. As discussed in the SAR (Ref. 1), radioactive solid wastes that are generated are temporarily stored in clearly labeled containers located throughout the facility, and are periodically removed to the AFRRI radiological waste facility for long-term storage and preparation for shipment. The AFRRI Safety and Health Department is responsible for the preparation of solid waste for shipment to an NRC-authorized disposal facility. The "O2" byproduct license requires that all waste handling disposal conducted under the "O2" license be in accordance with NRC regulations. Shipment of waste is done in accordance with 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," and the U.S. Department of Transportation regulations, including regulations for waste packaging and labelling.

The NRC staff reviewed the information above and finds that packaging and labelling of radioactive wastes at the AFRRI reactor facility will be conducted in compliance with applicable regulations. Therefore, based on the information above, the NRC staff concludes that the licensee's radioactive waste packaging and labelling is acceptable.

3.2.3 Release of Radioactive Waste

As discussed in Section 3.1.1.1 of this SER, gaseous wastes (particularly Ar-41) from the AFRRI reactor facility are diluted and discharged through the facility stack while being monitored by the SGM. The licensee demonstrated that these releases would not cause the dose limits in 10 CFR Part 20 to be exceeded.

As discussed in Sections 3.1.1.2 and 3.2.1 of this SER, liquid wastes from the reactor facility are transferred to the AFRRI "O2" byproduct material license, and comingled with the liquid radioactive waste from the entire AFRRI complex in one of two liquid waste subsystems, depending on the level of activity. The "O2" byproduct license requires that all liquid waste handling and disposal conducted under the "O2" license be in accordance with NRC regulations. The waste is diluted as necessary, and sampled and analyzed to ensure compliance with 10 CFR 20.2003(a) solubility requirements and activity concentration limits before being released to sanitary sewerage (Refs. 1 and 21).

As also discussed in Section 3.1.1.3 and 3.2.1 of this SER, solid wastes from the AFRRI reactor facility are also typically transferred to the AFRRI "O2" byproduct material license. The "O2" byproduct license requires that all solid waste handling and disposal conducted under the "O2" license be in accordance with NRC regulations. Handling of these wastes is done by the AFRRI Safety and Health Department, and, following temporary storage, all solid radioactive wastes are ultimately shipped to an NRC-authorized offsite disposal facility.

The NRC staff reviewed the information above, and also reviewed AFRRI's annual operating reports for the years 2008 through 2015 (Ref. 25) and NRC IRs for the years 2008 through 2016 (Ref. 26) for the facility. The NRC staff noted that all waste was released in compliance with regulatory requirements, and that there were no significant issues related to radioactive waste releases. Therefore, based on the information above, the NRC staff concludes that the releases of radioactive waste from the AFRRI reactor facility are acceptable, and that there is reasonable assurance that such releases will not pose a significant risk to the public health and safety or to the environment.

3.3 Conclusions

Based on its review of the information in the SAR, as supplemented, and its observations and review of the licensee's operations, the NRC staff concludes the following regarding the licensee's radiation protection program and radioactive waste management:

- AFRRI's radiation protection program complies with the requirements in 10 CFR 20.1101(a) and 10 CFR 20.1101(c), is acceptably implemented, and provides reasonable assurance that the facility staff, the environment, and the public are protected from unacceptable radiation exposures. The licensee has acceptably staffed and equipped its radiation protection program. The radiation protection staff has acceptable lines of authority and communication to carry out the program.

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

- The licensee has adequately identified and described potential radiation sources. The licensee also sufficiently controls radiation sources.

- AFRRI's ALARA program complies with the requirements in 10 CFR 20.1101(b). The program applies the guidance of ANSI/ANS-15.11 (Ref. 71) by implementing time, distance, and shielding to reduce radiation exposures. Based on a review of historical radiation doses and current controls for radioactive material at the facility, there is reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA.

- The radiation monitoring and surveying program at AFRRI helps ensure compliance with 10 CFR 20.1501 and 10 CFR 20.1502. The results of radiation surveys carried out at AFRRI, doses to the persons issued dosimetry, and the results of the environmental monitoring program help confirm that the implementation of the radiation protection and ALARA programs are effective.

- 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 exposures. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas provide reasonable assurance that doses to AFRRI staff and the public will be below the applicable limits in 10 CFR Part 20.

- The facility radioactive waste management program provides reasonable assurance that radioactive waste produced at the facility will be controlled and handled in accordance with applicable regulations, and its release will not pose an unacceptable radiation risk to the environment and the public.

The NRC staff reviewed the AFRRI radiation protection program and radioactive waste management program as described in the SAR, as supplemented. The NRC staff finds that the licensee implements adequate and sufficient measures to minimize radiation exposure to facility workers and the public. Therefore, the NRC staff concludes that there is reasonable assurance that the AFRRI radiation protection and radioactive waste management programs will provide acceptable radiation protection to AFRRI staff, the public, and the environment.

4. ACCIDENT ANALYSES

4.1 Accident-Initiating Events and Scenarios

The accident analysis presented in the Armed Forces Radiobiology Research Institute's (AFRRI) safety analysis report (SAR) (Ref. 1), as supplemented, helped establish safety limits (SLs) and limiting safety system settings that are imposed on the AFRRI reactor through the technical specifications (TSs). The licensee analyzed potential reactor transients and other hypothetical accidents. The licensee also analyzed the potential effects of natural hazards as well as potential accidents involving the operation of the reactor. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's analytical assumptions, methods, and results. In addition, the NRC staff performed independent calculations and obtained independent analysis of accidents with other Training, Research, Isotopes, General Atomics (TRIGA) reactors (Refs. 63 and 72) and compared those results with accidents analyzed by the AFRRI licensee. As discussed below, none of the potential accidents considered in the SAR, as supplemented, would lead to significant occupational or public radiation exposure.

NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," (Ref. 39), suggests that each licensee consider the applicability of each of the following accident scenarios:

- maximum hypothetical accident (MHA)
- loss-of-coolant accident (LOCA)
- accidental insertion of reactivity
- loss-of-coolant flow
- mishandling or malfunction of fuel
- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment

4.1.1 Maximum Hypothetical Accident

In its January 17, 2012, response (Ref. 12) to an NRC staff request for additional information (RAI), the licensee stated that the failure of a fueled experiment in air was the MHA for the AFRRI reactor facility, and provided an analysis of this MHA. The MHA assumes that the fueled experiment fails and that all noble gases and halogen fission products accumulated inside the experiment capsule are directly released into the reactor room air without radioactive decay. The fueled experiment failure is assumed to include one gram of 19.75-percent low-enriched uranium (LEU) irradiated in the AFRRI reactor for 42 minutes at 1 megawatt thermal (MWt) and was exposed to a neutron flux level at the sample location of 1×10^{13} neutrons per square centimeter per second. The 42-minute sample irradiation time at 1 MWt (the typical power level for full-power operation at AFRRI) was assumed because it allows the fueled experiment to reach the TS 3.6, Specification b, limit of 1 curie (Ci) for iodine (I) isotopes I-131 through I-135. The fission product inventory of the fueled experiment was calculated by the ORIGEN code. The released radionuclides were assumed to diffuse in the air of the reactor room and would ultimately be released to the unrestricted area. Because research reactors do not have specific accident-related regulations, the NRC staff compared calculated dose values for accidents to standards in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation," amendments to 10 CFR 20.1001, "Purpose," through

10 CFR 20.2402, “Criminal penalties”; and to the appendices in 10 CFR Part 20. These standards and appendices became effective on January 1, 1994. These amendments changed the dose limits for occupationally exposed persons and members of the public, and the concentrations of radioactive material that are allowed in effluents released from licensed facilities. The licensee must follow the requirements in 10 CFR Part 20, as amended, for all aspects of facility operation. The doses for all accidents analyzed and discussed later in this section are within the limits required by the current version of 10 CFR Part 20.

The licensee performed the dose analyses for the MHA by considering two ventilation system scenarios. Under normal operating conditions, TS 3.4, “Ventilation System,” requires the operation of the ventilation system during reactor operations. If the reactor deck continuous air particulate monitor (CAM) detects a release of airborne activity, the reactor room ventilation is designed to be automatically secured through the closure of system isolation dampers.

Scenario 1 assumes that the ventilation system isolation function fails (dampers remain open) following release of radioactive materials to the reactor room. These assumed conditions result in the prompt release of the radioactive materials through the AFRRI stack outside the reactor room and the building housing the AFRRI reactor, to the unrestricted area. Therefore, scenario 1 results in higher dose to members of the public and helps ensure that the accident dose analysis for the members of the public is conservative.

Scenario 2 assumes that the CAM detects the elevated airborne activity, and the reactor room ventilation system is automatically secured through the closure of system isolation dampers for the reactor room. In scenario 2, the release is confined to the reactor room, resulting in a greater dose to the persons in the reactor room and adjacent areas of the building, and helping ensure that the accident dose analysis for doses in these locations is conservative.

The licensee calculated the dose to AFRRI reactor staff, who are occupational radiation workers, in the reactor room for both scenarios. For scenario 2, the licensee also calculated the doses to other AFRRI staff (receptors A, B, and C discussed below), some of whom are occupational radiation workers but others of whom may be members of the public, who are located outside the reactor room in other areas of AFRRI.

Additionally, for the release in scenario 1, the licensee calculated the dose to members of the public in the unrestricted area at various distances from the AFRRI facility stack. Although the scenario 1 release point is the AFRRI stack, 13 meters (m) (43 feet (ft)) above the ground, the licensee conservatively assumed a ground release. The licensee also assumed the release of 100 percent of the noble gas and 25 percent of the halogen fission products from the failed experiment to the unrestricted area. This release is assumed to occur with a single reactor room air change. Based on the reactor room volume of 917 m^3 (32,384 ft^3) and the normal ventilation exhaust flow rate of 1.68 m^3 per second (59.3 ft^3 per second), the radiological release duration is 9.1 minutes, which conservatively maximizes the dose rate to members of the public exposed to the accident plume. If the reactor room isolation dampers close, as assumed in scenario 2, the fission product release to the unrestricted area would be slower and would be based on the leakage rate from the reactor room. The isolation of the reactor room dampers will result in a lower public dose and would be bounded by public dose results calculated in scenario 1.

The AFRRI MHA also assumes the following conditions:

- The analysis of both scenarios assumes an occupational dose to the operating staff inside the reactor room for an evacuation period of 5 minutes.

- The analysis for scenario 2 assumes a dose to AFRRI staff members, identified as receptors A, B, and C, from direct external exposure from the released airborne radioactive materials in the reactor room. By electronic mail dated September 27, 2016 (Ref. 83), the licensee clarified that receptors A and B would be occupational radiation workers, while receptor C could potentially be a member of the public. The receptors are located as follows:

  – Receptor A is 3 ft (0.91 m) from any reactor room perimeter wall, but not within the reactor room. This condition represents an AFRRI reactor staff member closest to the reactor perimeter with an assumed evacuation time of 5 minutes.

  – Receptor B is 20 ft (6.1 m) from any reactor room perimeter wall, with one additional concrete block wall between receptor B and the reactor perimeter wall. This condition represents an AFRRI staff member (who is not necessarily a reactor staff member, but who is an occupational radiation worker (Ref. 83)) at a location closest to the controlled access area with an assumed evacuation time of 20 minutes.

  – Receptor C is 100 ft (30 m) from any reactor room perimeter wall, with two additional concrete block walls between receptor C and the reactor perimeter wall. This condition represents the location of the closest emergency evacuation assemblage point where AFRRI staff, who could be members of the public, are assumed to remain for 2 hours. In its August 5, 2015, RAI response (Ref. 21), the licensee clarified that that the actual emergency evacuation assemblage points are at the extremes of the AFRRI site as far from the reactor building as possible, over 100 ft (30 m) from any reactor wall. The area within the AFRRI site boundary is within the scope of the AFFRI Reactor Facility Emergency Plan (Ref. 1) for reactor facility emergencies.

- Twenty-five percent of the halogens released from the failed experiment do not plate out and are ultimately released to the unrestricted area. In its August 5, 2016 (Ref. 21), RAI response, the licensee clarified that the doses calculated for operating staff in the reactor room and for AFRRI facility occupants (receptors A, B, and C) include the external dose contribution from the radiation emitted by the 75 percent of the halogens that plate out in the reactor room.

- One hundred percent of the noble gases from the failed experiment capsule are released to the unrestricted area.

- All halogen and noble gas activity released to the reactor room is instantly mixed uniformly with the reactor room air.

- All the reactor room air containing the radioactive gases is available for release to the unrestricted area.

- The dose rates calculated are given for the time of failure of the fueled experiment and do not include decay corrections for the duration of the evacuation times.

- A Gaussian plume dispersion model with ground release that assumed the most stable atmospheric class (Pasquill F) with a wind speed of 1 m/second (3.3 ft/second) and neglected deposition and meandering processes was used to compute the doses external to the reactor building.

- The model assumed that the prevailing wind for the duration of the release was always in the direction of the receptor.

The NRC staff reviewed the licensee's radionuclide inventory generated using the ORIGEN code for the MHA, including the assumptions and boundary conditions. The NRC staff finds that the licensee's radionuclide inventory is accurately represented for the fuel used in the fueled experiment. Based on the evacuation performance demonstrated during routine evacuation drills conducted by the licensee at the AFRRI facility, the NRC staff considers the use of a 5-minute exposure to occupationally exposed workers in the reactor room and the 5-, 20- and 120-minute exposure to receptors A, B, and C, respectively, to be conservative and reasonable.

Table 4-1 presents the licensee's MHA analysis results for the occupational dose [total effective dose equivalent (TEDE) millirem (mrem)] to persons in the reactor room, for scenarios 1 and 2.

Table 4-1 MHA, Occupational Staff, TEDE Dose

Scenario	Exposure Criteria	TEDE (mrem)
1	Ventilation system on	401
2	Ventilation system off	508

Table 4-2 presents the licensee's results for the external dose to AFRRI facility occupants (receptors A and B, who are occupational radiation workers, and receptor C, who is assumed to be a member of the public) from shine from the radioactive material released into the reactor room, for scenario 2 in which all radioactive material is assumed to remain in the reactor room.

Table 4-2 MHA, AFRRI Facility Occupant, TEDE Dose

<table>
<tr><th>Exposure Criteria</th><th>Receptor</th><th>Evacuation Time (minutes)</th><th>TEDE (mrem)</th></tr>
<tr><td rowspan="3">Scenario 2—
Ventilation system off</td><td>A</td><td>5</td><td>9.9</td></tr>
<tr><td>B</td><td>20</td><td>7</td></tr>
<tr><td>C</td><td>120</td><td>4</td></tr>
</table>

Table 4-3 presents the licensee's results for members of the public located outside the facility in unrestricted areas, for scenario 1 in which radioactive material is released from the reactor facility stack (but for which a ground release was conservatively assumed). Table 4-3 also presents the results of confirmatory calculations (discussed below) that the NRC staff performed for doses to members of the public for scenario 1.

Table 4-3 MHA, Member of Public, TEDE Dose

Exposure Criteria	Distance (meters)	TEDE (mrem)	
		AFRRI	NRC
Scenario 1—Ventilation system on	10	76	–*
	50	33	65
	100	11	57
	150	5	30
	200	3	17
	250	2	11

*The HOTSPOT code cannot calculate dose values close to the source.

The NRC staff reviewed the results of the licensee's MHA dose calculations, and compared the results to the dose limits in 10 CFR Part 20, which are as follows:

- 10 CFR 20.1201, "Occupational dose limits for adults"—5,000 millirem (mrem) whole-body dose TEDE

- 10 CFR 20.1301, "Dose limits for individual members of the public"—100 mrem TEDE

The licensee's results demonstrate that the occupational and public exposure are less than the limits in the current versions of 10 CFR 20.1201 and 10 CFR 20.1301.

The NRC staff performed a confirmatory analysis to verify the licensee's results for doses to members of the public for scenario 1. The NRC staff used the HOTSPOT computer code with the licensee's assumptions on release rates and atmospheric stability criteria. The NRC-calculated public dose values listed in Table 4-3 differ from the licensee-calculated values due to the differences in the calculation methodology and assumptions used by the licensee and the NRC staff. However, both the licensee's and the NRC staff's calculation results are below the maximum exposure limits for members of the public in 10 CFR Part 20. Because the HOTSPOT code cannot calculate dose values very close to a release point, the NRC staff's confirmatory analysis does not include a calculation of the dose at 10 m (32.8 ft) from the release point. However, given the ground release assumption used in the calculations, any calculation of the dose at 10 m (32.8 ft) would be overestimated. Since an actual release would be from an elevated release point (the facility stack), most of the released material would pass overhead a receptor at 10 m (32.8 ft), and the dose to that receptor would result primarily from shine from the overhead plume, rather than from submersion in a cloud of radioactive material.

The NRC staff noted that for scenario 2, although the ventilation system would be off and the dampers closed, some radioactive material could potentially still leak from the reactor room into other areas of the AFRRI facility. Since the calculated doses to AFRRI facility occupants (receptors A, B, and C) in Table 4-2 only account for shine from material in the reactor room and do not account for any material that may leak from the reactor room to the receptor locations, the actual doses to receptors A, B, and C for scenario 2 could be higher than the Table 4-2 values. However, given that only a portion of the material in the reactor room would be expected to leak to the receptor A and B locations during the 5- and 20-minute evacuation times assumed for those locations, the concentrations of radioactive material at those locations would be below that in the reactor room. The dose to receptor A, who has a 5-minute evacuation time, would be bounded by the 508 mrem dose calculated for a worker who remains in the reactor room for 5 minutes for scenario 2. Since receptor A is an occupational radiation worker, this dose would therefore continue to be well below the regulatory limit of 5,000 mrem. Receptor B (also an occupational radiation worker) has a longer 20-minute evacuation time, but if the 508 mrem dose were multiplied by a factor of 4 to account for the evacuation time and generate an conservative estimate of 2,032 mrem for the dose to Receptor B, that estimate would still be well below the regulatory limit. For receptor C, an AFRRI staff member outside the building who could potentially be a member of the public, the dose would be bounded by the sum of the 4 mrem external shine dose for scenario 2 in Table 4-2 and the 76 mrem dose to a member of the public 10 meters from the stack for scenario 1 in Table 4-3. The 76 mrem dose is greater than any dose that would be received by receptor C from reactor room leakage, because (1) the 76 mrem dose assumes that all airborne material in the reactor room (except iodines that plate out) are released, and that all of the material moves in the direction of the receptor, while for receptor C, only a portion of the material would be released and move in the direction of the receptor; and, (2) receptor C is greater than 10 m (32.8 ft) from the reactor room, allowing for more dispersion of the material. However, the sum of these doses, 80 mrem, would still be below the 100 mrem regulatory limit for dose to members of the public.

The NRC staff reviewed the licensee's analysis of the consequences of the MHA. The NRC staff finds that the licensee's analyses used qualified methodologies with an acceptable radiation source term and incorporated conservative or justifiable assumptions on other boundary conditions. In conducting the MHA evaluation, the NRC staff used the dose limits in 10 CFR Part 20. The NRC staff finds that the calculated occupational and public radiation exposures for the MHA are within the limits applied for licensing of the AFRRI reactor facility and also within the limits in 10 CFR Part 20. Based on the information above, the NRC staff concludes that the potential MHA calculated doses are within acceptable limits.

4.1.2 Fuel Element Cladding Failure

The licensee analyzed a fuel element cladding failure in air in the SAR, as supplemented in its June 20, 2011 (Ref. 8), response to an RAI, to ensure that the consequences of the designated MHA (fueled experiment failure) bounds the fuel cladding failure accident. The fuel element cladding failure event assumes the instantaneous release of noble gases and halogen fission products directly into the reactor room air without radioactive decay. Boundary conditions and assumptions include the use of a conservative fuel element power density during operation of 20.2 kilowatts thermal (kWt) in a fuel element with saturated inventories for all released isotopes except krypton-85.

The licensee did not present a mechanism for the occurrence of the accident; however, it assumed that the cladding of the fuel element fails and that fission products accumulated in the gap between the fuel and clad are released into the reactor room. These nuclides would diffuse in the air of the reactor room and would ultimately be released to the unrestricted area. The NRC staff reviewed the licensee's radionuclide inventory generated using the ORIGEN code and the assumptions and boundary conditions used for the fuel element cladding failure accident analysis and finds that they are consistent with the guidance in NUREG-1537 (Ref. 39) and are representative of the LEU fuel used in the AFRRI reactor.

The licensee analyzed doses for the fuel element cladding failure. As was the case for the MHA, the fuel element cladding failure assumes the same two scenarios related to the ventilation system. Scenario 1 assumes the ventilation system fails to secure (continues to operate with isolation dampers open). Scenario 1 helps to ensure that the accident dose analysis for the members of the public is conservative. Scenario 2 assumes the ventilation system isolation dampers close as designed. Scenario 2 helps to ensure that the accident dose analysis for occupational doses is also conservative.

The doses are calculated for the same occupational and public receptors that were considered for the MHA. The analysis is based on the same general assumptions that were used for the MHA, with the following changes or additions:

- Fission product inventory was calculated for a TRIGA LEU fuel element at 20.2 kWt reaching saturated conditions for the radioactive halogen and noble gas fission products.

- To calculate the fission product release fraction, the licensee assumed that the fuel element has an average temperature of 500 degrees Celsius (°C), 932 degrees Fahrenheit (°F) which is 60 °C (140°F) higher than the calculated peak fuel temperature of 440 °C (824 °F). The licensee applied a release fraction of 1.22×10^{-4} to arrive at the total gap activity. The fission product release fraction is based on experiments performed by General Atomics (GA) on TRIGA fuel (Ref. 72).

Table 4-4 presents the licensee's MHA analysis results for the occupational dose to persons in the reactor room, for scenarios 1 and 2.

Table 4-4 Fuel Element Failure, Occupational Staff, TEDE Dose

Scenario	Exposure Criteria	TEDE (mrem)
1	Ventilation system on	93
2	Ventilation system off	122

Table 4-5 presents the licensee’s results for the external dose to AFRRI facility occupants (receptors A and B, who are occupational radiation workers, and receptor C, who is assumed to be a member of the public) from shine from the radioactive material released into the reactor room, for scenario 2 in which all radioactive material is assumed to remain in the reactor room.

Table 4-5 Fuel Element Failure, AFRRI Facility Occupant, TEDE Dose

Exposure Criteria	Receptor	Evacuation Time (minutes)	TEDE (mrem)
Scenario 2—Ventilation system off	A	5	0.65
	B	20	0.43
	C	120	0.26

Table 4-6 presents the licensee’s results for members of the public located outside the facility in unrestricted areas, for scenario 1 in which radioactive material is released from the reactor facility stack (but for which a ground release was conservatively assumed).

Table 4-6 Fuel Element Failure, Member of Public, TEDE Dose

Exposure Criteria	Distance (meters)	Whole-Body Dose (TEDE) (mrem)
Scenario 1—Ventilation system on	10	22
	50	10
	100	4
	150	1
	200	<1
	250	<1

For all cases, the TEDE is below the MHA (i.e., fueled experiment failure accident) dose values. The NRC staff find that total doses to AFRRI facility occupants (receptors A, B, and C) for scenario 2, which could include the external shine doses in Table 4-5 as well as the dose from material leaking from the reactor room, if estimated as discussed in the MHA section above, would also be below MHA values. The NRC staff finds that the calculated occupational and public radiation exposures for the fuel element failure are bounded by the MHA and are within the limits of 10 CFR Part 20 and, therefore, are acceptable.

4.1.3 Insertion of Excess Reactivity

TS 3.1.2 includes a limiting condition for operation for the reactor pulse mode. This specification states that the maximum step insertion of reactivity shall be $3.50 (2.45 percent delta k/k [% $\Delta k/k$, excess reactivity in percent]) in the pulse mode of operation to help ensure the peak fuel temperature is less than 830 °C (1,526 °F). TRIGA fuel-operating experience has demonstrated that fuel damage could occur as a result of hydrogen gas accumulation and redistribution in the hydride fuel if the reactor is pulsed with a large reactivity addition at fuel temperatures exceeding 830 °C (1,526 °F) after an extended period of operation at 1 MWt or greater. Because of this physical property of the fuel, the pulse maximum reactivity insertion is limited to \$3.50, and the allowable initial power level at the instant of pulse initiation is limited to less than or equal to 1 kWt.

The licensee analyzed step insertions of reactivity up to \$4.00 using the BLOOST transient analysis computer code. This code performs combined reactor kinetics heat transfer calculations using a point kinetics model that analyzes reactor core power and fuel temperature reactivity transients with a variable-temperature fuel heat capacity model. The fuel element is reasonably considered adiabatic for the short duration of the neutronic pulse. The BLOOST model predicts the core average fuel temperature response and can use power-peaking factors to model the highest power density locations in the reactor core. The NRC staff considers the BLOOST code an acceptable methodology for analyzing reactivity insertion events in TRIGA reactors and that it provides a conservative prediction of peak power, energy release, and fuel temperature.

As discussed in Section 2.6 of this SER, the licensee presented BLOOST results that show the peak fuel temperature is 666 °C (1,230.9 °F) for a reactivity insertion of \$3.00 and 831 °C (1,527.8 °F) for a reactivity insertion of \$4.00. The maximum step insertion of reactivity specified in TS 3.1.2 is \$3.50 (2.45% Δk/k), which results in a corresponding peak fuel temperature well below the pulsing temperature limit of 830 °C (1,526 °F). The licensee demonstrated that the reactor can be pulsed by inserting a maximum reactivity of \$3.50 without exceeding the 830 °C (1,526 °F) limit.

Step reactivity insertions of a large magnitude could only occur by accidental pulsing of the reactor or by insertions of a high-reactivity experiments. TS 3.2.2, Table 3, “Minimum Reactor Safety System Interlocks,” includes a requirement for a pulse initiation interlock that prevents pulsing the reactor at a reactor power of greater than 1 kWt. This interlock helps to ensure that a step reactivity insertion remain less than \$3.50, which prevents exceeding the 830 °C (1526 °F) fuel temperature limit. The 1 kWt interlock prevents air from being applied to the transient rod piston at steady-state reactor power levels above 1 kWt, thereby preventing the pulsing of the reactor. To minimize the possibility of a step reactivity insertion event from the insertion of a high-reactivity worth experiment, TS 3.6, “Limitations on Experiments,” Specification e, limits the worth of experiments as follows:

> The sum of the absolute reactivity worth of all experiments in the reactor and in the associated experimental facilities shall not exceed \$3.00 (2.1% Δk/k). This includes the total potential reactivity insertion that might result from experiment malfunction, accidental experiment flooding or voiding, and accidental removal or insertion of experiments. The absolute reactivity worth of any single secured experiment shall not exceed \$3.00 (2.1% Δk/k). The absolute reactivity worth of any single moveable or unsecured experiment shall be less than \$1.00 (0.70% Δk/k). The combined absolute reactivity worth of multiple moveable or unsecured experiments in the reactor and associated experimental facilities at the same time shall be less than \$1.00 (0.70% Δk/k);

Additionally, reactor operators complete prestartup checks and are required to use and follow approved written procedures for pulsing the reactor and for the insertion and removal of experiments in the reactor or other experimental facilities.

In its September 30, 2016 (Ref. 84), letter, the licensee provided additional analysis to supplement the SAR and earlier RAI responses dated February 9, 2016 (Ref. 19), and

April 20, 2012 (Ref. 13), related to a ramp reactivity insertion event. In its February 9, 2016, RAI response (Ref. 19), the licensee stated that there are two independent, redundant power and scram channels. If the first scram channel failed to mitigate the power increase resulting from a ramp reactivity insertion, then the second channel would terminate the reactivity insertion at the same power level, 1.09 MWt. For the purpose of the analysis, the event mitigation provided by the interlock preventing rod withdrawal on a short-period condition (less than or equal to a 3-second period) is ignored, and a maximum 0.5-second delay from scram initiation to control rod insertion (to close relay contacts and bleed the magnetic field or air pressure) is assumed based on the slowest responding instrument channel.

The largest average insertion rate is the transient rod drive, 0.0997 $/second. Initiating the insertion of excess reactivity at 1.0 MWt, the sequence and results would be as follows:

(1) The reactor reaches the scram initiation setpoint of 1.09 MWt.

(2) The reactor scram signal is initiated.

(3) The control rod continues to drive out for 0.5 seconds (instrumentation response time delay).

(4) The resultant additional reactivity insertion would be $0.05.

(5) The resultant positive period would be 210 seconds.

(6) The resultant peak power would be 1.093 MWt.

(7) The resultant maximum temperature reached would be 416 °C (780.8 °F).

The licensee performed a similar analysis for the other three control rods. The maximum temperatures that would be reached for the safety, shim, and regulating control rods were 415 °C (779 °F), 416 °C (780.8 °F), and 416 °C (780.8 °F), respectively.

At an initial steady-state power of 100 watts (W) thermal, the maximum reactivity insertion caused by the withdrawal of the transient rod is $0.98 before it reaches the reactor scram setpoint of 1.09 MWt. The maximum reactivity insertion of $0.98 is well below the limiting reactivity insertion of $3.50 at which the fuel temperature stays below the SL. Initiating the transient control rod withdrawal scenario from the steady-state power of 1 MWt limits the maximum reactivity insertion to $0.13 before initiation of a reactor scram. Again, this limit is well below the $3.50 limit.

In its September 30, 2016, RAI response (Ref. 84), the licensee provided additional analysis to supplement the SAR and RAI responses dated April 20, 2012 (Ref. 13), February 9, 2016 (Ref. 19), and August 5, 2016 (Ref. 21), related to a ramp reactivity insertion event caused by the simultaneous and continuous withdrawal of three control rods. This event was considered since it is possible to control all three standard control rods simultaneously on the servo. This is considered more limiting than the single control rod continuous rod withdrawal event. In this scenario, the licensee assumed that one of two independent and redundant power and scram channels would fail to perform its protective function and the second would terminate the reactivity insertion at a power level of 1.1 MWt. The event mitigation provided by the interlock preventing rod withdrawal on a short-period condition (less than or equal to a 3-second period) is assumed to limit the period to 3-seconds; a 1.0-second delay from scram initiation to control rod insertion (to close relay contacts and bleed the magnetic field or air pressure) was assumed based on the TS 3.2.1, Specification c of less than 1-second from scram initiation to full rod insertion.

The event was assumed to initiate from a critical reactor at a power of 100 Wt. Although the event models the withdrawal of three control rods, the reactor period interlock of 3 seconds limits the reactivity addition rate. The event progresses as follows:

(1) The reactor reaches the limiting condition for operation setting of 1.1 MWt.

(2) The reactor scram signal is initiated.

(3) The control rod continues to drive out for 1.0 seconds (instrumentation response time delay).

(4) The resultant peak power would be 1.52 MWt.

(5) The resultant peak power is less than 1.99 MWt which corresponds to a departure from nucleate boiling ratio (DNBR) of 1.0.

This analysis result does not exceed 1.99 MWt, which has been determined by calculation to result in a DNBR of 1.0 (See SER Section 2.6). Therefore, this event poses no significant challenge to the reactor fuel integrity.

The licensee also analyzed a postulated improper fuel loading scenario in which an operator mistakenly inserts a fuel element in a core that is critical at low power (less than 1 watt). The limiting case for the improper fuel loading scenario assumes the rapid insertion of a fuel element in the B ring that results in a reactivity insertion of $0.95 (0.67% Δk/k), which would not result in the exceedance of reactor fuel temperature limits. This scenario is unlikely to occur because the B ring does not have an open location in which to insert a fuel element. A more probable improper fuel loading scenario would be the insertion of a fuel element in a peripheral fuel element location. This scenario results in a reactivity addition of $0.30 (0.21% Δk/k), initiating a mild transient in which the reactor would continue to operate at a higher power level of 50 kWt.

The NRC staff concludes that fuel damage events from accidental reactivity insertions are unlikely. In SAR Section 4.5.3, “The Core Excess Reactivity” (Ref. 2), as supplemented, the licensee provides the results of analyses that demonstrates that both the physical characteristics of the fuel and the TS limitations help ensure that fuel integrity would be maintained and that no fission products would be released from the fuel by accidental reactivity insertions.

The NRC staff reviewed the licensee’s evaluation of a postulated reactivity insertion event, as described in the SAR, as supplemented, and finds that (1) the licensee’s analysis is comprehensive and (2) the results demonstrate that the maximum fuel temperature reached during the positive reactivity addition event would remain well below the TS SL of 1,000 °C (1832 °F) and the pulse mode fuel temperature limit of 830 °C (1526 °F), nor does it result in a DNBR of 1.0 or less. The NRC staff finds that the licensee’s postulated step and ramp scenarios that could result in a potential positive reactivity insertion event to be realistic examples for a pulsing TRIGA research reactor. The NRC staff also finds that the licensee’s stated design features and administrative controls (1) would render the postulated positive reactivity insertion events unlikely, (2) are consistent with the controls and design features found at other pulsing TRIGA reactors, and (3) can be maintained, tested, and inspected by the licensee to help ensure operability and TS compliance. Based on the information described above, the NRC staff concludes that the licensee has sufficient design features and administrative restrictions in place to make accidental pulsing or removal of secured experiments during reactor operation to be unlikely, and that the TS SL would not be exceeded if either event were to occur.

4.1.4 **Loss-of-Coolant Accident**

The licensee provides the results of the loss of coolant accident (LOCA) in SAR Section 13.2.1 (Ref. 2), as supplemented by its August 5, 2016, RAI response (Ref. 21). Two scenarios could result in a significant loss of coolant from the reactor pool—(1) the pumping or draining of the water from the reactor pool from installed components or (2) a pool penetration failure. Pumping or draining all of the coolant water in the pool from installed components is not likely because of the location and orientation of the piping and the installed siphon breaks.

Pool-level instrumentation and the AFRRI staff would provide early detection of tank failures caused by corrosion or other failures that lead to a slow loss of water. A drop in pool level will set off a low-level alarm, which will promptly initiate a corrective response by AFRRI staff in accordance with approved operations procedures or standing orders. The licensee stated that the reactor pool is continually monitored for leakage. The operator must measure pool water level on a daily basis. During nonduty hours, AFRRI's guard force has standing orders to notify the on-call reactor staff and the safety and facilities staff upon detection of water leaks or abnormal accumulation of water in the reactor areas. Most leaks could be replenished by adding water to the reactor pool, since there are three independent systems available to fill the pool in an emergency, and the largest of these systems consists of a 4 inch (in) (10.1 centimeters (cm)) diameter fire hose at 100 pounds per square inch pressure which can be connected directly to the pool.

The licensee calculated the consequences of the case in which all water is assumed removed from the pool. If the reactor pool is completely drained of coolant, air circulation would cool the fuel elements. Calculations and experiments performed by GA have shown that air circulation would adequately prevent fuel damage by removing the decay heat of the fuel (Ref. 72).

A more likely scenario than the complete draining of the reactor pool is the case in which the core is only partially uncovered. For a partial LOCA, GA performed experiments that concluded that the temperature rise for a partial loss of coolant is less severe than that for a complete loss of coolant, and that fuel damage would not result (Ref. 72).

To prevent fuel damage that would cause fission product release, stress on the fuel element cladding must not exceed the ultimate strength of the clad. The fuel cladding stress primarily depends on the fuel and clad temperatures. Assuming that the fuel and clad are at the same temperature and that the hydrogen-to-zirconium ratio is 1.6 to 1.7, the equilibrium hydrogen pressure at approximately 950 °C (1742 °F) produces a stress on the clad equal to its ultimate strength (Ref. 62). Based on this stress limit, GA established and the NRC staff has accepted a fuel temperature limit of 950 °C (1742 °F) when the cladding temperature is greater than 500 °C (932 °F), which could be expected during a LOCA with air cooling (Ref. 61).

In Section 13.2.1.3 of the SAR, as supplement by RAI responses dated September 21, 2012 (Ref. 14), and June 28, 2013 (Ref. 15), the licensee calculated the maximum fuel temperature during the LOCA versus two fuel element power density assumptions, including infinite operating time (300,000 years) at 1.1 MWt and operation at 72 hours per week for 40 years at a power level of 1 MWt. Assuming infinite operation, the licensee presents a bounding analysis based on GA Report No. E-117-196, "TRIGA Four-Rod Cluster Loss-of-Coolant Accident Analysis," issued October 1972 (Ref. 73). This report discusses the results of GA research that demonstrates the ability of natural convection of air to maintain fuel cladding temperature below 900 °C (1652 °F) following an instantaneous coolant loss as long as a power density of 21 kWt per element is not exceeded. The licensee's calculations show that the assumed infinite operation of the AFRRI TRIGA at a full power of 1.1 MWt yields a peak power density of 19.4 kWt in the B04 fuel element position. Given that the B04 location represents the highest power density within the AFRRI core, this scenario is bounded by the GA analysis.

The infinite operation of the AFRRI reactor is a purely hypothetical scenario. Fission product buildup within the AFRRI core limits full-power operation to less than 24 continuous hours, at which point the negative reactivity from fission product poisons prevents operation at full power. Historically, the AFRRI has operated an average of approximately 500 kilowatt-hours (equivalent to 0.5 hours of operation at 1.0 MWt) per week. To provide a bounding analysis of a LOCA at the AFRRI reactor, the licensee assumed a LOCA following 1 MWt operation for 72 hours per week for 40 years. This analysis is conservative, given the stated physical limitations introduced by the accumulation of fission product poisons and the resulting additional safety margin. The licensee assumed two cases, including LOCAs with an instantaneous loss of pool water and the loss pool water over a 15-minute period. The peak fuel temperatures for the two cases are 548 °C (1018.4 °F) and 477 °C (890.6 °F), respectively. These temperatures provide an adequate margin to the limit of 950 °C (1742 °F) for fuel not immersed in water (Ref. 61). These calculations used the GA TAC2D thermal-hydraulic code and assumed natural convection with air as the coolant, beginning either at the time of scram or 15 minutes after scram. The TAC2D computer code is generally accepted as valid for analyzing TRIGA reactor thermal hydraulics (Ref. 74). The NRC staff concludes that the LOCA scenario does not result in exceeding the 950 °C (1742 °F) limit for air cooling. Based on the above analysis and experimental results, the NRC staff concludes that neither full nor partial LOCAs would result in damage to the reactor fuel.

The reactor pool water also provides significant shielding to limit radiation exposure from the reactor core during reactor operation to the AFRRI staff and members of the public. In the event of a LOCA, AFRRI personnel and members of the public could be subject to direct or scattered gamma radiation from the exposed reactor fuel inside the reactor tank. The licensee used the Monte Carlo N-Particle Transport (MCNP) code to construct a three-dimensional model for calculating the dose rate. The calculations assumed prior operation of the reactor at 1 MWt for time periods up to 100 days. The calculations assumed that the gamma radiation energy was 1 million electron volts. Dose rates are presented for 100 days of operation as a function of time after shutdown at various axial and radial distances from the reactor core. The MCNP code is generally accepted for gamma radiation dose calculations in TRIGA reactors.

The only credible LOCA for the AFRRI would involve a large crack or hole in the core projection (nodes), which could potentially empty the pool within a short time frame (15 minutes). While the reactor is operating, the reactor roof area is not accessible to AFRRI personnel or members of the public. Emergency preparedness exercise experience has demonstrated that the reactor area can be evacuated within 1 minute and the building within 10 minutes.

After the loss of 6 inches (15.2 cm) of pool water, a building evacuation alarm is automatically sounded. As the pool continues to drain (assuming that no action would be taken to mitigate the loss of water), the building would be evacuated before the loss of enough shielding (reactor coolant) to produce measurable dose to members of the public within the AFRRI building. The emergency evacuation assemblage points are at the extremes of the AFRRI site as far from the reactor building as possible. The area within the AFRRI site boundary is within the scope of the AFFRI Reactor Facility Emergency Plan (Ref. 1) for reactor facility emergencies. The licensee stated that any non-AFRRI members of the public would be more than 60 m (197 ft) from the reactor room, with several intervening shielding walls such that any dose to members of the public would be below the 100 mrem limit of 10 CFR 20.1301. Extrapolating from unshielded dose rate calculations from the 2010 SAR LOCA analysis at 20 m (66 ft) and 30 m (98 ft), the licensee determined that the dose rate at 60 meters from the reactor would be less than 0.065 mrem per hour.

Assuming that the core is completely uncovered at 15 minutes post shutdown, an occupationally exposed individual standing on the core shroud would be standing in a radiation field of approximately 28 roentgen equivalent man (rem) per hour. Assuming evacuation time from the 50-foot-square reactor room of 2 minutes, the dose received would be approximately 1 rem. In the instantaneous LOCA scenario in which all shielding provided by the pool water is eliminated in 1 second, the dose rate above the core was calculated to be 126 rem per hour. Assuming a 2-minute stay time above the reactor core, a dose of 4.2 rem would be received, still within the 5 rem annual occupational exposure limit in 10 CFR 20.1201.

The NRC staff reviewed the licensee's LOCA analysis for the AFRRI TRIGA core and finds that the LOCA does not result in damage to the reactor fuel because the maximum fuel temperature is below the air-cooled limit of 950 °C (1742 °F). In addition, doses within the reactor facility staff are within the limits that would allow timely recovery operations to proceed. Dose to members of the public are below the limits in 10 CFR Part 20. As a result, the NRC staff concludes that the results for the LOCA analysis are acceptable and that the LOCA does not pose significant risk to public health and safety or to personnel.

4.1.5 **Loss of Coolant Flow**

As described in the SAR (Ref. 1), as supplemented on March 4, 2010 (Ref. 2), and by RAI responses dated September 27, 2010 (Ref. 4), and August 5, 2016 (Ref. 21), the AFRRI TRIGA reactor core is cooled by natural convection of pool water up between the fuel elements. The pool water is cooled by a primary cooling system that circulates primary coolant (pool water) through a heat exchanger. The heat is removed from the pool water by a secondary cooling system that circulates secondary coolant to a cooling tower.

In the event of a possible blockage of a coolant channel created by a foreign object lodged in the lower grid plate, the open fuel element lattice and coolant crossflow would facilitate adequate cooling of all fuel elements. Based on the fuel and reactor design, the licensee stated that loss of coolant flow from blocked fuel cooling channels is not a credible scenario for the AFRRI reactor.

Because the reactor fuel is cooled by natural circulation, no pumps are employed in forcing water through the core. Therefore, coolant flow is not susceptible to coolant pump failures or loss of electrical power. The absence of pumped flow helps eliminate concerns that heavier objects that have settled below the core can be drawn up into the core region. Additionally, large foreign material that could result in blockage of coolant flow is prevented from reaching the core by (1) the presence of a carriage above the core, (2) a 3/16-inch-thick aluminum shroud surrounding the core, and (3) the presence of a chain fence around the reactor pool, which prevents individuals from inadvertently walking too close to the pool while carrying large objects that could be inadvertently dropped into the pool. A skimmer near the pool water surface removes any buoyant particulates that enter the pool. Finally, tack mats at the entrance trap small debris before it can be tracked into the pool area and thus keep the room clean. The NRC staff accepts that it is extremely unlikely that a coolant blockage would develop in the core and cause fuel element damage.

The loss of heat removal, regardless of cause, would be evident to operators using various indicators (reactor power, for example) at the control console, given the negative coefficients of reactivity of the coolant and fuel. If no action is taken by the operators to mitigate the heatup of pool water, the power could decrease to a point the reactor would be subcritical. In the scenario in which secondary cooling is lost, natural convection of the pool water would continue to safely cool the reactor core while the bulk reactor pool temperature would slowly heat up at a rate of 14.1 °C (25 °F) per hour at a power of 1 MWt (Ref. 21). Because of the significant inventory of pool water, this event would progress slowly, allowing several hours for operators to take corrective action or shutdown the reactor. NRC confirmatory thermal-hydraulic analysis confirmed safe full-power operation of the reactor with coolant temperatures of 60 °C (140 °F) or less. Even if the pool water should completely evaporate and leave the core uncovered, the fuel would be adequately cooled as described in SER Section 4.1.4, “Loss of Coolant Accident.”

The NRC staff finds that a loss of either primary or secondary coolant flow would retain adequate cooling of the core. Therefore, the NRC staff concludes that these accident scenarios would not pose any danger to public health and safety or to personnel, and that the licensee’s analysis of these accidents is acceptable.

4.1.6 **Mishandling or Malfunction of Fuel**

As described in the SAR (Ref. 1), as supplemented on March 4, 2010 (Ref. 2), February 7, 2011 (Ref. 7), September 6, 2011 (Ref. 9), and January 17, 2012 (Ref. 12), all possible accidents involving the mishandling or damage of fuel elements, including dropping of the elements, other mechanical damage, or other malfunctions, are bounded by the analysis carried out for the fuel element cladding failure in air, which is further bounded by the MHA. The licensee has an established procedure for handling fuel, has no recorded incidents of mishandling its fuel elements, and has no incidents of cladding failure. TS 3.7 requires the surveillance of fuel elements by visual inspection and annual measurement to verify the continuing integrity of the fuel element cladding. A facility procedure controls the removal of a fuel element from the reactor pool.

If fuel is dropped while it is submerged or if the fuel suffers damage because of a manufacturing defect or through corrosion, the consequences of the incident would be bounded by the fuel element cladding failure accident scenario and the MHA. Moreover, scrubbing by the pool water surrounding the fuel elements would significantly mitigate any fission product release beyond the assumptions made in the MHA analysis. Fuel elements that have failed at other reactors have shown that such failure would have no significant effect on the building occupants or members of the public.

The NRC staff finds that events of mishandling or malfunctioning of fuel are unlikely and that any potential fission product release and dose to the public would be bounded by the fuel element cladding failure scenario and the MHA. Therefore, the NRC staff concludes that these accident scenarios would not pose any significant danger to public health and safety or to personnel, and that the licensee’s analysis of these accidents is acceptable.

4.1.7 **Experiment Malfunction**

SAR Section 13.1.5 (Ref. 2) describes the licensee's controls for the prevention of the occurrence of a postulated experimental malfunction. TS 3.6 and TS 4.6 place limits on experiments installed in the reactor and associated experimental facilities (see Sections 5.3 and 5.4 of this SER). The objectives of these limits are to help prevent damage to the reactor and to limit any potential releases of radioactive materials and resulting exposures to personnel in the event of an experimental failure. These TSs limit reactivity worth, mass of explosive materials, and other experiment materials to prevent or minimize accidental reactivity insertions, damage to reactor components, and release of radioactivity.

In addition, TS 6.4 requires the Reactor and Radiation Facilities Safety Subcommittee (RRFSS), Reactor Facility Director, and Health Physics Department to review and approve all new or previous experiments with substantial modification. TS 6.2 requires the RRFSS to review whether new or modified experiments can be done without prior NRC approval under the requirements of 10 CFR 50.59, "Changes, tests and experiments" (see SER Sections 5.6.2 and 5.6.4). The limits on experiments are also analyzed and approved to help ensure that releases are within 10 CFR Part 20 limits.

The NRC staff reviewed the information in the SAR (Ref. 1), as supplemented on March 4, 2010 (Ref. 2), and TSs, and finds that the licensee's limitations, controls, and procedures for experiments are in place, and are adequate to help to minimize the potential occurrence of an accidental experiment malfunction, such as the overpressurization of an experiment or an excessive step reactivity insertion. The design of experimental irradiation facilities has also been reviewed and found acceptable. If an experiment were to malfunction, the TS controls, which limit the reactivity worth, mass of explosive materials, and other experiment materials, would limit the accidental reactivity insertions, damage to reactor components, and release of radioactivity.

The licensee stated in its RAI responses dated September 6, 2011 (Ref. 9), and January 17, 2012 (Ref. 12), that the failure of a fueled experiment in air is the MHA for the AFRRI reactor facility, as discussed in Section 4.1.1 of this SER. The MHA assumes that the fueled experiment fails and that all noble gases and halogen fission products accumulated inside the experiment capsule are directly released into the reactor room air without radioactive decay. The fueled experiment failure is assumed to include one gram of 19.75-percent LEU irradiated in the AFRRI reactor for 42 minutes at 1 MWt and exposed to a neutron flux level at the sample location of 1×10^{13} neutrons per square centimeter per second. The 42-minute sample irradiation time was assumed because it is the time required to reach the TS limit of 1 Ci for iodine isotopes I-131 through I-135. The fission product inventory of the fueled experiment was calculated by the ORIGEN code. The released radionuclides were assumed to diffuse in the air of the reactor room and would ultimately be released to an unrestricted area.

The NRC staff finds that the calculated occupational and public radiation exposures for the MHA are within the limits in 10 CFR Part 20 and bound other credible experiment failures. Therefore, based on the information provided above, the NRC staff concludes that the licensee has proper controls established to minimize the potential occurrence of an accidental experiment malfunction and to help ensure that the radiation dose consequences would not be more severe than the limits in 10 CFR Part 20, and that the licensee's analysis of experiment malfunctions is acceptable.

4.1.8 **Loss of Normal Electrical Power**

The license described the AFRRI reactor response to a loss of normal electrical power in the SAR (Ref. 1), as supplemented on March 4, 2010 (Ref. 2), and September 27, 2010 (Ref. 4). The licensee stated in the SAR that in the event of loss of normal power supplies, uninterruptable power supplies located in the main reactor room supply power to all electrical components listed in TS Section 3.0, "Limiting Conditions for Operations." The uninterruptable power supply system is designed to allow for an organized, orderly shutdown. Any loss of power causes a reactor shutdown. An emergency electrical power system is not necessary to safely shut down the reactor and is not required to ensure public health and safety.

In the event of a loss of electrical power without backup power, all control rods and the transient rod would be inserted into the core automatically by gravity. The gravity insertion of the control rods initiates upon the loss of power to the electromagnets for the standard control rods and to the three-way solenoid valve that maintains air pressure to hold the transient rod in position. The reactor operators can visually verify using flashlights that the control rods are in the full inserted position. Portable battery powered radiation detection instruments are available to measure radiation levels. The primary and secondary coolant pumps, which normally provide pool cooling, would stop. Reactor decay heat removal would continue through natural circulation of the primary coolant. The reactor pool has sufficient heat capacity in the coolant to absorb the decay heat from the reactor without the need for the primary or secondary cooling systems. As described in the licensee's SAR Section 13.2.1, "Loss of Coolant Accident" (Ref. 2), all coolant can instantaneously be lost and air cooling of the core will prevent fuel temperature from exceeding the 950 °C (1742 °F) SL for air cooling (Ref. 61).

The NRC staff reviewed the information in the SAR, as supplemented, and finds that, upon a loss of electrical power, the reactor will shut down and there is sufficient heat capacity in the coolant in the reactor pool to absorb the decay heat from the reactor without the need for the primary or secondary cooling system operation during the reactor shutdown. Therefore, based on the information above, the NRC staff concludes that loss of normal electrical power poses little risk to the health and safety of the public or to AFRRI staff, and that the licensee's analysis of a loss of normal electrical power is acceptable.

4.1.9 **External Events**

In the SAR, Chapter 2 (Ref. 1), as supplemented in RAI responses dated February 7, 2011 (Ref. 7), and August 5, 2016 (Ref. 21), the licensee stated that the likelihood of external events, such as hurricanes, floods, and tornadoes, is considered insignificant. Since AFRRI is located on a military installation, and the airspace is controlled, external events involving aircraft are considered unlikely. Although helicopters take off and land at the hospital near AFRRI, if a helicopter were to crash into the facility, the mechanical impacts would not directly affect the reactor, as discussed below. The likelihood of an earthquake is also considered extremely low, and the licensee's calculations show that the fuel can be cooled in air as long as the reactor is shut down. The potential consequences of external events would be bounded by a LOCA and, for fission product releases, the MHA.

Hurricanes and tornadoes should not affect the structural integrity of the reactor building and the reactor pool. The licensee stated that the most severe result of a hurricane or tornado would be the loss of electrical power that would result in a controlled reactor shutdown.

No earthquake ever recorded within a 50-mile radius of the AFRRI facility has caused any damage to the facility. The licensee stated that the probability of significant earthquakes near the facility is very remote since it is situated in a zone where the probablility for seismic activity is low; therefore, it does not consider earthquakes a credible hazard.

Mechanical impact would not directly affect the reactor. The reactor is located in a reactor pool that is more than 14 feet deep, and the pool is horizontally surrounded by a thick concrete pool wall. Any external explosion or toxic release that would have an effect on reactor operation is not considered credible because no manufacturing or chemical facility is located near AFRRI.

The NRC staff reviewed the licensees' analysis of accidents initiated by an external event and finds that the probability of this type of potential accident is extremely low. The NRC staff also finds that the fuel is protected by the facility design from external event-initiated accidents. Therefore, based on the information above, the NRC staff concludes that members of the public are not subject to undue radiological risk as a result of an external event, that external events do not pose a risk to the health and safety of the public and to the AFRRI staff that would exceed the results for fission product release of the MHA, and that the licensee's analysis of external events is acceptable.

4.1.10 **Mishandling or Malfunction of Equipment**

In its RAI response of September 27, 2010 (Ref. 4), the licensee described six categories of accident initiators associated with the mishandling or malfunction of equipment and provided an analysis for each.

These categories include the following accident initiators:

- Operator error at the controls
- Malfunction or loss of safety-related instruments or controls, such as amplifiers or power supplies
- Electrical fault in control/safety rod systems
- Malfunction of confinement
- Rapid leak of contaminated liquid, such as primary coolant
- Malfunction of lead shield doors

<u>Operator Error at the Controls</u>

The licensee stated that unanticipated operator error is accounted for by control system design with rod withdrawal prevents (RWPs) and scram logic circuitry. The system incorporates RWPs to prevent any further positive reactivity from being inserted into the core until specific conditions are satisfied, but these RWPs do not prevent a control rod from being lowered or scrammed.

The licensee stated that the scram logic circuitry assures that a specific set of reactor core and operational conditions must be satisfied for reactor operation to occur or continue in accordance with the TSs. The scram logic circuitry involves a set of open-on-failure logic relay switches in series. Any scram signal or component failure in the scram logic results in a loss of standard control rod magnet power and a loss of air to the transient rod cylinder, resulting in an automatic, orderly shutdown of the reactor.

Malfunction or Loss of Safety-Related Instruments or Controls

The licensee stated that the AFRRI reactor core is monitored by a variety of detectors. Thermocouples from each of the two instrumented fuel elements comprise two of the detectors. Fission detectors, ion chambers, and a Cherenkov detector comprise the remaining reactor monitoring channels. These detectors are used to provide (1) at least five independent channels that monitor the power level and fuel temperature of the core during steady-state operation and (2) at least three independent channels that monitor the power level and fuel temperature of the core during pulse operations. These channels are composed of a detector, interconnecting cables or lines, amplifiers, and output devices that measure a specific variable. Signals are sent to the reactor console digital and analog displays. The licensee stated that failure of any channel would not result in an accident, and failure of most channels would result in either a reactor shutdown or control rod RWP.

Electrical Fault in Control/Safety Rod Systems

The licensee stated that in the case of a standard control rod drive, an electrical fault de-energizes the electromagment releasing the armature section, resulting in the standard control rod dropping by gravity to its fully inserted position. For the transient (pneumatic) rod drive, a solenoid valve acts as an on/off switch controlling whether or not air is supplied to the pneumatic cylinder. De-energizing the solenoid valve vents the compressed air supply and relives the pressure in the cylinder, resulting in the piston dropping by gravity to its lower limit, fully inserting the transient rod into the core. This design ensures that the transient rod is fully inserted into the core except when compressed air is supplied.

Malfunction of Confinement

The licensee stated in its January 17, 2012, RAI response (Ref. 12) that the failure of a fueled experiment in air is the MHA for the AFRRI reactor facility. In the evaluation of MHA scenario 1, the ventilation system isolation function is assumed to fail on detection of the radioactive release to the reactor room. This assumption is functionally equivalent to the elimination of confinement because such conditions would result in the rapid release of the radioactive materials to the unrestricted area. The licensee provided MHA dose calculations for this scenario (discussed in SER Section 4.1.1) to demonstrate that the potential occupational and public exposures from the MHA were less than the limits in 10 CFR 20.1201, “Occupational dose limits for adults,” and 10 CFR 20.1301, “Dose limits for individual members of the public.”

Rapid Leak of Contaminated Liquid, including Potential Malfunction of the Reactor Pool

The licensee stated that the reactor tank itself is constructed with a minimum wall thickness of 0.25 in (0.64 cm) and a floor thickness of 0.5 in (1.3 cm). Any leakage beyond the aluminum tank should be encapsulated by the concrete surrounding the tank and should not pose any environmental or personnel hazards exceeding 10 CFR Part 20 limits. The licensee stated that were a leak to occur elsewhere in the primary coolant system, drains are provided such that coolant would drain to AFRRI’s radioactive waste water system, and would be retained in hold-up tanks until it could be sampled before being released to the environment in compliance with 10 CFR Part 20 limits.

Malfunction of Lead Shield Doors

The licensee stated that the reactor core dolly and lead shield door drive systems have slip clutches installed. The purpose of the slip clutches is to prevent accidental damage to the doors or core shroud by stopping movement if any obstruction should be encountered. If these clutches should fail, core travel speed is approximately 2.25 ft per minute (0.686 m per minute). This speed has been determined to be insufficient to cause damage to the 0.25 (0.64 cm) aluminum shroud surrounding the core itself.

The licensee stated that the facility interlock system is designed to eliminate the possibility of accidental radiation exposure of personnel working in the exposure rooms or the preparation area, and to prevent interference (i.e., contact or impact) between the reactor tank lead shield doors and reactor core shroud. These interlocks prevent rotation (i.e., opening or closing) of the reactor tank shield doors and the operation and movement of the reactor core between different regions of the pool unless specific operating conditions are satisfied. The interlock system is tested periodically to ensure proper operation.

The licensee stated that both exposure rooms are equipped with emergency stop buttons. An emergency stop button is also located on the reactor console. Pressing any of these emergency stop buttons causes an immediate reactor scram and gives a scram indication to the reactor operator at the control console. Magnetic power to the standard control rods and the air supply to the transient rod cylinder cannot be obtained in an emergency stop scram condition without resetting the reactor console key, which, when performed, automatically initiates a time delay with horns sounding in the appropriate exposure rooms. Therefore, the emergency stop circuit provides an independent means for an individual accidentally trapped in an exposure room to prevent an unsafe condition involving operation from occurring, while also providing a positive indication to the reactor operator that someone could be trapped in an exposure room.

NRC Staff Conclusion on Mishandling or Malfunction of Equipment

The NRC staff reviewed the information provided by the licensee and discussed above in SER Sections 4.1.10.1 through 4.1.10.6, and finds that the physical limitations of the AFRRI reactor design are such that the mishandling or malfunction of equipment would lead to consequences that are bounded by the MHA for fission product release or by the LOCA. Therefore, the NRC staff concludes that the consequences of the mishandling or malfunction of equipment would not pose a significant risk to public health and safety or to the AFRRI staff, and that the licensee’s analysis of accidents involving mishandling or manfunction of equipment is acceptable.

4.2 Conclusions

The NRC staff reviewed the licensee’s analyses of potential accidents at the reactor facility. The NRC staff concludes that the licensee has postulated and analyzed sufficient accident-initiating events and scenarios. On the basis of its review of the information in the licensee’s SAR, as supplemented, the NRC staff concludes the following:

- The licensee considered the expected consequences of a broad spectrum of 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 analyzed the most significant credible accidents and the MHA and determined that, under conservative assumptions, the most significant credible accidents and the MHA will not result in occupational radiation exposure of the AFRRI staff or radiation exposure to a member of the public in excess of the applicable 10 CFR Part 20 limits.

- The licensee has employed appropriate methods in performing the accident and consequence analyses.

- For accidents involving insertions of excess reactivity, the licensee has demonstrated that a pulse reactivity limit of $4.00 will result in peak fuel temperatures below the SL (1,000 °C) (1832 °F). The licensee has established an administrative limit for pulse reactivity insertions of $3.50.

- The TS 3.1.2 limit for pulse reactivity insertions is specified as $3.50 to ensure that during pulse operation the peak fuel temperature will stay below the pulsing temperature limit of 830 °C (1526°F). For accidents involving insertions of excess reactivity, the licensee has demonstrated that a pulse reactivity limit of $3.50 will result in peak fuel temperatures below the pulsing temperature limit (830 °C) (1526 °F).

- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures. The reactor can be safely cooled with all fuel elements in an air environment. Doses to individuals evacuating the reactor room and at the site boundary are calculated to be below the limits in 10 CFR Part 20.

- External events that would lead to fuel disruption are unlikely.

- The accident analysis confirms the acceptability of the licensed power of 1.1 MW, including the response to anticipated transients and accidents.

- The accident analysis confirms the acceptability of the assumptions stated in the individual analyses in the SAR, as supplemented.

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

The NRC staff reviewed the radiation source term and MHA calculations for the AFRRI reactor. 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 AFRRI reactor are in conformance with the requirements in 10 CFR Part 20. The licensee reviewed the postulated accident scenarios in NUREG-1537 (Ref. 39) and did not identify any other accidents with fission product release consequences not bounded by the MHA. The AFRRI reactor design features and administrative restrictions in the TSs help to prevent the initiation of accidents and mitigate associated consequences. Therefore, on the basis of its review, the NRC staff concludes that (1) there is reasonable assurance that no credible accident would cause significant radiological risk and (2) the continued operation of the AFRRI reactor poses no undue risk to the AFRRI staff, the public, or the environment.

5. TECHNICAL SPECIFICATIONS

In this section of the safety evaluation report (SER), the U.S. Nuclear Regulatory Commission (NRC) staff evaluates the licensee's proposed technical specifications (TSs). The Armed Forces Radiobiology Research Institute (AFRRI) TSs define specific features, characteristics, and conditions governing the safe operation of the AFRRI facility. TSs are explicitly included in the renewal license as Appendix A. The NRC staff reviewed the format and content of the TSs for consistency with the guidance in Chapter 14 and Appendix 14.1 to NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," issued February 1996 (Ref. 39), and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors" (Ref. 64). The NRC staff specifically evaluated the content of the proposed TSs to determine if it meets the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical Specifications." The NRC staff also relied on the references in NUREG-1537 and Interim Staff Guidance-2009-001, "Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors," dated October 15, 2009 (Ref. 42), to perform this review.

5.1 TS 1.0 Technical Specification Definitions

The licensee proposed the following definitions of frequently used terms in order to provide uniform interpretation of terms and phrases uses in the TSs. The definitions are intended to be generally consistent with the standard definitions and/or guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The licensee's proposed definitions include minor modifications to some standard definitions provided in the guidance, and also include some additional facility-specific definitions not covered in the guidance. The definitions in the TSs are as follows:

TS 1.0, "Definitions," states the following:

1.1. ALARA

The ALARA program (As Low As Reasonably Achievable) is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environment as low as reasonably achievable.

1.2. CHANNEL

A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

1.3. CHANNEL CALIBRATION

A channel calibration is 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 be deemed to include a channel test.

1.4. CHANNEL CHECK

A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same parameter.

1.5. CHANNEL TEST

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

1.6. CONFINEMENT

Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

1.7. CONTROL ROD

A control rod is a device fabricated from neutron absorbing material or fuel, or both, that is used to establish neutron flux changes and to compensate for routine reactivity losses. Scrammable control rods can be quickly uncoupled from their drive units to rapidly shutdown the reactor if needed.

1.8. CORE CONFIGURATION

The core configuration includes the number, type, or arrangement of fuel elements and standard control rods/transient rod occupying the core grid.

1.9. CORE GRID POSITION

[See the definition evaluated and found acceptable in Section 2.2 of this SER.]

1.10 EMERGENCY STOP

Emergency Stop is a scram designed to prevent or cease reactor operations. Emergency stop buttons are provided in Exposure Room 1, Exposure Room 2 and on the console

1.11. EXCESS REACTIVITY

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

1.12. EXPERIMENT

Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate nonroutine reactor characteristics or that is intended for irradiation within an experimental facility. Hardware rigidly secured to the core or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.

1.13. EXPERIMENTAL FACILITIES

The experimental facilities associated with the AFRRI TRIGA reactor shall be:

a. Exposure Room #1

b. Exposure Room #2

c. Reactor Pool

d. Core Experiment Tube (CET)

e. Portable Beam Tubes

f. Pneumatic Transfer System

g. In-core Locations

1.14. FUEL ELEMENT

[See the definition evaluated and found acceptable in Section 2.2 of this SER.]

1.15. HIGH FLUX SAFETY CHANNEL

A high flux safety channel is a power measuring safety channel in the reactor safety system, NP and NPP.

1.16. INITIAL STARTUP AND APPROACH TO POWER

Intentionally left blank

1.17. INSTRUMENTED FUEL ELEMENT

[See the definition evaluated and found acceptable in Section 2.2 of this SER.]

1.18. LONG TERM STORAGE

Long-term storage of fuel applies to fuel that has been taken out of service with no plans for use for more than one fuel measurement cycle.

1.19. MEASURED VALUE

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

1.20. MOVABLE EXPERIMENT

A movable experiment is one where it is intended that all or part of the experiment may be moved near the core or into and out of the core while the reactor is operating.

1.21. ON CALL

A person is considered on call if:

a. The individual has been specifically designated and the operator knows of the designation;

b. The individual keeps the operator posted as to their whereabouts and telephone number;

c. The individual remains at a reachable location and is capable of getting to the reactor facility within 60 minutes under normal circumstances; and

d. The individual remains in a state of readiness to perform their duties.

1.22. OPERABLE

Operable means a component or system is capable of performing its intended function.

1.23 OPERATIONAL CHANNEL

Operational Channel: The Operational Channel is a power measuring channel used during steady state and square wave operations

1.24. OPERATING

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

1.25. POWER LEVEL MONITORING CHANNEL

A power level monitoring channel is defined to be a channel that is intended to provide real time power level readings to the operator.

1.26 PROTECTIVE ACTION

Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified set point.

1.27. PULSE MODE

[See the definition evaluated and found acceptable in Section 2.5.1 of this SER.]

1.28. REACTIVITY WORTH OF AN EXPERIMENT

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

1.29. REACTOR OPERATING

[See the definition evaluated and found acceptable in Section 2.5.1 of this SER.]

1.30. REACTOR OPERATOR

A reactor operator is an individual who is licensed to manipulate the controls of a reactor.

1.31. REACTOR SAFETY SYSTEMS

Reactor safety systems are those systems, including their associated input channels that are designed to initiate a reactor scram for the primary purpose of protecting the reactor or to provide information for initiation of manual protective action.

1.32. REACTOR SECURED

[See the definition evaluated and found acceptable in Section 2.5.1 of this SER.]

1.33. REACTOR SHUTDOWN

[See the definition evaluated and found acceptable in Section 2.5.1 of this SER.]

1.34. REFERENCE CORE CONDITION

[See the definition evaluated and found acceptable in Section 2.2 of this SER.]

1.35. SAFETY CHANNEL

A safety channel is a high flux safety channel with scram capability.

1.36. SCRAM TIME

Scram time is the elapsed time between the initiation of a scram signal and the full insertion of the control rod.

1.37. SECURED EXPERIMENT

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

1.38. SENIOR REACTOR OPERATOR

A senior reactor operator is an individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

1.39. 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” is used to denote permission, neither a requirement nor a recommendation.

1.40. SHUTDOWN MARGIN

Shutdown margin is 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 in the most reactive position, and that the reactor will remain subcritical without further operator action.

1.41. STANDARD CONTROL ROD

[See the definition evaluated and found acceptable in Section 2.2 of this SER.]

1.42. STEADY STATE MODE

Operation in the steady state mode shall mean operation of the reactor either by manual operation of the control rods or by automatic operation of one or more control rods at power levels not exceeding 1.1 MW. Square wave mode is a subset of the steady state mode of operation.

1.43. SURVEILLANCE INTERVALS

Allowable surveillance intervals shall not exceed the following:

a. Biennial – interval not to exceed 30 months

b. Annual – interval not to exceed 15 months

c. Semi-annual – interval not to exceed 7.5 months

d. Quarterly – interval not to exceed 4 months

e. Monthly – interval not to exceed 6 weeks

f. Weekly – interval not to exceed 10 days

1.44. TRANSIENT ROD

[See the definition evaluated and found acceptable in Section 2.2 of this SER.]

1.45. TRUE VALUE

The true value is the actual value of a parameter.

1.46. UNSCHEDULED SHUTDOWN

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

The NRC staff reviewed the definitions above. As noted above, some definitions in the AFRRI TSs were evaluated and found acceptable in other sections of this SER, and those definitions are stated in the identified SER sections. TS 1.21, "On Call," which is listed above, allows an "On Call" staff member to get to the reactor facility within 60 minutes under normal circumstances, which is 30 minutes longer than stated in the guidance in ANSI/ANS-15.1-2007 (Ref. 64); this was changed by a previous license amendment, considering the unique traffic conditions that can occur in the Washington, D.C. area. Other definitions listed above are either standard definitions used in research reactor TSs and in the guidance provided in NUREG-1537 (Ref. 39) and ANS/ANSI-15.1-2007, or are facility-specific definitions that the NRC staff finds to be consistent with those guidance documents. Therefore, based on the information above, the NRC staff concludes that the licensee's TS definitions are acceptable.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 TS 2.1 Safety Limit: Fuel Element Temperature

TS 2.1 was evaluated and found acceptable in Section 2.5.3 of this SER.

5.2.2 TS 2.2 Limiting Safety System Settings for Fuel Temperature

TS 2.2 was evaluated and found acceptable in Section 2.5.3 of this SER.

5.3 Limiting Conditions for Operations

5.3.1 TS 3.1 Reactor Core Parameters

TS 3.1 was evaluated and found acceptable in Sections 2.2, 2.2.2, and 2.5.1 of this SER.

5.3.2 TS 3.2 Reactor Control and Safety System

5.3.2.1 TS 3.2.1 Reactor Control System

TS 3.2.1, "Reactor Control System," applies to the channels monitoring the reactor core as follows:

Specifications

a. The reactor shall not be operated unless the measuring channels listed in Table 1 are operable for the specific mode of operation.

(…)

Table 1. Minimum Measuring Channels

Measuring Channel	Effective Mode	
	Steady State	Pulse
Fuel Temperature Safety Channel	2	2
Linear Power Channel	1	0
Log Power Channel	1	0
High-Flux Safety Channel	2	1

(1) Any Linear Power, Log Power, High-Flux Safety or Fuel Temperature Safety Channels may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.

(2) If any required measuring channel becomes inoperable while the reactor is operating for reasons other than that identified in the previous footnote (1) above, the channel shall be restored to operation within five minutes or the reactor shall be immediately shutdown.

TS 3.2.1, Specification b, was evaluated and found acceptable in Section 2.2.2 of this SER.

TS 3.2.1 ensures that, during the normal operation of the AFRRI Training, Research, Isotopes, General Atomics (TRIGA) reactor in the specified mode of operation (i.e., steady-state or pulse), sufficient information is available to the operator to ensure safe operation of the reactor. The minimum number of operable measuring channels shown in Table 1 of the TSs will provide the operator with safety measures (i.e., fuel temperature, linear, log, high flux safety, and power pulsing level monitors) to ensure that the reactor power level is adequately monitored for both steady-state and pulsing modes of operation.

The fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit (SL) (see TS 2.1 for the fuel temperature SL). The power level channels assure that radiation-indicating reactor core parameters are adequately monitored for both steady-state and pulsing modes of operation. The specifications for reactor power level indication are included in this section, because the power level is related to the fuel temperature.

Footnote 1 of this TS allows AFRRI to take these measuring channels off line, briefly, for the purpose of a check, test, or calibration because in certain situations, it is necessary to run the reactor to perform the check, test, or calibration. When one measuring channel is off line, the other channels are required to be working. Footnote 2 states that if a channel becomes inoperable, it shall be restored to operation within 5 minutes, or the reactor shall be immediately shut down. Operations may continue for 5 minutes while a determination is made as to whether a channel is operational or not. In its August 5, 2016, RAI response (Ref. 21), the licensee discussed the applicability of footnote 1 to the Pulsing Power Channel and explained that for all modes of operation, the final stage of repair or calibration is a test operation. For the purposes of testing and calibration, limited operations are an allowed condition of operations in order to complete repairs or calibrations. For pulse mode operations, three measuring channels with scram capability are normally available. During a test, two channels are still operational, providing ample redundancy.

The NRC staff reviewed the information in the safety analysis report (SAR), as supplemented, and finds that TS 3.2.1, Specification a, helps ensure that the reactor will not be operated unless the required minimum number of measuring channels is available to the operator to ensure safe operation of the reactor. The NRC staff also finds that TS 3.2.1, Specification a meets 10 CFR 50.36(c)(2), which requires that TSs include limiting conditions for operation (i.e, the lowest functional capability or performance levels of equipment required for safe operation of the facility). On the basis of its review of the information provided, the NRC staff concludes that TS 3.2.1, Specification a, is acceptable.

5.3.2.2 TS 3.2.2 Reactor Safety System

TS 3.2.2, "Reactor Safety System," applies to the reactor safety systems as follows:

Specification

The reactor shall not be operated unless the safety systems described in Tables 2 and 3 are operable for the specific mode of operation.

Table 2. Minimum Reactor Safety System Scrams

Channel	Maximum Set Point	Effective Mode	
		Steady State	Pulse
Fuel Temperature	600°C	2	2
Percent Power, High Flux	1.1 MW	2	0
Console Manual Scram Button	Closure switch	1	1
High Voltage Loss to Safety Channel	20% Loss	2	1
Pulse Time	15 seconds	0	1
Emergency Stop	Closure switch	3	3
(1 in each exposure room, 1 on console)			
Pool Water Level	14 feet from the top of the core	1	1
Watchdog (DAC to CSC)	On digital console	1	1

Table 3. Minimum Reactor Safety System Interlocks

Action Prevented	Effective Mode	
	Steady State	Pulse
Pulse initiation at power levels greater than 1 kW		X
Withdrawal of any control rod except transient		X
Any rod withdrawal with count rate below 0.5 cps as measured by the operational channel	X	X
Simultaneous manual withdrawal of two standard rods	X	
Any rod withdrawal if high voltage is lost to the operational channel	X	X
Withdrawal of any control rod if reactor period is less than 3 seconds	X	
Application of air if the transient rod drive is not fully down. This interlock is not required in square wave mode.	X	

* Reactor safety system interlocks shall be tested daily whenever operations involving these functions are planned.

TS 1.23, “Operational Channel,” defines an operational channel as a power-measuring channel used during steady-state and square wave operations.

TS 3.2.2, Tables 2 and 3, ensure that, during the normal operation of the AFRRI TRIGA reactor in the specified mode of operation (i.e., steady-state or pulse), the minimum number of reactor safety system channels or interlocks required for safe operation of the reactor are operable. The minimum number of operable reactor safety channels or interlocks shown in Tables 2 and 3 of the TS will provide the following safety measures:

- Table 2:

  - Fuel temperature scram, as measured by an instrumented fuel element thermocouple, and power level scrams give protection to ensure that the reactor can be shut down before the safety limit on the fuel element temperature has been exceeded.

  - The manual scram allows the operator to shut down the system at any time if an unsafe or abnormal condition occurs.

  - In the event of failure of the power supply for the high-flux safety channels (defined in TS 1.15 as power measuring safety channels in the reactor safety system), operation of the reactor without adequate instrumentation is prevented.

  - The preset timer ensures that the reactor power level will reduce to a low level after pulsing.

  - The emergency stop allows personnel trapped in a potentially hazardous exposure room or the reactor operator to scram the reactor through the system.

- The pool water level ensures that a loss of biological shielding would result in a reactor shutdown.

- The watchdog scram will ensure reliable communication between the data acquisition computer and the control system computer.

- Table 3:

  - The interlock preventing the initiation of a pulse at a critical level above 1 kilowatt (kWt) assures that the pulse magnitude will not allow the fuel element temperature to approach the SL.

  - The interlock that prevents movement of standard control rods in pulse mode will prevent the inadvertent placing of the reactor on a positive period while in pulse mode.

  - Requiring a count rate to be seen by the operational channels ensures sufficient source neutrons to bring the reactor critical under controlled conditions.

  - The interlock that prevents the simultaneous manual withdrawal of two standard control rods limits the amount of reactivity added per unit time.

  - The interlock that prevents rod withdrawal if high voltage is lost to the operational channel helps ensure that the channel provides accurate power indication .

  - The interlock that prevents withdrawal of any control rod if the reactor period is less than 3 seconds ensures that the reactor operator has control of the reactor.

  - The interlock that prevents the application of air if the transient rod drive is not fully down helps prevent a reactor pulse in steady-state mode.

Since the fuel element temperature is an important parameter in ensuring fuel element integrity, TS 3.2.1 and TS 3.2.2 require fuel element temperature measuring channels and fuel element temperature safety system scram channels. To help ensure that the fuel element temperature is properly monitored, TS 4.2.3 defines the surveillance requirements of the fuel element temperature measuring channels and fuel element temperature safety channels.

The licensee stated that TS 3.2.2, Table 2, lists the minimum required reactor safety system scrams, including three emergency stop buttons (two located in each exposure room and one on the reactor console), each of which allow independent scram of the reactor.

The NRC staff notes that the licensee has not proposed a scram for primary coolant bulk temperature, which is a limiting parameter in the thermal-hydraulic analysis. TS 3.3, Specification a (see SER Section 2.3), prohibits reactor operation if the bulk water temperature exceeds 60 degrees Celsius (°C) (140 degrees Fahrenheit (°F)). In general, reactors should have scram channels for parameters which are inputs to the safety analysis, in accordance with the guidance in NUREG-1537 (Ref. 39). For the AFRRI reactor, however, bulk coolant temperature changes slowly with time, about 14 °C (25 °F) per hour with the pool cooling system turned off and the reactor at full power. Additionally, operation with the pool cooling system off is not the normal operating configuration of the reactor. It is the responsibility of the licensee to not exceed the bulk water temperature by reducing reactor power or shutting down the reactor before reaching the temperature limit. The NRC staff finds that controlling bulk pool temperature by TS 3.3, Specification a, is acceptable to protect the assumptions in the thermal-hydraulic analysis, and is consistent with what NRC has found acceptable at other TRIGA reactors. The NRC staff also finds that because a scram for primary coolant bulk temperature is controlled by operator action and not equipment, 10 CFR 50.36(c)(2) does not require a limiting condition for operation TS for such a scram. Therefore, the NRC staff finds it acceptable that TS 3.2.2 does not require a scram for primary coolant bulk temperature.

The NRC staff reviewed TS 3.2.2, and finds that (except as noted and justified above) the safety channels and interlocks in TS 3.2.2, Tables 2 and 3, are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). These safety channels and interlocks provide a comprehensive and diverse method to help ensure that the AFRRI TRIGA reactor will be operated safely. The NRC staff also finds that TS 3.2.2 is consistent with 10 CFR 50.36(c)(2), which requires that TSs include limiting conditions for operation. Based on the information above, the NRC staff concludes that TS 3.2.2 is acceptable.

5.3.2.3 <u>TS 3.2.3 Facility Interlock System</u>

TS 3.2.3 was evaluated and found acceptable in Section 2.4 of this SER.

5.3.3 TS 3.3 Coolant System

TS 3.3 was evaluated and found acceptable in Section 2.3 of this SER.

5.3.4 TS 3.4 Ventilation System

TS 3.4, “Ventilation System,” states the following:

<u>Specification</u>

a. The reactor shall not be operated unless the facility ventilation system is operating, except for periods of time not to exceed two continuous hours to permit repair, maintenance, or testing. The ventilation system is designed such that if operable there is negative pressure in the reactor room. In the event of a release of airborne radioactivity in the reactor room above routine reactor operation and normal background values, the ventilation system to the reactor room shall be automatically secured via closure dampers by a signal from the reactor deck continuous air particulate monitor.

b. The reactor shall not be operated in exposure room 1 or 2

   1. If the relative air pressure in the exposure room in use is greater than the reactor prep area (room 1105) except for periods of time not to exceed two continuous hours to permit repair, maintenance, or testing when the dampers shall be closed.

      or;

   2. The prep area RAM E3 or E6 is alarming.

The licensee stated that the purpose of this specification is to ensure that the ventilation system is operable and in operation to limit the consequences of possible releases of radioactive materials resulting from reactor operation. In its September 21, 2016, RAI response (Ref. 78), the licensee stated that the setpoints are set as low as possible taking into account background radiation during normal operation such that spurious alarms are avoided. Damper closure is verified by an alarm signal introduced to the continuous air monitor by a check source on each day operations are planned.

TS 3.4 specifies that the ventilation system must be operable during reactor operations. It helps ensure the isolation of the reactor room via isolation dampers when above-normal levels of airborne radioactive material are present in the reactor room, limiting the release of airborne radioactive material. TS 3.4 helps ensure that, during the normal operation of the AFRRI TRIGA reactor, the concentration of gaseous radioactive isotopes argon-41 (Ar-41) and nitrogen-16 (N-16) are kept below limits for occupational and public dose in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection against Radiation." TS 3.3.2 also helps ensure that occupational and public exposures during accident conditions at the AFRRI will be kept below the limits in 10 CFR Part 20. In the event of an accident, installed radiation monitors in the reactor room and in the stack would detect high levels of radioactivity and signal for automatic closure of the ventilation system dampers, thus isolating the source of the radiation from the environment.

One branch of the ventilation system services selected areas of the reactor facility to control possible airborne radiation hazards that may arise as a result of normal and abnormal operation of the AFRRI TRIGA reactor. During normal operation, continuous ventilation ensures that Ar-41 is continuously released from the reactor room; the downstream concentrations are low enough to ensure compliance with the public dose limit allowed by 10 CFR Part 20.

The NRC staff reviewed TS 3.4 and finds that TS 3.4 helps ensure the operability of the ventilation system and is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff also finds that TS 3.4 meets 10 CFR 50.36(c)(2), which requires TSs on limiting conditions for operation. On the basis of the information above, the NRC staff concludes that TS 3.4 is acceptable.

5.3.5 TS 3.5 Radiation Monitoring System and Effluents System

5.3.5.1 TS 3.5.1 Monitoring System

TS 3.5.1 was evaluated and found acceptable in Section 3.1.4 of this SER.

5.3.5.2 TS 3.5.2 Effluents: Argon-41 Discharge Limit

TS 3.5.2 was evaluated and found acceptable in Sections 3.1.1.1 and 3.1.7 of this SER.

5.3.6 TS 3.6 Limitations on Experiments

TS 3.6, "Limitations on Experiments," states the following:

Specifications

The following limitations shall apply to the irradiation of experiments:

a. If the possibility exists that a release of radioactive gases or aerosols may occur;

    1. The amount and type of material irradiated shall be limited to ensure yearly compliance with Table 2, Appendix B, of 10 CFR Part 20, assuming that 100% of the gases or aerosols escape;

    2. The ventilation system shall be operational while the samples are being transferred from the pool or the reactor core.

b. Each fueled experiment shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.0 curies, and the maximum strontium-90 inventory is not greater than 5.0 millicuries;

c. Known explosive materials shall not be irradiated in the reactor in quantities greater than 25 milligrams. In addition, the pressure produced in the experiment container upon detonation of the explosive shall have been determined experimentally, or by calculations, to be less than half the design failure of the container;

d. Samples shall be doubly contained when release of the contained material could cause corrosion of the experimental facility or damage to the reactor;

e. The sum of the absolute reactivity worth of all experiments in the reactor and in the associated experimental facilities shall not exceed \$3.00 (2.1% $\Delta k/k$). This includes the total potential reactivity insertion that might result from experiment malfunction, accidental experiment flooding or voiding, and accidental removal or insertion of experiments. The absolute reactivity worth of any single secured experiment shall not exceed \$3.00 (2.1% $\Delta k/k$). The absolute reactivity worth of any single movable or unsecured experiment shall be less than \$1.00 (0.70% $\Delta k/k$). The combined absolute reactivity worth of multiple movable or unsecured experiments in the reactor and associated experimental facilities at the same time shall be less than \$1.00 (0.70% $\Delta k/k$);

f. In calculations regarding experiments, the following assumptions shall be made:

    1. If the effluent exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of the particles produced can escape; and

    2. For a material whose boiling point is above 55°C and where vapor formed by boiling the material can escape only through an undisturbed column of water above the core, up to 10% of the vapor can escape;

g. If an experiment container fails and releases materials that could damage the reactor fuel or structure by corrosion or other means, physical inspection of the reactor fuel and structure shall be performed to identify damage and potential need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Facility Director and shall be determined to be satisfactory before operation of the reactor is resumed; and

h. Experiments shall be designed such that failure of one experiment shall not contribute to the failure of any other experiment. All operations in an experimental facility shall be supervised by a member of the reactor operations staff. All experiments shall be either secured or observed for mechanical stability to ensure that unintended movement will not cause an unplanned reactivity change in excess of $1.00.

This specification applies to experiments installed in the reactor and its experimental facilities. The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment malfunction, so that airborne concentrations of activity averaged over a year do not exceed Appendix B to 10 CFR Part 20.

TS 3.6, Specifications a and f, limit the radioactive material produced in experiments that may release airborne radioactive material and gives conditions to be used in the safety analysis of the experiment. The TS also requires the ventilation system to be operational while samples are being transferred from the pool or the reactor core in case an experiment fails and releases airborne radioactivity. The release will follow an analyzed pathway from the reactor room. The purpose of TS 3.6, Specifications a and f, is to help ensure that potential releases of radioactive material from experiments are bounded by the exposure limits in 10 CFR Part 20 for AFRRI staff and members of the public. This includes failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment. TS 3.6, Specification f, regarding calculations about malfunctions of experiments, specifies conservative assumptions to be made in the calculations with respect to the release of vapors and particulates to ensure that the results of the calculations are conservative. The assumptions in TS 3.6, Specifications a and f, are standard for research reactor TSs and ensure that source term calculations are conservative. The NRC staff finds that TS 3.6, Specifications a and f, are acceptable because they help ensure that releases of radioactive material from experiments are within 10 CFR Part 20 release limits, both during normal reactor operations and under potential accident conditions.

TS 3.6, Specification b, limits the allowed inventories of iodine and strontium isotopes in any experiment to ensure the health and safety of the facility workers in the event of an accident that would result in the release of these isotopes due to an accident in the experiment. The licensee analyzed the failure of a fueled experiment in its January 17, 2012, RAI response (Ref. 12). The fueled experiment is the designated maximum hypothetical accident (see Chapter 4 of this SER). Given the fission product yields reported in NUREG/CR-2387, “Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors,” issued April 1982 (Ref. 63), the 1.0-curie limit of radioiodines iodine-131 (I-131) through I-135 is reached before 5 millicuries of strontium-90 in fueled experiments. The NRC staff finds that TS 3.6, Specification b, is acceptable because it limits exposure from potential fueled experiment failure or malfunction to 10 CFR Part 20 limits and is within the analysis of the maximum hypothetical accident.

TS 3.6, Specification c, requires that no explosive material in excess of 25 milligrams (mgs) is allowed to be irradiated in the reactor. TS 3.6, Specification c, assures that an experiment involving potential explosives will be properly contained within an experimental vessel and not damage any reactor components in the event of an accident. It limits the detonation pressure to no more than half the design pressure of the container, thereby ensuring the safety factor of 2 recommended by Regulatory Guide (RG) 2.2, “Development of Technical Specifications for

Experiments in Research Reactors," issued November 1973 (Ref. 75). The 25-mg limit is a longstanding limit also discussed in RG 2.2. Because the TS 3.6, Specification c, requirement for experiments containing explosive materials is consistent with the recommendation of RG 2.2, the NRC staff finds that it is acceptable.

TS 3.6, Specification d, requires that samples be doubly contained when release of the contained material could cause corrosion of the experimental facility or damage to the reactor. Because TS 3.6, Specification d, helps ensure protection of reactor components against unanticipated corrosion in the event that an experiment containing corrosive materials were to fail, the NRC staff finds that it is acceptable.

TS 3.6, Specification e, establishes an upper limit on the absolute reactivity worths of all experiments in the reactor and in the associated experimental facilities to assure that the reactor cannot achieve a power level that could exceed the core temperature safety limit should they be inadvertently moved. TS 4.1, Specification b, which is evaluated and found acceptable in SER Section 5.4, requires estimation and measurement of experiment worths to help ensure compliance with TS 3.6, Specification e. TS 6.4, which is evaluated and found acceptable in SER Section 5.6.4, requires the review and written approval (for changes greater than $0.25) of new experiments for radiological safety to also help ensure compliance with TS 3.6, Specification e. TS 3.6, Specification e, helps ensure the licensee limits the reactivity of one single or the sum of all multiple movable or unsecured experiments to less than $1.00 to prevent prompt criticality, which is substantially below the analyzed maximum allowable pulse size of $3.50.

TS 3.6, Specification e, also establishes the reactivity limit of $3.00 for any secured experiments. Because the experiment is held stationary in the reactor, the likelihood that it would fall away from the core to produce an undesirable step increase in reactivity is minimized. The reactivity limit of $3.00 for secured experiments is designed to be below the analyzed maximum allowable pulse size of $3.50. TS 3.6, Specification e also limits the sum of the absolute value reactivity worths of all individual experiments to be less than $3.00. Failure of one experiment, or multiple experiments simultaneously, will not result in a reactivity insertion greater than $3.00, which is less than the analyzed pulse magnitude of $3.50. This value is permissible if it does not violate the TS on excess reactivity and shutdown margin. The $3.00 limit among research reactors is common practice to prevent excessive positive and negative reactivity loading and is acceptable to the NRC staff. Because the experimental reactivity limits in TS 3.6, Specification e, keep the reactivity of experiments within bounds shown to be safe in the SAR, the NRC staff finds that TS 3.6, Specification e, is acceptable.

TS 3.6, Specification g, requires performing a physical inspection of the reactor fuel and structures if an experiment container fails to determine (1) the consequences of the failure of a capsule that releases materials that could damage the reactor fuel or structure and (2) the need for corrective action. Because TS 3.6, Specification g, helps ensure that the consequences of any potential experiment failure are evaluated, and also helps ensure that any necessary corrective actions are identified, the NRC staff finds it acceptable.

TS 3.6, Specification h, requires that experiments be properly designed and that the placement of experiments will be conducted properly and with the appropriate supervision. Because TS 3.6, Specification h, establishes controls that help ensure experiments are conducted safety and helps reduce the likelihood and consequences of any potential experiment failure, the NRC staff finds that it is acceptable.

The NRC staff also evaluated the licensee's limitations on experiments in TS 3.6 against the TS limitations on experiments that are recommended by NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The technical content of TS 3.6 is consistent with guidance and provides a measure of performance against which proposed experiments can be evaluated. Additionally, these specifications contain acceptable controls to protect the reactor, facility workers, and the public in the event of a malfunction of an experiment. The NRC staff also finds that TS 3.6 meets 10 CFR 50.36(c)(2), which requires TSs on limiting conditions for operation. Therefore, based on its review of the information described above, the NRC staff concludes that the licensee's limitations on experiments in TS 3.6, Specifications a through h, are acceptable.

5.4 Surveillance Requirements

5.4.1 TS 4.0 Surveillance Requirements

TS 4.0, "Surveillance Requirements," helps ensure that the quality of systems and components will be maintained to the original design and fabrication specifications, or, if to new specifications, that those specifications have been reviewed. TS 4.0 states the following:

Specifications

Surveillance requirements may be deferred during reactor shutdown (except TS 4.4, TS 4.5.1 and TS 4.5.2) however; they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.

	TS	Possible to defer during shutdowns?	Required prior to routine operations?
1.	4.1 Reactor core parameters	Yes	Yes
2.	4.2.1 Reactor Control Systems	Yes	Yes
3.	4.2.2 Reactor Safety Systems	Yes	Yes
4.	4.2.3 Fuel Temperature	Yes	Yes
5.	4.2.4 Facility Interlock System	Yes	Yes
6.	4.3 Coolant Systems	Yes	Yes
7.	4.4 Ventilation Systems	No	N/A
8.	4.5.1 Monitoring System	No	Yes
9.	4.5.2 Effluents	No	N/A
10.	4.6 Reactor Fuel Elements	Yes	Yes
11.	4.2.2 Low Pool Water Scram	Yes	No

TS 4.0 governs the conduct of surveillance required to allow operational flexibility that does not impact safety. The surveillance items in the TS 4.0 table are listed to specify which surveillances may be deferred during times when required equipment is out of service or reactor operations are suspended. The NRC staff finds that TS 4.0 provides appropriate surveillance practices and is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff also finds that TS 4.0 meets 10 CFR 50.36(c)(3), which requires that TSs include surveillance requirements that help ensure that limiting conditions for operation will be met. On the basis of its review of the information above, the NRC staff concludes that TS 4.0 is acceptable.

5.4.2 TS 4.1 Reactor Core Parameters

TS 4.1, “Reactor Core Parameters,” states the following:

Specifications

a. TS 4.1, Specification a, was evaluated and found acceptable in Section 2.2 of this SER.

b. The reactivity worth of an experiment shall be estimated before reactor power operation with the experiment the first time it is performed. If the absolute reactivity worth is estimated to be greater than $0.25, the worth shall be measured at a power level less than 1 kW.

c. TS 4.1, Specification c, was evaluated and found acceptable in Section 2.2 of this SER.

d. TS 4.1, Specification d, was evaluated and found acceptable in Section 2.5.2 of this SER.

TS 4.1, Specification b, helps ensure that the requirements of TS 3.6, Specification e, which limit the reactivity worths of experiments (see SER Section 5.3.6), are met. TS 4.1, Specification b, is consistent with NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff also finds that TS 4.1, Specification b, is consistent with 10 CFR 50.36(c)(3), which requires TSs on surveillance requirements that help ensure that limiting conditions for operation will be met. Therefore, on the basis of its review of the information above, the NRC staff concludes that TS 4.1, Specification b, is acceptable.

5.4.3 TS 4.2 Reactor Control and Safety Systems

5.4.3.1 TS 4.2.1 Reactor Control Systems

TS 4.2.1 was evaluated and found acceptable in Section 2.2.2 of this SER.

5.4.3.2 TS 4.2.2 Reactor Safety Systems

TS 4.2.2 was evaluated and found acceptable in Sections 2.3 and 2.5.1 of this SER.

5.4.3.3 TS 4.2.3 Fuel Temperature

TS 4.2.3 was evaluated and found acceptable in Section 2.2.1 of this SER

5.4.3.4 TS 4.2.4 Facility Interlock System

TS 4.2.4 was evaluated and found acceptable in Section 2.4 of this SER.

5.4.3.5 TS 4.2.5 Reactor Fuel Elements

TS 4.2.5 was evaluated and found acceptable in Section 2.2.1 of this SER.

5.4.4 TS 4.3 Coolant System

TS 4.3 was evaluated and found acceptable in Section 2.3 of this SER.

5.4.5 TS 4.4 Ventilation System

TS 4.4, “Ventilation System,” states the following:

Specification

1. The operating mechanism of the ventilation system dampers in the reactor room shall be verified to be operable and visually inspected monthly, not to exceed 6 weeks.

2. The relative air pressure in the reactor room and exposure room to be used shall be verified to be negative each day operations in the affected exposure room are planned.

3. The reactor exhaust damper flow failure closure system shall be tested each day that reactor operations are planned.

TS 4.4 requires the monthly verification of the operability and proper operation of the ventilation system positive sealing dampers; the verification is considered adequate to ensure their proper operation. Negative air pressure in the exposure rooms is verified for those days where the exposure rooms are to be used to help ensure that Ar-41 produced in the rooms during use will be exhausted through an analyzed air flow path. The reactor exhaust damper flow failure system is tested to help ensure that the ventilation system responds as assumed in the SAR to flow failure. The testing meets the recommendations of ANSI/ANS-15.1-2007 (Ref. 64), Section 4.5(1). The NRC staff reviewed TS 4.4 and finds that the surveillance requirements in TS 4.4 provide the necessary testing to help ensure that the ventilation system is maintained and available. The NRC staff also finds that TS 4.4 meets 10 CFR 50.36(c)(3), which requires TSs on surveillance requirements that help ensure that limiting conditions for operation will be met. Based on the information above, the NRC staff concludes that TS 4.4 is acceptable.

5.4.6 TS 4.5 Radiation-Monitoring System and Effluents

1.1.1.1 TS 4.5.1 Monitoring System

TS 4.5.1 was evaluated and found acceptable in Section 3.1.4 of this SER.

5.4.6.1 TS 4.5.2 Effluents

TS 4.5.2 was evaluated and found acceptable in Sections 3.1.1.1 and 3.1.7 of this SER.

5.4.7 TS 4.6 Reactor Fuel Elements

TS 4.6 was evaluated and found acceptable in Section 2.2.1 of this SER.

5.5 Design Features

5.5.1 TS 5.1 Site and Facility Description

TS 5.1, “Site and Facility Description,” states the following:

Specifications

a. The reactor building, as a structurally independent building in the AFRRI complex, shall have its own ventilation system branch. The effluent from the reactor ventilation system shall exhaust through absolute filters to a stack having a minimum elevation that is 18 feet above the roof of the highest building in the AFRRI complex.

b. The reactor room shall contain a minimum free volume of 22,000 cubic feet.

c. The ventilation system air ducts to the reactor room shall be equipped with dampers which automatically close off ventilation to the reactor room upon a signal from the reactor room continuous air particulate monitor.

d. The reactor room shall be designed to restrict air leakage when the ventilation system dampers are closed.

e. The reactor areas exhausting through the reactor ventilation system shall include the Controlled Access Area (CAA) and the Reactor Control Area (RCA). The specific rooms included in each of those areas shall be listed in the Physical Security Plan for the AFRRI TRIGA Reactor Facility.

f. The reactor is housed in building #42 of the AFRRI complex and the restricted areas are located within that structure. The restricted areas are described in the SAR for the AFRRI reactor facility section 1.3.1 including figures 2-2 through 2-4 which describe the location of the reactor in the AFRRI complex. Figures 3-1 through 3-4 are the floor plan layouts which identify the reactor areas

TS 5.1 specifies the design features of the AFRRI facility, the reactor building construction and dedicated ventilation system, the ventilation stack, the reactor room, and the reactor confinement volume. The systems have been designed to maintain a negative pressure with respect to the atmosphere, so that effluent discharge will be controlled at all times. The construction of the facility is such as to restrict leakage of air into or out of the reactor room. The stack provides for an elevated release of discharged effluents to give adequate dilution and dispersion, as well as to preclude a ground release in the event of any malfunction. The separate ventilation system branch ensures a dedicated air flow system for reactor effluents. TS 5.1, Specification e, describes the reactor exhaust area. TS 5.1, Specification f, specifies the restricted areas (areas under the NRC TRIGA license), which are described in the SAR.

The NRC staff finds that TS 5.1 is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff also finds that TS 5.1 meets 10 CFR 50.36(c)(4), which requires TSs on design features such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety. The NRC staff concludes that the design of the reactor building and associated ventilation system components is adequate to ensure the health and safety of the facility workers and the public, and that TS 5.1, Specifications a through f, are acceptable.

5.5.2 TS 5.2 Reactor Core and Fuel

5.5.2.1 TS 5.2.1 Reactor Fuel

TS 5.2.1 was evaluated and found acceptable in Section 2.2.1 of this SER.

5.5.2.2 TS 5.2.2 Reactor Core

TS 5.2.2 was evaluated and found acceptable in Section 2.2 of this SER.

5.5.2.3 TS 5.2.3 Control Rods

TS 5.2.3 was evaluated and found acceptable in Section 2.2.2 of this SER.

5.5.3 TS 5.3 Fuel Storage

TS 5.3, "Fuel Storage," states the following:

> Specification
>
> All fuel elements not in the reactor core shall be stored and handled in accordance with applicable regulations. Irradiated fuel elements and fueled devices shall be stored in an array that permits sufficient natural convective cooling by water or air and that prevents the fuel element or fueled device temperature from exceeding design values. Storage shall be such that stored fuel elements and fueled devices remain subcritical under all conditions of moderation and reflection in a configuration where k_{eff} is not greater than 0.90.

TS 5.3 helps ensure that the k_{eff} value is limited to 0.9, which is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64).

In its RAI response dated October 20, 2011 (Ref. 10), the licensee stated that all spare fuel elements are stored in the reactor tank under water in standard 12-element TRIGA fuel storage racks. The licensee performed an analysis showing that, if a fully loaded rack should fail, the resulting configuration of the 12 fuel elements lying on the bottom of the pool could not reach criticality. The analysis indicated that, in the worst case configuration, k_{eff} is less than 0.746, which is below the value of $k_{eff} < 0.9$ recommended in ANSI/ANS-15.1-2007 (Ref. 64). The licensee's analysis demonstrates that the licensee can remain in compliance with TS 5.3 under any physically possible conditions of fuel storage.

The NRC staff reviewed TS 5.3 and concludes that TS 5.3 helps ensure that adequate cooling by natural convection, either by water or air, of stored irradiated fuel elements and fueled devices will be incorporated into the design of any storage array. TS 5.3 also helps ensure that the fuel will be stored in a configuration that provides an adequate criticality safety margin, and the licensee's criticality safety analysis demonstrates the licensee's ability to comply with the k_{eff} limit in TS 5.3. The NRC staff finds that this design feature TS is acceptable to protect the fuel element cladding and fission product barrier, and helps prevent any inadvertent criticality of stored fuel. The NRC staff also finds that TS 5.3 is meets 10 CFR 50.36(c)(4), which requires

TSs on design features. Therefore, based on the information above, the NRC staff finds that TS 5.3 is acceptable.

5.6 Administrative Controls

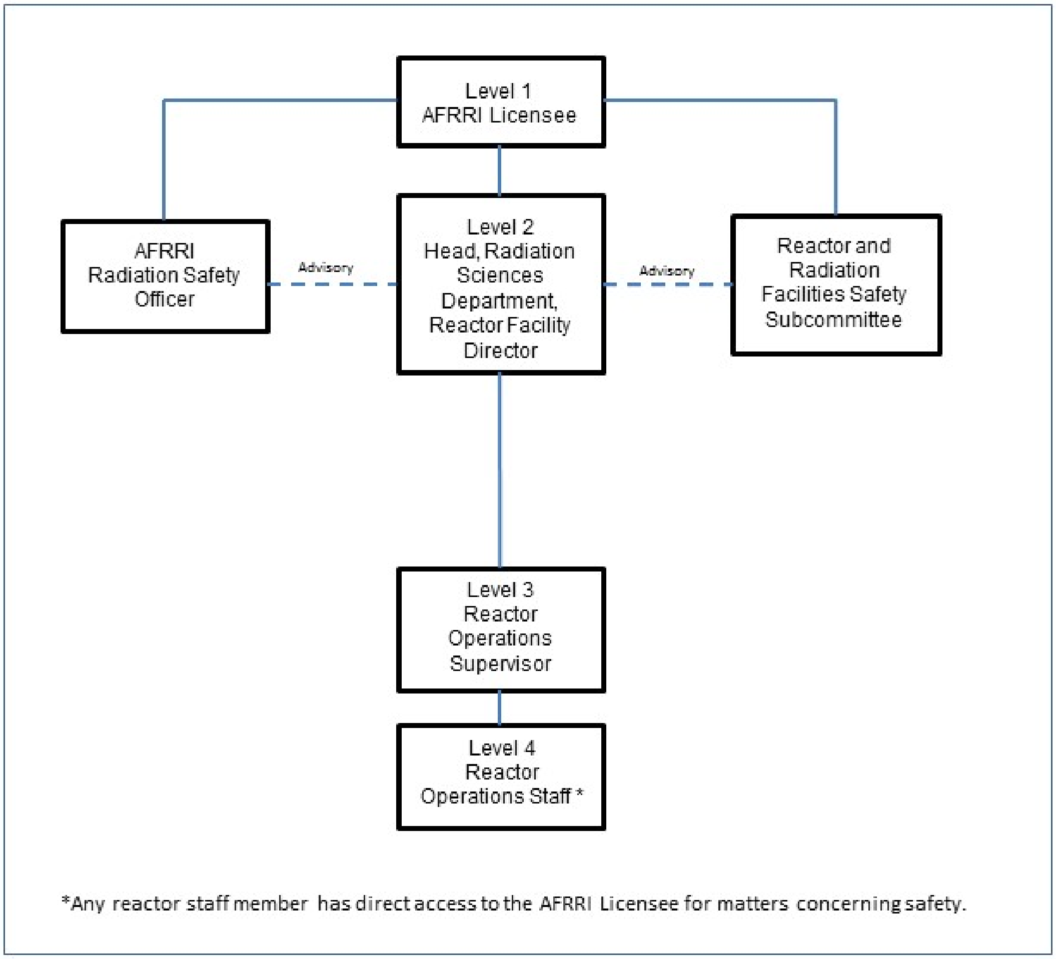
TS 6.0, "Administrative Controls," includes requirements for the conduct of operations for the AFRRI TRIGA reactor. The administrative controls in TS 6.0 include responsibilities, facility organization, staff qualifications, training, the safety committee, operational review and audits, procedures, required actions, and reports and records. The primary guidance for the development of administrative control TS for research reactors is ANSI/ANS-15.1-2007 (Ref. 64).

5.6.1 TS 6.1 Organization

5.6.1.1 TS 6.1.1 Structure

TS 6.1.1, "Structure," discusses the organizational responsibilities identified in the organizational chart as follows:

Figure 1. Organization of Personnel for Management and Operation of the AFRRI Reactor Facility



TS 6.1.1, "Structure," states the following:

> The organizational structure of the reactor facility is depicted below.
>
> MANAGEMENT LEVELS
>
> Level 1: AFRRI Director: Responsible for the facility license.
>
> Level 2: Reactor Facility Director: Responsible for reactor facility operations and administration shall report to Level 1.
>
> Level 3: Reactor Operations Supervisor: Responsible for the day-to-day operation of the reactor and shall report to Level 2.
>
> Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3 for matters involving reactor operations.

In TS 6.1.1, the licensee has identified the responsibilities and organization of individuals associated with the facility. TS 6.1.1 helps ensure that the AFRRI organization structure and responsibilities are maintained. The NRC staff reviewed TS 6.1.1 and finds that the AFRRI organizational structure described in TS 6.1.1 and shown in TS 6.1.1, Figure 1 is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.1.1 meets 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to organization and management necessary to assure safe operation. On the basis of its review of the information above, the NRC staff concludes that TS 6.1.1 is acceptable.

5.6.1.2 TS 6.1.2 Responsibility

TS 6.1.2, "Responsibility," presents the responsibilities and organization of the individuals identified in the TS 6.1.1, Figure 1, organizational chart as follows:

> The AFRRI Licensee shall have license responsibility for the reactor facility. The Reactor Facility Director (RFD) shall be responsible for administration and operation of the reactor facility and for determination of applicability of procedures, experiment authorizations, maintenance, and operations. The Reactor Facility Director may designate an individual who meets the requirements of Technical Specification 6.1.3.1.a to discharge these responsibilities during an extended absence. During brief absences (periods less than 4 hours) of the Reactor Facility Director and his designee, the Reactor Operations Supervisor shall discharge these responsibilities. The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the AFRRI TRIGA reactor. The requirements of the radiation safety program are established in 10CFR20. The program shall comply with the requirements in 10CFR20. Additional guidelines from ANSI/ANS-15.11-1993;R2004 "Radiation Protection at Research Reactor Facilities" should be considered.

As shown on the organizational chart and described in TS 6.1.2, the AFRRI Radiation Safety Officer (RSO) is responsible for implementing the radiation protection program at the reactor and throughout the AFRRI facility. The RSO advises the Reactor Facility Director (Level 2) and reports to the AFRRI Director (Level 1) for all matters relating to radiation safety. All operations at the reactor are performed under the AFRRI ALARA program developed by the RSO. The licensee stated that AFRRI has developed and implemented a radiation protection program that meets the requirements of 10 CFR 20.1101(a) through (c), ANSI/ANS-15.1-2007 (Ref. 64), and ANSI/ANS-15.11 (Ref. 71).

TS 6.1.2 helps ensure that the AFRRI-specific organization levels and responsibilities are maintained. The NRC staff reviewed TS 6.1.2 and finds that the organizational responsibilities stated in TS 6.1.2 are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.1.2 meets 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to organization and management. On the basis of the information above, the NRC staff concludes that TS 6.1.2 is acceptable.

5.6.1.3 TS 6.1.3 Staffing

TS 6.1.3, "Staffing," presents the qualification, minimum staffing levels, and training requirements. TS 6.1.3.1, "Selection of Personnel," states the following:

a. AFRRI Licensee

The AFRRI Licensee is the AFRRI Director. The AFRRI Director has management responsibility for adhering to the terms and conditions of the AFRRI reactor license R-84, the AFRRI O2 byproduct license, the AFRRI Technical Specifications and for protecting the health and safety of the facility staff and members of the public.

b. Reactor Facility Director

At the time of appointment to this position, the Reactor Facility Director shall have six or more years of nuclear experience. The individual shall have a baccalaureate or higher degree in an engineering or scientific field. The degree may fulfill up to four years of experience on a one-for-one basis. The Reactor Facility Director shall have held a USNRC Senior Reactor Operator license on the AFRRI reactor for at least one year before appointment to this position.

c. Reactor Operations Supervisor

At the time of appointment to this position, the Reactor Operations Supervisor shall have three years nuclear experience. Higher education in a scientific or engineering field may fulfill up to two years of experience on a one-for one basis. The Reactor Operations Supervisor shall hold a USNRC Senior Reactor Operator license on the AFRRI reactor. In addition, the Reactor Operations Supervisor shall have one year of experience as a USNRC licensed Senior Reactor Operator at AFRRI or at a similar facility before the appointment to this position.

d. Reactor Operators/Senior Reactor Operators

At the time of appointment to this position, an individual shall have a high school diploma or equivalent, and shall possess the appropriate USNRC license.

e. Additional reactor staff as required for support and training

At the time of appointment to the reactor staff, an individual shall possess a high school diploma or equivalent.

TS 6.1.3.2, “Operations,” states the following:

a. Minimum staff when the reactor is not secured shall include:

1. A licensed Senior Reactor Operator on call, but not necessarily on site;

2. Radiation control technician on call, but not necessarily on site;

3. At least one licensed Reactor Operator or Senior Reactor Operator present in the control room; and

4. Another person within the AFRRI complex who is able to carry out written emergency procedures, instructions of the operator, or to summon help in case the operator becomes incapacitated.

5. One licensed Senior Reactor Operator may fill both the on call and control room positions simultaneously. In that case, the minimum staff is three persons.

b. An Senior Reactor Operator shall be present at the reactor during the following operations:

   1. All fuel or control rod relocations within the reactor core region (control rod movement associated with routine reactor operation is not considered to be a relocation);

   2. Initial reactor startup and approach to power;

   3. Recovery from an unplanned or unscheduled shutdown.; and

   4. Relocation of any experiment with reactivity worth greater than $1.00.

c. A list of the names and telephone numbers of the following personnel shall be readily available to the operator on duty:

   1. Management personnel (Reactor Facility Director, AFRRI Licensee) or designee;

   2. Radiation safety personnel (AFRRI Radiation Safety Officer) or designee; and

   3. Other operations personnel (Reactor Staff, Reactor Operations Supervisor)

TS 6.1.3.3, "Training of Personnel," states the following:

> Training and retraining program shall be maintained to ensure adequate levels of proficiency in persons involved in the reactor and reactor operations. The training and retraining program will be consistent with the NRC-approved reactor requalification plan.

TS 6.1.3.1 specifies the selection of personnel. The licensee stated that it complies with the guidance in ANSI/ANS-15.4-2007, "Selection and Training of Personnel for Research Reactors" (Ref. 51), for the selection of personnel, and the NRC find that TS 6.1.3.1 is consistent with this guidance. Therefore, the NRC staff finds that TS 6.1.3.1 is acceptable.

TS 6.1.3.2, Specification a, describes the minimum staffing when the reactor is not secured. The licensee stated that the unsecured reactor will always be under the control of a Reactor Operator (RO) or Senior Reactor Operator (SRO). 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." The regulation, 10 CFR 50.54(m)(1) states, in part, "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 […]." The NRC staff finds that TS 6.1.3.2, Specification a, is consistent with 10 CFR 50.54(k) and 10 CFR 50.54(m)(1). Therefore, the NRC staff finds that TS 6.1.3.2, Specification a, is acceptable.

TS 6.1.3.2, Specification b, specifies certain reactor operations requiring the presence of an SRO. The regulation, 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 TS 6.1.3.2, Specification b, is consistent with 10 CFR 50.54(m)(1). Therefore, the NRC staff finds that TS 6.1.3.2, Specification b, is acceptable.

TS 6.1.3.2, Specification c, requires contact information for support personnel be available to the operating staff, which is consistent with the guidance in ANSI/ANS-15.1-2007 (Ref. 64) and NUREG-1537 (Ref. 39). Therefore, the NRC staff finds TS 6.1.3.2, Specification c acceptable.

TS 6.1.3.3 establishes the criteria for the training program for operations personnel. The licensee stated that ANSI/ANS-15.4-2007 (Ref. 51) is used as guidance for the selection and training of personnel. The NRC staff finds that TS 6.1.3.3 is consistent with the guidance in NUREG-1537 (Ref. 39), ANSI/ANS-15.1-2007 (Ref. 64) and ANSI/ANS-15.4-2007. On this basis, the NRC staff finds that TS 6.1.3.3 is acceptable.

The NRC staff reviewed TS 6.1.3, and finds that it satisfies the requirements of 10 CFR 50.54(k) and 10 CFR 50.54(m)(1), and is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.1.3 meets 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to organization and management. Therefore, on the basis of the information above, the NRC staff concludes that TS 6.1.3 is acceptable.

5.6.2 TS 6.2 Review and Audit—The Reactor and Radiation Facilities Safety Subcommittee (RRFSS)

NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64) specify that the purpose of a Reactor and Radiation Facilities Safety Subcommittee (RRFSS) is to provide independent oversight of reactor operations.

TS 6.2 describes the attributes of the RRFSS as follows.

5.6.2.1 <u>TS 6.2.1 Composition and Qualifications</u>

TS 6.2.1.1, "Composition," states the following:

a. Regular RRFSS Members (Permanent Members)

    1. The following shall be members of the RRFSS:

        a. AFRRI Radiation Safety Officer

        b. AFRRI Reactor Facility Director

    2. The following shall be appointed to the RRFSS by the AFRRI Licensee:

        a. Chairman

        b. One to three non-AFRRI members who are knowledgeable in fields related to reactor safety. At least one shall be a Reactor Operations Specialist or a Health Physics Specialist.

b. Special RRFSS Members (Temporary Members)

    1. Other knowledgeable persons to serve as alternates in section 6.2.1.1.a.2.b above as appointed by the AFRRI Licensee.

    2. Voting ad hoc members, appointed by the AFRRI Licensee to assist in review of a particular problem.

c. Nonvoting members as appointed by the AFRRI Licensee.

TS 6.2.1.2, “Qualifications,” states the following:

> The minimum qualifications for a person on the RRFSS shall be six years of professional experience in the discipline or specific field represented. A baccalaureate degree may fulfill four years of experience.

TS 6.2.1 helps ensure that the RRFSS composition, qualifications, and operation are properly delineated. TS 6.2.1 describes the composition of the RRFSS and requires the RRFSS to include experts who are non-AFRRI members. As described in the licensee’s RAI response dated August 5, 2016 (Ref. 21), the composition of the permanent RRFSS committee shall be the RSO, the Reactor Facility Director, the Chairman, and at least one to three non-AFRRI members, which is a minimum of four members. The NRC staff reviewed TS 6.2.1, and finds that the requirements in TS 6.2.1 are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.2.1 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to review and audit functions necessary to assure safe operation. On the basis of the information above, the NRC staff concludes that TS 6.2.1 is acceptable.

5.6.2.2 <u>TS 6.2.2 Function and Authority</u>

TS 6.2.2.1, “Function,” states the following:

> The RRFSS shall be directly responsible to the AFRRI Licensee. The subcommittee shall review all radiological health and safety matters concerning the reactor and its associated equipment, the structural reactor facility, and those items listed in Section 6.2.4.

TS 6.2.2.2, “Authority,” states the following:

> The RRFSS shall report to the AFRRI Licensee and shall advise the Reactor Facility Director in those areas of responsibility specified in Section 6.2.4.

TS 6.2.2 describes the function and authority of the AFRRI RRFSS, which is responsible to the AFRRI licensee. The facility organizational structure (as shown in Figure 1) indicates that the RRFSS reports to the AFRRI licensee. The RRFSS’s function to perform reviews and an independent audit is in agreement with the recommendations of NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.2.2 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to review and audit functions. On the basis of the information above, the NRC staff concludes that TS 6.2.2 is acceptable.

5.6.2.3 TS 6.2.3 Rules

TS 6.2.3.1, “Alternates,” states the following:

> Alternate members may be appointed in writing by the RRFSS Chairman to serve on a temporary basis. No more than two alternates shall participate on a voting basis in RRFSS activities at any one time.

TS 6.2.3.2, “Meeting Frequency,” states the following:

> The RRFSS shall meet at least two times during a calendar year. Any member of the RRFSS may submit a written request to the RRFSS Chairman to convene a special meeting of the RRFSS to discuss urgent matters.

TS 6.2.3.3, “Quorum,” states the following:

> A quorum of the RRFSS for review shall consist of a minimum of four members that can vote and occupy the following positions; the Chairman (or designated alternate), the Reactor Facility Director (or designated alternate), the Radiation Safety Officer (or designated alternate), and one or more non-AFRRI member. A majority of those present shall be regular members. The operating staff shall not constitute a majority. A member may occupy two positions but may only vote once.

TS 6.2.3.4, “Voting Rules,” states the following:

> Each regular RRFSS member shall have one vote. Each special RRFSS member shall have one vote. The majority is 51% or more of the regular and special members present and voting and concurrence between the Radiation Safety Officer and the Reactor Facility Director.

TS 6.2.3.5, “Minutes,” states the following:

> a. Draft minutes of the previous meeting should be available to regular members at least one week before a regular scheduled meeting.
>
> b. Once approved by the subcommittee, final minutes shall be submitted to level one management for review within 3 months.

TS 6.2.3 establishes the RRFSS meeting and voting rules and describes the operation of the RRFSS, which is responsible for an independent audit of the AFRRI activities and conducts its review and audit functions in accordance with a written charter. It includes provisions for meeting frequency, voting rules, quorums, alternates, and minutes.

NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64) specify that the purpose of the committee is to provide independent oversight and that the operating staff should not constitute the majority of a quorum. The licensee clarified that under TS 6.2.3.3, four people must be present for a quorum. Depending on the number of outside regular members under TS 6.2.1.1.a(2)(b), the total regular membership could be four to six people. In certain cases, special voting members may be appointed under TS 6.2.1.1.b. The NRC staff reviewed TS 6.2.3 and finds that the composition and qualifications for the RRFSS stated in TS 6.2.3 are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64).

Additionally, the NRC staff finds that TS 6.2.3 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to review and audit functions. Based on the information above, the NRC staff concludes that TS 6.2.3 is acceptable.

5.6.2.4 TS 6.2.4 Review Function

TS 6.2.4 states the following:

> The RRFSS shall review*:*
>
> a. Safety evaluations for (1) changes to procedures, equipment, or systems having safety significance and (2) tests or experiments conducted without NRC approval under provisions of Section 50.59 of 10 CFR.
>
> b. Changes to procedures, equipment, or systems that change the original intent or use, are non-conservative, or those that meet any of the applicable criteria in Section 50.59 of 10 CFR;
>
> c. Proposed tests or experiments that could affect reactivity or result in the uncontrolled release of radioactivity, or those that might meet any of the applicable criteria in Section 50.59 of 10 CFR;
>
> d. Proposed changes in Technical Specifications, the Safety Analysis Report, or other license conditions;
>
> e. Violations of applicable statutes, codes, regulations, orders, technical specifications, license requirements, or of internal procedures or instructions having safety significance;
>
> f. Operating abnormalities having safety significance;
>
> g. Events that have been reported to the NRC; and
>
> h. Audit reports of the reactor facility operations.

TS 6.2.4, Specifications a through h, describe the items to be reviewed by the RRFSS, which are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). TS 6.2.4, Specifications a and c, contain the requirements for the review and approval of experiments. TS 6.2.4, Specification h, requires the RRFSS to review audit reports. TS 6.2.4, Specifications a and h, contain the requirements for the review of changes to procedure, equipment, or systems; changes to the TSs, license, or SAR; violations; operating abnormalities with safety significance; and reportable events. The NRC staff finds that TS 6.2.4 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to review functions. Based on the information above, the NRC staff concludes that TS 6.2.4 is acceptable.

5.6.2.5 <u>TS 6.2.5 Audit Function</u>

TS 6.2.5 states the following:

> Audits of reactor facility operations shall be performed under the cognizance of the RRFSS, but in no case by the personnel responsible for the item audited. The audits shall be performed annually, not to exceed 15 months. A report of the findings and recommendations resulting from the audit shall be submitted to the AFRRI Licensee within three months after the report has been received. Deficiencies uncovered that affect reactor safety shall immediately be reported to level one management. Audits may be performed by one or more individuals who need not be RRFSS members. These audits shall examine the operating records and the conduct of operations, and shall encompass the following:
>
> a. Conformance of facility operation to the Technical Specifications and the license;
>
> b. Performance, training, and qualifications of the reactor facility staff;
>
> c. Results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect safety;
>
> d. Facility emergency plan and implementing procedures;
>
> e. Facility Physical Security Plan;
>
> f. Any other area of facility operations considered appropriate by the RRFSS or the AFRRI Licensee; and
>
> g. Reactor Facility ALARA Program. This program may be a section of the total AFRRI program.

TS 6.2.5, Specifications a through g, require the RRFSS to annually audit reactor operations and events, the emergency plan, the training and requalification program, the physical security plan, and the ALARA program. TS 6.2.5 also requires that individuals performing the annual audits not be the same individuals who are responsible for the items audited. The NRC staff reviewed TS 6.2.5 and finds that the RRFSS audit functions, as stated in TS 6.2.5, are adequate in scope, reflect the requisite independence, and are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.2.5 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to audit functions. On the basis of the information above, the NRC staff concludes that TS 6.2.5 is acceptable.

5.6.3 TS 6.3 Procedures

The licensee has specified in TS 6.3 the type of written procedures that must be prepared, approved, and used as follows:

> Written procedures for certain activities shall be approved by the Reactor Facility Director and reviewed by the RRFSS. The procedures shall be adequate to ensure safe operation of the reactor, but shall not preclude the use of independent judgment and

action as deemed necessary. Operational procedures shall be used for the following items:

a. Conduct of irradiation and experiments that could affect the operation and safety of the reactor;

b. Surveillance, testing, maintenance, and calibration of instruments, components, and systems involving nuclear safety;

c. Personnel radiation protection consistent with 10 CFR Part 20;

d. Implementation of required plans such as the Physical Security Plan and Emergency Plan, consistent with restrictions on Safeguards information;

e. Fuel loading, unloading, and movement within the reactor core; and

f. Reactor startup checklist, standard operations, and securing the facility.

Although substantive changes to the above procedures shall be made only with approval by the Reactor Facility Director, temporary changes to the procedures that do not change their original intent may be made by the Reactor Operations Supervisor. All such temporary changes shall be documented and subsequently reviewed and approved by the Reactor Facility Director.

TS 6.3 specifies the items that must be covered by standard operating procedures. TS 6.3 also specifies that new procedures and substantive changes to procedures must be approved by the Reactor Facility Director and the RRFSS. The licensee also stated that the operating procedures are required to be used and followed to help ensure effective procedure adherence.

The licensee stated that, under TS 6.3, the Reactor Facility Director can make substantive changes to procedures, and that the Reactor Operations Supervisor can make temporary changes. Both substantive changes and temporary changes are reviewed in advance with respect to 10 CFR 50.59, “Changes, tests and experiments,” documented, and reviewed by the RRFSS at its next scheduled meeting.

TS 6.3 helps ensure that procedures are written, reviewed, and approved before performance of the activities listed in the specification. In addition, TS 6.3 helps ensure that the procedures will be followed regarding irradiations, experiments, maintenance, radiation protection, physical security, emergency planning, fuel handling, and reactor operations. TS 6.3 is consistent with ANSI/ANS-15.1-2007 (Ref. 64), Section 6.4, “Procedures,” except that the standard suggests that this TS also separately list (as “g”) an item requiring procedures for the use, receipt and transfer of byproduct material. The AFRRI reactor license authorizes AFRRI to conduct activities under the 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material” involving the 3-Curie startup source as well as byproduct material produced by operation of the reactor or non-fueled reactor experiments. For receipt and transfer of byproduct materials, AFRRI relies on the procedures for byproduct material receipt, possession, handling, transfer, shipping, and disposal required by its “O2” license (NRC Materials License No. 19-08330-02, Docket 030-04545) (Although the last two digits of the license number are 02, the licensee refers to this license as the “O2” license). Specifically, AFRRI also internally transfers byproduct material for disposal under its “O2” license. The NRC staff finds that TS 6.3, Specification f requires procedures for the use of the 3-Curie americium-beryllium startup source because use of the source for reactor startup is part of standard operations. With the exception of the 3-Curie americium-beryllium source, TS 6.3, Specification a requires procedures for the use of byproduct materials in irradiation and experiments and Specification e requires such procedures for any byproduct material present in fuel in the reactor. The NRC staff notes that, if a replacement sealed 3-Curie americium-beryllium source were needed in the future, AFRRI could not transfer the startup source to the “O2” license unless the “O2” license possession limits were changed via a license amendment.

The “O2” license also requires that procedures related to the receipt and transfer of byproduct materials be documented, reviewed, and approved by AFRRI’s Radiation Safety Committee and the licensee’s audit program evaluate changes to the “O2” procedures. The “O2” license requires that all byproduct material handling, shipping, and disposal conducted under the “O2” license be performed in accordance with NRC regulations.

The NRC staff finds that TS 6.3 has the requisite items and procedure controls for safe operation of the reactor. Because TS 6.3, Specifications a, e, and f, and the “O2” license collectively require procedures for the use, receipt and transfer of byproduct material and require that byproduct material is used and handled in compliance with NRC regulations, the NRC staff finds that TS 6.3 need not include a separate requirement for procedures for byproduct materials receipt, use, and transfer. Additionally, the NRC staff finds that TS 6.3 meets 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs that include administrative controls related to procedures, and is consistent with NUREG-1537 which also recognizes that byproduct material is handled under a materials license at some RTRs. Based on the information above, the NRC staff concludes that TS 6.3 is acceptable.

5.6.4 TS 6.4 Review and Approval of Experiments

TS 6.4 provides the experimental review and approval requirements as follows:

> Before issuance of a reactor authorization, new experiments shall be reviewed for radiological safety and approved by the following:
>
> a. Reactor Facility Director
>
> b. Health Physics Department
>
> c. Reactor and Radiation Facilities Safety Subcommittee (RRFSS)
>
> Prior to its performance, an experiment shall be included under one of the following types of authorizations:
>
> a. <u>Special Reactor Authorization</u> for new experiments or experiments not included in a Routine Reactor Authorization. These experiments shall be performed under the direct supervision of the Reactor Facility Director or designee.
>
> b. <u>Routine Reactor Authorization</u> for approved experiments safely performed at least once. These experiments may be performed at the discretion of the Reactor Facility Director and coordinated with the Health Physics Department, when appropriate. These authorizations do not require additional RRFSS review.

c. Reactor Parameters Authorization for routine measurements of reactor parameters, routine core measurements, instrumentation and calibration checks, maintenance, operator training, tours, testing to verify reactor outputs, and other reactor testing procedures. This shall constitute a single authorization. These operations shall be performed under the authorization of the Reactor Facility Director or the Reactor Operations Supervisor.

Substantive (> $0.25) changes to previously approved experiments shall be made only after review by the RRFSS and after approval (in writing) by the Reactor Facility Director or designated alternate to ensure that the change does not impact compliance with TS 3.6, LIMITATIONS ON EXPERIMENTS. Minor changes that do not significantly alter the experiment (< $0.25) may be approved by the Reactor Operations Supervisor. Approved experiments shall be carried out in accordance with established procedures.

TS 6.4 helps ensure acceptable management control over AFRRI experiments. TS 6.4 specifies the different types of experiments that require review and approval before being performed at AFRRI. It further specifies the extent of the analysis that should be submitted for review. TS 6.4 ensures that all experiments start off initially having a high level of review and that changes to existing approved experiments must undergo a level of review that is appropriate to the significance of the change. TS 6.2.4, "Review Functions," specifies that the RRFSS shall review safety evaluations for tests or experiments conducted under 10 CFR 50.59 and for any experiments that might meet any criteria in 10 CFR 50.59. Therefore, included in TS 6.4 is the determination of whether the new or modified experiments can be performed without prior NRC approval under the requirements of the regulations in 10 CFR 50.59. TS 6.4, as well as TS 6.1.2, "Responsibility," help ensure that limits on experiments are analyzed and approved to help ensure compliance with the requirements of 10 CFR Part 20 limits and 10 CFR 50.59. The NRC staff reviewed TS 6.4 and finds that TS 6.4 is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.4 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to review functions. On the basis of the information above, the NRC staff concludes that TS 6.4 is acceptable.

5.6.5 TS 6.5 Required Actions

TS 6.5 defines the required actions to be taken in the event of a safety limit violation and a reportable occurrence.

5.6.5.1 TS 6.5.1 Actions to be Taken in Case of Safety Limit Violation

TS 6.5.1 states the following:

a. The reactor shall be shut down immediately, and reactor operation shall not be resumed without authorization by the USNRC.

b. The safety limit violation shall be reported to the USNRC, the AFRRI Licensee, and the RRFSS not later than the next working day.

c. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the RRFSS, and shall describe (1) applicable circumstances preceding the violation, when known, the cause and contributing factors (2) effects of the violation on facility components, structures, or systems, the health and safety of personnel and the public and (3) corrective action taken to prevent or reduce the probability of recurrence.

d. The Safety Limit Violation Report shall be submitted to the USNRC, the AFRRI Licensee, and the RRFSS within 14 days of the violation.

TS 6.5.1, Specification a, requires the facility to shut down when the SL is exceeded and not resume operations until authorized by the NRC.

TS 6.5.1, Specification b, requires that the violation be reported to the AFRRI licensee, the RRFSS, and the NRC within 24 hours.

TS 6.5.1, Specification c, specifies the content of the SL violation written report, including the evaluations and corrective actions to be taken.

TS 6.5.1, Specification d, requires submitting the written report to the AFRRI Director, the RRFSS, and the NRC within 14 days.

The NRC staff finds that TS 6.5.1 meets the requirements of 10 CFR 50.36(c)(1) for actions to be taken if a safety limit is exceeded. Additionally, the NRC staff finds that TS 6.5.1 is consistent with the 10 CFR 50.36(c)(7) and 10 CFR 50.36(c)(8) requirements related to initial event notifications and written reports, respectively. The NRC staff also finds that TS 6.5.1 is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). On the basis of its review of the information above, the NRC staff concludes that TS 6.5.1 is acceptable.

5.6.5.2 TS 6.5.2 Reportable Occurences

TS 6.5.2 states the following:

> The types of events listed below shall be reported as soon as possible by telephone and confirmed in writing by facsimile, e-mail, or similar transmission to the USNRC no later than the following working day after confirmation of the event, with a written follow-up report within 14 days. The report shall include (as a minimum) the circumstances preceding the event, current effects on the facility, and status of corrective action. The report shall contain as much supplemental material as possible to clarify the situation. Supplemental reports may be required to fully describe the final resolution of the occurrence.
>
> a. Operation with any safety system setting less conservative than specified in Section 2.2, Limiting Safety System Setting for Fuel Temperature.
>
> b. Operation in violation of any Limiting Condition for Operation, Section 3, unless prompt remedial action is taken as permitted in section 3.
>
> c. Malfunction of a required reactor safety system component during operation that renders or could render the system incapable of performing its intended safety function unless the malfunction or condition is caused by maintenance.

d. Any unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips resulting from a known cause are excluded.

e. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of a condition that could result in operation of the reactor in a manner less safe than conditions covered in the Safety Analysis Report.

f. The release of fission products from a fuel element through degradation of the fuel cladding. Possible degradation may be determined through an increase in the background activity level of the reactor pool water.

g. Abnormal and significant degradation of the reactor coolant boundary or confinement boundary (excluding minor leaks).

h. A release of radioactivity that exceeds or could have exceeded the limits allowed by 10 CFR Part 20, or these Technical Specifications.

i. Unscheduled conditions arising from natural or man-made events that, as a direct result of the event, require operation of safety systems or other protective measures required by Technical Specifications.

j. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the bases for the Technical Specifications that have or could have permitted reactor operation with a smaller margin of safety than in the original analysis.

k. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analysis Report or Technical Specifications bases, or discovery during facility life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

The NRC staff finds that the reporting requirements in TS 6.5.2 are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS 15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.5.2 is consistent with the 10 CFR 50.36(c)(7) and 10 CFR 50.36(c)(8) requirements related to initial event notifications and written reports, respectively. Therefore, the NRC staff concludes that TS 6.5.2 is acceptable.

5.6.5.3 TS 6.5.3 Actions to be Taken in Case of Reportable Occurrences

TS 6.5.3 states the following:

a. Reactor conditions shall be returned to normal, or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Reactor Facility Director or designated alternate.

b. The occurrence shall be reported to the Reactor Facility Director or designated alternate and to the USNRC.

c. The occurrence shall be reviewed by the RRFSS at its next scheduled meeting.

TS 6.5.3, Specification a, requires the reactor to be returned to normal or shut down in case a reportable occurrence requires corrective action. TS 6.5.3, Specification a, also states that the Reactor Facility Director must authorize restart of the reactor after a reportable occurrence.

TS 6.5.3, Specifications b and c, require that the occurrence must also be reported to the Reactor Facility Director and the NRC, and the occurrence must be reviewed by the RRFSS.

The NRC staff finds that TS 6.5.3 is consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64) and, therefore, is acceptable. The NRC staff finds that TS 6.5.3 gives acceptable notification of the Reactor Facility Director and the NRC, provides for a review by the RRFSS, and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information above, the NRC staff concludes that TS 6.5.3 is acceptable.

5.6.6 TS 6.6 Operating Reports

TS 6.6 lists the type of required reports that must be prepared for various situations that may occur during AFRRI operations as follows:

> In addition to the applicable reporting requirements of Title 10 of the Code of Federal Regulations, the following reports shall be submitted to USNRC Document Control Desk.
>
> a. Annual Operating Report: Routine operating reports covering the operation of the reactor during the previous calendar year shall be submitted by March 31 of each year. The Annual Operating Report shall provide a comprehensive summary of the operating experience having safety significance during the year, even though some repetition of previously reported information may be involved. References in the Annual Operating Report to previously submitted reports shall be clear.
>
> Each Annual Operating Report shall include:
>
> 1. A brief narrative summary of:
>
>   a. Changes in facility design, performance characteristics, and operating procedures related to reactor safety that occurred during the reporting period;
>
>   b. Results of surveillance test and inspections;
>
> 2. A tabulation showing the energy generated by the reactor on a monthly basis, the cumulative total energy since initial criticality, and the number of pulses greater than $2.00;
>
> 3. List of the unscheduled shutdowns for which corrective action was required to ensure safe operation of the reactor, including the reasons and the corrective actions taken;

4. Discussion of the major safety-significant corrective and/or preventative maintenance performed during the period, including the effects (if any) on the safe operation of the reactor, and the reasons for the corrective maintenance required;

5. A brief description of:

    a. Each change to the facility to the extent that it changes a description of the facility in the Safety Analysis Report;

    b. Changes to the procedures as described in the Safety Analysis Report;

    c. Any new experiments or tests performed during the reporting period that is not encompassed in the Safety Analysis Report;

6. A summary of the safety evaluation made for each change, test, or experiment not submitted for Commission approval pursuant to Section 50.59 of 10 CFR Part 50. The summary shall show the reason leading to the conclusion that the criteria in paragraph (c)(2) of that Section were not met and that no change to the Technical Specifications was required;

7. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as determined at or prior to the point of such release or discharge. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient.

    a. <u>Liquid Waste</u> (summarized on a monthly basis)

        i. Radioactivity discharged during the reporting period;

        Total radioactivity released (in curies);

        Concentration limits used and isotopic composition for fission and activation products

        Total radioactivity of each nuclide released during the reporting period and, based on representative isotopic analysis, average concentration at point of release during the reporting period;

        ii. Total volume of effluent water (including diluents) during periods of release;

    b. <u>Gaseous Waste</u> (summarized on a quarterly basis)

    Radioactivity discharged during the reporting period for:

        Argon-41;

        Particulates with half-lives greater than eight days;

c. Solid Waste (summarized on a quarterly basis)

   Total cubic feet and combined activity in curies of materials in solid form disposed of under license R-84;

8. A description of the results of any environmental radiological surveys performed outside the facility;

9. A list of exposures greater than 25% of the allowed 10 CFR Part 20 limit received by reactor personnel or visitors to the reactor facility;

b. Other Reports: A report shall be submitted to the USNRC within 30 days describing:

   1. Any permanent change of either the AFRRI Licensee or the Reactor Facility Director; or

   2. Significant changes in the transient or accident analysis described in the SAR.

TS 6.6 lists the operating reports that must be prepared, as well as their schedules for completion and submittal. TS 6.6, Specification a, items 1 through 9, help ensure that adequate annual reporting information is maintained. TS 6.6, Specification a, Items 1 through 9, give requirements for the status of the facility, major changes, radiation exposures, and other pertinent information to be provided to the NRC. The NRC staff finds that TS 6.6, Specification a, items 1 through 9, set AFRRI annual operating report requirements that are consistent with guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). TS 6.6, Specification b, items 1 and 2, help ensure that special reporting requirements are adequately delineated.

The NRC staff reviewed TS 6.6 and finds that the reporting requirements as specified in TS 6.6 are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). Additionally, the NRC staff finds that TS 6.6 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to reporting requirements. (Note that TS 6.6, Specification a, item 7, is also discussed in SER Section 3.2.1.) On the basis of its review of the information above, the NRC staff concludes that TS 6.6 is acceptable.

5.6.7 TS 6.7 Records

TS 6.7 lists the types of required records that must be retained at the facility and specifies the duration of the retention period. This ensures that important documents remain available consistent with the guidance in ANSI/ANS-15.1-2007 (Ref. 64).

TS 6.7.1, “Records that Shall be Retained for a Period of at Least Five Years,” lists the following items:

a. Normal reactor operations;

b. Principal maintenance operations;

c. Reportable occurrences;

d. Surveillance activities required by Technical Specifications;

e. Reactor facility radiation and contamination surveys;

f. Experiments performed with the reactor;

g. Changes to operating procedures;

h. Fuel inventories and fuel transfers;

i. Records of meetings of the RRFSS.

TS 6.7.2, "Records to be Retained for at Least One Certification Cycle," states the following:

> Records of retraining and requalification of licensed operators and senior reactor operators shall be retained at all times the individual is employed or until the license is renewed.

TS. 6.7.3, "Records that Shall be Retained for the Life of the Facility," lists the following items:

a. Gaseous and liquid radioactive effluents released to the environs;

b. Appropriate offsite environmental monitoring surveys;

c. Radiation exposures for all reactor personnel monitored; and

d. Drawings of the reactor facility.

e. Reviews and reports pertaining to a violation of the safety limit, limiting safety system setting (LSSS) or limiting condition of operation (LCO).

TSs 6.7.1 through 6.7.3 list the types of required records that must be retained and specify the duration of that retention for various functions and also situations that may occur during AFRRI operations. This ensures that important information is made available in a timely manner for proper management oversight.

TS 6.7.1 helps ensure that record retention requirements for records to be retained for at least 5 years are properly delineated in the TS. The NRC staff finds that the record requirements in TS 6.7.1 are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff also finds that TS 6.7 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to recordkeeping requirements. On the basis of its review of the information above, the NRC staff concludes that TS 6.7.1 is acceptable.

TS 6.7.2 helps ensure that the licensee maintains training records of licensed operators while they are employed or until the license is renewed. The NRC staff reviewed TS 6.7.2 and finds that the record retention requirements in TS 6.7.2 are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff also finds that TS 6.7 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to recordkeeping requirements. On the basis of the information above, the NRC staff concludes that TS 6.7.2 is acceptable.

TS 6.7.3, help ensure that AFRRI facility records retention requirements for records that need to be retained for the lifetime of the AFRRI facility are appropriately delineated. The NRC staff reviewed TS 6.7.3, and finds that TS 6.7.3, Specifications a through d, for record retention requirements are consistent with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64). The NRC staff also finds that TS 6.7 is consistent with 10 CFR 50.36(c)(5), which requires, in part, that licensees have TSs on administrative controls related to recordkeeping requirements necessary to assure safe operation. On the basis of the information above, the NRC staff concludes that TS 6.7.3 is acceptable.

5.7 Conclusions

The NRC staff reviewed and evaluated the TS as part of its review of the application for license renewal of Facility Operating License No. R-84 (NRC Docket No. 50-170). The TSs define certain features, characteristics, and conditions that govern the operation of the AFRRI TRIGA reactor. The TS are explicitly included in the renewed license as Appendix A. The NRC staff reviewed and evaluated the content of the TSs to determine whether the TSs meet the requirements in 10 CFR 50.36, "Technical specifications." Based on its review, the NRC staff concludes that the AFRRI TRIGA TSs meet the requirements in the regulations. The NRC staff also reviewed the format and content of the proposed TSs for consistency with the guidance in NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64) and finds that the TSs are consistent with the guidance. The NRC staff based this conclusion on the following findings:

- To satisfy the requirements in 10 CFR 50.36(a), the licensee provided proposed TSs with the application for license renewal. As required by the regulation, the TSs (other than those covering administrative controls) included appropriate summary bases. Those summary bases are not part of the TSs as required by 10 CFR 50.36(a)(1).

- The AFRRI is a facility of the type described in 10 CFR 50.21(c); therefore, as required by 10 CFR 50.36(b), the facility operating license will include the TSs. To satisfy the requirements in 10 CFR 50.36(b), the licensee provided proposed TSs derived from analyses in the AFRRI TRIGA reactor license renewal SAR, as supplemented.

- The proposed TSs acceptably implement the recommendations of NUREG-1537 (Ref. 39) and ANSI/ANS-15.1-2007 (Ref. 64), by using definitions that are acceptable.

- To satisfy the requirements in 10 CFR 50.36(c)(1), the licensee proposed TSs that specify a safety limit for the fuel temperature and a limiting safety system setting for the reactor protection system to ensure that the safety limit is not reached.

- The proposed TSs contain limiting conditions for operation on each item that meets one or more of the criteria in 10 CFR 50.36(c)(2)(ii).

- The proposed TSs contain surveillance requirements that satisfy the requirements in 10 CFR 50.36(c)(3).

- The proposed TSs contain design features that satisfy the requirements in 10 CFR 50.36(c)(4).

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

The NRC staff reviewed the proposed TSs and finds the proposed AFRRI TRIGA TSs acceptable. The NRC staff concludes that normal operation of the AFRRI TRIGA within the limits of the proposed TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or for workers' occupational exposures. The NRC staff concludes that the proposed TSs provide reasonable assurance that the AFRRI reactor will be operated as analyzed in the SAR, as supplemented, and that adherence to the proposed TSs will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 4, "Accident Analyses" of this SER.

6. CONCLUSIONS

Based on its evaluation of the license renewal application as discussed in the previous chapters of this safety evaluation report, the U.S. Nuclear Regulatory Commission (NRC) staff concludes the following:

- The application for license renewal dated June 24, 2004, as supplemented by letters dated March 4 (Ref. 2), August 13 (Ref. 3), September 27 (Ref. 4), October 21 (Ref. 5), and December 15, 2010 (Ref. 6); February 7 (Ref. 7), June 20 (Ref. 8), September 6 (Ref. 9), October 20 (Ref. 10), and November 28, 2011 (Ref. 11); January 17 (Ref. 12), April 20 (Ref. 13), and September 21, 2012 (Ref. 14); June 28 (Ref. 15) and August 27, 2013 (Ref. 16); December 4, 2014 (Ref. 17); March 30, 2015 (Ref. 18); and February 9 (Ref. 19), February 29 (Ref. 20), August 5 (Ref. 21), September 12 (Refs. 76 and 77), September 21 (Ref. 78), September 26 (Ref. 79), September 27 (Ref. 83), September 30 (Ref. 84), and November 16, 2016 (Ref. 36), complies with the standards and requirements in the Atomic Energy Act of 1954, as amended (AEA), and the U.S. Nuclear Regulatory Commission (NRC) rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR).

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

- There is reasonable assurance that (1) the activities authorized by the renewed 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 facility will continue to be useful in the conduct of research and development activities.

- 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 NRC.

- The applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," have been satisfied.

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

- The issuance of this license is in accordance with the NRC's regulations in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," and all applicable requirements have been satisfied.

- The receipt, possession, and use of byproduct and special nuclear materials as authorized by this facility operating license will be in accordance with NRC regulations in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

7. REFERENCES

1. Armed Forces Radiobiology Research Institute, letter to U.S. Nuclear Regulatory Commission, Application for Renewal of Facility Operating License R-84, Docket Number 50-170, dated June 24, 2004:

    - Application for Renewal (ADAMS Accession No. ML041800067)
    - Safety Analysis Report (ADAMS Accession No. ML101650415, redacted version)
    - Environmental Report and Technical Specifications (ADAMS Accession No. ML041800068)
    - Reactor Operator Requalification Program, Financial Qualifications, and Decommissioning Cost Estimate (ADAMS Accession No. ML041800071)
    - AFRRI Emergency Plan (ADAMS Accession No. ML16258A465 redacted version)

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