Safety Evaluation Report

Renewal of the Facility Operating License for the Board of Regents of the University of California, University of California, Irvine Nuclear Reactor Facility

License No. R-116
Docket No. 50-326

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

July 2016

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to a timely application filed by the Board of Regents of the University of California (the licensee) for a 20-year renewal of Facility Operating License No. R-116 to continue operating the University of California, Irvine Nuclear Reactor Facility (UCINRF). In its safety review, the NRC staff considered information the licensee submitted, including past operating history recorded in the licensee's annual reports to the NRC, inspection reports NRC personnel prepared, as well as firsthand observations. On the basis of its review, the NRC staff concludes that the 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, UCINRF staff, or the environment.

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ABBREVIATIONS AND ACRONYMS

$	dollar(s) (unit of reactivity or United States currency)
°C	degree(s) Celsius
°C/min	degree(s) Celsius per minute
°F	degree(s) Fahrenheit
°F/min	degree(s) Fahrenheit per minute
μCi	microcurie(s)
μCi/cm^3	microcurie(s) per cubic centimeter
μCi/mL	microcurie(s) per milliliter
μmho/cm	micromho(s) per centimeter
μs	microsecond(s)
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
ALARA	as low as reasonably achievable
ALI	annual limit on intake
Am-Be	americium-beryllium
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar-41	argon-41 (isotope of argon)
ATR	adjustable transient rod
CAM	continuous air particulate monitor
cc	cubic centimeter(s)
cfm	cubic foot (feet) per minute
CFR	*Code of Federal Regulations*
CHF	critical heat flux
Ci	curie(s)
cm	centimeter(s)
cpm	count(s) per minute
DAC	derived air concentration
DCF	dose conversion factor
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
EFPD	effective full-power day
EH&S	Environmental Health and Safety
EP	emergency plan
ft	foot (feet)
ft^3	cubic foot (feet)

FTR	fast transient rod
FY	fiscal year
g	acceleration due to gravity
g	gram(s)
GA	General Atomics
gal	gallon(s)
H	hydrogen
HEPA	high-efficiency particulate air or high-efficiency particulate arrestance
hr	hour
HV	high voltage
HVAC	heating, ventilation, and air conditioning
I	iodine
I-129	iodine-129 (isotope of iodine)
IFE	instrumented fuel element
in	inch(es)
IR	inspection report
ISG	interim staff guidance
k_{eff}	effective multiplication factor
kg	kilogram(s)
Kr	krypton
Kr-85	krypton-85 (isotope of krypton)
kW	kilowatt(s)
kWt	kilowatt(s) thermal
L	liter(s)
lb	pound(s)
LC	license condition
LCC	limiting core configuration
LCO	limiting condition(s) for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LOCF	loss of coolant flow
LRA	license renewal application
LSSS	limiting safety system setting
m	meter(s)
m/s	meter(s) per second
m^3/s	cubic meter(s) per second
MACCS	MELCOR Accident Consequence Code System
MCNPX	Monte Carlo N-Particle eXtended (computer code)
mg	milligram(s)
MHA	maximum hypothetical accident
mph	mile(s) per hour
mrad/hr	millirad(s) per hour
mrem	millirem(s)
mrem/hr	millirem(s) per hour
mrem/yr	millirem(s) per year

msec	millisecond(s)
MW	megawatt(s)
MWh	megawatt-hour(s)
MWt	megawatt(s) thermal
N-16	nitrogen-16 (isotope of nitrogen)
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NVLAP	National Voluntary Laboratory Accreditation Program
pH	potential of hydrogen
PSP	physical security plan
PTS	pneumatic transfer system
RAI	request(s) for additional information
RAM	radiation area monitor
rem	roentgen equivalent(s) man
rem/hr	roentgen equivalent(s) man per hour
RG	regulatory guide
ROC	Reactor Operations Committee
RSC	Radiation Safety Committee
RSO	radiation safety officer
RSP	Radiation Safety Program
RTR	research and test reactor
s	second(s)
SAR	safety analysis report
SDM	shutdown margin
SER	safety evaluation report
SL	safety limit
SNM	special nuclear material
SOI	statement(s) of intent
SRM	staff requirements memorandum/memoranda
SRO	senior reactor operator
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TNT	trinitrotoluene
TRIGA®	Training, Research, Isotopes, General Atomics
TS	technical specification(s)
U	uranium
U-235	uranium-235 (isotope of uranium)
U-ZrH	uranium-zirconium hydride
UCI	University of California, Irvine
UCINRF	University of California, Irvine Nuclear Reactor Facility
wt%	weight percent

Zr	zirconium
$\Delta k/k$	absolute reactivity

SECTION 1

INTRODUCTION

1.1 Overview

By letter dated October 18, 1999 (Ref. 1), as supplemented on April 24 and June 2, 2000 (Refs. 2 and 3); January 27 (Ref. 4), March 23 (Ref. 5), May 17 (Ref. 6), July 14 (Ref. 7), August 25 (two letters) (Refs. 31 and 32), October 20 (two attachments) (Ref. 8), and October 29, 2010 (Ref. 28); June 7 (two letters) (Refs. 9 and 10), June 24 (Ref. 12), July 7 (Ref. 11), August 1 (Ref. 13), October 3 (Ref. 14), and December 2, 2011 (two letters) (Refs. 15 and 16); January 12 (Ref. 17), March 1 (Ref. 18), and September 11, 2012 (Ref. 60); February 26 (Ref. 83), March 5 (Ref. 68), March 7 (Ref. 84), July 11 (Ref. 69), and October 8, 2014 (two letters) (Refs. 48 and 70); December 22, 2015 (two attachments) (Ref. 64); and April 22 (Ref. 76), April 29 (Ref. 81), and May 13, 2016 (Ref. 86), the Board of Regents of the University of California (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) a license renewal application (LRA) for a 20-year renewal of the Class 104c Facility License No. R-116 (the license), NRC Docket No. 50-326, for the University of California, Irvine Nuclear Reactor Facility (UCINRF).

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.51(a) states that "[e]ach license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from date of issuance" (Ref. 19). The Board of Regents of the University of California holds the license for the UCINRF that was issued on November 25, 1969. The term of the license was for a period of 30 years expiring on November 24, 1999. Because of the timely renewal provision contained in 10 CFR 2.109(a), the licensee is permitted to continue operating UCINRF under the terms and conditions of the current license until the NRC staff completes action on the renewal request. A renewal would authorize the licensee to continue operation of the UCINRF for an additional 20 years.

The NRC staff based its review of the request to renew the UCINRF license on the information contained in the LRA, as well as in supporting supplements in response to the NRC staff's request for additional information (RAI). During the review process, the NRC staff issued RAI in letters dated December 3, 2009 (Ref. 21); May 26, 2010 (Ref. 22); April 5, 2011 (Ref. 23); and January 30 (Ref. 71), May 8 (Ref. 72), May 28 (Ref. 73), July 15 (Ref. 74), and October 1, 2014 (Ref. 75). The LRA includes the license renewal safety analysis report (SAR), technical specifications (TS), and an environmental report. The licensee also provided financial qualifications, decommissioning information, and an operator requalification plan. The licensee stated that the security plan and emergency plan (EP) were not revised as part of the LRA request. Discussion on these plans is provided below. The NRC staff also conducted site visits on September 10, 2012, and February 5-6, 2015, to observe facility conditions and to discuss the RAI and TS. As part of its review, the NRC staff also examined annual reports of facility operation submitted by the licensee from 2009 to 2015 and the inspection reports (IRs) prepared by NRC personnel from 2009 to 2016.

With the exception of the security plan and portions of the EP, material pertaining to this review may be examined or copied, for a fee, at the NRC's Public Document Room, at One White Flint North, 11555 Rockville Pike, Rockville, MD. The NRC staff maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Documents related to this license renewal may be accessed through the NRC's Public Library on the Internet at http://www.nrc.gov. The physical security plan (PSP) is protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements," and 10 CFR 2.390(d). Portions of the EP is also withheld from public disclosure because both plans are considered security-related information. Because parts of the SAR and RAI responses from the licensee contain security-related information and are protected from public disclosure, only redacted versions are available to the public.

The "References" section of this safety evaluation report (SER) 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 application against the requirements of the regulations, including 10 CFR Part 20, "Standards for Protection against Radiation," 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The NRC staff also considered recommendations of applicable regulatory guides and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The NRC staff also considered the recommendations contained in NUREG-1537, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (Ref. 24). Because no specific accident-related regulations exist for research reactors, dose values for accidents are compared against the requirements in 10 CFR Part 20 (i.e., the standards for protecting employees and the public against radiation).

In SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," dated October 24, 2008 (Ref. 25), the NRC staff provided the Commission with information regarding plans to revise the review of LRAs for research and test reactors (RTRs). The Commission issued its staff requirements memorandum (SRM) for SECY-08-0161 on March 26, 2009 (Ref. 26). 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 directed the NRC staff to put into place a graded approach with a scope commensurate with each facility's risk. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 of SECY-08-0161. In the alternative safety review approach, the NRC staff should consider the results of past NRC staff reviews when determining the scope of the review. A basic requirement, as stated in the SRM, is that licensees must be in compliance with applicable regulatory requirements.

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

The NRC staff reviewed the LRA using the guidance in the final ISG, dated October 15, 2009 (Ref. 27), and, because the licensed power level for the UCINRF is less than 2 MWt, the NRC staff performed a focused review of the LRA. Specifically, the review focused on reactor design and operation, accident analysis, TS, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to the facility after submitting the application.

The NRC staff reviewed the PSP for UCINRF, titled "Physical Security Plan, Nuclear Reactor Facility, University of California, Irvine," Revision 4, dated February 2014, submitted by letter dated March 5, 2014 (Ref. 68). The NRC staff issued RAI to the licensee in letters dated May 28 and October 1, 2014 (Refs. 73 and 75), and the licensee responded by letters dated July 11, 2014, and October 23, 2014, respectively, including a revised PSP. The NRC staff reviewed the revised PSP, Revision 4.1.1, finds that it meets the applicable regulations, and based on that finding, concludes that the security plan, dated October 2014, is acceptable. The licensee maintains the program to provide the physical protection of the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials." Changes to the PSP can be made, by the licensee, 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 PSP. The NRC staff's review of the IRs for the past several years identified one Severity Level IV violation. However, corrective actions planned and taken to correct the violation and prevent recurrence were adequately addressed during the inspection and documented in the IR (IR 50-326/2016-201) (Ref. 85).

The NRC staff reviewed the EP for UCINRF, entitled "Emergency Plan for the University of California, Irvine Nuclear Reactor Facility," Revision 4.1, dated February-March 2013, submitted by letter dated May 23, 2013 (Ref. 82). The NRC staff issued RAI to the licensee in a letter dated January 30, 2014 (Ref. 71), and the licensee provided its responses by letters dated February 26, 2014 (Ref. 83), and March 7, 2014 (Ref. 84), including a revised EP. The NRC reviewed the revised EP for UCINRF, Revision 4.1, and finds that it meets the applicable regulations, and based on that finding, concludes that the EP, dated March 7, 2014, is acceptable. Additionally, the licensee is required to maintain the EP, in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," which reasonably ensures 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 EP, and no violations have been identified in recent years.

The licensee provided a revised "UCINRF Operator Requalification Program," dated April 24, 2000 (Ref. 2), and submitted an updated Operator Requalification Program by letter dated October 29, 2010 (Ref. 28), in response to RAI dated October 1, 2010. The NRC staff reviewed and approved the Operator Requalification Program on April 15, 2011 (Ref. 28).

The NRC staff separately evaluated the environmental impacts of the renewal of the license for UCINRF 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 February 13, 2012 (77 FR 7610), which concluded that renewal of the UCINRF 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 of the UCINRF safety review and to delineate the technical details considered in evaluating the radiological safety aspects of continued operation. This report provides the basis for renewing the UCINRF license at a steady-state power level up to and including 250 kWt. The license also authorizes short-duration power pulses with insertions not to exceed $3.00 of reactivity. These pulses are calculated not to raise the fuel temperature at the hottest core location above 830 degrees Celsius (°C) (1,526 degrees Fahrenheit (°F)).

This SER was prepared by Michael F. Balazik, Patrick G. Boyle, A. Jason Lising, A. Francis DiMeglio, and Walter A. Meyer, project managers in the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking, Research and Test Reactors Licensing Branch, and Lois Kosmas, a financial analyst in the NRC's NRR, Division of Inspection and Regional Support, Financial Analysis and International Projects Branch. URS Washington Safety Management Solutions, a contractor of the NRC, also provided input to this SER.

1.2 Summary and Conclusions regarding the 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 reports to the NRC, as well as IRs prepared by the NRC staff. On the basis of this evaluation and resolution of the principal issues reviewed for the UCINRF, the NRC staff made the following findings:

- The design and use of the reactor structures, systems, and components important to safety during normal operation discussed in SAR Chapter 4 (Ref. 20), 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 has considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA), emphasizing those that could lead to a loss of integrity of fuel element cladding and a release of fission products. The licensee analyzed the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed the dose limits specified in 10 CFR Part 20 for unrestricted areas.
- The licensee's management organization, training, and research activities, in accordance with the TSs, are adequate to ensure safe operation of the facility.
- The systems provided for the control of radiological effluents, when operated in accordance with the TSs, are adequate to ensure that releases of radioactive materials from the facility are within the limits of NRC regulations and are as low as is reasonably achievable.
- The licensee's TSs, which specify limits that control operation of the facility, offer a high degree of assurance that the facility will be operated safely and reliably. No significant degradation of the reactor has occurred, as discussed in the SAR Chapter 4 (Ref. 20), 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 it's 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 it's SNM.
- The licensee maintains an EP in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, which reasonably ensures 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 personnel who can safely operate the reactor.

On the basis of these findings, the NRC staff concludes there is reasonable assurance that the licensee can continue to operate the UCINRF in accordance with the Atomic Energy Act of 1954, as amended (AEA), NRC regulations, and the renewed facility operating license without endangering public health and safety, UCINRF staff, 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 UCINRF is in the basement of Rowland Hall on the University of California, Irvine (UCI), main campus and is operated by the Department of Chemistry. Rowland Hall is a multipurpose building that contains classrooms, laboratories, and other educational facilities. The UCINRF licensed area consists of the reactor room, two associated laboratories, and a control room on a single level in the basement of Rowland Hall. Although other research laboratories and work areas are in the general vicinity of the UCINRF, none share common walls with the reactor facility.

Facility Operating License R-116 authorizes the licensee to operate the UCINRF at a maximum steady-state power level of 250 kWt and to pulse the reactor with a maximum reactivity insertion of $3.00.

The UCINRF is an in-ground pool-type Training, Research, Isotopes, General Atomics (TRIGA®) Mark I reactor. The UCINRF operates with TRIGA® fuel, which is uranium-zirconium hydride, loaded to 8.5 weight percent uranium (U) of less than 20 percent uranium-235 (U-235) enrichment. The core is surrounded by a graphite reflector assembly contained in an aluminum casing. The pool water also serves as a reflector. The reactor assembly rests on an aluminum platform about 2 ft (0.610 m) above the bottom of a reinforced concrete pool with an aluminum liner.

Zirconium hydride that is homogeneously combined with the U fuel is the primary provider of neutron moderation. The pool water between the fuel elements also supplies some neutron moderation. The reactor has a strong, prompt negative temperature coefficient of reactivity that limits reactor power in the event of a power excursion. The core is cooled by natural convection of pool water. The reactor coolant water in the pool is maintained below 25 °C (77 °F) by circulation through a heat removal system. The reactor's experimental facilities include a pneumatic transfer system, rotary sample rack, and central thimble irradiation facility, as well as locations in and around the reactor core. Because of the in-ground pool, the reactor facility does not incorporate beam ports.

Four control rods are used to control the reactivity of the reactor core. Two motor-driven control rods serve as a shim and a regulating rod. The third and fourth rods are an adjustable transient rod (ATR) and a fast transient rod (FTR). The ATR can be motor driven or pneumatically driven. The FTR is only pneumatically driven. The FTR serves as a safety rod during steady-state operation of the reactor. The transient rods (ATR and FTR) can be quickly removed from the core by air pressure to allow the reactor to operate in pulse mode. Each rod can be disengaged from its drive (pneumatic or electric) and will drop into the core by gravity, immediately returning the core to a subcritical configuration.

1.4 Shared Facilities and Equipment

The UCINRF comprises five rooms all contained in a separate area, within a five-story building that contains minimal shared facilities, which include electric power, heating, cooling, water, and sewerage. The shared facilities affecting the UCINRF remain under the control of UCINRF staff. During the site visit on September 10, 2012, the NRC staff noted no other shared facilities or equipment.

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

The UCINRF is integrated with the common building ventilation (Ref. 64). This ventilation system maintains a negative pressure difference between the UCINRF and outside areas during both normal and emergency operation of the ventilation system. Manual or solenoid-operated valves, or both, provide isolation of this system to maintain containment integrity. In the event of a high-radiation reading from the continuous air particulate monitor (CAM) in the facility, normal ventilation is secured, and a dedicated emergency purge exhaust is actuated.

The UCI chilled water system supplies secondary cooling water for the shell side of the pool water heat exchanger. Although the safety analysis specifies no safety function for this system, a dedicated heat exchanger can remove heat from the pool generated by reactor operation. A water purification system that maintains pool water quality. Frequent monitoring of pool conductivity and system differential pressures by operators are used in combination to assist in identifying a possible leak from the secondary (UCI chilled water) to the primary (pool water) boundary of the heat exchanger.

1.5 Comparison with Similar Facilities

The first TRIGA® reactor went into operation at the General Atomics Laboratories in San Diego, CA, on May 31, 1958. Since then, over 60 TRIGA® reactors have been built and safely operated in over 20 countries. Pulsing TRIGA® reactors are in operation at Oregon State University, University of Wisconsin, Washington State University, University of Texas, and several other locations. Instruments and controls used in the UCINRF are similar in design and operation to most TRIGA® research reactors licensed by the NRC. The TRIGA® fuel typically has no performance-related issues as long as the well-established operating and water quality limits are maintained (Ref. 34).

1.6 Summary of Operations

The UCINRF was initially licensed for operation on November 23, 1969. During the past 47 years, the reactor facility has been used for education and training, nuclear research, and a range of irradiation services for UCI and other institutions. The reactor has accumulated 1,637.76 MWh of power operation through June 2015 and has been pulsed 1,010 times without a fuel rod failure. During the year ending June 30, 2015, over 2,800 samples were irradiated. The licensee expects to maintain or increase the previous rate of use for the period of license renewal.

The NRC staff review considered UCINRF annual reports from 2005 through 2015 and IRs from 2005 through 2016. The annual report summaries did not indicate significant degradation of fuel element integrity, control rod operability issues, or radiological exposure concerns. The NRC staff IRs identified three Severity Level IV violations: (1) a failure to conduct the required annual onsite emergency drill (Violation No. 50-326/2010-201-01), (2) a failure to maintain adequate reactor operator staffing in the control room during reactor operation (Violation No. 50-326/2011-201-01), and (3) exceedance of the recurrent training requirement for shipping hazardous material (Violation No. 50-326/2011-201-02).

The Severity Level IV violations are the least severe of the cited violations. They are considered more regulatory or safety significant than minor violations, but result in no or relatively inappreciable potential regulatory or security consequences. Corrective action to preclude recurrence is required to resolve a Severity Level IV violation, which the licensee provided in its response to the Notice of Violation, by letters dated February 2, 2011 (ADAMS Accession No. ML110460119), for Violation No. 50-326/2010-201-01, and February 14, 2012 (ADAMS Accession No. ML12073A327), for Violation No. 50-326/2011-201-01 and Violation No. 50-326/2011-201-02.

The NRC staff reviewed the licensee’s February 2, 2011, letter and finds that the licensee’s actions planned and taken to correct the violation and prevent reoccurrence were addressed. The NRC staff documented its finding in the IR (IR 50-326/2011-201) dated January 17, 2012. The NRC staff also reviewed the licensee’s February 14, 2012, letter and finds that actions planned and taken to correct the violation and prevent reoccurrence were addressed. The NRC staff plans to document its finding in the next IR.

The licensee stated that the facility will continue to be used as a source of neutrons and gamma radiation for basic and applied research and educational uses. Based on its review of the information provided in the application, the NRC staff finds, consistent with 10 CFR 50.21(c), that the facility will continue to be useful in the conduct of research and development activities.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research reactor, that the applicant reach an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level wastes and spent nuclear fuel.

In accordance with a letter from DOE to the NRC dated May 3, 1983 (Ref. 29), it has been determined that all universities and other government agencies operating nonpower reactors have entered into a contract with DOE that provides that DOE retain title to the fuel and is obligated to take the spent fuel or high-level waste for storage or reprocessing. An email from DOE to the NRC (Ref. 30) reconfirms this obligation with respect to the fuel at UCINRF (DOE Contract No. 78759, valid from March 1, 2009 - December 31, 2017). By entering into this contract with DOE, the licensee has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

A review of the modifications made to the facility during the license period indicates that most were technological upgrades to instrumentation, addition of experimental facilities, or minor changes to the design that either enhanced capabilities or improved reactor operations. The licensee stated that all modifications were evaluated under 10 CFR 50.59, "Changes, Tests and Experiments," to ensure that the safety of the UCINRF was not affected and prior NRC approval was not required.

The Rowland Hall ventilation system was extensively modified in 2011. The licensee indicated that the ventilation system was upgraded by the addition of three high-plume dilution exhaust fans designed to carry and dilute fumes from the chemistry laboratories and avoid the recycling of exhaust into the building. The licensee stated that this modification also results in additional dilution of any radioactive effluents produced by reactor operation. The licensee also installed a new well and pumping system near the reactor tank to reduce the likelihood of ground water intrusion adjacent to the reactor tank.

The licensee did not request any changes to its facility as part of this license renewal request.

1.9 Financial Considerations

1.9.1 Financial Ability To Operate the Reactor

The regulations in 10 CFR 50.33(f) state:

> 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.

The licensee does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Furthermore, 10 CFR 50.33(f)(2) states that "[a]pplicants to renew or extend the term of an operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license."

The NRC staff determined that, under 10 CFR 50.33(f), the licensee is subject to a full financial qualification review. Therefore, the licensee must supply information demonstrating that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the period of the license. The licensee must also submit estimates for the total annual operating costs for each of the first 5 years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover those costs.

By letters dated October 8, 2014 (two letters) (Refs. 48 and 70), the licensee submitted its projected operating costs for the UCINRF for each of the fiscal years (FYs) 2015 through 2019, which estimated a range from $22,000 in FY 2015 to $24,761 in FY 2019. The licensee stated that the projected operating costs are for "other" costs (e.g., supplies and maintenance) and that the administration of the reactor is the responsibility of teaching faculty paid from the University of California instructional funds. According to the licensee, its primary sources of funding to cover the UCINRF operating costs will come from the State-funded budget and recovery of costs from a variety of other sources, such as external and internal charges for facility use. The licensee expects that the funding sources will continue for the years mentioned. The NRC staff reviewed the licensee's estimated operating costs and projected source of funds and finds them to be reasonable.

The NRC staff finds that the licensee has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs for the UCINRF for the period of the renewed license. Accordingly, the NRC staff determines that the UCINRF meets the financial qualification requirements of 10 CFR 50.33(f) and is financially qualified to engage in the proposed activities regarding the UCINRF.

1.9.2 Financial Ability To Decommission the Facility

The regulations in 10 CFR 50.33(k) state, "[A]n application for an operating license … for a production or utilization facility, [must provide] information in the form of a report, as described in § 50.75 of this part, indicating how reasonable assurance will be provided that funds will be available to decommission the facility."

The regulations in 10 CFR 50.75(d)(1) require that "[e]ach non-power reactor applicant for or holder of an operating license … shall submit a decommissioning report as required by § 50.33(k) of this part." The decommissioning report must contain a cost estimate for decommissioning the facility, an indication of the funding method(s) used to ensure funding for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are specified in 10 CFR 50.75(e)(1).

In the October 8, 2014, letters (Refs. 48 and 70), the licensee updated its decommissioning cost estimate to $2 million (2014 dollars), based on the Decommissioning Plan for the University of Arizona TRIGA® Reactor, which was prepared by Enercon Services, Inc. The University of Arizona TRIGA® Reactor is similar to the UCINRF. The decommissioning cost estimate provides costs by tasks for decommissioning (e.g., reactor component removal, radiological waste packaging, transportation and disposal, preparation and approval of plans and procedures, University oversight) and includes a 25-percent contingency factor. According to the licensee, for the purpose of projection, the estimate assumes that the total cost to decommission the UCINRF is 50 percent labor and 50 percent other costs. The licensee stated that it will update the decommissioning cost estimate periodically in the future using inflation based on the licensee-established escalation rates. The NRC staff reviewed the information provided by the licensee concerning decommissioning of the UCINRF as well as the cost estimate based on a comparison with the costs of decommissioning the University of Arizona reactor. Based on its review, the NRC staff concludes that the decommissioning approach and decommissioning cost estimates are reasonable.

The licensee 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 indicate that funds for decommissioning will be obtained when necessary.

The licensee provided an SOI, dated October 8, 2014 (Ref. 48), that states, "[T]he University will make such funds [the current cost estimate for the UCINRF is $2 million in 2014 dollars] available for decommissioning when necessary."

To support the SOI and the licensee's qualifications to use an SOI, the application states that the licensee is a State of California Government entity and includes documentation to corroborate this statement. The application also offers information supporting the licensee's representation that the decommissioning funding obligations for the UCINRF are backed by the full faith and credit of the State of California. The licensee also provided information verifying that Michael P. Clark, Interim Provost and Executive Vice Chancellor (the signatory of the SOI), is authorized to execute contracts on behalf of the licensee.

The NRC staff reviewed the licensee's information on decommissioning funding assurance as described above, and under 10 CFR 50.75(e)(1)(iv), finds that the licensee is a State Government licensee, the SOI is acceptable to provide financial assurance, the decommissioning cost estimate is reasonable, and the licensee's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable to indicate that funds will be obtained when necessary. The NRC staff notes that any adjustment of the decommissioning cost estimate should incorporate, among other things, changes in costs resulting from the availability of disposal facilities, consistent with the requirement in 10 CFR 50.9, "Completeness and Accuracy of Information."

1.9.3 Foreign Ownership, Control, or Domination

Section 104(d) of the AEA, 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 regulation 10 CFR 50.38, "Ineligibility of Certain Applicants," similarly states this prohibition. According to the application, the licensee is a State Government entity within California and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

1.9.4 Nuclear Indemnity

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

1.9.5 Financial Consideration Conclusion

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

1.10 Emergency Power

The licensee's TSs do not include the emergency power TS (TS 3.6 and TS 4.6) as recommended by NUREG-1537 and ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors" (Ref. 33). In SAR Sections 3.7 and 13.8 (Ref. 20), the licensee stated that it has an available emergency power source. The licensee stated that radiation monitors (radiation area monitor and CAM systems) and alarms and the security monitoring and alarm system are wired to a building emergency power generator designed to power these loads automatically on failure of regular electric power. However, the emergency electric power system is not necessary to safely shut down the reactor and is not required to ensure protection of public health and safety. In the event of a loss of normal electric power, all control rods would automatically insert into the core by gravity. Operators can confirm the rods are seated down by visual observation with a flashlight. The reactor decay heat would be dissipated through natural circulation of the pool water.

The NRC staff reviewed the information provided by the licensee and finds that there is sufficient coolant in the reactor pool to absorb the decay heat from the reactor indefinitely. The NRC staff also finds that loss of normal electric power poses little risk to the health and safety of the public or to the UCINRF staff. On the basis of its review, the NRC staff concludes that emergency power and, therefore, TSs controlling emergency power are not required for UCINRF.

1.11 Facility Operating License Possession Limits and License Changes

The NRC staff reviewed the current UCINRF facility license and finds that the license contains conditions that control the receipt, possession, and use of by-product material and SNM in accordance with 10 CFR Parts 30 and 70. However, the format used in some license conditions (LCs) were out of date.

The renewal of the Facility Operating License No. R-116 for the UCINRF authorizes the receipt, possession, and use of special nuclear and by-product materials. SNM consists of such material as the U-235 in the reactor fuel, SNM in fission chambers, fission plates, foils, solutions, and SNM produced by operation of the reactor. By-product material consists of such material as activation products produced by operation of the reactor in the fuel, experiments, and reactor structure, and a 3-Ci sealed americium-beryllium neutron startup source. The restricted area is defined in TS 5.1 and SAR Section 3.4, and all activities performed within this area fall under the jurisdiction of the 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.

As is current practice, the NRC staff added LCs to prevent the separation of SNM and to clarify the by-product material possession requirements to prevent the separation of by-product material other than by-product material produced in experiments. The NRC staff reformatted the LCs to make them easier to read and understand. 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.

By letter dated April 29, 2016 (Ref. 81), the licensee requested that it needed an increase in the SNM possession limits of 0.5 kg. The current possession limit for Facility Operating License No. R-116 is to receive, possess, and use in connection with the operation of the facility up to a total 4.5 kg of U-235. The licensee stated that a total SNM possession limit of 5.0 kg is needed to sufficiently perform its research needs (Ref. 81).

In addition, the license requested a limit of 25 g of U-235 in the form of fission detectors, fission plates, foils, chemical compounds, or solutions enriched to any enrichment. Based on the information described above, the NRC staff concludes that a total SNM possession limit requested by the licensee is reasonable for facility operation and is acceptable.

SECTION 2

REACTOR DESCRIPTION

2.1 Summary Description

2.1.1 Introduction

The University of California, Irvine reactor, is a General Atomics (GA) Training, Research, Isotopes, General Atomics (TRIGA®) Mark I reactor that is licensed for a maximum steady-state power level of 250 kWt and to perform power pulsing with a maximum reactivity insertion of $3.00 to limit pulses to those that will not raise the fuel temperature at the hottest core location above 400 degrees Celsius (°C) (752 degrees Fahrenheit (°F)).

The reactor core is surrounded by a graphite reflector assembly contained in an aluminum casing. The reactor core and reflector assembly rests on an aluminum platform near the bottom of a cylindrical, water-filled aluminum tank, 10 ft (3.04 m) wide by 15 ft (4.57 m) long and 25 ft (7.62 m) deep, supported by a reinforced concrete shield. Concrete measuring 2 ft 6 in (0.76 m) surrounds the tank, and 4 ft 6 in (1.37 m) of concrete is below the tank. The approximately 20-ft (6.10-m) column of water above the core provides axial shielding as well as coolant. The control rod drives are mounted above the tank on a bridge structure spanning the diameter of the tank.

The reactor uses solid uranium-zirconium hydride (U-ZrH) fuel containing 8.5 weight percent (wt%) uranium (U) of less than 20-percent enrichment TRIGA® fuel arranged in a circular array. The fuel cladding is stainless steel. Inserting or withdrawing neutron-absorbing control rods regulates the reactor power.

The inherent safety of TRIGA® reactors has been demonstrated by the extensive experience gained from similar designs used throughout the world. The TRIGA® fuel is characterized by inherent safety, high-fission product retention and the ability to withstand water quenching at temperatures as high as 1,150 °C (2,102 °F). The safety of the fuel arises from the strongly negative prompt temperature coefficient characteristic of $U\text{-}ZrH_x$ fuel-moderator elements. The "x" represents the hydrogen (H)-to-zirconium (Zr) stoichiometry ratio in the $U\text{-}ZrH_x$ nomenclature. As the fuel temperature rises, this coefficient immediately compensates for reactivity insertions. The reactor fuel temperature safety limit (SL) (Technical Specification (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 425 °C (797 °F) as measured in the instrumented fuel element (IFE) in specific locations of the core (Section 2.5.3 of this safety evaluation report (SER) contains a discussion of the SL and LSSS). A series of GA and U.S. Nuclear Regulatory Commission (NRC) reports discusses such features as reactor kinetic behavior (GA-7882, "Kinetic Behavior of TRIGA® Reactors," dated March 31, 1967 (Ref. 46)), fission product retention (NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA® Reactors," issued August 1987 (Ref. 38), and GA-4314, "The $U\text{-}ZrH_x$ Alloy: Its Properties and Use in TRIGA® Fuel," issued in 1980 (Ref. 34)), and accident analysis (NUREG/CR-2387, "Credible Accident Analyses for TRIGA® and TRIGA®-Fueled Reactors," issued April 1982 (Ref. 51)).

Reactivity control is provided by motor-driven control rods: one shim, one regulating, one adjustable transient rod (ATR), and one fast transient rod (FTR). The ATR and the FTR can also be driven pneumatically and provide pulsing capability. All four control rods incorporate scram capability and act as safety rods during steady-state operation.

The instrumentation and control system consists of one instrumentation channel to monitor reactor fuel temperature in one of five possible core positions and three instrument channels that monitor the full range of power levels during licensed reactor operation.

The core is cooled by the natural convection of pool water purified by a filtration and demineralizer system. The pool water is cooled by a coolant system incorporating a heat exchanger in which the primary water is cooled by the University of California, Irvine (UCI), chilled water system.

2.1.2 Experimental Facilities

The experimental facilities are described in the safety analysis report (SAR), Section 10.0. The UCINRF has been designed with multiple in-core irradiation facilities to promote a broad range of potential experimental activities. These facilities include a rotary specimen rack, central thimble, pneumatic transfer systems (PTS), and individual fuel element grid locations.

The central thimble, at the center of the core, provides space for the irradiation of small samples at the point of maximum flux. The thimble is an aluminum tube, 1.5 in (3.81 cm) outside diameter with a wall thickness of 0.083 in (0.211 cm), which extends from the bridge and runs through the central grid plate holes to the reactor core. The tube is constructed so samples rest at approximately core centerline. The curves in the tube prevent radiation streaming to the room and a padlock on the entry funnel cap prevents placement of samples in the system without permission from the reactor operator.

The UCINRF has two PTS. A moderate-speed PTS extends from a receiver-sender station in the laboratory adjacent to the reactor room into a terminus in the core. A blower exhausts air from the system through a filter into the reactor room exhaust duct. The vacuum created by the blower is used to draw samples into or out of the in-core terminus. The terminus is in one of the holes in the bottom reactor grid plate and is supported by the bottom grid plate. The specimen capsule will come to rest vertically at about the midplane of the core. A fast, pneumatically operated transfer system is similarly constructed and installed, except the maximum sample capsule size is much smaller and solenoid control valves are used to port compressed nitrogen gas into and out of the system. The send-receive unit for the fast PTS is in the reactor room and directly attached to a radiation detector system. All systems are fixed in place so they cannot be inadvertently removed from the core.

The UCINRF has a rotary specimen rack, commonly called a “lazy Susan,” which is integral to the radial graphite reflector assembly. The rack supports 40 aluminum tubes that serve as receptacles for 1.12-in (2.84-cm) diameter specimen tubes. The rack may be rotated (repositioned) manually or electrically by a crank or by a motor and slip clutch, respectively. The specimen containers are inserted and removed through a tube extending from the rack housing to the bridge. This tube is offset with large bends to avoid direct line radiation streaming from the core.

TS 3.8.3, Failures or Malfunctions, states:

Specification(s).

a. Where the possibility exists that the failure of an experiment under (a) normal operating conditions of the experiment or the reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor room or an unrestricted area, the quantity and type of material in the experiment shall be limited in activity such that exposures of the reactor personnel to the gaseous activity or radioactive aerosols in the reactor room or control room shall not exceed the occupational dose limits in 10 CFR § 20.1201. Additionally, exposures to members of the public to these releases in the unrestricted area shall not exceed the dose limits in 10 CFR § 20.1301. In calculating potential consequences, one or more of the following assumptions shall be made, as applicable.

    i. 100% of the gases or aerosols escape from the experiment.

    ii. If the effluent from an experimental facility exhausts through a hold-up tank, which closes automatically on high radiation levels, 10% of the gaseous activity or aerosols will escape.

    iii. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, 10% of the aerosols will escape.

    iv. For materials whose boiling point is above 55 °C and where vapors formed by boiling this material are radioactive and could escape only through an undisturbed column of water above the core, 10% of these vapors escape.

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

TS 3.8.3 addresses the potential for failures and malfunctions of experiments by requiring assumptions for experiments that will help ensure that the source term calculations are conservative such that, if an experiment failure or malfunction should occur, the gases or aerosols released will not result in exceeding limits in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection against Radiation.” The TS also helps ensure that the reactor will not be operated with damage to fuel or structure if an experiment failure occurs with the potential to cause damage.

TS 3.8.3, Specification a, sets limits on the radioactive products produced in experiment materials that may release airborne radioactive particles and helps provide conditions to be used in the safety analysis of the experiment. The purpose of TS 3.8.3, Specification a, is to help ensure that potential releases of radioactive material from experiments are bounded by the exposure limits in 10 CFR Part 20 for UCINRF staff and members of the public. This includes experiment failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment. The NRC staff reviewed TS 3.8.3 and finds that the assumptions in TS 3.8.3, Specification a, are consistent with those made for other research reactors and help ensure that source term calculations are conservative.

The NRC staff reviewed TS 3.8.3, Specification a, and finds that these specifications help to limit doses from potential experiment failure or malfunction from exceeding 10 CFR Part 20 limits. Based on the information provided above, the NRC staff concludes that TS 3.8.3, Specification a, is acceptable.

TS 3.8.3, Specification b, helps ensure that any container failures are inspected to determine the consequences and need for corrective action. The NRC staff finds that the requirements of TS 3.8.3, Specification b, help ensure that the reactor does not operate without inspecting for possible damage to reactor fuel or structure from a failed experiment. Based on the information provided above, the NRC staff concludes that TS 3.8.3, Specification b, is acceptable.

The NRC staff finds that the experimental facilities are typical of TRIGA® reactors and their use is properly controlled by the TS 3.8.1, TS 3.8.2 (discussed in Sections 5.3.8.1 and 5.3.8.2 of this SER, respectively), and TS 3.8.3. The NRC staff reviewed TS 3.8.3 and finds that TS 3.8.3 adequately implements the guidance provided in NUREG-1537, Appendix 14.1, Section 3.8.3. Furthermore, on the basis of the information provided above, the NRC staff concludes that the experimental facilities and TS 3.8.3 are acceptable.

2.2 Reactor Core

The core is described in SAR Sections 4.1 and 4.2. The core uses solid fuel elements in which the zirconium-hydride moderator is homogeneously combined with enriched U fuel (U-ZrH_x fuel). The reactor core is a lattice of stainless-steel-clad U-$ZrH_{1.65}$ fuel-moderator elements and aluminum-clad graphite elements. The H-to-Zr stoichiometry ratio is represented by the "x" in the U-ZrH_x nomenclature. The H content is important because it influences many attributes of fuel behavior. Section 2.2.1 of this SER discusses the fuel elements. Approximately one-third of the core volume is water, which serves as the coolant and as additional moderator. Neutron reflection in the radial direction is provided by water, graphite dummy elements, and approximately 1 ft (0.30 m) of graphite enclosed in an aluminum housing.

Reactor core components are contained between top and bottom aluminum grid plates. The top grid plate has 126 positions for core components arranged in six concentric rings around a central thimble, which can be used for high-flux irradiations. The upper grid plate has sections that can be removed to provide larger in-core experiment locations. One location takes the place of three fuel positions and can accommodate a 2.4-in (6.1-cm) diameter experiment. The second variable location replaces seven fuel positions and can accommodate a 4.4-in (11.17-cm) diameter experiment. TS 5.3.1.b, "Reactor Core," states that the core fuel shall be kept in a close-packed array in core lattice positions except for control rods, single- or three-element or seven-element positions occupied by in-core experiments, irradiation facilities (including transfer system termini), graphite dummy elements, and a central dry tube.

The reactor core consists of 80 U-ZrH fuel elements, each with a central zirconium rod for structural integrity, an IFE, a fuel-followed shim control rod, a fuel-followed regulating control rod, an air-followed ATR, and an air-followed FTR. A graphite reflector block, on the periphery of the grid plate, is used for reflection of neutrons. There are two locations in the core where a startup neutron source may be positioned. The remaining positions contain experimental facilities. Section 2.2.1 of this SER discusses the IFE. Section 2.2.2 of this SER discusses the control rods.

The reactor core is near the bottom of an aluminum-lined concrete pool containing demineralized water. The radial shielding for the reactor consists of water and concrete and, in some directions, graphite. Specification a of TS 3.3.1, “Pool Water Level,” requires the pool water level to be at least 24 ft (7.32 m) above the tank floor (1 ft (0.30 m) below the tank edge) to provide for sufficient cooling and shielding.

TS 5.3.1, Reactor Core, states:

> Specification(s).
>
> a. The core assembly shall consist of TRIGA® standard 8.5/20 stainless steel clad fuel elements.
>
> b. The core fuel shall be kept in a close-packed array in core lattice positions except for control rods, single- or three-element or seven-element positions occupied by in-core experiments, irradiation facilities (including transfer system termini), graphite dummy elements, and a central dry tube.
>
> c. Reflection of neutrons shall be provided by combinations of graphite and water, with the graphite in sealed containment with aluminum cladding, either in the form of rods occupying grid positions, or in a larger reflector structure surrounding the core.
>
> d. An Am-Be neutron source shall be provided in one of two specific locations provided in the upper grid plate to provide start-up neutrons. It may be removed for maintenance purposes.

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

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

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

TS 5.3.1, Specification d, helps ensure that the startup source remains in its proper location and is handled in accordance with the SAR.

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

2.2.1 Reactor Fuel

General Atomics began development and use of TRIGA® U-ZrH fuels in 1957, and over 6,000 fuel elements have been fabricated for the more than 66 TRIGA® research reactors in various countries around the world. The TRIGA® U-ZrH fuel has unique safety features, including a large negative prompt temperature coefficient of reactivity such that, if excess reactivity were suddenly inserted into the core, the resulting fuel temperature increase would tend to limit the power excursion. These safety features also include high-fission product retention and chemical stability when quenched from high temperatures in water and dimensional stability over a large temperature range. Experiments performed in the late 1950s demonstrated that Zr hydride possesses the basic mechanism needed to produce these desired characteristics (Ref. 34). Over 25,000 pulses have been performed domestically and abroad with TRIGA® fuel elements, with fuel temperatures reaching peaks of about 1,150 °C (2,102 °F) without failure.

The fuel is described in the SAR. The UCINRF uses typically sized TRIGA® fuel elements with fueled sections 15 in (38.1 cm) long by 1.43 in (3.63 cm) in diameter, clad with 0.02-in (0.05-cm) thick 304 stainless steel. The cylindrical stainless-steel cladded fuel elements in which the fuel is a solid homogeneous mixture of U-ZrH alloy nominally contain 8.5 wt% U enriched to less than 20-percent uranium-235. To facilitate hydriding, a 0.18-in (0.46-cm) diameter hole is drilled through the center of the active section; a Zr rod is inserted in this hole after hydriding is complete. The ZrH stoichiometry ratio of the fuel is approximately 1.65. The fuel elements contain graphite reflectors at each end of the fuel and stainless steel top and bottom end fixtures to facilitate placement within the core support structure.

The UCINRF uses one IFE, which is identical to the standard fuel element with the exception of three thermocouples embedded in the fuel. The sensing tips are halfway between the vertical centerline and the outer radius and at the horizontal center, 1 in (2.54 cm) above and 1 in (2.54 cm) below the horizontal center. The IFE allows the temperature of the fuel to be directly measured.

TS 5.3.3, Reactor Fuel, states:

> Specification(s). Standard TRIGA® fuel elements shall have the following characteristics:
>
> a. The total uranium content shall be nominally 8.5% by weight, enriched to less than 20% ^{235}U.
>
> b. The hydrogen to zirconium atom ratio in the zirconium hydride shall be a nominal 1.65 hydrogen atoms to 1.0 zirconium atom.
>
> c. The cladding shall be 304 stainless steel, nominally 0.020 inches thick.
>
> d. An upper fitting with an engraved unique serial number shall be designed to fit a latching tool for fuel movement.

TS 5.3.3, Specification a, provides the nominal weight percent and maximum enrichment of the TRIGA® fuel and helps ensure that the fuel requirement is consistent with the analysis provided in the SAR Section 4.2.

TS 5.3.3, Specification b, provides the fuel stoichiometry to help ensure that it is consistent with the fuel used to develop the SL established in TS 2.1 and the LSSS in TS 2.2. The SAR contains an analyses based on the nominal stoichiometry of 1.65 for the fuel, which support the thermal margins developed for the bases to TS 2.1 and 2.2. The NRC staff finds the allowable fuel stoichiometry in TS 5.3.3, Specification b, acceptable.

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

TS 5.3.3, Specification d, helps ensure safe remote movement of each fuel element and that each fuel element is properly labeled for purposes of fuel accountability.

TS 5.3.3, Specifications a through d, help ensure that important design features of the reactor fuel are maintained as described in the SAR. TS 5.3.3 supports the bases for the SL in TS 2.1 and TS 2.2 and is consistent with the guidance provided in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” and ANSI/ANS-15.1-2007, “The Development of Technical Specifications for Research Reactors.” On the basis of the information provided above, the NRC staff concludes that TS 5.3.3 is acceptable.

Inspection of Reactor Fuel

During normal operation, fuel element growth and deformation can occur as described in NUREG-1537 (Ref. 24) and GA Report E-117-833 (Ref. 34). 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 time the fuel spends over a temperature threshold of approximately 750 °C (1,382 °F).

As described in the SAR, the maximum fuel temperature is well below this threshold for operations at 250 kWt, minimizing the possibility of swelling. Also, the licensee stated that the maximum fuel temperature during a $3.00 pulse has been calculated to be less than 400 °C (752 °F), and the time at which elevated temperature may be reached during pulsing is so short that pulse operations should not cause fuel swelling. In addition, the fuel is inspected regularly for defects, growth, and deformation.

As described in the SAR, the fuel cladding will be inspected to detect gross failure or visually observed deterioration. Attributes inspected include the fuel rod transverse bend and length, bulges, and any other visible cladding defects.

TS 3.1.6, Fuel Element Inspection Parameters, states:

> Specification(s). The reactor shall not be operated with any fuel element identified to show damage. An exception is made for operation up to a power level at which a leak becomes detectable solely in order to be able to identify the leaking element. A fuel element shall be identified as showing damage and be removed from core if:
>
> a. the transverse bend exceeds 1/16th inches (0.0625 in) over the length of the element; or

b. the growth in length over original measurements exceeds 1/8th inch (0.125 in); or

c. a cladding defect is suspected by a finding of release of any fission products; or

d. visual inspection identifies unusual pitting, bulging, or corrosion; or

e. burn-up of the element is estimated to exceed 50%.

TS 3.1.6, Specifications a through e, establish inspection requirements to detect gross failure or visual deterioration of the fuel. Specifications a through e help ensure that the reactor is operated only with fuel that has an effective cladding barrier to the release of any potential fission products. The fuel element attributes inspected include the fuel element transverse bend and length and a visual inspection for bulges or other cladding defects. The limit for transverse bend reduces the likelihood of difficulty in disassembling the core and helps to maintain cooling channel geometry. The elongation limit helps ensure that the cladding material will not be subjected to stresses that could cause a loss of fuel cladding integrity. The fuel burnup limit helps minimize H migration to help prevent excessive H pressures within the cladding. The reactor designer, GA, recommended these parameters. The NRC staff reviewed TS 3.1.6 and finds that the TS 3.1.6 limits on transverse bend and length are consistent with the values provided in NUREG-1537. The burnup limit in TS 3.1.6, Specification e, is consistent with the guidance provided in NUREG-1537.

There are cases where a cladding defect exists but is detectable only when the reactor is operating. In the unlikely event that a cladding defect is detected (normally by a small increase in the pool water radioactive material concentration or airborne radioactive material concentration), the TS allows the licensee to operate the reactor only in order to locate the damaged fuel element. All operation would need to meet regulatory requirements in 10 CFR Part 20 for release of radioactive material and occupational and public doses. On the basis of the information provided above, the NRC staff concludes that TS 3.1.6, Specifications a through e, are acceptable.

An important parameter in ensuring fuel element integrity is the fuel element temperature. TS 3.2.2, “Reactor Measuring Channels,” and TS 3.2.3, “Reactor Safety System,” require a fuel element temperature measuring channel and a fuel element temperature safety channel. TS 4.2 defines the periodicity at which the surveillance requirements are performed (discussed in Section 5.4.2 of this SER). Additional surveillance requirements for fuel elements and fuel follower control rods are contained in TS 4.1, “Reactor Core Parameters” (discussed in Section 2.5.2 of this SER).

The NRC staff reviewed the SAR Section 4.1 and Section 4.2, which describe the fuel elements used in the UCINRF, the design limits, and the technological and safety-related bases for these limits. The NRC staff finds that the licensee adequately discussed the constituents, materials, and components of the fuel elements. The NRC staff also finds that compliance with the applicable TS will help ensure uniform core operating characteristics and adherence to the design bases and safety-related requirements. On the basis of the information provided above, the NRC staff concludes that the fuel elements and the associated TSs are acceptable.

2.2.2 Control Rods

The control rods are described in SAR Section 4.3. The UCINRF uses boron carbide control rods that are characteristic of most TRIGA® reactors. The UCINRF uses two motor-driven (shim and regulating) and two transient control rods. The transient rods consist of a motor/pneumatically driven control rod (ATR) and a completely pneumatically driven control rod (FTR). All of the control rods have scram capabilities in accordance with UCINRF TS 5.3.2.

The shim and regulating rods consist of four sections: (1) the top section is graphite, (2) the control section is 15 in (38.1 cm) of graphite impregnated with powdered boron carbide (borated graphite), (3) the fuel-follower section is 15 in (38.1 cm) of 8.5 wt% U-ZrH fuel, and (4) the bottom section is 6.5 in (16.51 cm) of graphite. These control rods are clad with stainless steel. The two transient rods consist of an ATR and an FTR. The ATR consists of two sections: the top control section is 15 in (38.1 cm) of borated graphite, and the bottom section is 21 in (53.34 cm) of air-filled follower. The FTR has a double-length borated graphite section. The transient rods are clad with aluminum.

The control rod drive systems are all mechanically independent from one another, and a malfunction in one would not affect insertion or withdrawal of any other. The shim and regulating control rods are positioned with motor-driven rack-and-pinion drives connected to the control rod through an electromagnet. When the reactor is scrammed, the electromagnet is deenergized, and the shim and regulating rods drop into the core by gravitational force. The control rod drive positions are provided on the operator display panel. The ATR uses a pneumatic-electromechanical drive system to allow ejection of a predetermined amount of the transient rod from the core for pulsed operation. For a scram or loss of power, the three-way air supply solenoid is deenergized, relieving the pressure in the cylinder so that the transient rod drops into the core by gravity. A system of limit switches is used to indicate the position of the air cylinder and transient rod. The FTR is driven by a pneumatic cylinder with scram capability by removal of air pressure, but no travel adjustment capability.

Also, in the SAR, the licensee stated that a safety plate is welded to the extension of the inner reflector liner about 16 in (40.6 cm) below the bottom grid plate. This plate prevents the possibility of a control rod from dropping out of the core. The reactor control rod system is designed to help ensure safe reactor control and shutdown under all operating conditions.

TS 5.3.2, Control Rods, states:

Specification(s).

a. The SHIM and REG rods shall be motor driven with scram capability and solid boron compounds in a poison section, with fuel followers of standard TRIGA® fuel meeting the same specifications as in Technical Specification 5.3.3.

b. The ATR transient rod shall be motor and pneumatically driven, have scram capability, and contain solid boron compounds in a poison section. The ATR shall have an adjustable upper travel limit to provide variable pulse insertion capability. The FTR transient rod shall be pneumatically driven and have scram capability, and contain solid boron compounds in a poison section. The ATR and FTR shall incorporate air filled followers.

TS 5.3.2, Specifications a and b, present the design requirements for the shim, regulating, and both transient rods which help ensure that control rods are fabricated to reliably perform their intended control and safety function. TS 5.3.2 requires that all control rods employ a solid boron absorber. TS 5.3.2 also requires that all of the control rods be able to be scrammed. The NRC staff reviewed TS 5.3.2 and finds solid boron has well-established nuclear and material characteristics. The NRC staff also finds that TS 5.3.2 characterizes important design features of the control rods and is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 5.3.2 is acceptable.

TS 3.2.1, Control Rods, states:

> Specification(s). The reactor shall not be operated unless the REG, SHIM, ATR and FTR control rods are operable. Control rods shall not be considered operable if:
>
> a. damage or deterioration is apparent to any rod or drive assembly that could affect operation; or
>
> b. the scram time for any control rod is greater than 1 second for 90% reactivity insertion; or
>
> c. the total reactivity worth of the two transient control rods (ATR and FTR) is greater than $3.00.

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

TS 3.2.1, Specification b, helps ensure that the control rod insertion time is 1 second or less for 90-percent reactivity insertion. The requirement helps ensure a sufficient amount of reactivity insertion is inserted rapidly enough to prevent fuel damage. The NRC staff finds that the 1-second scram insertion time is typical of TRIGA®s and is consistent with the guidance provided in NUREG-1537, Appendix 14.1, Section 3.2(1).

TS 3.2.1, Specification c, helps limit the total worth of the two transient rods to restrict the pulse size to that analyzed in the SAR.

The NRC staff finds that TS 3.2.1 helps ensure that, during the normal operation of the UCINRF, the time required for the control rods to be fully inserted, from the instant that a safety channel variable reaches the safety system setting, is rapid enough to prevent fuel damage. Compliance with this specification helps ensure that the reactor will be promptly shut down when a scram signal is initiated. For the range of transients anticipated for a TRIGA® reactor, the specified scram time is adequate to ensure the safety of the reactor. The FTR is either positioned full-out or full-in for pulsing purposes. On the basis of the information provided above, the NRC staff concludes that TS 3.2.1 supports the design-basis requirements provided in the SAR to prevent fuel damage and is acceptable.

Section 2.5.1 of this SER discusses the reactivity worth of the control rods.

TS 4.1 and TS 4.2 define the surveillance requirements for the measurement and verification of rod worth, performance, and operability of the control rods (discussed in Sections 2.5.2 and 5.4.2 of this SER, respectively).

The NRC staff finds that the control rods conform to the applicable design bases as described in SAR Section 4.3 and can shut down the UCINRF from any operating condition or applicable accident scenario. The control rod design for the UCINRF includes reactivity worths that can control the excess reactivity planned for the UCINRF, including the assurance of an acceptable shutdown reactivity and margin. The licensee has justified appropriate TS design limits, limiting conditions for operation, and surveillance requirements for the control rods. On the basis of the information provided above, the NRC staff concludes that the control rods and applicable TSs are acceptable.

2.2.3 Neutron Moderator and Reflector

The SAR Section 4.4 describes the moderator and reflector. UCINRF uses TRIGA® U-ZrH fuel in which the zirconium hydride acts as the primary moderator. This combination of materials is sufficient to moderate the neutron fission spectra and thermalize neutrons. Because the fuel and moderator are alloyed together, any sudden increase in power heats the fuel and moderator simultaneously, providing the prompt negative temperature coefficient characteristic of the U-ZrH fuel. The water in the pool acts as both a moderator and reflector, in addition to being used as a coolant.

The reactor core is surrounded by a ring-shaped block of graphite reflector encased in a leak-tight, welded aluminum can. The graphite has a radial thickness of about 11.20 in (28.4 cm), with an inside diameter of approximately 20.88 in (53.0 cm) and a height of about 22.5 in (57.15 cm). A well in the top of the reflector provides space for the rotary specimen rack (lazy Susan), which does not penetrate the leak-tight aluminum can. The reflector assembly rests on an aluminum platform about 6.4 ft (2.0 m) from the bottom of the pool.

The fuel elements contain two sections of graphite, each 3.5 in (0.89 cm.) long above and below the fuel, which serve as axial reflectors for the core. The fuel itself provides a significant contribution to moderation of the neutrons through scattering with the fuel matrix. In addition, grid positions not filled by fuel moderator elements and other core components may be filled by graphite moderator reflector elements.

The NRC staff reviewed the information in the SAR and finds that the moderator and reflector elements used in the UCINRF are consistent with other TRIGA® reactors. The NRC staff also reviewed the constituents, materials, and components for the reflector elements and finds that they are in agreement with the description provided in the SAR. On the basis of the information provided above, the NRC staff concludes that there is reasonable assurance that the neutron moderator and reflectors will function safely in the core for the renewal period without adversely affecting public health and safety.

2.2.4 Neutron Startup Source

The SAR Section 4.4.4 describes the facility’s startup source. The neutron source is 3 Ci of americium-beryllium (Am-Be), doubly encapsulated in stainless steel and contained in a cylindrical aluminum holder that can be positioned in either of two diametrically opposite holes in the top grid plate between the F and G rings (a description of the rings is in Section 2.2.5).

The SAR states that the primary function of the neutron source is to provide an adequate neutron-induced signal on the reactor instrumentation before and during startup to show that the reactor power level channels are functioning properly. TS 3.2.3 (discussed in Section 5.3.2.3 of this SER) requires an interlock to prevent control rod withdrawal when the power level is less than or equal to 1×10^{-7} percent of full power. In addition, TS 5.3.1 (discussed in Section 2.2 of this SER) requires that an Am-Be neutron source be provided in a specific location in the core to provide startup neutrons. A neutron source clad failure would be detected during the routine analysis of pool water as required by TS 4.3 (discussed in Section 2.3 of this SER) to periodically measure the radioactive content of the reactor pool water.

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

2.2.5 Core Support Structures

The SAR, Section 4.0, provides the description and dimensions of the core support structure. The top grid plate is an aluminum plate supported by a ring welded to the top inside surface of the reflector container. The bottom grid plate is an aluminum plate that supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. Six adapters are bolted to pads welded to a ring that is, in turn, welded to the reflector container.

The core components are contained between top and bottom aluminum grid plates. The plates have 127 total positions in six concentric rings (labeled B to G) around a central port (used for high-flux irradiations). The SAR states that these holes in the two grid plates provide accurate lateral positioning for core components, including fuel elements, control rods, and experimental facilities. Small holes at various positions in the top grid plate permit insertion of wires and other small devices into the reactor for in-core measuring purposes, and two other holes provide positioning for the neutron source. Several cutouts in the top plate allow for the insertion of larger experimental facilities. Interstitial coolant passages in the bottom core plate provide coolant flow. The arrangement helps ensure a stable and reproducible core configuration. Penetrations in the core plates allow for sufficient coolant flow.

The SAR also states that a safety plate is below the bottom grid plate to prevent the possibility of a control rod from dropping out of the core. A steel bridge, which supports the rod drives and the loading and drive mechanisms for the rotary specimen rack, spans the reactor pool. The bridge will support the weight of a 3,000-lb (1,361-kg) fuel element cask. The portions of the pool not covered by the bridge are covered by aluminum grates that can support the weight of people. The SAR indicates that the bridge and core support structure have maintained their structural integrity and that visual inspections during reactor core and experiment changes will be sufficient to recognize significant degradation.

The NRC staff reviewed the information provided in the SAR as described above and finds that the reactor core components are typical of TRIGA® reactors. These reactor core components are capable of positioning and aligning the fuel elements for all anticipated operating conditions and will provide adequate coolant flow to each fuel element. On the basis of its review of the information provided in the SAR, the NRC staff concludes that there is reasonable assurance that the reactor bridge core support structure will function safely for the renewal period without adversely affecting public health and safety.

2.3 Reactor Tank or Pool

The SAR, Sections 3.4.1 and 4.1, and request for additional information (RAI) responses (Refs. 7, 8, 12, and 64), describe the reactor pool. The reactor pool is formed by an aluminum tank liner supported by reinforced concrete. During construction, the aluminum tank was double-wrapped with hot tarred felt to provide a water barrier between the aluminum liner and the surrounding concrete. The tank and concrete shield were constructed to GA design specifications. The tank was subjected to quality control testing, including radiography of 20 percent of the welds and leak testing; additional leak testing was performed after installation at the reactor facility. There are no penetrations in the reactor tank; all piping for coolant, pneumatic transfer tubes, and experimental facilities accesses the pool from the top. The pool water level is normally maintained more than 15 ft (4.57 m) above the top of the reactor core. The pool water level is continuously monitored by a water-level monitoring device that provides an alarm at the reactor console and a monitored remote location when water level drops to 1 ft (0.30 m) below the tank edge. There is no automatic makeup of water loss from the pool; thus, manual makeup of water is required. The amount of makeup water is recorded on a daily startup reactor checklist. The licensee stated that, to detect deviations from expected water loss from evaporation, records are examined on each occasion when makeup water is added to the pool. The TS design features establish the basic requirements for the reactor coolant system and pool water.

TS 5.2, Reactor Coolant System, states:

> Specification(s).
>
> a. The reactor core shall be cooled by natural convection water flow.
>
> b. Inlet and outlet pipes that lead to the heat exchanger or demineralizer shall be equipped with siphon breaks not less than 14 feet above the upper core grid plate.
>
> c. A pool water level indication shall be provided at the control console with an alarm at the control console and an alarm to a central monitoring station.
>
> d. A pool water temperature indication shall be provided at the control console.
>
> e. A pool water conductivity measurement instrument shall be provided in the reactor room.
>
> f. Gamma and beta radiation spectrometry equipment shall be provided for water sample radioactivity assay.

TS 5.2, Specification a, helps ensure proper cooling to the core. The NRC staff reviewed the information provided in the SAR and finds that the core can be adequately cooled by natural convection flow without the need for forced cooling which is in agreement with GA design assumptions.

TS 5.2, Specification b, helps ensure that water cannot be drained by a failure that creates a siphon. TS 5.2, Specification b also helps ensure an adequate quantity of water for cooling the core and provide radiation shielding.

TS 5.2, Specification c, helps ensure operators are informed of water loss in the event of a leakage path developing that could drain the reactor pool. TS 5.2, Specification c, also helps limit the consequences of a loss-of-coolant accident.

TS 5.2, Specifications d through f, help ensure pool water quality by providing instrumentation and requirements for measuring pool water temperature, conductivity, and radioactivity assays. TS 5.2, Specifications d through f, also help ensure the pool water is capable of performing its stated functions.

The NRC staff reviewed information provided in the SAR and finds that TS 5.2 will help ensure the coolant system remains consistent with the assumptions and design bases as described in the SAR. The NRC staff also finds that TS 5.2 characterizes important design features of the UCINRF and is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that the design of the reactor tank and TS 5.2 are acceptable.

TS 3.3.1, Pool Water Level, states:

> Specification(s).
>
> a. The reactor shall not be operated unless the pool water level is at least 24 feet above the tank floor (1 foot below the tank edge).
>
> b. An audible alarm, with reporting to a monitored remote location if not locally silenced by an operator, shall operate continuously to alert personnel if the water level in the reactor pool falls below the above limit. Visual checking of water level shall be substituted every 10 hours during periods when the alarm is found to be inoperable and no substitute level device has been implemented.
>
> c. Records shall be maintained of the date, time and quantity of all make up water added to the pool.

TS 3.3.1, Specification a, specifies the minimum amount of water above the reactor core to help ensure the reactor core remains covered with water, provides radiological shielding to UCINRF staff, and sets a limit for pool water level as an aid in detecting loss of pool water from evaporation or pool leakage. The NRC staff finds that this water level is consistent with the level assumed in the thermal-hydraulic analysis (discussed in Section 2.6 of this SER).

TS 3.3.1, Specification b, helps ensure UCINRF staff are aware of a loss-of-pool-water level by requiring an audible alarm indication both locally and at a continuously monitored remote location (which can be silenced by an operator if the local alarm is attended) if the water level drops below the tank edge by 1 ft (0.30 m). The NRC staff reviewed TS 3.3.1, Specification b, and finds it acceptable.

TS 3.3.1, Specification c, helps ensure water inventory tracking by requiring maintenance of records for pool water additions. The NRC staff reviewed TS 3.3.1, Specification c, and finds that these records provide information to the operating staff to assist in determining whether the water loss is by evaporation or pool leakage.

TS 3.3.2, Pool Water Temperature, states:

> Specification(s). The pool water temperature shall be maintained between 17 °C and 25 °C

TS 3.3.2 helps ensure that the bulk water temperature limit is maintained to preserve the assumptions made in the thermal-hydraulic analysis and to prevent the breakdown of resins important to water chemistry. The lower temperature limit is maintained to minimize pool stress from overcooling. The NRC staff reviewed TS 3.3.2 and finds that this TS is consistent with the assumptions used in the thermal-hydraulic analysis and is supported by information provided by the licensee in an RAI response (Ref. 11).

TS 3.3.3, Pool Water Conductivity, states:

> Specification(s). The pool water conductivity level shall be maintained less than 3 micromhos/cm. Make-up water shall meet this specification before being added to the pool.

TS 3.3.4, Pool Water Radioactivity, states:

> Specification(s). The average pool water radioactivity level shall be maintained within limits for sewer disposal as established by 10 CFR § 20, Appendix B, Table 3 for radionuclides with half-lives longer than 24 hours.

The NRC staff reviewed TS 3.3.4 and finds that TS 3.3.4 follows the guidance provided in NUREG-1537, Section 5.2, and that the average pool water radioactivity value is limited by the values in 10 CFR Part 20, Appendix B, Table 2, are consistent with radiological conditions in the reactor coolant of other TRIGA® reactors.

The NRC staff completed an analysis (Ref. 47) that demonstrated that a conductivity limit no greater than 5 μmho/cm will ensure that the pH range is limited to 5.6 to 5.8, which is consistent with the guidance provided in NUREG-1537 to maintain the pH range of 5.0 to 7.5. Because the licensee chose a conductivity limit of 3 μmho/cm in TS 3.3.3, there was no need for a TS requirement to limit the reactor pool water pH.

The NRC staff reviewed TS 5.2, TS 3.3.1, TS 3.3.2, TS 3.3.3, and TS 3.3.4 and finds that these TSs help ensure water quality, minimize corrosion of facility components in contact with coolant, allow for early identification of pool water leakage, and control the radioactive content of the pool water. The NRC staff also finds that the requirements set forth in these TSs are consistent with the guidance provided in NUREG-1537. Based on the information provided above, the NRC staff concludes that TS 5.2, TS 3.3.1, TS 3.3.2, TS 3.3.3, and TS 3.3.4 are acceptable.

Pool Water Leakage

The licensee has reported that ground water from rain and irrigation of areas outside the reactor building has been observed in fuel element storage pits in the floor of the reactor room. These pits were designated for either wet or dry storage of fuel elements. Any fuel stored in these pits is secured in racks designed to limit reactivity and raise the elements above any potential ground water levels. Two of the total of five pits have been subjected to water incursion. The licensee stated that these two storage pits would be used only as a last resort. Because the reactor pool is in-ground, the licensee has considered the potential effect of this ground water incursion on the reactor pool liner (Refs. 7, 8, 12, and 64).

The reactor tank, described in the SAR Section 4.0 and RAI responses (Ref. 12), was installed in an excavated pit and, before backfilling, was double-wrapped with hot tarred felt to protect the aluminum pool liner from its environment. Thick concrete surrounds the in-ground portions of the tank.

The licensee stated that, to monitor and reduce the accumulation of water around the reactor pool, a well with an automatic sump pump was drilled in the reactor room floor, extending to 4 ft (1.22 m) below the bottom of the concrete tank (about 8.5 ft (2.59 m) below the aluminum pool tank liner). Water that may accumulate in this well is pumped by an automatic system, which provides an indication if water accumulating in the well is not pumped out. If water is present, samples are taken monthly of the water to determine whether there is any radioactivity in the water. If radioactivity is present, the samples are used to determine if the radioactive content is within 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” Table 3, limits. Pump operation would be secured if the limits were to exceed those in 10 CFR Part 20, Appendix B, Table 3 (Ref. 64). Furthermore, as required by TS 3.3, the water level in the pool is monitored to check for potential leakage, and sampling of pool water for radioactivity is required to ensure radioactivity levels are below the 10 CFR Part 20, Appendix B, Table 3, limits and to ensure any water accumulating in the wells is also below these limits.

The licensee also stated that no radioactivity has been detected in these samples, which indicates that there has been no leakage from the tank. The SAR states that no evidence of corrosion of the reactor pool tank from the outside has been detected, as evidenced by the lack of pool water radioactivity detected in the monitoring wells. The licensee attributed this to the techniques used in construction to protect the tank liner from its environment.

The pool water level alarm is used in determining if the pool is leaking primary coolant water to the environment. The pool water level is not automatically corrected to make up for evaporative pool water loss. The level monitor reports the level in approximately 0.469-in (1.19-cm) increments. For the pool, 0.984 in (2.50 cm) corresponds to 80 gal (302.83 L) of water. The correct level is maintained by manual means. Because the temperature of the pool water remains between approximately 18 °C and 24 °C (64 °F and 75 °F, respectively), the average value of relative humidity in the area is relatively low. The local and central alarm level is set at about 12 in (30.48 cm) below the pool rim, corresponding to a water loss of 480 gal (1,820 L), which is less than 2 percent of total pool capacity. The alert level is set at about 2 in (5.08 cm) above that level and indicates that water should be added to preserve dashpot action on the REG and SHIM rods. Records maintained of pool water makeup would alert operating staff of abnormal water loss. The licensee stated (Ref. 64) that a loss rate of greater than 1-in (2.54-cm) decrease in pool level, or approximately 80 gal (302.83 L), in a week, is considered an unusual drop and could indicate pool leakage. The licensee stated that an unusual drop in water level is procedurally required to be reported and investigated.

The NRC staff reviewed the information provided as described above and finds that the method of treating the potential ingress of ground water is acceptable since (1) the system installed will prevent ground water from reaching the reactor pool and will be monitored for effectiveness, and (2) the reactor pool will be monitored for pool leakage and radioactivity content.

Based on the information provided above and a history of acceptable tank performance, the NRC staff concludes that the reactor pool design has acceptable features that will minimize the potential for a loss of integrity, which could lead to a loss of coolant or other malfunction (Ref. 7). In addition, the NRC staff concludes that the pool water level instrumentation and the water quality requirements are adequate to help ensure that the water level above the core upper grid plate is at least 15 ft (4.57 m) during operation and that the water quality standards are appropriate. The pool water level is monitored, and the UCINRF operating staff would investigate leakage.

The NRC staff reviewed the UCI chill water system heat exchanger and finds that the release of reactor pool coolant to the UCI chilled water system is not anticipated because the UCI chilled water (secondary cooling of the heat exchanger shell-side) system pressure is monitored and maintained at a higher pressure than the reactor pool cooling (primary cooling of the heat exchanger tube-side). On this basis, the NRC staff concludes that the likelihood of a significant release to the environment resulting from pool leakage is extremely low. If a limited amount of primary coolant were released to the environment by a pool leak, the requirement in TS 3.3.4 to maintain primary coolant radioactivity levels below 10 CFR Part 20 limits for release to the environment would help ensure doses to the public would be within acceptable limits.

2.4 Biological Shield

In SAR Section 4.0, the licensee stated that the reactor is in a belowground aluminum-lined tank surrounded by reinforced concrete and earthen backfill. There are no penetrations (e.g., experimental beam ports) in the pool walls below the water level; therefore, the only potential for radiation exposure is at the location above the reactor pool.

Shielding above the reactor is provided by the pool water, which is normally maintained at 15 ft (4.57 m) above the top grid plate, with alarms as specified in TS 3.3.1, "Pool Water Level," (discussed in Section 2.3 of this SER). Chapter 6 of the SAR (Ref. 20) specifies that the measured dose rate directly above the core at the pool surface during full-power (250-kWt) operation is 2.0 mrad/hr (equivalent to 2.0 mrem/hr). Additionally, the ceiling slab directly above the reactor was constructed of 17 in (43.18 cm) of reinforced concrete to minimize the dose to individuals in portions of the building above the reactor room. Safety analysis report Chapter 6 also describes the approximate dose rate at the floor of the room above as less than 0.1 mrad/hr (0.1 mrem/hr) during full-power reactor operation.

The NRC staff inspection program routinely reviews the licensee's radiation protection program, including routine independent verification of radiation levels in the facility as well as observation of radiation measurements obtained by the licensee's operating staff.

Based on a review of the information provided in the SAR and results from the NRC staff inspection program, the NRC staff finds that the biological shield components are typical of TRIGA® reactors and consistent with the guidance provided in NUREG-1537, Appendix 4.1. On this basis, the NRC staff concludes that there is reasonable assurance that, during the renewal period, the biological shield will limit exposures from the reactor and reactor-related sources of radiation so that the limits of 10 CFR Part 20 will not be exceeded.

2.5 Nuclear Design

The reactor design bases, as described in SAR Section 4.5 and RAI responses (Refs. 7, 9, 11, and 64), are established by the maximum operational capability for the fuel elements and fuel element configurations. The TRIGA® reactor system has five major areas that define the reactor design bases:

(1) fuel temperature
(2) prompt fuel temperature coefficient
(3) control rod worths
(4) thermal-hydraulics and heat transfer (pool water temperature)
(5) reactor power

The safety limit for the UCINRF is based on the fuel temperature, which, because of the strongly negative temperature coefficient of reactivity of the TRIGA® fuel, contributes to the inherent safety of the TRIGA® reactor. A limit on reactor power ensures operation within the SAR design analysis as well as below the fuel temperature SL and pool water temperature limits. The information discussed in this section establishes the design bases for the content of other chapters in this SER.

2.5.1 Normal Operating Conditions

Steady-State Operation

The UCINRF is licensed to operate at a steady-state maximum power level of 250 kWt. The thermal-hydraulic calculations (Ref. 11) show by analysis that operation at a steady-state power level of 250 kWt corresponds to a peak fuel temperature of 245 °C (473 °F) with a departure from nucleate boiling ratio (DNBR) greater than 7.27. In addition, the licensee stated that experience has shown that operation of TRIGA®reactors at a thermal power level of 2.3 MWt will not result in damage to the fuel. Reactors of this type have operated successfully for many years at power levels up to 2.3 MWt.

TS 3.1.1, Steady-State Operation, states:

> Specification(s). The reactor power level in steady-state operation shall not exceed 250 kilowatts.

TS 3.1.1 specifies a steady-state power level of 250 kWt to help ensure that adequate cooling is provided for the fuel rods by natural convection of pool water. The licensee performed thermal-hydraulic calculations for the UCINRF (Ref. 9), which indicate that, even at a power level of 300 kWt and inlet coolant temperature of 25 °C (77 °F), the maximum fuel element temperature is 253 °C (488 °F) and the minimum DNBR is 6.67. NUREG-1537, Part 1, Appendix 14.1, recommends a minimum DNBR of 2.0. The NRC staff reviewed the licensee’s analysis, compared the results against TRIGA®s of similar design and power levels, and finds the licensee’s analysis acceptable. The NRC staff also finds the stated maximum fuel temperatures acceptable and consistent with other TRIGA®-designed reactors. Also, as discussed in Section 2.5.3 of this SER, the peak fuel temperature is significantly less than the fuel temperature SL of 1,000 °C (1,832 °F). On the basis described above, the NRC staff concludes that the requirement in TS 3.1.1 that reactor power level not exceed 250 kWt during steady-state operation is acceptable.

Pulse Mode Operation

The UCINRF 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), and the pulse may not be initiated from a core power in excess of 1 kWt. Pulsing from a power level greater than 1 kWt is prevented by a required interlock, which prevents the transient rods from firing. TS 3.1.4 limits the maximum possible reactivity insertion at $3.00, which has been shown to produce a peak fuel temperature in the reactor of 400 °C (752 °F), well below the limit of 830 °C (1,526 °F). The fuel temperature limit of 830 °C (1,526 °F) during pulsing is recommended by GA to help ensure that no fuel damage occurs because of internal pressure caused by H migration (Ref. 36). The limit of 830 °C (1,526 °F) was based on an evaluation of the fuel damage experience at the Texas A&M University TRIGA® reactor. In that instance, the damage was limited to swelling of the fuel, but the cladding maintained its integrity.

TS 3.1.4, Pulse Mode Operation, states:

> Specification(s). The reactor shall not be operated in the pulse mode unless in addition to the other requirements of Technical Specification 3.1.2 and 3.1.3:
>
> a. the steady-state power level of the reactor is less than 1 kilowatt; and
>
> b. the total reactivity worth of the two transient control rods (ATR + FTR) is measured to not exceed $3.00.

TS 3.1.4, Specification a, which establishes the maximum power level from which a pulse may be initiated, helps ensure the pulse will be initiated from a low fuel temperature so that the maximum fuel temperature during the pulse does not exceed established analyses in the SAR with sufficient margin.

TS 3.1.4, Specification b, which limits the combined worth of the two transient rods, helps ensure the reactivity addition for any pulse is limited to $3.00 and that the assumptions for the analyzed condition in the SAR are not exceeded.

The NRC staff reviewed the information in the SAR, as supplemented by responses to RAI, and finds that the fuel temperature rise during a pulse transient has been calculated conservatively using an adiabatic model. Insertion with the power level below 1 kWt helps ensure that the starting temperature for a pulse is below 25 °C (77 °F). The temperature rise from a $3.00 reactivity insertion pulse is thus calculated to bring the peak fuel temperature to less than 400 °C (752 °F), well below the SL and well below the maximum fuel temperature limit for pulsing of 830 °C (1,526 °F) recommended by GA, the fuel designer.

The NRC staff also finds that the design and functional description of the transient rod system as presented in the licensee's SAR provides reasonable assurance that pulses will be limited to values that maintain fuel integrity as determined by the analyses in SAR Chapter 13 (Ref. 20). In addition, TS 3.2.3 requires an interlock to be operable that prevents pulse operation unless the power level is less than 1 kWt, as well as two other interlocks to prevent pulses in the steady-state mode of operation. One interlock prevents application of air to the FTR unless all other rods are fully inserted; the second prevents application of air to the ATR unless the air cylinder is fully down. In SAR Section 13.3 (Ref. 20), the licensee analyzed the effect of initiating a pulse when the reactor is operating at full power. The calculation shows that, from an initial full-power level of 250 kWt and fuel temperature of 240 °C (464 °F) in the B-ring, a hypothetical pulse of $3.00 results in a maximum fuel temperature of 570 °C (1,058 °F), again below the limit of 830 °C (1,526 °F). By this analysis, the licensee demonstrated that, even with a failure of the 1-kWt interlock combined with operator error, this reactivity insertion accident would result in a fuel temperature well below the temperature limit for pulsing and the SL.

On the basis of its review, the NRC staff concludes that TS 3.1.4 is acceptable as the peak fuel temperature limit and the associated maximum reactivity addition limit for pulsing helps to ensure that the reactor can be safely pulsed with no fuel damage.

Core Configuration

The reactivity values for various core components provided in the licensee's SAR (Ref. 20) are a combination of measured values and estimated values based on measurements made on other TRIGA® reactors of the same generic type. Estimates of the reactivity effects of flooding or voiding of all experimental facilities are presented in the SAR (Ref. 20), as well as the effects of fuel versus water for the central grid hole and three-element triangular cutout facilities. The bounding reactivity worth change for flooding or voiding any of the experimental facilities is well within the $3.00 limitation on experiment reactivity worth required by TS 3.8.1, "Reactivity Limits," for a single secured experiment. Accident analysis in SAR Chapter 13 demonstrates that a worst-case hypothetical pulse of $3.00 from an initial full-power level of 250 kWt would not exceed the fuel temperature SL. Estimated fuel element worths compared with water are presented as a function of reactor ring location, with the highest worth of $1.16 (0.81 percent Δk/k) in the B-ring grid position. A curve of the estimated reactivity loss as a function of power shows that, based on the power coefficient of reactivity, full-power operation at 250 kWt results in a reactivity loss of approximately $1.50.

The NRC staff reviewed the information in the SAR, as supplemented by responses to RAI, and concludes that the possible core configurations available for experiments would help ensure that fuel failure would not occur.

Control Rod Worths

The control rod worths for each of the four rods are presented as measured values in RAI response dated January 9, 2015 (Ref. 67). The measured total reactivity worth of the four rods is given as $8.93, with the highest-worth rod being the shim rod, worth $3.52. The two transient rods together are worth $2.41, which is below the limit of $3.00 specified in TS 3.1.4. The licensee provided Monte Carlo N-Particle eXtended (MCNPX) computer code calculated values for rod worth, which agreed within acceptability limits with the measured values. The measured and calculated rod reactivity worth agreed within 6 percent. The calculated total reactivity worth of the four rods was $9.52, with the highest-worth rod being the shim rod, worth $3.85. The calculated reactivity worth of the two transient rods together was $2.52.

An interlock prevents simultaneous withdrawal of the regulating, shim, and transient rods. During pulsing, one or both transient rods together are quickly ejected from the core, while an interlock prevents movement of the regulating and shim rods while in pulse mode. Control rod movement is, therefore, controlled by interlock. An insertion of excess reactivity (failure of interlock) accident is analyzed in Section 4.1.2 of this SER.

Based on its review and consideration of the factors described above, and the MCNPX simulation confidence interval, the NRC staff finds that the calculated rod reactivity worth compare acceptably with the measured values.

Excess Reactivity

The maximum reactivity in the reference core condition is required to be less than 2.1 percent Δk/k, equivalent to $3.00. The excess reactivity, reported in the SAR and RAI response (Ref. 10), was $2.66 measured and $2.82 calculated by the MCNPX core model.

TS 3.1.3, Core Excess Reactivity, states:

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

TS 3.1.3 establishes a limit on excess reactivity, allowing operational flexibility while limiting the reactivity available for reactivity addition accidents. The maximum excess reactivity helps establish a basis for ensuring that an adequate shutdown margin (SDM) is available by control rod insertion. The licensee stated that, because the fuel has no burnable poisons, the excess reactivity of the core is reduced by reactor operation. The licensee provided data in the 2015 UCINRF Annual Report (Ref. 67) that indicate a measured value of core excess of $2.87, which results in an SDM of $2.54. These values are supported by licensee analysis in the SAR and RAI response (Ref. 10).

The NRC staff reviewed the information in SAR Section 1.2, as supplemented by responses to RAI, and finds that the licensee has selected the maximum excess reactivity in TS 3.1.3 that will allow the reactor to operate in accordance with the design features provided in the SAR, while allowing for the operational flexibility to conduct various experiments. The NRC staff also finds that the reported values are in general agreement and consistent with the guidance provided in NUREG-1537. On the basis of the information provided above, the NRC staff concludes that TS 3.1.3 is acceptable.

Shutdown Margin

TS 3.1.2 establishes the limit on SDM to be $0.55. The total reactivity worth of the four rods, as measured and presented in UCINRF 2015 Annual Report (Ref. 67), is $8.93, with the highest rod worth equal to $3.52. For a core with the TS maximum excess reactivity and the highest worth rod stuck in the full-out position, the SDM calculated by the NRC staff is $2.41, well in excess of the TS requirement of $0.55. The SDM is based on the “stuck rod” criterion found in the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff evaluated the SDM requirements for the reactor and finds them to be comparable to requirements for similar reactors and sufficient to allow the reactor to be shut down safely and remain subcritical without risk of fuel damage.

TS 3.1.2, Shutdown Margin, states:

> Specification(s). The reactor shall not be operated unless the shutdown margin provided by the control rods is greater than $0.55 with:
>
> a. irradiation facilities and experiments in place and the total worth of all unsecured experiments in their most reactive state; and
>
> b. the most reactive control rod fully withdrawn; and
>
> c. the reactor in the reference core condition.

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

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

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

The NRC staff reviewed the analysis presented in SAR Section 6.2 (Ref. 20), RAI responses (Refs. 10 and 11), and the licensee's annual report (Ref. 67). The NRC staff finds that the licensee has used input parameters justified by analysis presented in the SAR and further justified by comparison with the known reactor characteristics. The NRC staff also finds that the licensee has adequately analyzed the reactivity effects of individual core components (Ref. 10). The TS related to the normal operating conditions of the reactor core include limits on excess reactivity, the minimum SDM, allowable core configurations, and surveillance requirements for the core reactivity parameters and reactivity worth of the control rods (Ref. 76). These TSs are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that the analysis presented in the SAR and in response to RAI adequately justifies these TSs and shows that normal reactor operation will not lead to the release of fission products from the fuel.

Based on the information provided above, the NRC staff concludes that the licensee has adequately analyzed expected normal reactor operation during the period of the renewed license and that TS 3.1.2 is acceptable.

2.5.2 Reactor Core Physics Parameters

The UCINRF staff performed three-dimensional calculations for the reactor physics parameters (Ref. 10). The licensee, with assistance from GA, used Monte Carlo codes to conduct its nuclear analysis. The calculations produced operational parameters that were consistent with the actual measured values from the operations conducted at the UCINRF.

The basic core physics parameters are presented in RAI responses (Ref. 10) for the reactor as follows:

Prompt Neutron Lifetime

The prompt neutron lifetime was computed by the inverse velocity (1/v) absorber method to be 98.5 μs.

Effective Delayed Neutron Fraction

The effective delayed neutron was computed from Monte Carlo calculations to be 0.0079.

Prompt Fuel Temperature Coefficient of Reactivity

The prompt fuel temperature coefficient of reactivity was calculated over a range of fuel temperature from 20 °C (68 °F) to 927 °C (1,700 °F). The temperature coefficient of reactivity ranged from −7.0E−05 to −1.11E−04 Δk/k per ° C.

Coolant Void Coefficient

The overall estimated coolant void coefficient worth is negative $0.06 per 1 percent water void.

Moderator Coefficient

The moderator coefficient, calculated over a temperature range from 24 °C (75 °F) to 727 °C (1,340 °F), varies from −8.8E−05 to −3.7E−05 Δk/k per ° C.

Power Peaking

The power produced in each fuel rod was calculated establishing a reactor radial power peaking map. The maximum fuel element power with the reactor at a 250-kWt power level is 4.519 kWt while the average fuel element power is 3.125 kWt. From these data, the hottest rod radial power factor is calculated to be 1.446 (4.519 kWt/3.125 kWt) which can be found in the fuel element in core position C6. The product of the radial peaking factor and the axial peaking factor (1.352) gives a maximum power peaking factor of 1.955 (Ref. 10).

TS 4.1, Reactor Core Parameters, states:

Specification(s).

a. The total reactivity worth of each control rod shall be measured annually or following any significant change (>$0.25) in core configuration.

b. The core excess reactivity shall be determined using control rod position data prior to each day's operation, or prior to each operation extending more than one day, or following any significant change (>$0.25) in core configuration.

c. The shutdown margin shall be determined at each day's startup, or following any significant change (>$0.25) in core configuration.

d. All core fuel elements shall be inspected (under water) to assure compliance with Technical Specifications 3.1.6.a through 3.1.6.d quinquennially, but at intervals separated by not more than 500 pulses of magnitude greater than $1.00 of inserted reactivity. Fuel follower control rods shall be visually inspected and measured for bend at the same time interval. Such surveillance shall also be performed for elements in the B and C rings in the event that there is indication that fuel temperatures greater than the limiting safety system setting on temperature may have been exceeded.

e. Prior to resumption of routine pulse mode operations following a period of no pulse mode operations for more than 1 year, a test of pulsing performance with a pulse insertion of $1.50 shall be performed to assure pulsing power and fuel temperature response is as predicted from prior experience.

f. Full core, fuel and control rod surveillance shall be conducted before further reactor operation if significant changes are observed in any measured parameters such that it could be concluded that the fuel element or control rod integrity has been compromised or fuel element or control rod damage has occurred.

TS 4.1, Specification a, provides an interval for when total reactivity worth for individual control rods should be calculated to help ensure viability of the control rods.

TS 4.1, Specification b, helps ensure the determination of core excess reactivity as required to support TS 3.1.3. TS 4.1, Specifications a and b, establish the requirement for ensuring that changes to the core are subject to limitation of the LCC defined in the SAR and RAI responses.

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

TS 4.1, Specification d, addresses inspection of the fuel to detect gross failure or visual deterioration. The guidance in NUREG-1537 states that this could be accomplished by inspecting 20 percent of the fuel per year over 5 years. The licensee performs a single inspection of all the fuel every 5 years. The NRC staff previously evaluated and approved a license amendment request (SER for Amendment 5, dated November 8, 2000) (Ref. 35) changing the periodicity for this surveillance from 36 months to 60 months (i.e., quinquennially).

TS 4.1, Specification e, provides a requirement for pulsing operations, when pulsing operations are not performed for an extended period of time, to help ensure proper pulsing performance.

TS 4.1, Specification f, provides trigger events for the inspection of all fuel elements and control rods if significant changes are detected in such reactor parameters as pool water conductivity, pool water radioactivity, or control rod drop times.

The staff reviewed TS 4.1 and finds that these surveillance intervals are acceptable to detect changes in core behavior and are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that these surveillances will help ensure that the SDM and core excess reactivity are within their TS limits and concludes that TS 4.1 is acceptable.

The NRC staff reviewed TS 5.3.1, TS 3.1.2, TS 3.1.3, and TS 3.1.7, related to the normal operating conditions of the reactor core, including limits on the steady-state operating power level, allowable core configuration, SDM, and excess reactivity. TS 4.1 relates to the corresponding surveillance requirements and control rod reactivity worth. The NRC staff finds that the analyses presented in the SAR and RAI responses (Refs. 10 and 17) justify these TSs and that normal operation will not lead to the release of fission products from the fuel. The NRC staff also finds that the applicant has considered appropriate core physics parameters that are similar to those found acceptable at other NRC-licensed TRIGA® reactors. Based on these considerations, the NRC staff concludes that the licensee has adequately analyzed the core for expected operation, and the values of the core physics parameters are, therefore, acceptable.

2.5.3 Operating Limits

The provisions of in 10 CFR 50.36(c)(1) require reactor licensees to specify SLs and LSSSs in the TSs. The regulation defines SLs as limits upon important process variables necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity. LSSSs for nuclear reactors are defined as settings for automatic protective devices related to those variables that have significant-safety functions. Where an LSSS is given for a variable on which an SL is placed, the setting must be chosen so that automatic protective actions will correct an abnormal situation before an SL is exceeded.

The principal physical barrier to the release of fission products for TRIGA® reactors is the fuel element cladding, and the most important parameter to maintain the fuel cladding integrity is the fuel element temperature. A loss in the integrity of the fuel rod cladding may occur if there is a buildup of excessive pressure between the fuel-moderator and the cladding. This excessive pressure occurs if the fuel temperature exceeds the SL. The presence of air, fission product gases, and H from the dissociation of the H and Zr in the fuel moderator at elevated temperatures causes such pressure. The fuel temperature and the ratio of H to Zr in the alloy determine the magnitude of this pressure.

The SL for the stainless-steel-cladded, high-hydride TRIGA® fuel is based primarily on experimental evidence obtained during high-performance reactor tests on this fuel. These data indicate that the stress in the cladding caused by H pressure from the dissociation of ZrH and from other gases will remain below the cladding stress limit. The TSs establish a value of 1,000 °C (1,832 °F) as the SL. Based on the theoretical and experimental evidence (Ref. 34), the NRC staff concludes that this SL represents a conservative value to provide confidence that the integrity of the fuel elements will be maintained and that no damage to cladding will occur.

The licensee’s TS include an LSSS to help ensure that there is a considerable margin of safety before the SL specified above is reached.

TS 2.1, Safety Limit—Fuel Element Temperature, states:

> <u>Specification(s).</u> The temperature in a stainless steel clad, high hydride fuel element shall not exceed 1000 °C under any condition of operation.

TS 2.1 specifies the SL for the fuel elements used in the UCINRF. The licensee stated that an important parameter for a TRIGA® reactor is the fuel rod temperature. This is well-suited to be an SL specification because fuel temperature can be measured using an IFE. NUREG-1282 (Ref. 38) identifies the SL for TRIGA® fuel elements with stainless-steel cladding based on the stress to the cladding that results from H pressure from the dissociation of the ZrH. This stress will remain below the yield strength of the stainless-steel cladding if the fuel temperature is below 1,150 °C (2,102 °F) when the fuel cladding temperature is below 500 °C (932 °F) (Ref. 34). During operation, fission product gases and dissociation of the H and Zr build up a gas inventory in internal components and spaces of the fuel elements. Limiting the maximum fuel temperature prevents an excessive internal pressure that heating the gases could generate. The Zr to H nominal ratio of 1.65 (TS 5.3.3, Specification b) influences many attributes of fuel behavior, including internal pressure produced by H. Fuel growth and deformation can occur during normal operation, as described in GA-4314 (Ref. 34). The UCINRF TS 2.1 limit of 1,000 °C (1,832 °F) satisfies the established limit of 1,150 °C (2,102 °F) and ensures that any stress created by internal pressure generated from heated gases related to a Zr-to-H ratio of

1.65 remains below the yield strength of the stainless-steel cladding. The UCINRF SL is set at the temperature established in NUREG-2387.

The NRC staff finds that TS 2.1 is consistent with the guidance provided in NUREG-1537, Appendix 14.1, Section 2.1, and supported by research conducted by the fuel manufacturer. On the basis of the information provided above, the NRC staff concludes that TS 2.1 is acceptable.

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 cladding failure of the fuel. A power level limit is calculated that ensures that the fuel temperature SL will not be exceeded and that film boiling will not occur. As discussed in Section 2.6 of this SER, the design-basis analysis has shown that, for operation at 250 kWt, across a broad range of core and coolant inlet temperatures with natural convection flow, film boiling will not occur.

TS 2.2, Limiting Safety System Settings, states:

> Specification(s). For a core composed entirely of stainless steel clad, high hydride fuel elements, a limiting safety system setting applies to the standard instrumented fuel element (IFE) which shall be located in the B- or C-ring as indicated in the following table:

Location	Limiting Safety System Setting
Core lattice positions B2, B4, C5, C6, or C7	≤425 °C

TS 2.2 specifies the acceptable locations and the temperature limit for the IFE. The licensee's calculations have shown (Ref. 12) that the grid locations selected for the IFE represent the locations of highest power density. The 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 LSSS of 425 °C (797 °F) protects the fuel element cladding at the hottest point in the core. Exceeding this temperature will cause a reactor scram and protect the fuel temperature from exceeding the SL in TS 2.1. The NRC staff finds that limiting the temperature in these locations to 425 °C (797 °F) will ensure that the temperature SL is not exceeded even when allowing for measurement and analytical uncertainties. On this basis, the NRC staff concludes that TS 2.2 is acceptable.

The NRC staff finds that the UCINRF nuclear design analysis described in the SAR and RAI responses is typical of TRIGA® reactors and is properly controlled and implemented in the applicable TSs. The NRC staff also finds that the fuel temperature SL and LSSS for the UCINRF are based on acceptable analytical and experimental investigations that are consistent with those approved by the NRC staff and used at other TRIGA®-type reactors. On the basis of the information provided above, the NRC staff concludes that the UCINRF nuclear design and applicable TSs are acceptable.

2.6 Thermal-Hydraulic Design

The important parameter in the thermal-hydraulic design of a reactor is the maximum fuel temperature, which may be influenced by critical heat flux (CHF), the heat flux associated with DNB. The parameter of interest is the DNBR, which is the ratio of the CHF to the maximum heat flux at full power. The guidance in NUREG-1537, Section 4.6, states that a DNBR greater than or equal to 2.0 is acceptable.

The UCINRF operates at a steady-state operating power of 250 kWt. The limiting parameter necessary to guarantee structural integrity of the fuel is maintaining peak fuel temperature less than 1,000 °C (1,832 °F), as specified in the SL (TS 2.1). Above this temperature, H pressure within the fuel rod may reach a level that could compromise the fuel cladding integrity.

Steady-State Operation

SAR Section 4.5.2 and RAI response (Ref. 11) present the thermal-hydraulic design of the UCINRF. For steady-state operation, the thermal-hydraulics analysis determined the hot element maximum fuel temperature, hottest element maximum clad temperature, core average temperature, hot element outlet flow temperature, average element outlet flow temperature, hot element flow rate, and average element flow rate.

The fuel elements are cooled by natural convection of pool water. A natural circulation flow rate is established, balancing the driving head against the core entrance and exit pressure losses and frictional, acceleration, and hydrostatic head losses in the core flow channels. A thermal evaluation for the reactor core operating with natural convection water flow through the core was made for the reactor operating at 100 kWt, 250 kWt, 275 kWt, and 300 kWt, with a reactor pool temperature of 25 °C (77 °F).

The RELAP5-Mod 3.3 computer code was used to calculate the steady-state natural convection flow through a vertical water coolant channel adjacent to the fuel element heat source and the radial heat fluxes through and from the fuel element to the natural convection flow at discrete axial points of the fuel element. The code determines the clad, fuel, and fuel centerline temperature and the axial distribution of the natural convection flow. The SAR and RAI response (Ref. 11) present details of the application of the code to the reactor core.

The RELAP5-Mod 3.3 code is widely used in the determination of thermal-hydraulic characteristics of research reactors and has been benchmarked against several research reactors. The details of the validity of using the code in determining the thermal-hydraulic characteristics of the reactor core are presented in the RAI response (Ref. 11).

Although the licensed reactor power level is 250 kWt, calculations were performed at a bounding scenario of 300 kWt. The results of the thermal-hydraulic analysis for the reactor core operating at 300 kWt are as follows:

Parameter

Number of fuel elements	80
Hot rod total peaking factor	1.955
Reactor pool temperature, °C (°F)	25 (77)
Coolant saturation temperature, °C (°F)	114 (237)
Exit coolant temperature, average, °C (°F)	46.33 (115.4)
Exit coolant temperature, maximum, °C (°F)	53.56 (128)
Peak fuel temperature in average fuel element, °C (°F)	214 (418)
Maximum wall temperature in hottest element, °C (°F)	123 (254)
Peak fuel temperature in hottest fuel element at 300 kWt, °C (°F)	253 (488)
Core average fuel temperature, °C (°F)	164 (327)
Minimum DNB ratio at 300 kWt	6.67

The licensee stated that, based on the temperatures given in the table above, some subcooled nucleate boiling is expected to occur. During subcooled boiling, bubbles form on the cladding surface and then detach, entering the flow stream. These bubbles increase the coolant buoyancy and enhance mixing in the coolant flow, improving heat transfer away from the cladding surface. The bubbles eventually collapse since the surrounding coolant is below saturation temperature.

The licensee used the results of its RELAP analysis as an input to the Bernath correlation to determine the DNBR. Using the Bernath correlation is consistent with NRC staff guidance and produces conservative results. The predicted value of 6.67 for DNBR at 300 kWt is consistent with similar TRIGA® reactor analyses at this power level and provides significant margin to the 2.0 acceptance criteria.

The licensee also determined the maximum reactor power level at which DNB would occur. For a reactor pool temperature of 25 °C (77 °F) and 20 ft (6.1 m) of water above the core, the maximum power level for a DNBR of 1.15 is 2.0 MWt, corresponding to a hot rod power of 36.2 kWt/element. The analysis demonstrates that, for the reactor operating at 250 kWt and a pool water temperature of 25 °C (77 °F), there is a satisfactory margin to the limits and guidance provided in NUREG-1537 for the minimum DNBR of 2.0.

The thermal-hydraulic analysis performed by the licensee demonstrates that a significant margin to the maximum fuel temperature SL is expected during steady-state operations.

The NRC staff reviewed the licensee analysis and finds that the licensee has used qualified calculation methods and conservative and justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with measurements obtained from the reactor and other reactors of similar design. On the basis of its review of the information provided, the NRC staff concludes that the analysis in the thermal-hydraulic analysis report (Ref. 11) demonstrates acceptable safety margins in regard to thermal-hydraulic parameters.

Pulse Mode Operation

The limiting condition for pulsed operation is the peak fuel temperature. The licensee stated that the peak fuel temperature is limited to 425 °C (797 °F) by TS 3.2.3. On the basis of early experience with TRIGA® fuel, temperatures above 830 °C (1,526 °F) demonstrated that fuel damage could occur because of hydrogen gas accumulation and redistribution in the hydride fuel if the reactor is pulsed after an extended period of operation at steady-state power of 1 MWt (Ref. 36). The reactor operates at only 250 kWt, with a corresponding average core temperature of 150 °C (302 °F) (Ref. 11), thereby providing a sufficient safety margin to 830 °C (1,526 °F).

The licensee presented a thermal-hydraulic analysis of a reactor pulse with a $3.00 insertion of reactivity, the TS limit for the combined worth of the two pulsing control rods. In SAR Section 4.5.6 (Ref. 20), using experimental data derived from pulsing and calculations using the Fuchs-Nordheim model (the standard model for TRIGA® pulsing analysis), the licensee derived values for the peak power level, reactor period, pulse width, total energy released, and peak fuel temperature as follows:

Parameter

Reactivity insertion, $	3.00
Peak power level, MWt	~1,200
Reactor period, msec	~3.1
Pulse width, msec	~11
Total energy release, MWt-sec	~16
Fuel temperature, °C	<450

The NRC staff reviewed the licensee's analyses and finds that it was performed with qualified calculation methods and conservative or justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with measurements obtained at the UCINRF and other reactors, indicating that the analytical results are conservative. On this basis, the NRC staff concludes that the pulsing limit of $3.00 provides an acceptable safety margin for limiting the peak fuel temperature.

2.7 Conclusions

Based on the above considerations, the NRC staff concludes that the licensee has presented adequate information to demonstrate its ability to configure and operate the reactor. The NRC staff concludes that the thermal-hydraulic analysis in the SAR and RAI responses demonstrates that the core has acceptable safety margins for thermal-hydraulic conditions.

The NRC staff reviewed the licensee's analysis for pulsed operation and finds that, with pulse sizes up to $3.00, the maximum core fuel temperature will remain below the limit set by the known mechanical and thermal properties of the fuel. On this basis, the NRC staff concludes that the reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely in accordance with the TSs.

On the basis of its review, the NRC staff concludes that there is reasonable assurance that continued operations consistent with the proposed TS not to exceed 250 kWt would not pose undue radiological risk to the health and safety of the public, UCINRF staff, or the environment. The NRC staff also concludes that there is reasonable assurance that the licensee is capable of safe operation of the UCINRF, as limited by the TSs, for the period of the renewed license.

SECTION 3

RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 <u>Radiation Protection</u>

Activities involving radiation at the University of California, Irvine Nuclear Reactor Facility (UCINRF), are controlled under the radiation protection program, which 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 specify, 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). 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 NRC staff inspection program routinely reviews radiation protection and radioactive waste management at the UCINRF. The NRC staff also observed activities in these areas during site visits to support the license renewal review. The licensee's historical performance in this area and the safety analysis report (SAR) provide documentation that measures are in place to minimize radiation exposure to UCINRF staff and the public and to provide adequate protection against operational releases of radioactivity to the environment.

3.1.1 Radiation Sources

Radiation sources at the UCINRF are discussed in SAR Section 6.3.2. The NRC staff review considered the descriptions provided of potential radiation sources, including the inventories of each physical form and their locations. This review of radiation sources included identification of potential radiation hazards as presented in Chapters 6 and 13 of the licensee's SAR (Ref. 20) and supplemented by request for additional information (RAI) responses (Ref. 15) in which the licensee provided verification that the hazards were accurately depicted and comprehensively identified.

3.1.1.1 Airborne Radiation Sources

In SAR, Sections 6.3.2.1 and 6.3.2.2, as supplemented by RAI response (Ref. 15), the licensee stated that, during normal operations of the UCINRF, the primary airborne sources of radiation are argon-41 (Ar-41) and nitrogen-16 (N-16). Ar-41 results from irradiation of the argon in air in experimental facilities and dissolved air in the reactor pool water. N-16 is produced when oxygen in the pool water is irradiated by the reactor core. The N-16 radionuclide has a half-life of 7.13 seconds. The NRC staff reviewed the licensee's analysis of the production and release of routine airborne radioactive effluents and the resultant doses to both the UCINRF staff and members of the public.

As described in SAR Section 6.3.2.1, neutron irradiation of the oxygen in the reactor cooling water creates N-16. The N-16 rises to the surface of the pool with the thermal plume of buoyant water heated by the core. When the diffuser is not in operation, the dose rate at the top of the reactor tank is mostly from N-16, with a small contribution from Ar-41. The N-16 diffuser is used to direct reactor cooling water across the top of the core, breaking up the buoyant water column from the core and significantly delaying the transport of radioactive N-16 to the top of the pool surface. Because of the short half-life of N-16, operation using the diffuser reduces the dose to UCINRF staff from N-16 in the reactor room. The measured dose level at the pool water surface without the diffuser is 20 mrem/hr (Ref. 64). When the diffuser is in operation, this level is significantly reduced by the vertical flow of water downward over the core resulting in a negligible release of N-16 to the atmosphere. Thus, Ar-41 is the main radiological dose contributor at the pool surface. The SAR, Section 6.3.1, Table 6-3, lists a dose rate at the pool surface of 2 mrem/hr with the reactor operating at full power and the diffuser in operation. Technical Specification (TS) 3.7.1 requires that radiation monitoring equipment be in operation and that the equipment provide information about radiation levels to the reactor operator (discussed in Sections 3.1.1.1, 3.1.4, and 3.1.7 of this safety evaluation report (SER)). Because of the short half-life of N-16 and the length of time for it to reach the pool surface, and access to the bridge above the core being limited during reactor operations, concentration of N-16 results in a negligible dose to UCINRF staff at the pool water surface. Given the short half-life and the additional time for N-16 that leaves the reactor pool to be released to the environment, public exposure from N-16 is negligible. The NRC staff reviewed the licensee's calculations of the production and release of N-16 and the resultant doses to the UCINRF staff and members of the public and finds these limits to be below the limits in 10 CFR 20.1201 and 20.1301, respectively.

As described in SAR Section 6.3.2.2, Ar-41 is produced through the irradiation of air in the experimental systems of the reactor. Air is present in the pneumatic transfer system (PTS), rotary rack, and central thimble tube. Because there is no flushing of the rotary rack and central thimble tube, the Ar-41 produced in these systems diffuses into the reactor room air either directly or through the pool water. Operation of the PTS results in the release of irradiated air, which is discharged directly to the reactor room exhaust duct. These features limit the exposure of UCINRF staff to Ar-41.

The licensee determined the occupational dose level in the reactor room from Ar-41 during normal operation (Ref. 13). Using measured data, the licensee determined that the maximum occupational exposure concentration for Ar-41 in the reactor room is limited to 1×10^{-8} μCi/mL. This exposure concentration is less than 0.4 percent of the derived air concentration (DAC) limiting value of 3×10^{-6} μCi/mL established in Table 1 (occupational values) of 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. If the reactor operated continuously with this Ar-41 concentration level for 2,000 hours per year, the licensee concluded that the occupational exposure would be less than 0.4 percent of the limiting DAC value.

Because of the renovation of the ventilation system in December 2010, the licensee assessed the potential dose from Ar-41 to offsite members of the public and provided this analysis through RAI responses (Refs. 12 and 13). The exhaust duct from the reactor room is combined with other building exhaust in a common header that exits the building on the roof of Rowland Hall 90 ft (27.4 m) above ground level in an area not accessible to the public. Normal dilution of the reactor facility exhaust stream with the other building ventilation exhausts helps to ensure that the reactor exhaust of Ar-41 is well below the annual limit of 1×10^{-8} μCi/mL in 10 CFR Part 20, Appendix B, Table 2 (effluent concentrations). The exit velocity of the exhaust and the exhaust blower design creates an effective stack height of 100 ft (30.5 m) above the building for a 10-mph wind speed. Although atmospheric dispersion would further reduce the ground-level concentration of the effluents, the licensee's analysis demonstrated that this additional dilution is not required to comply with 10 CFR Part 20, Appendix B, Table 2 (effluent concentrations), concentration limits for Ar-41.

The licensee stated that the dose to offsite members of the public from Ar-41 is primarily from the Ar-41 produced in the PTS, which is discharged directly into the main reactor room exhaust duct along with the reactor room ventilation air (Ref. 13). The maximum release of Ar-41 into the reactor ventilation system when the PTS is operating has been measured at a concentration of 7×10^{-8} μCi/mL. When the PTS is not operating, the maximum release of Ar-41 into the ventilation system has been measured at a concentration of 1×10^{-8} μCi/mL.

In the SAR and supplemented RAI response (Ref. 13), the licensee stated that the ventilation system provides a dilution of at least a factor of 100. This dilution resulted from mixing large quantities of clean air with the reactor exhaust air during the discharge. At the point of discharge well above the roof of the building housing the reactor, the licensee calculated the Ar-41 concentration to be approximately 7×10^{-10} μCi/mL. The NRC staff reviewed the licensee's calculations, performed confirmatory calculations, and found that the concentration of Ar-41 at the point of discharge above the roof was well within the air effluent concentration limits of 1×10^{-8} μCi/mL in 10 CFR Part 20, Appendix B, and below the ALARA limit in 10 CFR 20.1101 of 10 mrem/yr.

TS 3.7.2, Effluents, Specification a, states:

> Specification(s)
>
> a. The annual average concentration of ^{41}Ar released to the environment shall not exceed 1×10^{-8} microcuries per milliliter (μCi/mL).
>
> (…)

The licensee establishes the actual exposure of the public by placing environmental dosimeter packs at the point of effluent release. TS 3.7.1, Specification b, and TS 4.7, Specification d, establish the requirements for the environmental monitoring and the frequency with which the environmental monitors will be evaluated.

TS 3.7.1, Radiation Monitoring Systems, Specification b, states:

> Specification(s).
>
> (…)
>
> b. Environmental monitoring dosimeter packs, exchanged at least quarterly, shall be placed at the primary exhausts of the facility at all times, except when undergoing exchange. Additional packs shall be located in adjacent buildings and in a more remote control location for comparison.

TS 4.7, Radiation Monitoring System and Effluents, Specification d, states:

Specification(s).

(…)

d. The environmental monitoring dosimeters required by Technical Specification 3.7.1.b including those monitoring exhaust effluents, shall be evaluated quarterly.

(…)

The NRC staff reviewed TS 3.7.1, Specification b, TS 3.7.2, Specification a, and TS 4.7, Specification d, and finds that these TSs help ensure adequate monitoring of airborne release of radioactive effluents during normal operation of the UCINRF. These TSs also help ensure that the airborne radioactive release of Ar-41 will not pose a significant risk to the health and safety of the public or the environment.

Based on its review of the SAR, the NRC staff finds that the production and control of UCINRF airborne radiation sources and atmospheric effluent releases of Ar-41 and N-16 from normal operation are within the limits of 10 CFR Part 20 requirements. The NRC staff finds that TS 3.7.2, Specification a, and information that the licensee provided in the SAR indicate that, during normal operation of the UCINRF, airborne releases will result in doses to the maximally exposed member of the public on the order of 1 mrem/yr or less. Based on the information provided above, the NRC staff concludes that normal operation of the UCINRF is within the limits of 10 CFR Part 20 and that TS 3.7.1, Specification b, TS 3.7.2, Specification a, and TS 4.7, Specification d, are acceptable.

3.1.1.2 Liquid Radiation Sources

In the SAR, Section 5.3 (Ref. 20), and RAI response (Ref. 4), the licensee indicated that impurities in the primary coolant become activated by neutrons as they pass through the reactor core. Most of this material is captured in the mechanical filtration or ion exchange in demineralizer resins and, therefore, is dealt with as solid waste. However, small quantities and concentrations of other than routine liquid radioactive waste may be generated from decontamination or maintenance activities. This liquid waste is transferred to the University Environmental Health and Safety Office (EH&S) for disposal. The SAR states that radiation exposures from these liquid radiation sources at the UCINRF are small and, as a result, do not present a significant hazard to either UCINRF staff or the public.

The SAR, Section 6.3.2, describes another potential low-level liquid radiation source being N-16 entrained in the pool water. As discussed in Section 3.1.1.1 of this SER, N-16 has a half-life of 7.13 s. Occupational exposure from the N-16 is very limited, as access to the platform above the core is minimal during reactor operations and the short half-life of N-16 makes exposure to radiation workers and the public negligible. A diffuser in the pool above the reactor core directs a small flow of reactor pool water downwards and across the top of the core area, significantly slowing the upward flow of heated water containing the N-16 and helping to ensure the dose rate from N-16 decay is reduced. Any N-16 in the pool water quickly decays after the reactor is shut down.

As required by TS 4.3, Specification d, (discussed in Section 5.4 of this SER), the licensee samples primary water for radioactive content on a quarterly basis to detect potential fission product leakage from the reactor fuel, leakage from sealed sources, or activation of materials in the coolant water to determine that the radioactive content is below the limits for release to the environment (as discussed in Section 2.3 of this SER).

TS 3.7.2, Effluents, Specification b, specifies the quantity of radioactive material in liquid form that is allowed to be released to the environment as follows:

> Specification(s).
>
> (…)
>
> b. The quantity of radioactivity in liquid effluents released from the facility to the sewer system shall be soluble and not exceed the limits of 10 CFR § 20, Appendix B, Table 3.

TS 4.7, Radiation Monitoring System and Effluents, Specification e, states:

> Specification(s).
>
> (…)
>
> e. Any liquid effluents to be released to the sewer system from the facility shall be verified to contain only soluble materials and analyzed for radioactive content prior to release.

In a response to RAI (Ref. 64), the licensee stated in an RAI response (Ref. 64) that all radioactive liquids discharged to the sewer system are monitored before release. The NRC staff reviewed TS 3.7.2, Specification b, and TS 4.7, Specification e, and finds that these TSs help ensure compliance with the limits in 10 CFR Part 20, Appendix B, Table 3 (releases to sewers) and control disposal of liquid sources of radiation. On the basis of its review, the NRC staff concludes that TS 3.7.2, Specification b, and TS 4.7, Specification e, are acceptable.

Based on a review of the information provided above, the NRC staff further finds that liquid radioactive sources from continued normal operation of the UCINRF are properly controlled and will not pose a significant hazard to public health or safety, or the environment. Any release of liquid radioactive effluents containing only soluble materials will be less than the limits established in 10 CFR Part 20 requirements. The NRC staff concludes that liquid radioactive sources from normal operation of the UCINRF are acceptable.

3.1.1.3 Solid Radioactive Sources

The SAR, Sections 6.3 and 11.8 (Ref. 20), and TS 5.3.1, indicate that solid radioactive sources include reactor fuel and a startup neutron source. The licensee stated that the fission products in the reactor fuel constitute the most significant solid radiation source at the UCINRF. These fuel elements are either in the reactor core or stored underwater in storage racks.

Solid waste includes low-level radioactive waste created because of facility operations (such as filters and resin) and experiments (such as gloves and discarded samples). Low-level solid radioactive waste is transferred to the University of California, Irvine (UCI), Office of EH&S radioisotope waste handling program. While in the reactor facility, the waste is stored in specially marked containers that are specified for use by the University Radiation Safety Program (RSP). Solid radioactive waste production at the UCINRF is minimal. The NRC staff reviewed the annual reports (Refs. 39-45, 61-63, and 67) that indicate a total of approximately 43 ft^3 (1.21 m^3) of dry solid waste had been transferred to the UCI Office of EH&S for the time period between 2005 through 2015.

Other nonfuel sources include activated reactor components, irradiated samples, and a sealed source in the reactor pool. The reactor contains a stainless-steel clad americium-beryllium startup source in the reactor core. Section 2.2.4 of this SER discusses the startup source and its location in the grid box.

Based on its review of the information described above, the NRC staff finds that solid radioactive sources from continued operation are properly controlled, have resulted in no significant exposures, and can be handled without endangering the safety of the UCINRF staff. The NRC staff concludes that the control of solid radioactive sources at the UCINRF is acceptable.

3.1.2 Radiation Protection Program

The regulation in 10 CFR 20.1101(a) requires each licensee to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities. The SAR, Section 11.0 (Ref. 20), describes the radiation protection strategy and methodology used at the UCINRF. According to the SAR, the program incorporates portions of 10 CFR, "Energy"; California Health and Safety Code, Division 20, "Miscellaneous Health and Safety Provisions"; and California Administrative Code, Title 17, "Public Health." The goal of the program is to limit radiation exposures and radioactivity releases to a level that is ALARA without restricting facility operations for education and research. SER Section 3.1.3 discusses the ALARA commitment and practices. The UCINRF radiation protection program consists of various elements that help ensure doses and releases are ALARA, including training of personnel, administrative controls, radiation monitoring and surveying, personnel dosimetry, program surveillance by the UCI EH&S Office, waste management, emergency supply accountability, and access control.

The ultimate responsibility for this program is vested in the Chancellor of UCI. The UCI campus radiation safety officer (RSO) is responsible for implementing radiation protection at the UCINRF as specified in TS 6.3, "Radiation Safety," while the day-to-day radiation protection activities within the reactor facility are performed by the operating staff, directed by the Reactor Supervisor.

UCINRF Standard Operating Procedure No. 5, "Radiological Safety Program," establishes the facility's radiation protection program. Procedures relating to the radiation protection program must be approved by the RSO and the Radiation Safety Committee (RSC) in addition to normal review by the Reactor Operations Committee as required by TS 6.4, "Operating Procedures." This TS specifically requires written procedures for personnel radiation protection and for shipping and transfer of radioactive materials.

TS 6.1.2, Responsibilities, Specification e, states:

> (…)
>
> e. There shall be a University of California, Irvine campus Radiation Safety Officer (RSO) responsible for the safety of operations from the standpoint of radiation protection. This position shall report to the Office of

> Environmental Health and Safety which is an organization independent of the reactor operations organization as shown in Figure 1. An independent campus-wide Radiation Safety Committee (RSC) shall be responsible for establishment and review of all policies involving radiation and radioactivity. Routine radiological safety requirements within the reactor facility shall be carried out by reactor operators and/or individual experimenters, all of whom shall be required by regulations, to have received training in radiological safety and be authorized for radiation use by the campus Radiation Safety Officer.

(…)

TS 6.3, Radiation Safety, states:

> As delineated in Technical Specification 6.1.2.e, the campus Radiation Safety Officer (RSO) shall be responsible for implementation of the radiological safety program at the reactor facility in accordance with applicable federal and state of California standards and regulations. The program should use the guidelines of ANSI/ANS 15.11-2004.
>
> The RSO shall be responsible for an annual audit of the radiation safety program.

TS 6.1.2, Specification e, and TS 6.3 establish responsibilities for implementing the radiation protection program. The NRC staff reviewed TS 6.1.2, Specification e, and TS 6.3 and finds these TSs consistent with the guidance provided in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” and ANSI/ANS-15.1-2007, “The Development of Technical Specifications for Research Reactors.” On this basis, the NRC staff concludes that TS 6.1.2, Specification e, and TS 6.3 are acceptable.

The NRC staff reviewed the UCINRF radiation protection program and finds that it complies with 10 CFR Part 20 regulations and is consistent with the guidance in ANSI/ANS-15.11-1993 (R2004), “Radiation Protection at Research Reactor Facilities” (Ref. 37). The NRC staff also reviewed the U.S. Nuclear Regulatory Commission (NRC) inspection reports (IRs) for the past 5 years, which include detailed review of the radiation protection program, and found no significant radiation safety issues. The NRC staff finds that the UCINRF radiation protection program complies with 10 CFR 20.1101(a), is implemented in an acceptable manner, and provides reasonable assurance that, for all facility activities, the program will protect the public, the UCINRF staff, and the environment from unacceptable radiation exposures. On the basis of its review, the NRC staff concludes that the radiation protection program is acceptable.

3.1.3 ALARA Program

To comply with the regulation in 10 CFR 20.1101(b), the licensee established and implemented a policy that all operations are to be planned and conducted in a manner to keep all exposures ALARA. UCINRF Standard Operating Procedure No. 5 establishes the facility’s ALARA program. This includes using methods and procedures that shield radiation sources and UCINRF staff; that increase the distance between an exposure point and a radiation source; that reduce the time a person might be exposed to a given dose rate; that contain sources; and that use careful, thoughtful advanced planning when working in an area that might contain a radiation field. Various administrative controls have been put into place to accomplish the

ALARA goals. According to the SAR, the UCI Radiation Safety Office maintains a suitable ALARA policy (Ref. 14). The policy and practice are enforced at the UCINRF and are delineated in Section G of the UCI "Radiation Safety Manual." This section outlines the UCI ALARA program and specifies trigger levels for the investigation of radiation exposures.

TS 6.4, Specification e, requires the facility to have written operating procedures that include provisions to maintain personnel exposures ALARA. TS 6.1.3, Specification c, requires training for experimenters, including training in the features of the ALARA program (discussed in Sections 5.6.4 and 5.6.1.3 of this SER).

The NRC staff reviewed IRs for the UCINRF, which include detailed oversight of the facility's ALARA program, and concluded that the program provided guidance for keeping doses ALARA and was consistent with the requirements in 10 CFR Part 20. The NRC staff reviewed these IRs and 5 years of annual reports with attention to radioactive effluents and personnel occupational exposure. The NRC staff finds that releases to the environment and radiation doses to UCINRF staff were consistent with those at other similar reactor facilities which demonstrates that the ALARA program is functioning adequately.

The NRC staff finds that the UCINRF ALARA program complies with the regulations in 10 CFR 20.1101, is consistent with the guidance in ANSI/ANS-15.11-1993 (R2004), and is established in accordance with TS 6.4, Specification e. Based on the information provided above, the NRC staff concludes that the ALARA program is functioning adequately and provides reasonable assurance that radioactive effluents and UCINRF staff and public doses will continue to be minimized during the renewed license period. On this basis, the NRC staff concludes that the UCINRF ALARA program is acceptable.

3.1.4 Radiation Monitoring and Surveying

SAR Section 3.7 and Section 11.0 provide an overview of the UCINRF radiation monitoring and surveying program. The radiation monitoring program for the reactor is structured to ensure that all three categories of radiation sources – air, liquid, and solid – are detected and assessed promptly. Area monitors in the reactor bay and in the gaseous effluent pathway are in continuous use. Contamination surveys with a Geiger counter or equivalent and area dose rates with ion chambers, scintillation detectors, or energy-compensated Geiger counters are performed periodically.

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

(1) may be necessary for the licensee to comply with the regulations in this part
(2) are reasonable under the circumstances to evaluate
  - (i) the magnitude and extent of radiation levels
  - (ii) concentrations or quantities of residual radioactivity
  - (iii) the potential radiological hazards of the radiation levels and residual radioactivity detected

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

The licensee maintains numerous fixed and portable radiation detection instruments throughout the facility. Fixed gamma area radiation monitors are throughout the reactor room and adjacent laboratory space to alert UCINRF staff of changing radiation conditions.

TS 3.7.1, Radiation Monitoring Systems, Specification a, states:

Specification(s).

a. The reactor shall not be operated unless the following minimum radiation monitoring instruments are operating:

	Minimum Number Operating
Radiation Area Monitors (RAM):	2
Continuous Air Particulate Monitor (CAM):	1

(…)

TS 3.7.1, Specification a, requires two radiation area monitors (RAMs) to be operable that provide indications of radiation levels in the facility. TS 3.7.1, Specification a, also requires one continuous air particulate monitor (CAM) to be operable that can detect the presence of airborne fission products or release of radioactivity in the vicinity of the reactor bridge. All RAMs have their readouts at a station adjacent to the reactor control console. The CAM is in the reactor room adjacent to the reactor pool. The monitor draws air from a tube beneath the reactor bridge and would provide the first indication of any fission product leak from the fuel. The readout for the CAM is adjacent to the reactor control console. The RAMs and the CAM are all equipped with audio and visual alarms, and all three transmit high-level alarms to a remote monitoring station.

A reading on the CAM at the alarm level of 5,000 cpm could indicate a fission product release from a defective fuel element or a release of radioactivity in the vicinity of the reactor bridge. The alarm from the CAM will cause the ventilation system to switch from the normal mode to the emergency mode of operation.

TS 3.5.1, Ventilation System, Specification b and Specification c, state:

Specification(s).

(…)

b. The reactor shall not be operated unless it is verified that the ventilation system goes into the emergency mode upon manual actuation or a signal of high radiation activity from a Continuous Air Particulate Monitor (CAM) measuring air from above the pool as described in Technical Specification 3.5.2. Verification shall be by observing the emergency flow rate is at least 240 cubic feet per minute (cfm), the absence of regular exhaust flow, and the pressure differential reading between the reactor area and the outside is negative.

c. Radioactive by-product materials shall not be handled in the facility unless the ventilation system is operating as described in Technical Specifications 3.5.1.a. and 3.5.1.b. above.

TS 3.5.1, Specification b, helps ensure that, in case of accidental release of particulate or gaseous radioactivity into the reactor room, the ventilation system in the emergency mode provides sufficient airflow to maintain a negative pressure differential with the areas outside the reactor room. The negative pressure helps to ensure that the radioactivity in the reactor room is directed to the exhaust system so the effluent is released from the stack. In emergency mode, the release is exhausted slowly to allow dilution and filter radioactive particles. In this mode, air is exhausted from the reactor room at a rate of about 250 cfm through a Magnamedia Beryllium-collector style filter. At the same time, supply valves to the reactor room and laboratories are closed and the regular reactor room exhaust and the fume hood exhaust control valves are closed. The PTS blower is disabled, and audio and visual alarms indicate that the emergency condition has been actuated. Flow rates can be monitored at the control room ventilation system controller to assure this has been accomplished (Ref. 13). The NRC staff reviewed TS 3.5.1, Specification b, and finds that this specification helps ensure that the ventilation is set to emergency mode, which uses a high-efficiency particulate air (HEPA) filtered purge exhaust to limit the release of radioactivity into the environment should a radiological release occur. The NRC staff finds that this mode of operating the ventilation system is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. TS 3.5.1, Specification b, will help ensure that the ventilation system is operable to reduce potential effects of the release of volatile chemicals during by-product handling.

TS 3.5.1, Specification c, ensures the ventilation is in operation when handling by-product material, to reduce the potential impact of release of volatile chemicals into the facility. On this basis, the NRC staff concludes that Specifications b and c of TS 5.3.1 are acceptable.

TS 3.5.2, Ventilation during Emergency Situations, states:

Specification(s). A signal of high radiation activity alarm from a Continuous Air Particulate Monitor (CAM) measuring air from above the pool or manual operation from the control room shall carry out the following functions:

a. close off inflow air by closing dampers; and

b. close off outflow air by closing dampers in exhaust ducts and removing power from relevant exhaust fans and fume hood; and

c. remove power from pneumatic transfer system so it can no longer operate to transfer air through any core region; and

d. open outflow damper in a small "purge" exhaust duct system equipped with a HEPA filter.

TS 3.5.2 specifies the ventilation system actions that take place during emergency situations when the CAM reaches the alarm point. The ventilation system actions limit the exposure to the public in the event of a radiological release by releasing airborne radioactivity in a controlled, filtered manner while maintaining a negative pressure in the reactor room so airborne radioactivity does not infiltrate the remainder of the building. The NRC staff reviewed TS 3.5.2 and finds that TS 3.5.2 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 3.5.2 is acceptable.

TS 4.7, Radiation Monitoring System and Effluents, Specifications a, b, and c state:

Specification(s).

a. A channel test of the area radiation monitoring systems required by Technical Specification 3.7.1.a shall be performed monthly. This shall include verification of the alarm set points.

b. A channel check of the Continuous Air Particulate Monitor (CAM) required by Technical Specification 3.7.1.a shall be performed daily. This shall include verification of the alarm set point.

c. A channel calibration of the radiation monitoring systems required by Technical Specification 3.7.1.a. shall be performed annually.

(…)

The NRC staff reviewed the surveillance intervals required by TS 4.7, Specifications a through c, and finds these specifications help ensure the operability of the radiation monitoring systems is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff finds that TS 4.7, Specifications a through c, are acceptable.

TS 4.5, Ventilation System, Specification c, states:

Specification(s).

(…)

c. A channel test of the function of the Continuous Air Particulate Monitor (CAM) alarm and the control room manual switch to properly set the ventilation system into emergency mode shall be performed daily.

TS 4.5, Specification c, requires a daily test of the ability of the CAM to perform its intended function, upon receipt of an alarm, to set the ventilation to the emergency mode. The NRC staff reviewed the surveillance interval required by TS 4.5, Specification c, and finds that Specification c is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 4.5, Specification c, is acceptable.

The NRC staff finds that the licensee's equipment is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility. The instruments are operated appropriately and calibration frequency is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 to help ensure compliance with 10 CFR 20.1501(a) and 1501(c). The NRC staff finds that TS 3.7.1, Specification a, requires sufficient monitors to evaluate potential radiation hazards and helps maintain routine effluent releases within regulatory limits. The NRC staff concludes that the radiation monitoring and surveillance activities performed at the UCINRF provide reasonable assurance that radiation from the reactor and from radioactive materials will be appropriately measured and characterized during the renewed license period and that the CAM will effectively shift the ventilation into the emergency mode when required. On the basis of its review, the NRC staff concludes that the UCINRF radiation monitoring and surveying program is acceptable.

3.1.5 Radiation Exposure Control and Dosimetry

SAR Section 3.0 and Section 11.0 describe the components of the licensee's radiation exposure control and dosimetry program. Radiation exposure control depends on factors such as the facility's design features, operating procedures, training, and equipment. Design features include shielding, ventilation, containment of the inventory within the fuel, entry control, protective equipment, personnel dosimetry, and estimates of annual doses at various locations.

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

The regulation, 10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose," requires monitoring of workers likely to receive an annual dose from sources external to the body, in excess of 10 percent of the limits described in 10 CFR 20.1501, "General." Section 20.1502 specifically requires monitoring of individuals entering a high or very high radiation field in which an individual could receive a dose equivalent of 0.1 rem in 1 hour.

The NRC staff reviewed the annual reports and the NRC staff IRs for the prior 5 years and found no identified concerns regarding the radiation exposure control and dosimetry program at the UCINRF.

The ventilation system maintains the reactor room at negative pressure with respect to outside areas and controls the exposures from Ar-41 and N-16 to levels below the limits required in 10 CFR 20.1201, "Occupational Dose Limits for Adults." The licensee uses thermoluminescent dosimeters (TLDs) to monitor personnel whole-body exposure. TLDs are assigned to individuals who have the potential to be exposed to radiation. The TLDs are also used to monitor extremities.

TLDs are processed quarterly to provide measurements of whole-body and extremity exposure of UCINRF staff. A vendor accredited by the National Voluntary Laboratory Accreditation Program (NVLAP) in accordance with 10 CFR 20.1501(d) supplies them. Programs accredited by NVLAP have been audited and tested to ensure that dosimeters respond accurately to radiation. TLDs are supplemented by pocket dosimeters. Tour groups and other visitors may be issued representative dosimeters rather than individual badges.

Radiation exposure is controlled through the use of training, postings, and physical barriers to higher levels of radiation. The licensee uses survey meters to measure dose rates from radiation fields, and these measured rates are posted where required. The NRC staff review finds that radiation exposure controls are in appropriate areas of the facility and are adequate to alert workers to radiation hazards within the facility. These provisions help ensure that external and internal radiation monitoring of all individuals required to be monitored meet the requirements of 10 CFR Part 20 and the goals of the facility ALARA program. The NRC staff review of the most recent annual operating report for 2015 revealed that the average annual whole-body exposure was 18.3 mrem. Previous annual operating reports indicate similar exposure numbers below the limits specified in 10 CFR 20.1201.

Based on its review of information provided in the SAR, the NRC staff finds that the licensee has the appropriate equipment and procedures for control of radiation exposure. The NRC staff also finds that UCINRF staff exposures are controlled through satisfactory radiation protection and ALARA programs and that UCINRF staff have had historically low radiation doses. Furthermore, based on the review of the results of annual reports from 2010 through 2015, annual occupational exposures and radiological releases to the environment are far below NRC allowable limits. The NRC staff review of IRs from 2007 through 2015 found no contradictory findings. On the basis of the information provided above, the NRC staff concludes that the licensee's control of personnel exposures and dosimetry is acceptable.

3.1.6 Contamination Control

Contamination control at the UCINRF is addressed through a combination of personnel monitoring and area surveys for contamination. The licensee performs contamination surveys periodically depending on the frequency with which radioactive material is used or handled. Written procedures control handling of radioactive material within the UCINRF. Workers are trained for working with radioactive material, including how to limit its spread during entry and exit to potentially contaminated areas, and are required to monitor both themselves and any hand-carried items for contamination as soon as practicable. NRC IRs show that areas of potential contamination have been posted and marked acceptably. According to the licensee, the instrumentation used for contamination control has the appropriate range and sensitivity to detect the types of radiation present at the UCINRF.

As discussed in SAR Section 11.5 (Ref. 20), area surveys for contamination are performed periodically. Any contamination found is isolated and decontaminated as soon as practicable. Confirmatory surveys after decontamination are then performed. Wipe tests for removable contamination are routinely performed in at least 15 key locations. The UCI EH&S Office independently performs contamination surveys quarterly.

TS 6.8.1 requires that records of reactor facility radiation and contamination surveys be retained for a period of at least 5 years or for the life of the component involved if less than 5 years. The NRC staff reviewed the annual reports from 2010 through 2015 and NRC IRs from 2010 through 2015 and finds that these reports have not identified concerns regarding the contamination control program at the UCINRF.

Based on the information provided above and the NRC staff's review of the SAR and historic performance of the facility's program for contamination control, the NRC staff concludes that adequate contamination controls exist to prevent the spread of contamination at the UCINRF. Furthermore, reasonable assurance exists that the program will continue to minimize the hazard from contamination to personnel for the renewed license period.

3.1.7 Environmental Monitoring

SAR Sections 3.7 and 11.6 provide information on the facility's environmental monitoring program. Environmental monitoring is conducted at the UCINRF to ensure compliance with Subpart F of 10 CFR Part 20 and the UCINRF TSs. Installed monitoring systems include area radiation monitors and air monitors that have been managed and maintained in a comprehensive program. TS 3.7.1, Specification a, requires a minimum of one CAM and two RAMs to be running while the reactor is in operation. With the exception of Ar-41, the SAR describes no other pathways for radioactive materials from the UCINRF to enter the unrestricted environment during normal operations. Calibration of the monitors is accomplished as required by the TS 4.7 and in accordance with facility procedures (discussed in Section 3.1.4 of this SER).

TS 3.7.1, Specification b, requires environmental dosimeter packs be placed in the effluent stack. Additional environmental packs are placed in 10 other locations, including unrestricted areas adjacent to the reactor facility and a number of campus locations. The environmental monitoring packs contain three calcium sulfate/dysprosium TLDs, which are averaged for exposure readings at a particular location. The packs are exchanged and read quarterly.

As required by 10 CFR 20.1501, contamination surveys are conducted to help ensure compliance with regulations reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive material, and potential radiological hazards. The NRC staff reviewed annual reports and NRC IRs from 2007 through 2015 and finds that the operation of the UCINRF has had an insignificant effect on the radiation exposure in and around the facility.

The NRC staff finds that the environmental monitoring program can properly assess the effect of day-to-day operation of the facility on the environment. The NRC staff further finds that the reporting requirements and the record retention requirements for environmental monitoring data are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review of the information described above, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological effect of the UCINRF on the environment and is acceptable.

3.2 <u>Radioactive Waste Management</u>

The purpose of the radioactive waste management program is to help ensure that radioactive waste materials are identified, assessed, controlled, and disposed of in conformance with all applicable regulations and in a manner to protect the health and safety of the public and the environment. The SAR, Section 11.8, provides an overview of the facility's radioactive waste management program.

3.2.1 Radioactive Waste Management Program

SAR Section 11.8 (Ref. 20) describes the movement, process, and release practices for radioactive waste. The objectives of the UCINRF radioactive waste management program are to minimize and properly handle, store, and dispose of waste. The Reactor Supervisor is responsible for the radioactive waste management program with support from the UCI RSP administered by the EH&S Office. Low-level radioactive waste, both liquid and solid, generated at the UCINRF is packaged in approved containers and transferred from the reactor facility to the control of the UCI RSP for storage and disposition.

The NRC staff reviewed SAR Section 11.8 and finds that acceptable controls are in place to prevent uncontrolled personnel exposures from radioactive waste operations and provide the necessary accountability to prevent unauthorized release of radioactive waste. Based on the information provided above, the NRC staff concludes that the UCINRF radioactive waste management program is acceptable.

3.2.2 Radioactive Waste Controls

SAR Section 11.8.1 describes the radioactive waste controls at the UCINRF and differentiates between radioactive waste and radioactive effluents, notably Ar-41. Waste volumes at the UCINRF historically have been small and of known characterization. When possible, radioactive waste is segregated at the point of origin from items not considered to be radioactive waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the projected future need for the materials involved. Consumable supplies, such as absorbent materials or protective clothing, are declared radioactive waste if radioactivity above background is found to be present. Radioactive waste generated by the reactor facility is transferred to the UCI State of California license for disposal. At present, all waste is retained for decay at the UCINRF under the UCI State of California license. Disposal of liquids to the sanitary sewer system is permitted in accordance with 10 CFR 20.2003, “Disposal by Release into Sanitary Sewerage.” TS 3.7.2, Specification b, establishes assurance that any release of radioactive materials contained in liquids released to the sewer system does not exceed the limits required by regulations.

Based on its review of the information in SAR Section 11.8, as described above, the NRC staff finds that UCINRF has adequate radioactive waste controls in place to monitor the radiation exposure from radioactive waste, perform required handling operations, and prepare for transfer to offsite disposal. Based on the information provided above, the NRC staff concludes that the UCINRF radioactive waste controls are acceptable.

3.2.3 Release of Radioactive Waste

The SAR, as supplemented by responses to RAI, indicates that normal operation of the UCINRF does not produce significant quantities of radioactive waste and that the only exception is gaseous radioactive effluents, notably Ar-41, which is regulated under 10 CFR Part 20. As discussed in Section 3.1.1.1 of this SER, gaseous effluents are monitored using air monitors and dosimeter packs to ensure compliance with the regulatory limits (i.e., the allowable effluent concentration for Ar-41 is 1×10^{-8} μCi/cm^3).

As discussed in Section 3.1.1.2 of this SER, if contaminated liquids are produced, they are discharged to the sanitary sewer after sufficient sampling and monitoring to ensure all releases meet the applicable regulatory requirements. The licensee monitors and records effluent measurements of all releases. TS 6.8.3 requires retention of records of effluents for the life of the facility. This is in conformance with the requirement of 10 CFR 20.2108, “Records of Waste Disposal.” Solid radioactive waste and laboratory liquid waste are disposed of by transferring the waste to the UCI EH&S radioisotope waste handling program for disposition. Waste is packaged and transported as required by appropriate NRC regulations, Department of Transportation regulations, and applicable State licenses. The licensee also provides a summary of the release of solid and liquid waste in its annual report required by TS 6.7.1, Annual Operating Report.

The NRC staff reviewed the annual reports for the period 2009 through 2015 (Refs. 43-45, 61-63, and 67) and NRC IRs for the period 2009 through 2015 (Refs. 49, 50, and 77-80) and finds that the effluent concentration of Ar-41 was well below the TS 3.7.2 limits and the release of solid and liquid radioactive waste was in accordance with regulatory requirements.

Based on the above information, the NRC staff concludes that there is reasonable assurance that releases of all forms of radioactive waste from the UCINRF will meet NRC requirements during the renewed license period.

3.3 Conclusions

On the basis of its evaluation of the information presented in the SAR, as supplemented, observations of the licensee's operations, and results of the NRC staff inspection program, the NRC staff concludes the following regarding the UCINRF radiation protection program and waste management:

- The UCINRF radiation protection program complies with the requirements of 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the UCINRF staff, the environment, and the public are protected from unacceptable radiation exposures. The radiation protection organization has acceptable lines of authority and communication to carry out the program.
- The systems provided for the control of radiological effluents, when operated in accordance with the 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.
- The UCINRF ALARA program complies with the requirements of 10 CFR 20.1101(b) and 1101(c). The program applies the guidelines of ANSI/ANS-15.11-1993 (R2004) implementing time, distance, and shielding to reduce radiation exposures. A review of historical radiation doses and current controls for radioactive material in the UCINRF provides reasonable assurance that radiation doses to the environment, the public, and UCINRF staff will be ALARA.
- The results of radiation surveys carried out at the UCINRF, doses to the persons issued dosimetry, and the environmental monitoring program help confirm that the implementation of the radiation protection and ALARA programs are effective and in compliance with the requirements of 10 CFR 20.1501(a).
- Facility design and operational procedures limit the production and release of Ar-41 and N-16 and control the potential for UCINRF staff and public radiation exposures. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas give reasonable assurance that doses to the UCINRF staff and public will be below applicable 10 CFR Part 20 limits.
- The facility radioactive waste management program provides reasonable assurance that radioactive waste released from the facility will be controlled and handled in accordance with applicable regulations and does not pose an unacceptable radiation risk to the environment and the public.

The NRC staff reviewed the UCINRF radiation protection program and waste management summary as described in the SAR Chapter 11, as supplemented. The NRC staff finds that the licensee has implemented adequate and sufficient measures to minimize radiation exposure to UCINRF staff and the public. Furthermore, the NRC staff concludes that there is reasonable assurance that the UCINRF radiation protection and waste management programs will provide acceptable radiation protection to the UCINRF staff, the public, and the environment.

SECTION 4

ACCIDENT ANALYSIS

The University of California, Irvine Nuclear Reactor Facility (UCINRF), safety analysis report (SAR), Chapter 13, provides accident analyses to demonstrate that the health and safety of the public and workers can be protected during analyzed reactor transients and other hypothetical accident scenarios. The accident analysis presented in this chapter demonstrated that no credible accident could lead to unacceptable radiological consequences to the UCINRF staff, the public, or the environment. Additionally, the licensee provided, consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," the consequences of the maximum hypothetical accident (MHA), which is an event involving the rupture of the cladding of the maximum irradiated fuel element in air. The MHA is considered the worst-case fuel failure fission product release scenario for a Training, Research, Isotopes, General Atomics (TRIGA®) reactor that would lead to the maximum potential radiation hazard to UCINRF staff and members of the public. The results of the MHA are used to evaluate the ability of the licensee to respond and mitigate the consequences of this postulated radioactive release. The NRC staff performed independent analysis of accidents with TRIGA®-fueled reactors (Refs. 51-53) and compared those results with accidents analyzed by the licensee. As demonstrated below, none of the potential accidents considered in the license renewal SAR (Ref. 20), as supplemented, would lead to significant occupational or public exposure. The potential consequences of the analyzed accidents meets the radiation exposure requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation."

NUREG-1537 suggests that each licensee consider the applicability of each of the following accident scenarios:

- MHA
- insertion of excess reactivity
- loss-of-coolant accident (LOCA)
- loss of coolant flow (LOCF)
- mishandling or malfunction of fuel
- experiment malfunction
- loss of normal electric power
- external events
- mishandling or malfunction of equipment

4.1 <u>Accident-Initiating Events and Consequences</u>

4.1.1 Maximum Hypothetical Accident

For the UCINRF, the MHA is defined as the rupture of the cladding of one fuel element in air. The scenario assumes that such an accident occurs with the reactor operated at full licensed power for a long enough time so that the inventories of all radionuclides in the scenario are at their maximum concentration. The analysis assumes that, at the time of fuel clad failure, the volatile fission products have accumulated in the gap and are released abruptly into the air with no radioactive decay; this includes the release of noble gases, halogens, and other volatile fission products. The licensee provided the MHA information in the SAR, supplemented by responses to requests for additional information (RAI) (Refs. 4-6 and 12-15) which the NRC staff used to evaluate the event.

Nuclide Inventory

For determining the radionuclide inventories, the licensee assumed the reactor had been operating continuously for 1,000 effective full-power days (EFPDs) at 250 kWt. This resulted in all of the halogens and noble gases reaching their saturation activities with the exception of very long-lived nuclides, such as krypton-85 (Kr-85) and iodine-129 (I-129), and any short-lived activation daughter products of which the licensee identified four. The source term for the reactor inventory was calculated using the SCALE 6.1 computer code package, which uses the ORIGIN-S code for isotope generation and depletion calculations. The gaseous fission product inventories by nuclide are presented in Table 13-1 of the licensee's SAR, as supplemented (Ref. 14). The MHA assumptions, as described below, included using saturated inventories in the released isotopes and did not account for the reduction in the iodine (I) isotopes that would occur through attachment to various surfaces or by the filtering processes. The NRC staff determined that the inventories of isotopes dominant to the MHA dose calculations (halogens and noble gases) were at the saturation (maximum) concentration for continuous full-power operation of the UCINRF with the exception of Kr-85 and I-129, which are long-lived isotopes. However, the NRC staff noted that the Kr-85 and I-129 dose contributions are small when compared with other more dominant contributors. The NRC staff reviewed the licensee's analytically generated radionuclide inventory for the MHA and the assumptions and boundary conditions used and finds that the nuclide inventories cited by the licensee are representative of the low-enriched uranium (LEU) fuel used in the UCINRF. Using information on fission yields and information from NUREG/CR-2387, "Credible Accident Analyses for TRIGA® and TRIGA®-Fueled Reactors" (Ref. 51), the NRC staff confirmed that the licensee's analysis and approach to determining radionuclide inventories were acceptable for the MHA dose calculations.

Release Fractions

Experiments have indicated that the release of radioactive nuclides from breach of the cladding in a TRIGA® fuel element would increase with the fuel temperature. Facilities such as the UCINRF have adopted a release fraction of 1.5×10^{-5} for fuel temperatures below 300 ° Celsius (°C) (572 ° Fahrenheit (°F)). This release fraction is derived from a 1980 General Atomics (GA) report based on experimental data from the TRIGA® reactor vendor (Ref. 34) and is consistent with release fractions used in other TRIGA® reactor analyses approved by the NRC staff. The licensee assumed that any noble gases released from the cladding would diffuse in the air until their radioactive decay because of their innate characteristic to not condense or combine chemically. The licensee conservatively did not account for reduction in iodine radioisotopes in its MHA; however, iodine radioisotopes are chemically active, and some of the radioactive iodine will be trapped by the materials it comes in contact with, such as water and reactor building structures. Most of these radioactive I radioisotopes will not become or remain airborne under accident scenarios. Because of this, the licensee conservatively assumed 100 percent of noble gases in the fuel gap are available to the reactor room and outside environment, in order to ensure upper limit dose estimates for the MHA.

The licensee's analysis used the following generally conservative assumptions:

- The source term represents the maximum curie inventory calculated in any fuel element during the full-power (1,000 EFPDs) extended operation at 250 kWt. Hence, the radioactive noble gas and halogen fission products are at saturation activity.
- No credit is given for decay time.
- The hottest fuel rod temperature is considered to be 300 °C (572 °F). This results in the application of a release fraction of 1.5×10^{-5} from the fuel to the gap.
- All of the gap activity is released to the reactor room and is instantly mixed uniformly with the air.
- All of the noble gases and I radioisotopes from the gap are assumed to be available for release to the environment through the emergency ventilation system.
- The most stable atmospheric class (Pasquill F) is assumed, with a wind speed of 1 m/s.
- It will take 5 minutes or less to evacuate persons from the reactor room and 1 hour or less to evacuate the areas in Rowland Hall in the vicinity of the reactor.

The licensee's analysis also assumes that the emergency ventilation system will be operating at a level where the ventilation system removes air from the reactor room at a minimum of 240 cfm (0.113 m^3/s). The reactor room emergency ventilation system discharges into the main building exhaust system at the roof level of Rowland Hall, which has three high plume dilution exhaust fans, each designed to create a flow of 80,000 cfm (37.8 m^3/s), further diluted to 128,000 cfm (60.4 m^3/s) with fresh air intake on the roof. The licensee stated that no backflow into the main building ventilation system is anticipated because of how the two ventilation systems are integrated and operational considerations (Ref. 64). The licensee reported a total exhaust flow of 384,000 cfm (181 m^3/s) above the building roof, but for conservatism in the analysis, only one building exhaust fan is considered to be operating during the analyzed scenario. Based on its review as discussed above, the NRC staff finds the licensee's method acceptable.

The occupational dose was calculated for an individual in the reactor room. The dose is also calculated for the public, which included a person at the nearest residence 656 ft (200 m) west of the UCINRF. In addition, the dose for the member of the public in the nonreactor area of Rowland Hall was calculated. The NRC staff reviewed the methodology and finds that the UCINRF analysis is consistent with the TRIGA® MHA methodology recommended in NUREG-1537 and adequate to calculate occupational and public radiation doses.

Occupational Exposures to Individuals Inside the Reactor Room

The licensee calculated the external dose rates and thyroid committed effective dose equivalent rates using the fission product release because of the MHA. The concentrations of radioisotopes were based on the fission product inventory uniformly released to the 6.6×10^{8}-cc volume of the reactor room. External dose conversion factors (DCFs) for submersion in air were obtained from Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 54). The DCFs for internal exposures were obtained from Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 55).

The licensee calculated the thyroid committed effective dose equivalent rate to be 4.9 rem/hr and the external dose rate from submersion to be 0.17 rem/hr, resulting in a total effective dose equivalent (TEDE) of 5.07 rem/hr. Therefore, the occupational TEDE to an individual as the result of inhalation and submersion from the radioactive species in the reactor room for a 5-minute exposure would be less than 422 mrem. This calculated occupational dose did not take credit for the plate-out of I radioisotopes and is within the regulatory limit given in 10 CFR 20.1201(a)(1)(i) of 5,000 mrem. The NRC staff finds that using a 5-minute exposure to UCINRF staff in the reactor room is reasonable based on the licensee's emergency planning zone and evacuation procedures. Table 4-1 below presents the licensee's results for the TEDE as well as the NRC staff's confirmatory TEDE calculations. The NRC staff reviewed the licensee's calculations and assumptions and finds that they are conservative and the methodology acceptable.

Potential Public Exposures to Individuals Outside the Rowland Hall Building

In the SAR and RAI response (Ref. 15), the licensee stated that the downwind dose calculations external to the building were performed using Gaussian plume dispersion relationships that are part of the "MELCOR Accident Consequence Code System (MACCS)," NUREG/CR-4691, Volume 2 (Ref. 56). The potential public exposure from the MHA scenario was calculated at the point of release above the roof of Rowland Hall and at the nearest resident to the UCINRF, which is at least 656 ft (200 m) west of the UCINRF.

The licensee calculated the exposures to an individual exposed to the rooftop and downwind concentrations respectively for infinite exposures. The licensee assumed that, at the time of the MHA, only one of the three exhaust fans was functioning. The TEDE for infinite exposure of an individual at the point of release was calculated to be 0.54 mrem. The TEDE for infinite exposure of an individual member of the public at the distance of the nearest residence was calculated to be 0.22 mrem. These doses are within the 100-mrem regulatory limit given in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

The NRC staff performed independent calculations for potential public exposure to confirm that the licensee's doses represent conservative values for the MHA. These calculations were performed using the same source term and radionuclide DCFs for inhalation and submersion dose as the licensee's calculation but with a more conservative dilution factor. The potential public TEDE was conservatively calculated for a member of the public located on the roof of Rowland Hall at the point of release above the building and is consistent with the licensee's calculation. The downwind calculation of the TEDE at the nearest resident for an infinite exposure was slightly lower than that calculated by the licensee. The NRC staff's results are in agreement with the licensee's results. Table 4-1 below presents the licensee's results for the TEDE as well as the NRC staff's confirmatory TEDE calculations.

Potential Public Exposures to Individuals in Rowland Hall Adjacent to the UCINRF

In the SAR and RAI response (Ref. 15), the licensee stated that because the reactor room is under negative pressure as compared to outside areas during normal and emergency operation of the ventilation system, there is no pathway for airborne radiation to reach individuals outside the UCINRF. As a result, the MHA in this scenario did not consider inhalation doses. The dose to a member of the public inside Rowland Hall adjacent to the UCINRF assumes that an individual will have only potential gamma-ray exposure from the radioactive source in the reactor room because the walls and windows are thick enough to absorb the beta radiation. The licensee assumed that a person outside the facility in the adjacent hallway at the "window" wall would be at least 5 ft (1.52 m) from the source.

The licensee further assumed that the external submersion dose rate experienced within the reactor room (51 mrem/hr from both beta and gamma sources) is a point source. Using a distance factor of 25, the licensee calculated a potential gamma dose rate to the individual outside the reactor room to be 2 mrem/hr, taking no credit for attenuation by wall materials or that the external submersion dose within the reactor room used as a premise of the calculation includes external beta dose as well.

Consistent with the emergency plan, the licensee stated that Rowland Hall can be evacuated in 15 to 30 minutes, even though persons adjacent to the UCINRF would be expected to be evacuated within 5 minutes. Thus, with a stay-time of 5 minutes, the same time as for the individuals in the reactor room, the estimated potential public exposure in the vicinity adjacent to the UCINRF would be 0.17 mrem. Even if it takes up to 1 hour to evacuate persons from Rowland Hall, only those nearest to the UCINRF would receive exposures approaching 2 mrem. These doses are well within the regulatory limits of 100 mrem as stated in 10 CFR Part 20. Table 4-1 presents the licensee's results for the TEDE as well as the NRC staff's confirmatory TEDE calculations.

Table 4-1 MHA Calculated TEDE

	UCI Dose Calculation Results (mrem)	NRC Staff Confirmatory Calculation Results (mrem)
Occupational	423	425
Public; at point of release	0.54	0.54
Public; nearest resident	0.22	0.13
Public; inside Rowland Hall	2	0.17

MHA Dose Calculation Conclusions

The NRC staff reviewed the MHA analysis for the scenarios presented, as well as the dose calculations and corresponding results, and finds that the licensee has used appropriate assumptions and analytical techniques and that its conclusions are appropriate and acceptable. The NRC staff's independent confirmatory dose calculations and results demonstrate that the licensee properly evaluated the postulated doses from the MHA scenarios. In addition, the doses from the postulated scenarios provided above demonstrate that the maximum TEDE doses were below the occupational limits in 10 CFR 20.1201, "Occupational Dose Limits for Adults," and the public exposure limits in 10 CFR 20.1301. Based on the results of the calculated doses provided above and by the NRC staff's independent calculations, the NRC staff concludes that the results of the licensee's MHA dose calculations meet the requirements in 10 CFR Part 20 and are acceptable.

4.1.2 Insertion of Positive Excess Reactivity

In the SAR, Section 13.3 (Ref. 20), the licensee provided its method for analyzing the positive reactivity insertion event. The licensee concluded that there was no credible means by which the control rods could be manipulated to promptly add reactivity without violating several conditions and procedures. The remaining credible option for the prompt insertion of positive reactivity was improper or unexpected movement of experiments comparable to a full pulse insertion made from steady-state power. The licensee analyzed the event in order to establish an extreme upper limit to the potential for fuel cladding stress at the facility. The licensee stipulated that the limiting accident is a rapid insertion of a large amount of reactivity ($3.00) into the reactor operating at full power by inserting the maximum worth of the two transient rods.

This $3.00 reactivity insertion corresponds to the maximum reactivity allowed in Technical Specification (TS) 3.8.1 (discussed in Section 5.3.8.1 of this safety evaluation report (SER)) that could be inserted by the unplanned removal of a secured experiment. The licensee noted that accidental pulsing of the transient rods while at full power would require the failure of the interlock that prevents the firing of the transient rods in the steady-state mode and the interlock that does not allow pulsing above a power level of 1 kWt. For this accident to occur, the operator would also have to fail to follow written procedures. The step reactivity insertion accident assumes an insertion of reactivity of $3.00 from the transient rod(s) at an initial reactor power level of 250 kWt. The measured fuel temperature for a B-ring fuel element at 250 kWt is 240 °C (464 °F). The licensee used the Fuchs-Nordheim model to calculate the temperature rise from a step reactivity insertion of $3.00. This calculation indicated that this reactivity insertion at full power would result in an increase in fuel temperature of 202.4 °C (396.3 °F). After accounting for a peak-to-average power ratio of approximately 2.0 and the temperature-dependent heat capacity of the fuel, the licensee calculated a final peak fuel temperature of 570 °C (1,058 °F), which is well below the safety limit (SL) of 1,000 °C (1,832 °F) given in TS 2.1 and the guidance provided in NUREG-1537 of 1,150 °C (2,102 °F). On this basis, the NRC staff finds that there are sufficient design features and administrative controls in place to make accidental pulsing or rapid removal of a secured experiment unlikely and that the SL would not be exceeded if either event were to occur.

Additionally, the licensee presented a scenario of a slow reactivity ramp insertion (Ref. 64) by assuming a continuous rod withdrawal malfunction of the most reactive control rod (SHIM) at low power. The event is terminated by the high-power scram. The high-power scram is assumed to occur 1 s after power reaches 275 kWt, which is 110 percent of the licensed power limit (250 kWt). This assumed event results in a maximum power level of 1.1 MW with a corresponding peak fuel temperature of 40.2 °C (104.4 °F). This power level is also below the 2.0-MWt value calculated by the licensee at which departure from nucleate boiling would occur.

The NRC staff reviewed the licensee’s methodology and assumptions for both scenarios and finds that the parameters are consistent with those provided in NUREG-1537. Furthermore, the NRC staff acknowledges that the Fuchs-Nordheim model employed by the licensee is a longstanding model used for this purpose. Both scenarios result in significant margin to the SL of 1,000 °C (1,832 °F) given in TS 2.1. Based on the information discussed above, the NRC staff concludes that the results of the insertion of excess reactivity scenario are acceptable.

4.1.3 Loss-of-Coolant Accident

The SAR, Section 13.4.1, and RAI response (Ref. 7) describe the licensee’s LOCA analysis. The licensee considered such a LOCA extremely unlikely because the reactor in-ground pool structure has no below-grade penetrations and because the coolant and purification system piping inlets and outlets are no more than 3 ft (1 m) below the surface of the pool. The licensee calculated the maximum fuel temperature resulting from a sudden LOCA by conservatively assuming continuous operation at 250 kWt with high-power density from using a 62-element core. In the licensee’s fuel temperature analysis, the water is assumed to be lost instantaneously after shutdown, and the hottest B-ring fuel element temperature was calculated to be less than 160 °C (320 °F) using a peak-to-average power ratio of 2.0. This fuel temperature was well below the TS 2.1 SL of 1,000 °C (1,832 °F); therefore, the LOCA is not expected to result in loss of cladding integrity. In addition, for standard 8.5/20 LEU fuel (TS 5.3.1, Specification a), the NRC staff previously accepted that no fuel damage is expected if the fuel temperature does not exceed 900 °C (1,652 °F) (Ref. 38) in cases where the cladding temperature is above 500 °C (932 °F). Although the cladding temperature is expected to stay below 500 °C (932 °F), the peak fuel temperature is below 900 °C (1,652 °F), so this limit is also met. Studies performed by Pacific Northwest Laboratory and GA for TRIGA® fuels support these conclusions (Refs. 51 and 53). The studies show that, in general, as long as the operating power is less than 1.5 MWt, the cladding should not be breached during a LOCA.

The SAR, as supplemented, provides radiation dose rates from the loss of all shielding water. The licensee estimated direct radiation doses at the top of the pool and classroom floor above the reactor after extended operations at 250 kWt. The calculated dose rates from 0.1 hour to 1 month after shutdown are given in Table 13-2 in an RAI response (Ref. 7). After shutdown for 24 hours, if the core is completely uncovered, the calculated dose rates are 200 rads/hr at the top of the pool and 0.48 rad/hr at the classroom floor above the reactor. The direct gamma dose rates from the unshielded core at the reactor bridge and areas directly above the core would require evacuation of the reactor room, Rowland Hall, and exclusion of the public from the vicinity of the facility boundary as required by the emergency plan. These precautions would help ensure that the 10 CFR Part 20 dose limits to the UCINRF staff and the public are satisfied. The NRC staff finds the LOCA estimated dose results and the required actions following such an event are acceptable to maintain reasonable safety and security of the public and the environment.

The licensee stated that instrumentation and the UCINRF staff will detect any tank failures because of corrosion or other failures that would lead to a slow loss of water. A drop in pool level equivalent to 480 gal (1,820 L), which is 2 percent of total tank capacity, will cause an alarm as described in Section 2.3 of this SER. A low-level alarm requires immediate corrective action by the UCINRF staff. TS 3.3.1, Specification b, requires the alarm be observable in the reactor control room and at a continuously monitored, remote location. The licensee estimated that it would take approximately 19 hours to completely drain the pool, assuming a worst-case scenario in which the pool drains directly to the soil below the reactor. The licensee stated that 19 hours is considered sufficient time to enact emergency procedures and evacuate people from the vicinity of the reactor. The NRC staff finds that it is reasonable for the licensee to use a firehose to provide replacement water to the reactor pool within 19 hours.

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

4.1.4 Loss of Coolant Flow

The SAR, Section 3.5, and RAI response (Ref. 4) discuss the LOCF. Because the UCINRF uses natural convection cooling, the geometry and design of the reactor core make it highly unlikely that local fuel element flow blockages would occur within the reactor core.

The University of California, Irvine (UCI), reactor is in a 23,000-gal (87,000-L) water-filled tank. The reactor assembly is cooled by natural convection using the pool water and the water in the primary cooling path. Heat generated from the reactor core is directly transferred to the pool water. Heat is removed from the pool by natural convection to the air of the reactor room at the surface of the pool and through a primary cooling path that passes pool water through a heat exchanger. The heat exchanger has a capacity of 880,000 British thermal units per hour with chilled water supplied by the UCI Central plant as the ultimate heat sink. Although the heat exchanger is maintained available, the reactor was designed to operate without any additional cooling capacity. A loss of heat exchanger cooling results in a slow increase in the temperature of the pool water, which is monitored by the operators, and results in the termination of reactor operations before exceeding the limit in TS 3.3.2 of 25 °C (77 °F). At 250 kWt steady-state power, the bulk pool water temperature increases adiabatically at a rate of 0.037 °C/min (.067 °F/min). The NRC staff finds that this slow rate of bulk water temperature increase would allow ample time for the UCINRF operators to identify and evaluate the condition, then implement corrective actions.

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

The NRC staff reviewed the description of the grid plates described in the SAR, Section 4.4.2, which indicates the cooling water passes through the differential area between the triangular spacer block on the top of each fuel element and the round holes in the grid plate. In addition, the grid plate provides spacing between the fuel elements. The NRC staff finds that, in the event of a possible blockage of a coolant channel created by a foreign object lodged in the grid plate, the open fuel element lattice would ensure sufficient continued cooling of all fuel elements because of crossflow.

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

4.1.5 Mishandling or Malfunction of Fuel

SAR Section 13.6 discusses the established procedures for handling fuel. Mishandling or malfunction of TRIGA® LEU fuel elements could result in physical damage to the fuel, although no damage to fuel has ever occurred at the UCINRF. TS 3.1.6 and TS 4.2 require surveillance of fuel rods by visual inspection and measurement in order to verify continuing integrity of the fuel rod cladding.

In the unlikely event of such a failure in air or water, the event consequences would be bounded by the results of the MHA scenario (discussed in Section 4.1.1 of this SER). The licensee has established procedures for handling fuel. The NRC staff reviewed the licensee's postulated mishandling or malfunction of fuel accident scenario and concludes that any potential fission product release resulting from mishandling of the fuel would be less than computed for the MHA analysis.

4.1.6 Experiment Malfunction

SAR Section 10.0, Section 12.0, and Section 13.7, discuss postulated experimental malfunction accidents. TS 3.8, TS 4.8, and TS 6.5 specify procedures, requirements, and limitations regarding the control and review of all reactor experiments. The licensee's review process of a proposed experiment includes a safety analysis that assesses the complete range of safety issues, such as the generation of radionuclides; the reactivity worth of the experiment; material properties such as chemical, physical, explosive, and corrosive characteristics of each experiment; and potential failures and malfunctions.

TS 6.5 requires the review and approval of all new experiments, during which the limits on experiments are analyzed and approved.

TS 3.8.1 helps prevent a step change in reactivity greater than $1.00 for any unsecured or moveable experiment and $3.00 for all experiments, values which are within the reactivity addition limit of TS 3.1.3. Potential reactivity malfunctions are bounded by the discussion of the insertion of reactivity accident (discussed in Section 4.1.2 of this SER).

TS 3.8.2, Specification b, establishes the requirement to limit using explosive material in the UCINRF to 25 mg of trinitrotoluene (TNT) equivalent and states that quantities less than or equal to 25 mg may be irradiated provided that the pressure produced in the experiment container shall be demonstrated to be less than half the design pressure of the container. The NRC staff finds this consistent with the guidance in Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” issued November 1973 (Ref. 57) and NUREG-1537.

TS 3.8.2, Specification c, limits the introduction of corrosive materials into the reactor coolant where it may damage reactor components by requiring double encapsulation. This TS helps ensure that a highly unlikely double encapsulation failure is required to have a release of such material into the coolant system and is consistent with the guidance in NUREG-1537. Additionally, this TS helps ensure that an anticipated failure of an experiment with corrosive material will not result in an unacceptable consequence.

TS 3.8.3 limits the quantity and type of material in the experiment so that, in the case of experiment failure, the airborne radioactivity in the reactor room or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR Part 20.

Based on the UCINRF TS limits for quantity and type of materials allowed in an experiment, the NRC staff concludes that the licensee’s consequences of an experiment malfunction leading to a radiological release are consistent with the guidance in NUREG-1537. Additionally, failures that could result in potential fuel damage will be bounded by the MHA analysis evaluated in Section 4.1.1 of this SER. Based on the information provided above, the NRC staff concludes that the performance of experiments within the restrictions of the TSs provides reasonable assurance that experiment malfunctions do not pose an undue risk to public health and safety, UCINRF staff, or the environment and are acceptable.

4.1.7 Loss of Normal Electric Power

SAR Section 13.8 evaluates the scenario for a postulated UCINRF accident involving the loss of normal electric power. UCINRF does not require emergency backup electric power to safely shut down the reactor, to maintain the reactor in a shutdown condition indefinitely, and to maintain long-term core cooling. The loss of normal electric power will cause the reactor to shut down through loss of voltage to the control rod drive mechanism and a reactor scram, as required by TS 3.2.3. Shutdown of the reactor during a loss of power is caused by the design of the reactor control system. The standard control rod system uses a magnetic coupling of the rods to the drive mechanisms such that a loss of power results in release and gravity-induced insertion of the control rods. The transient rods are inserted into the core by gravity on a loss of power because of pressure release from the pneumatic piston caused by the power-off position of the pneumatic solenoid valves. The loss of electric power would result in stopping the primary pumps and the heating, ventilation, and air conditioning (HVAC). However, reactor decay heat would be dissipated through natural circulation in the reactor pool, and the loss of HVAC function has no effect on the reactor pool. After reactor shutdown, the pool water has sufficient capacity to remove the generated decay heat indefinitely.

Although not required to safely shut down the reactor or to ensure public health and safety, the UCINRF has an emergency generator in Rowland Hall that provides backup electric power to radiation monitors and alarms, emergency lighting, and other equipment if normal electric power is lost.

The NRC staff reviewed the information in the SAR, as supplemented, that supports the licensee's postulated loss-of-electric-power event and determines the analysis to be acceptable. Based on the information provided above, the NRC staff concludes that loss of normal electric power poses little risk to the health and safety of the public or to the UCINRF staff.

4.1.8 External Events

The SAR, Sections 2.0 and 13.9, and RAI responses (Refs. 4, 12, and 13) describe the analysis of the potential effect to the UCINRF from external events. Floods, hurricanes, and tornadoes are not credible in the city of Irvine area and, therefore, are not considered to pose a threat to the reactor. The Los Angeles basin area of southern California has a history of considerable seismic activity, and many fault zones exist; therefore, buildings and structures are assembled to a high standard that must adhere to State regulatory requirements. Rowland Hall was upgraded, since originally built, to comply with later building codes. The UCINRF conforms to the Uniform Building Code Seismic Zone 4. This is the highest classification of the four zones in the United States, with the most stringent requirements for building design. The licensee stated that, traditionally, the Irvine region was considered relatively stable regarding seismic activity. Recent studies in 2009 by the city of Irvine and map overlay evaluation of faults by the State of California (Ref. 4) indicate that the UCINRF immediate locale is not identified as being subject to unusual hazards from landslides or liquefaction. The licensee updated the reactor safety system to include a seismic switch, required by TS 3.2.3, which will automatically shut down the reactor if motion of 3-percent gravity (g) (0.03g) is exceeded. The licensee provided updated seismic information in RAI responses that state that the general conclusion regarding levels of risk as presented at the time of original licensing has not changed. In an earthquake with significant severity, the consequences to the UCINRF are not expected to be more severe than the MHA and LOCA, since the design of the reactor pool limits the drain down rate and specific surveillances are used for early leak detection. Any fuel failure would most likely occur in water, which would reduce source terms as compared to the MHA.

Based on the information contained in the SAR, as supplemented, the NRC staff finds that severe storms, floods, and tornadoes are very unlikely for the area around the UCINRF site. The building, reactor foundation, shielding structure, reactor tank, and core support structure were designed in accordance with Uniform Building Code Seismic Zone 4 requirements. Meeting these requirements helps ensure that the reactor can be safely shut down following an earthquake likely to occur during the facility's lifetime. On this basis, the NRC staff concludes that the consequences of external events are bounded by the accidents discussed in this chapter of the SER, including the MHA analysis for fission product releases, and are acceptable.

4.1.9 Mishandling or Malfunction of Equipment

The SAR, Section 13.10, evaluates the potential mishandling or malfunction of equipment. The licensee stated that the reactor design includes appropriate control system interlocks and automatic protective circuits. TRIGA® fuel is designed to accept large-step reactivity insertion events without the loss of clad integrity. Therefore, events caused by operator errors during reactor operation would most likely result in reactor shutdown.

The NRC staff reviewed the licensee's postulated mishandling- or malfunction-of-equipment accident scenario and finds that fuel damage is unlikely. Based on its review, the NRC staff concludes that the consequences of mishandling or malfunction of equipment pose negligible risk to the health and safety of the public or UCINRF staff.

4.2 Conclusions

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

- The licensee has considered the expected consequences of a sufficiently broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a loss of integrity of fuel-element clad and a release of fission products.
- The licensee has performed analyses of the most serious credible accidents and the MHA. The licensee has used conservative assumptions in evaluating occupational and public exposure from releases resulting from an MHA. The MHA will not result in a radiation exposure to the UCINRF staff or to the public in excess of the applicable U.S. Nuclear Regulatory Commission limits in 10 CFR Part 20.
- The licensee has reviewed the postulated accident scenarios provided in NUREG-1537 and has not identified any other fission product release accidents having consequences not bounded by the MHA.
- The licensee has employed appropriate methods for accident analysis and consequence analysis.
- The administrative limit for pulse reactivity is conservatively specified as $3.00. For accidents involving insertion of excess reactivity, the licensee has demonstrated that a reactivity limit of $3.00 will result in a peak fuel temperature below the safety limit of 1,000 °C (1,832 °F) in TS 2.1. An insertion of excess reactivity resulting from the uncontrolled withdrawal of an experiment is limited to $3.00 by TS 3.8.1 and, therefore, does not pose a threat to fuel integrity.
- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures.
- Doses calculated from all credible accidents are below the limits of 10 CFR Part 20.
- The accident analysis for the UCINRF establishes the acceptability of the limiting core configuration defined and analyzed in the SAR, as supplemented.
- The accident analysis confirms the acceptability of the licensed power of 250 kWt, including the response to anticipated transients and accidents.

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

The NRC staff reviewed the radiation source term and MHA calculations for the UCINRF. The NRC staff finds the calculations, including the assumptions, demonstrate that the source term assumed and other boundary conditions used in the analyses are acceptable. The doses to the public and UCINRF staff are in conformance with the requirements in 10 CFR Part 20.
The UCINRF design features and administrative controls found in the TSs prevent the initiation of accidents and mitigate any consequences. On the basis of its review, the NRC staff concludes that there is reasonable assurance that no credible accident would cause significant radiological risk and the continued operation of the UCINRF poses no undue risk to the UCINRF staff, the public, or the environment.

SECTION 5

TECHNICAL SPECIFICATIONS

In this section of the report, the NRC staff provides its evaluation of the licensee's proposed technical specifications (TSs). The University of California, Irvine Nuclear Reactor Facility (UCINRF), TSs define specific features, characteristics, and conditions required for the safe operation of the UCINRF. The TSs are explicitly included in the renewal license as Appendix A. The NRC staff reviewed the format and content of the TSs for consistency with the guidance in NUREG-1537, Part 1, Chapter 14, and Appendix 14.1, and ANSI/ANS-15.1-2007, "The Development of Technical Specifications for Research Reactors." The NRC staff specifically evaluated the content of the proposed TSs to determine if they meet the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical Specifications." The NRC staff also relied on NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," and the interim staff guidance (Ref. 27) to perform its review.

5.1 Technical Specification Definitions

The licensee proposed the following definitions to be generally consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's proposed TSs include minor modifications to, and some additional facility-specific, definitions.

TS 1, Definitions, states:

> DEFINITIONS
>
> The following frequently used terms are defined to aid in the uniform interpretation of these specifications.
>
>> AUDIT is an examination of records, logs, procedures, or other documents to ascertain that appropriate specifications and guidelines are being followed in practice. An audit report is written to detail findings and make recommendations.
>>
>> CHANNEL is a combination of sensor, lines, amplifier and output device which are connected for the purpose of measuring the value of a parameter.
>>
>> 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.
>>
>> CHANNEL CHECK is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

CHANNEL TEST is an introduction of a signal into the channel to verify that it is operable.

CLOSE-PACKED ARRAY is a fuel loading pattern in which the fuel elements are arranged in the core by filling the inner rings first.

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

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

a. Regulating (REG): a rod having electric motor drive and scram capabilities. Its position may be varied manually or by an electronic controller. It shall have a fueled-follower section.
b. Shim (SHIM): a rod having electric motor drive and scram capabilities. Its position shall be varied manually. It shall have a fueled-follower section.
c. Adjustable Transient (ATR): a rod with scram capabilities that can be rapidly ejected from the reactor core using a pneumatic drive to produce a pulse. It has an electric motor drive to adjust its position or length of travel. It shall have a void follower.
d. Fast Transient (FTR): a rod with scram capabilities that can be rapidly ejected from the reactor core using a pneumatic drive to produce a pulse. Only fully UP or DOWN positions are available. It shall have a void follower.

CORE CONFIGURATION is a particular arrangement of fuel, control rods, graphite reflector elements, and experimental facilities inserted within the core grid plates.

CORE LATTICE POSITION is defined by a particular hole in the top grid plate of the core designed to hold a standard fuel element. It is specified by a letter, indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

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

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

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

b. UNSECURED EXPERIMENT is any experiment or component of an experiment that does not meet the definition of a secured experiment.

c. MOVEABLE EXPERIMENT is any experiment where it is intended that the entire experiment may be moved in or near the core or into or out of the core while the reactor is operating.

FUEL ELEMENT is a single TRIGA® fuel rod.

INITIAL STARTUP is the first start-up from reactor secured condition on any day when the reactor is to be operated in order to verify core excess and other instrument parameters.

INSTRUMENTED FUEL ELEMENT is an element in which one or more thermocouples are embedded for the purpose of measuring fuel temperature during reactor operation.

IRRADIATION FACILITIES are pneumatic transfer systems, central tube, rotary specimen rack, and the in-core facilities (including single element positions, three-element positions, and the seven element position) and any other facilities in the tank designed to provide locations for neutron or gamma ray exposure of materials.

MEASURED VALUE is the value of a parameter as it appears on the output of a channel.

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

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

OPERATIONAL CORE is a core configuration that meets all license requirements, including Technical Specifications.

PULSE MODE is any operation of the reactor with the mode switch in the PULSE position that satisfies all instrumentation and license requirements, including technical specifications, for pulse operation of the reactor.

REACTIVITY WORTH OF AN EXPERIMENT is the value of the reactivity change that results from the experiment being inserted or removed from its intended position.

REACTOR FACILITY is the physical area defined by rooms B64, B64A, B54, B54A, and B54B in the service level of Rowland Hall on the campus of the University of California Irvine.

REACTOR OPERATING is any time at which the reactor is not secured or shutdown.

REACTOR SAFETY SYSTEMS are those systems, including their associated input channels that are designed to initiate automatic reactor scram or to provide information for the manual initiation of a scram for the purpose of returning the reactor to a shutdown condition.

REACTOR SECURED is when:

Either

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

Or

(2) the reactor is shutdown and the following conditions exist:

   a. all neutron-absorbing control rods are fully inserted; and

   b. the console key switch is in the OFF position and the key is removed from the console lock; and

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

   d. no experiments are being moved or serviced that have a reactivity worth exceeding $1.00.

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

REFERENCE CORE CONDITION is when the core is at ambient temperature (cold) and the reactivity worth of xenon is zero.

REVIEW means a qualitative examination of audits, reports and records, procedures or other documents from which appropriate recommendations for improvements are made.

RING is one of six concentric bands in the grid locations surrounding the central opening of the core. The rings are designated by the letters B through G, with the letter B used to designate the innermost band.

SAFETY CHANNEL is a measuring channel in the reactor safety system.

SCRAM TIME is the elapsed time between the initiation of a scram signal and all control rods reaching their bottom limit.

SEVEN ELEMENT POSITION is a hexagonal section which can be removed from the upper grid plate for insertion of specimens up to 4.4 in in diameter after relocation of all six B-ring elements and removal of the central tube irradiation facility.

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.

SHUTDOWN MARGIN refers to 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 its most reactive position, and will remain subcritical without further operator action.

STEADY-STATE MODE is whenever the reactor is operating with the mode selector switch in the STEADY-STATE position.

SURVEILLANCE INTERVALS that are permitted are established as follows:

a. Quinquennial—interval not to exceed 6 years
b. Biennial—interval not to exceed 2-1/2 years
c. Annual—interval not to exceed 15 months
d. Semi-annual—interval not to exceed 7-1/2 months
e. Quarterly – interval not to exceed 4 months
f. Monthly—interval not to exceed 6 weeks
g. Daily—each day when the reactor is to be operated or before any operation extending more than one day

THREE ELEMENT POSITION is one of two triangular-shaped removable sections of the upper grid plate, one encompassing core lattice positions D5, E6 and E7 and the other D14, E18 and E19, designed to accommodate experiments. When fuel elements are placed in these locations, a special fixture shall be inserted to provide lateral support.

These definitions are either facility specific or are standard definitions used in research reactor TSs, consistent with NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that the licensee's TSs definitions are acceptable.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 TS 2.1 Safety Limit—Fuel Element Temperature

This TS is evaluated in Section 2.5.3 of this safety evaluation report (SER) and is acceptable.

5.2.2 TS 2.2 Limiting Safety System Settings

This TS is evaluated in Section 2.5.3 of this SER and is acceptable.

5.3 Limiting Conditions for Operation

5.3.1 Reactor Core Parameters

5.3.1.1 TS 3.1.1 Steady-State Operation

This TS is evaluated in Section 2.5.1 of this SER and is acceptable.

5.3.1.2 TS 3.1.2 Shutdown Margin

This TS is evaluated in Section 2.5.1 of this SER and is acceptable.

5.3.1.3 TS 3.1.3 Core Excess Reactivity

This TS is evaluated in Section 2.5.1 of this SER and is acceptable.

5.3.1.4 TS 3.1.4 Pulse Mode Operation

This TS is evaluated in Section 2.5.1 of this SER and is acceptable.

5.3.1.5 TS 3.1.5 This Section Intentionally Left Blank

The licensee left this section blank to allow for format consistency with NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff confirmed that the TS recommended in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.4, Reactor Core Parameters, are otherwise included. On this basis, the NRC staff finds this acceptable.

5.3.1.6 TS 3.1.6 Fuel Element Inspection Parameters

This information is in Section 2.2.1 of this SER and is acceptable.

5.3.1.7 TS 3.1.7 Core Configuration

TS 3.1.7, Core Configuration, states:

Specification(s).

a. The core shall be an arrangement of TRIGA 8.5/20 Low Enriched Uranium (LEU) fuel.

b. The core fuel elements shall include at least one 8.5/20 LEU fuel element with embedded thermocouples to enable monitoring of fuel element temperature.

c. The core fuel elements shall be kept in a close-packed array except for control rods, single- or three-element or seven-element positions occupied by in-core experiments, irradiation facilities (including transfer system termini), and a central dry tube.

d. The reflector, excluding experiments and experimental facilities, shall be graphite or a combination of graphite and water.

e. A control rod shall not be manually removed from the core unless calculations show that the core will be subcritical by ≥$0.55 excluding the worth of the rod being worked on and the worth of the most reactive remaining control rod.

TS 3.1.7, Specification a, limits the fuel in the core to Standard 8.5/20 low-enriched uranium (LEU) Training, Research, Isotopes, General Atomics (TRIGA®) fuel rods in order to help ensure that only the type of TRIGA® fuel elements that were analyzed in the safety analysis report (SAR) are used in the reactor core.

TS 3.1.7, Specification b, requires using at least one instrumented fuel element (IFE) which is necessary to help ensure proper temperature monitoring and to avoid exceeding the analyzed safety limit (SL). TS 2.2 identifies the core lattice position of the IFE.

TS 3.1.7, Specification c, helps ensure that there are no unoccupied grid positions in the central core lattice, thus controlling power peaking in the fuel elements to values assumed in the SAR. The specification also controls the location of experimental locations in the reactor core.

TS 3.1.7, Specification d, specifies the materials of the reflector to help ensure that reflectors used are those identified and found acceptable by the SAR so that predictable reflection of the core is provided.

TS 3.1.7, Specification e, is precautionary and requires the core to be subcritical by greater than the shutdown margin (SDM) of $0.55 without crediting the positions of the control rods before performing any manual control rod removal. This precaution helps prevent inadvertent criticality.

The NRC staff finds that TS 3.1.7, Specifications a through e, provide sufficient restrictions in arrangement of fuel elements and experiments to help ensure that excessive power densities will not be produced based on the analysis in the SAR and request for additional information (RAI) responses (Refs. 10, 11, and 20). The NRC staff finds that TS 3.1.7 is consistent with the guidance provided in NUREG-1537. On the basis of the information provided above, the NRC staff concludes that TS 3.1.7 is acceptable.

5.3.2 Reactor Control and Safety Systems

5.3.2.1 TS 3.2.1 Control Rods

This TS is evaluated in Section 2.2.2 of this SER and is acceptable.

5.3.2.2 TS 3.2.2 Reactor Measuring Channels

TS 3.2.2, Reactor Measuring Channels, states:

> Specification(s). The reactor shall not be operated in the specified mode unless the measuring channels described in Table 1 are operable.

Table 1. Minimum Measuring Channels

Measuring Channel	Operating Mode	
	Steady-state	Pulse
Fuel Element Temperature	1	1
Linear Power Level	1	-
Log Power Level	1	-
Power Level (%)	1	1 (peak power)
nvt circuit (integrated pulse energy)	-	1

> Note 1. Any single power level channel may be inoperable while the reactor is operating solely for the purpose of calibration and/or channel tests or checks on that channel.
>
> Note 2. Any single power level channel that is not required for safety scram purpose by TS 3.2.3 and ceases to be operable during reactor operation shall be returned to operating condition within 5 minutes or the reactor shall be shut down. For channels required by TS 3.2.3 the reactor shall be shut down immediately if the channel becomes inoperable.

SAR Section 7.2 describes the reactor core power measuring channels as using three types of neutron detectors: a fission counter, a compensated ion chamber, and an uncompensated ion chamber. The logarithmic (log) power level channel uses the signal from the fission counter; it has an adjustable trip level to assure sufficient neutrons are present for startup. The linear power level channel uses the compensated ion chamber, which provides a scram function as well as an additional adjustable bistable trip used for a “PULSE” interlock to prevent activating pulse mode circuitry for the reactor if the measured linear power level is too high. The power level percent channel uses the uncompensated ion chamber, which provides a scram function.

Note 1 provides the necessary conditions in order to perform surveillance for performing channel checks, channel tests, and channel calibrations with the reactor in operation. This is acceptable for the period of time required to perform the necessary surveillance as a redundant scram channel is available to provide a safety signal, if necessary. Note 2 provides a 5-minute period for the operator under the stated conditions to diagnose the cause of a loss of indication from a required measuring channel. Note 2 helps to avoid unnecessary reactor shutdowns if the cause of the loss of indication can be determined and corrected within the 5-minute period.

TS 3.2.2 states the reactor shall not be operated unless the measuring channels described in Table 1 are operable. The NRC staff finds that the TS 3.2.2 is consistent with the guidance in NUREG-1537, Appendix 14.1, Section 3.2(8), by providing redundant power measurement capability to the UCINRF. On the basis of the information provided above, the NRC staff concludes that TS 3.2.2 is acceptable.

5.3.2.3 TS 3.2.3 Reactor Safety System

TS 3.2.3, Reactor Safety System, states:

> Specification(s). The reactor shall not be operated unless the safety system channels described in Table 2 and the interlocks described in Table 3 are operable in the appropriate operating modes.

Table 2. Minimum Reactor Safety Channels

Safety Channel	Function and trip level setting	Operating Mode	
		Steady-state	Pulse
Fuel Element Temperature	Scram—≤425 °C (IFE)	1	1
Reactor Power level	Scram—≤110% of 250 kWt (≤275 kw)	2	-
Loss of HV on any power measuring channel	Scram	3	1
Manual Bar	Scram	1	1
Preset Timer	Scram pulse rods <15 seconds after pulse	-	1
Seismic Switch	Scram—if motion of ≥3% g (≤0.03g)	1	1
Pool Water Temperature	Manual Scram if >25 °C	1	1

Table 3. Minimum Interlocks

Interlock	Function	Operating Mode	
		Steady-state	Pulse
Wide Range Power Level Channel (Log)	Prevent control rod withdrawal when power level is $\leq 1\times10^{-7}$% of full power	1	-
REG, SHIM, ATR Control Rod Drives	Prevent application of air to fast transient rod when all other rods are not fully inserted	1	-
REG, SHIM, ATR Control Rod Drives	Prevent simultaneous withdrawal of more than one rod	1	-
REG, SHIM, ATR Control Rod Drives	Prevent movement of REG and SHIM rods and ATR drive in pulse mode	-	1
ATR Cylinder Drive	Prevent application of air to adjustable transient rod unless cylinder is fully down	1	-
Wide Range Linear Power Channel	Prevent ATR or FTR withdrawal unless power level ≤1 kilowatt	-	1

TS 3.2.3 establishes requirements to ensure that the minimum number of safety channels, interlocks, and their associated setpoints specified in TS Tables 2 and 3 are operable when the reactor is operating.

TS 3.2.3, Table 2, Minimum Reactor Safety Channels, provides the safety channels and their functions and trip setpoints. The SAR, as supplemented, describes that the Reactor Power Level scram at less than or equal to 110 percent of 250 kWt, or 275 kWt, and the Fuel Element Temperature scram at less than or equal to 425 °C (797 °F)) are redundant, diverse, and provide protection of the fuel SL. The fuel element temperature scram is at or below the limiting safety system setting (LSSS) setpoint. The power level scram setpoint less than or equal to 275 kWt is shown and discussed in Section 2.5.1 of this SER not to exceed the fuel temperature LSSS. These scrams help ensure that the reactor will be shut down before the SL on the fuel rod temperature will be exceeded. The setpoints are bounded by the SAR neutronics and thermal-hydraulics analyses performed for the limiting core configuration (Refs. 9-11). Acceptable SAR results are provided for the departure from nucleate boiling ratio, control rod worth, and radiation dose resulting from normal operation and accident analyses. The NRC staff finds that these limiting conditions for operation (LCO) for the Reactor Power Level and the Fuel Element Temperature setpoints are consistent with the guidance provided in NUREG-1537, Appendix 14.1, Section 3.2(4). On the basis of the information provided above, the NRC staff concludes that the TS 3.2.3 LCO scram setpoints of 275 kWt and 425 °C (797 °F) are acceptable.

The LCO setpoint for the automatic trip on loss of high voltage (HV) to the three detectors is provided in the SAR. This LCO helps ensure that the accuracy of reactor core flux measurement instruments that provide an input to the power level scram is maintained. The

NRC staff finds that the LCO setpoint for the automatic trip on loss of HV is consistent with the guidance in NUREG-1537. On the basis of the information provided above, the NRC staff concludes that TS 3.2.3 LCO on the loss of HV is acceptable.

TS 3.2.3, Table 2, also includes the following LCO:

- Manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition occurs.
- The preset timer for pulse operations ensures that the reactor power level will reduce to a low level after pulsing.
- The seismic switch shuts down the reactor in the event of major earth movement.
- Bulk pool coolant temperature indication is provided to alert the operator to scram the reactor when the bulk pool temperature is greater than 25 °C (77 °F). This value is consistent with pool coolant temperature used in the thermal-hydraulic analyses.

The NRC staff finds that these LCO are conservative and consistent with the guidance provided in NUREG-1537. On the basis of the information provided above, the NRC staff finds that TS 3.2.3, Table 2, "Minimum Reactor Safety Channels," is, therefore, acceptable.

TS 3.2, Table 3, Minimum Interlocks, provides interlock LCO and functions. The function of the Wide Range Power Level Channel (Log) interlock is to prevent control rod withdrawal with a neutron-induced signal less than 10^{-7} percent of full power. This interlock helps ensure that the Wide Range Power Level Channel is on scale before reactor startup. A REG, SHIM, ATR Control Rod Drives circuit interlock prevents the simultaneous withdrawal of two control rods. This limits the amount of reactivity change introduced by withdrawing the control rods. The NRC staff finds that these circuit interlocks are typical for TRIGA® facilities, are appropriate to UCINRF operation, have been properly considered in the SAR, are supported by appropriate bases, and are consistent with the guidance provided in NUREG-1537. On the basis of the information provided above, the NRC staff finds that the Wide Range Power Level Channel (Log) interlock and the REG, SHIM, ATR Control Rod Drives circuit interlock are acceptable.

TS 3.2.3, Table 3, also includes the following LCO:

- An interlock prevents the application of air to the fast transient rod (FTR) air cylinder in steady-state operating mode unless all other control rods are fully inserted to limit movement of the FTR to subcritical core conditions. This is because the FTR is either fully inserted or withdrawn from the core when used in steady-state mode.
- An interlock to prevent the initiation of a pulse above 1 kWt is to ensure that the magnitude of the pulse will not cause the fuel element temperature SL to be exceeded.
- An interlock to prevent withdrawal of the standard or regulating control rods or the adjustable transient rod (ATR) drive in the pulse mode is to prevent the reactor from being pulsed while on a positive period.
- An interlock to prevent introduction of air into the ATR cylinder in steady-state operating mode unless the ATR cylinder is fully down is to prevent inadvertent pulsing.

The NRC staff finds that these LCO are conservative and consistent with the guidance provided in NUREG-1537. On the basis of the information provided above, the NRC staff concludes that TS 3.2.3, Table 3, "Minimum Interlocks," is, therefore, acceptable.

5.3.3 TS 3.3 Coolant Systems

5.3.3.1 TS 3.3.1 Pool Water Level

This TS is evaluated in Section 2.3 of this SER and is acceptable.

5.3.3.2 TS 3.3.2 Pool Water Temperature

This TS is evaluated in Section 2.3 of this SER and is acceptable.

5.3.3.3 TS 3.3.3 Pool Water Conductivity

This TS is evaluated in Section 2.3 of this SER and is acceptable.

5.3.3.4 TS 3.3.4 Pool Water Radioactivity

This TS is evaluated in Section 2.3 of this SER and is acceptable.

5.3.4 TS 3.4 This Section Intentionally Left Blank

The licensee left this section blank. The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.4, Containment or Confinement, indicates that the licensee should specify the pressure or flow rate needed to maintain confinement. Because this licensee put these limits in TS 3.5, Ventilation, there was no need to repeat the specifications in TS 3.4. TS 3.5 is evaluated in Section 5.3.5 of this SER.

5.3.5 TS 3.5 Ventilation System

5.3.5.1 TS 3.5.1 Ventilation System

TS 3.5.1, Ventilation System, states:

> <u>Specification(s).</u>
>
> a. The reactor shall not be operated unless the ventilation system is operating as indicated by:
>
>   i. a minimum of 0.10 inches of water negative pressure difference between the reactor room and the control room and between the reactor room and the air outside the building; and
>
>   ii. a minimum total exhaust flow rate from the reactor area of 3600 cubic feet per minute (CFM) is present.
>
>   Note: The ventilation system may be inoperable for periods of time not to exceed two hours to allow repair, maintenance or testing of the system. During such an exception, no pulses shall be fired.
>
> (…)

TS 3.5.1, Specification a, helps ensure that the ventilation system is operable, as described in the SAR, and is capable of limiting the exposure to the UCINRF staff and public in the event of a radiological release (Ref. 15). Brief periods of system inoperability for the purpose of repair, maintenance, or testing of the ventilation system is acceptable because the capability of the continuous air monitor system in detecting a release of radioactivity is not reduced. The requirement not to pulse while the ventilation system is undergoing repair further reduces the remote likelihood of fuel element failure during such times. The NRC staff finds that TS 3.5.1 helps ensure that the ventilation system is operable when a potential for radiological release is present and that the ventilation setting is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 3.5.1 is acceptable.

TS 3.5.1, Specification b and Specification c, are evaluated in Section 3.1.4 of this SER and are acceptable.

5.3.5.2 TS 3.5.2 Ventilation during Emergency Situations

This TS is evaluated in Section 3.1.4 of this SER and is acceptable.

5.3.6 TS 3.6 This Section Intentionally Left Blank

The licensee left this section blank. The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.6, Emergency Power, states that emergency electric power should be analyzed in the SAR on a case-by-case basis. The licensee stated that the emergency electric power system is not necessary to safely shut down the reactor and is not required to ensure protection of public health and safety. This is evaluated in Section 1.10 of this SER and is acceptable.

5.3.7 TS 3.7 Radiation Monitoring Systems and Effluents

5.3.7.1 TS 3.7.1 Radiation Monitoring Systems

This TS is evaluated in Section 3.1.1.1, Section 3.1.4, and Section 3.1.7 of this SER and is acceptable.

5.3.7.2 TS 3.7.2 Effluents

This TS is evaluated in Section 3.1.1.1 and Section 3.1.1.2 of this SER and is acceptable.

5.3.8 TS 3.8 Limitations on Experiments

5.3.8.1 TS 3.8.1 Reactivity Limits

TS 3.8.1, Reactivity Limits, states:

> Specification(s). The reactor shall not be operated unless the following conditions governing reactivity worths exist:
>
> a. the reactivity worth of any unsecured or moveable experiment shall not exceed $1.00, and

b. the reactivity worth of an individual experiment shall not exceed $3.00, and

c. the sum of absolute values of reactivity worths of all experiments shall not exceed $3.00.

In TS 3.8.1, Specification a, the reactivity limit of less than $1.00 for a single unsecured or moveable experiment is designed to prevent inadvertent criticality from occurring and is substantially below the maximum allowable pulse size of $3.00. If this amount of reactivity is inadvertently added to the reactor, it will not have an unacceptable effect on the reactor system as demonstrated in the analysis of excess reactivity insertion in the SAR, Chapter 13.3.

TS 3.8.1, Specification b, establishes the reactivity limit of $3.00 for secured experiments. Because the experiment is held stationary in the reactor, the likelihood that it will fall away from the core to produce an undesirable positive step increase in reactivity is minimized. In TS 3.8.1.b, the reactivity limit is designed to be at or below the analyzed maximum allowable pulse size of $3.00. Similar to Specification a, if this amount of reactivity is inadvertently added to the reactor, it will not have an unacceptable effect on the reactor system as demonstrated in the analysis of positive excess reactivity insertion in the SAR, Chapter 13.3.

In TS 3.8.1, Specification c, the sum of the absolute value of the reactivity worth of all experiments shall be less than $3.00. This v

The NRC staff reviewed the reactivity limits established in TS 3.8.1, Specifications a, b, and c, above, and determined that the specifications include the determination of SDM and excess reactivity, as provided in TS 3.1.2 and TS 3.1.3. TS 3.8.1 helps to ensure that positive reactivity insertion events are properly controlled by the UCINRF staff during experiments. The NRC staff finds that TS 3.8.1, Specifications a, b, and c are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 (Ref. 33). On the basis of its review of the information provided above, the NRC staff concludes that TS 3.8.1 is acceptable.

5.3.8.2 TS 3.8.2 Materials

TS 3.8.2, Materials, states:

<u>Specification(s).</u> The reactor shall not be operated unless the following conditions governing experiments exist:

a. fueled experiments shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 0.02 curies, and the strontium-90 inventory does not exceed 1 microcurie; and

b. explosive materials in the amount of 25 milligrams of TNT (or equivalent) or lesser quantities may be irradiated provided that the pressure produced upon accidental detonation of the explosive has been calculated and/or experimentally determined to be less than half the design pressure of the container; and

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

TS 3.8.2, Specification a, follows the guidance in NUREG-1537, Appendix 14.1, Section 3.8.2, and requires an inventory limit of specific fission products, such as iodine (I) and strontium. Limiting the iodine content of a fueled experiment to 0.02 Ci and the strontium-90 content to 1 μCi, is acceptable to the NRC staff because it limits exposure to individuals during the evacuation of the reactor room and unrestricted areas from a potential fueled experiment failure or malfunction to within the limits of 10 CFR Part 20, "Standards for Protection against Radiation." The maximum hypothetical accident (MHA) (discussed in Section 4.1.1 of this SER) bounds the limits in Specification a. The NRC staff reviewed TS 3.8.2, Specification a, and finds this consistent with the guidance in Regulatory Guide (RG) 2.2 (Ref. 57).

TS 3.8.2, Specification b, limits the quantity of explosive material to 25 mg or less. Explosive material 25 mg or less may be irradiated, provided the pressure produced on detonation of the explosive has been calculated or experimentally demonstrated to be less than half the design pressure of the irradiation container. This specification helps ensure that no damage to the fuel cladding or reactor will result because of a failure of an experiment containing explosive material. The NRC staff reviewed TS 3.8.2, Specification b, and finds that this specification is consistent with the recommendations of RG 2.2, "Development of Technical Specifications for Experiments in Research Reactors" (Ref. 57). TS 3.8.2, Specification b, also implements the recommendations in NUREG-1537, Appendix 14.1, Section 3.8.2, regarding experiments that have explosive content by limiting the amount to an upper limit of 25 mg of trinitrotoluene or equivalent. The pressure produced on accidental detonation of the explosive was calculated (Ref. 64) to ensure proper design pressure of the container. The NRC staff reviewed the calculation and finds that the licensee can properly analyze samples of explosive material.

TS 3.8.2, Specification c, follows the guidance provided in NUREG-1537, Appendix 14.1, Section 3.8.2, and requires the double encapsulation of corrosive materials as a means to reduce the likelihood that the encapsulation could fail and the corrosive material damage the fuel cladding or other reactor components.

On the basis of its review, the NRC staff finds that TS 3.8.2 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 3.8.2 is acceptable.

5.3.8.3 TS 3.8.3 Failures or Malfunctions

This TS is evaluated in Section 2.1.2 of this SER and is acceptable.

5.3.9 TS 3.9 This Section Intentionally Left Blank

The licensee left this section blank because there are no facility-unique LCO. The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.9, Facility-Specific LCO, states that the licensee may have facility-unique LCO based on the design of the facility. Because the NRC staff has not identified any need for facility-unique LCO, the NRC staff finds this acceptable.

5.4 <u>Surveillance Requirements</u>

TS 4.0, General, states:

> <u>Specification(s).</u>
>
> a. Surveillance requirements may be deferred during prolonged (periods greater than 1 month) reactor shutdown (except Technical Specifications 4.2.j, 4.3.a, 4.3.b, 4.3.d, and 4.3.e.). However, they shall be completed prior to reactor start-up unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor start-up. Scheduled surveillance which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.
>
> b. All replacements, modifications, and changes to systems having a safety related function including the ventilation system, the core and its associated support structure, the pool, the pool coolant system, the control rod drive mechanisms, and the reactor safety system shall meet or exceed the requirements of the original system or component. A safety system shall not be considered operable until it has been properly tested to meet specifications.

TS 4.0, Specification a, specifies which TS can be deferred when the reactor is not operating and how the deferred surveillance will be performed when reactor operation resumes. This allows operational flexibility that does not affect reactor safety.

TS 4.0, Specification b, requires that changes to certain important systems be controlled to at least their original design and fabrication specifications and be tested for operability before use.

TS 6.2.3 requires the Reactor Operations Committee (ROC) to review and approve all proposed changes such as those listed in TS 4.0, Specification b.

TS 4.0, Specification a and Specification b, help ensure that the quality of systems and components will be maintained to their original design specifications. The NRC staff finds that TS 4.0 provides appropriate UCINRF surveillance practices and is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. Furthermore, TS 4.0 helps ensure that the quality of systems and components is maintained, the operation will be conducted within SL, and the LCO will be satisfied. Therefore, based on the information provided above, the NRC staff concludes that TS 4.0 is acceptable.

5.4.1 TS 4.1 Reactor Core Parameters

This TS is evaluated in Section 2.5.1 and Section 2.5.2 of this SER and is acceptable.

5.4.2 TS 4.2 Reactor Control and Safety Systems

TS 4.2, Reactor Control and Safety Systems, states:

Specification(s).

a. A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually or immediately following any significant (>$0.25) core configuration change.

b. Control rod scram times for all four control rods shall be determined annually or for individual rods immediately following any maintenance work involving that control rod or drive mechanism that may have affected rod scram performance.

c. All control rods shall be visually inspected for deterioration quinquennially.

d. The transient (pulse) rod pneumatic cylinders and the associated air supply systems shall be inspected annually, and cleaned and lubricated if necessary.

e. On each day that pulse mode operation of the reactor is planned, the transient (pulse) rods shall be verified to be operable before pulse operation is initiated.

f. A channel test of each of the reactor safety system channels and interlocks in Technical Specification 3.2 Tables 2 and 3, except for the pool water temperature measuring channel, shall be performed prior to each day's operation or prior to each operation extending more than one day.

g. A channel check of the functions of the seismic switch shall be performed annually or as soon as possible after an observed seismic event or one reported to be of sufficient magnitude to trip the switch.

h. A calibration of the pool water temperature measuring channel shall be performed annually to include verification of the alarm set point.

i. A calibration of the fuel temperature measuring channel shall be performed annually.

j. A calibration of the pool water level measuring channel shall be performed annually to include verification of the alarm set point.

TS 4.2, Specification a, requires that the reactor power measuring channels be calibrated annually using the calorimetric method to help ensure the accuracy of the linear, log, and percent power channels as required to support TS 3.2.2.

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

TS 4.2, Specification d and Specification e, help ensure the operability of the ATR and FTR to prevent unexpected malfunctions during pulse operation.

TS 4.2, Specification f, requires that a channel test of the reactor safety system and interlocks be performed before each day's operation to help ensure the operability of these safety channels as required by TS 3.2.3 and helps ensure the fuel temperature SL is not exceeded.

TS 4.2, Specification g, helps ensure the proper functioning of the seismic switch.

TS 4.2, Specification h and Specification j, require a channel check and annual calibration of the pool water temperature and water level measuring channels to help ensure the pool temperature and water level remain in specification as required to support TS 3.3.2 and 3.3.1.

TS 4.2, Specification i, requires an annual channel calibration of the fuel element measuring channel to help ensure the LSSS of 425 °C (797 °F) is not exceeded.

The NRC staff finds that the surveillances in TS 4.2 of reactor control and safety systems, including fuel element parameters and fuel element temperature safety channels and their intervals, are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds that these surveillance frequencies will help ensure performance and operability of the fuel elements, including the fuel element temperature measuring systems and components.

The NRC staff reviewed TS 4.2, Specification g, against the frequency recommended in NUREG-1537 and ANSI/ANS-15.1-2007 for backup shutdown systems. ANSI/ANS-15.1-2007 recommends an annual surveillance. Although the seismic trip function is not a backup shutdown system, it is a component of the reactor safety system that provides a reactor scram. The NRC staff has previously approved an annual surveillance frequency at two other research and test reactors (RTRs) (Idaho State University and the University of New Mexico) that incorporate seismic switches. The TSs for these RTRs specify annual performance of channel checks on the seismic switch. Based on the information provided above, the NRC staff concludes that TS 4.2, Specification g, is acceptable.

5.4.3 TS 4.3 Reactor Pool Water

TS 4.3, Reactor Pool Water, states:

Specification(s).

a. A channel check of the pool water level measuring channel shall be performed monthly to include verification of the alarm reporting system.

b. The pool water conductivity shall be measured at the end of each operating day, or at shutdown for a period of operation extending more than one day. For periods of extended shutdown, the conductivity measurement shall be made monthly.

c. The pool water temperature shall be monitored each hour during reactor operation.

d. The pool water radioactivity shall be measured quarterly.

e. The pool water loss rate shall be evaluated on each occasion when make-up water is added to the pool. Any unusual increase in loss rate shall be investigated as a possible pool leak before any further reactor operation.

TS 4.3, Specification a, provides for surveillance intervals that help to ensure that the equipment for monitoring the pool water level is functioning properly. In addition, TS 4.0, Specification a, prevents deferring of performance of TS 4.3, Specification a, surveillance during periods of prolonged reactor shutdown, to provide an early indication of a possible pool leak.

TS 4.3, Specifications b and d, provide for surveillance intervals for determining the quality of the pool water, including its radioactive content, and help ensure the pool water remains functional as a reflector and moderator for the reactor core. In addition, TS 4.0, Specification a, prevents deferring the performance of TS 4.3, Specifications b and d, surveillance during periods of prolonged reactor shutdown to provide an early indication of a possible pool leak or fuel clad failure.

TS 4.3, Specification c, provides for surveillance intervals for monitoring pool water temperature during reactor operation to help ensure that the pool water temperature limit is not exceeded.

TS 4.3, Specification e, requires monitoring of pool water makeup to provide an early indication of a possible pool leak. In addition, TS 4.0, Specification a, prevents deferring of performance of TS 4.3, Specification e, for continuous indication of a possible pool leak during a prolong reactor shutdown.

The NRC staff reviewed the requirements for TSs regarding pool water level and quality. The NRC staff finds that the water level and the water quality monitoring as required in TS 4.3 are adequate to help ensure that the water level required above the core is maintained, any significant pool water leakage is detected and corrected by the licensee staff, and the pool water quality is maintained. Based on the information provided above, the NRC staff also finds that the possibility of a significant release to the environment resulting from pool leakage is extremely low. On this basis, the NRC staff concludes that TS 4.3 is acceptable.

5.4.4 TS 4.4 This Section Intentionally Left Blank

The licensee left this section blank. The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.4, Containment or Confinement, indicates that the licensee should specify the surveillance requirement for pressure or flow rate needed to maintain confinement. Because this licensee put the surveillance requirements in TS 4.5, there is no need to include these surveillance requirements in TS 4.4. The NRC staff reviewed TS 4.5 and finds that the surveillance requirements for the ventilation system are consistent with the guidance in NUREG-1537 for confinement. This is evaluated in Section 5.4.5 of this SER.

5.4.5 TS 4.5 Ventilation Systems

TS 4.5, Ventilation Systems, states:

Specification(s).

a. A channel check of the existence of negative air pressure between the reactor room and the control room, and the reactor room and the outside air in both normal and emergency modes shall be performed daily.

b. A channel check of the exhaust flow rates from the reactor area in both normal and emergency modes shall be performed daily, to demonstrate that the ventilation system is operable in both normal and emergency modes by observation of flow rates, and valve/damper action.

(…)

TS 4.5, Specification a and Specification b, help ensure that the ventilation system is operational as described in the SAR, as supplemented, and satisfy the analysis assumptions of the SAR accident analysis and TS 3.5. The NRC staff reviewed TS 4.5, Specification a and Specification b, for the ventilation system and finds that TS 4.5, Specification a and Specification b, help to ensure the assumptions in SAR Section 3.6 and is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007.

Based on its review of the information provided above, the NRC staff concludes that TS 4.5, Specifications a and b, are acceptable.

TS 4.5, Specification c, is evaluated in Section 3.1.4 of this SER and is acceptable.

5.4.6 TS 4.6 This Section Intentionally Left Blank

The licensee left this section blank. The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 4.6, Emergency Electrical Power System, states that channel checks and operability checks should be performed. The licensee stated that the emergency electric power system is not necessary to safely shut down the reactor and is not required to ensure protection of public health and safety. This is evaluated in Section 1.10 of this SER and is acceptable.

5.4.7 TS 4.7 Radiation Monitoring System and Effluents

This TS is evaluated in Section 3.1.1.1, Section 3.1.1.2, and Section 3.1.4 of this SER and is acceptable.

5.4.8 TS 4.8 Experiment Limits

TS 4.8, Experiment Limits, states:

Specification(s).

a. No experiment shall be installed in the reactor unless a safety analysis has been performed and reviewed in accordance with Technical Specifications 3.8 and 6.5.

b. The reactivity worth of a new experiment shall be verified at a power level less than 2 watts, before reactor operation at higher power with the experiment.

TS 4.8, Specification a, requires that, before installation of an experiment in the reactor, a safety analysis be performed to show that the experiment meets the requirements of TS 3.8, “Limitations on Experiments,” and TS 6.5, “Experiment Review and Approval.”

TS 4.8, Specification b, requires that the calculated reactivity worth of new experiments be confirmed at low reactor power before proceeding with the experiment at higher power.

The NRC staff reviewed TS 4.8 surveillance requirements for controlling experiments and finds that TS 4.8 helps to ensure that the requirements of TS 3.8 for experiments are met. The NRC staff also finds that TS 4.8 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 4.8 is acceptable.

5.5 Design Features

5.5.1 TS 5.1 Site and Facility Description

TS 5.1, Site and Facility Description, states:

> Specification(s).
>
> The site shall be the reactor facility as described below.
>
> The reactor facility shall be a restricted access area consisting of a main area, two associated laboratory areas, and a control room on a single level in the basement of Rowland Hall, on the University of California Irvine campus. The minimum free air volume of the reactor area including the two associated laboratories shall be 23,000 cubic feet. Normal entry to these areas shall be restricted to a single doorway from the control room. Large doors shall be provided to the adjacent loading dock to provide emergency egress and/or access for incoming or outgoing large items. Full visibility shall be provided between the control room and the reactor area. The reactor shall be housed in a closed area designed to restrict leakage.

TS 5.1 and the definition of reactor facility define the licensed area of the UCINRF. This clarifies the physical boundary of U.S. Nuclear Regulatory Commission (NRC)-licensed space. TS 5.1 describes the UCINRF site boundary. The minimum free volume of 23,000 ft^3 maintained by the facility provides a dilution capacity. This free volume limits the dose to UCINRF staff from any released airborne radioactivity from reactor operations within the reactor room. Calculations in the licensee's SAR, and RAI responses (Refs. 12-15 and 20), demonstrate that occupational exposures for the MHA are kept below 10 CFR Part 20 limits under accident conditions. The licensee's annual reports between 2011 and 2015 submitted to the NRC indicate that airborne effluents, which contain mostly argon-41, released to the general environment and associated with reactor operation are significantly below the limits in Table 2 of 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."

The NRC staff reviewed TS 5.1 and finds that TS 5.1 provides important features of the physical design of the UCINRF and defines the boundaries of the facility being licensed. These specifications support the accident analysis performed to meet 10 CFR Part 20 requirements, define the operational and site-area boundaries for the facilities, and are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 5.1 is acceptable.

5.5.2 TS 5.2 Reactor Coolant System

This TS is evaluated in Section 2.3 of this SER and is acceptable.

5.5.3 TS 5.3 Reactor Core and Fuel

5.5.3.1 TS 5.3.1 Reactor Core

This TS is evaluated in Section 2.2 of this SER and is acceptable.

5.5.3.2 TS 5.3.2 Control Rods

This TS is evaluated in Section 2.2.2 of this SER and is acceptable.

5.5.3.3 TS 5.3.3 Reactor Fuel

This TS is evaluated in Section 2.2.1 of this SER and is acceptable.

5.5.4 TS 5.4 Fuel Storage

On May 13, 2016 (Ref. 86), the licensee requested a revision to proposed TS 5.4 submitted on April 22, 2016 (Ref. 76). The licensee added fueled devices to be stored in a geometrical array to TS 5.4, Specification a. The licensee also revised the irradiated fuel element and fueled device storage temperature limit to design limit in TS 5.4, Specification b. The revised TS 5.4 is provided below.

TS 5.4, Fuel Storage, states:

> Specification(s).
>
> a. All fuel elements and fueled devices shall be stored in a geometrical array where the k_{eff} is less than 0.80 for all conditions of moderation and reflection.
>
> b. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed design limit.
>
> c. Fuel elements or fueled devices showing evidence of damage (see Technical Specification 3.1.6) shall be stored separately from fuel not suspected to be damaged, and shall be checked for fission product leakage.

TS 5.4, Specification a, limits the effective multiplication factor (k_{eff}) value to 0.8, which is more restrictive than the k_{eff} value of 0.9 recommended in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff finds TS 5.4, Specification a, establishes a sufficient reactivity margin to guard against accidental criticality of fuel elements and fueled devices in storage and is consistent with NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff finds TS 5.4, Specification a, acceptable.

TS 5.4, Specification b, helps ensure that irradiated fuel elements and fuel devices be stored in an array that will permit sufficient natural convection cooling by water or air such that the temperature of the fuel element or fueled device will not exceed design. The NRC staff finds that TS 5.4, Specification b, is consistent with other TRIGA® facilities and the guidance provided in NUREG-1537. Based on the information provided above, the NRC staff concludes that TS 5.4, Specification b, is acceptable.

The NRC staff finds that TS 5.4, Specification c, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff finds TS 5.4, Specification c, acceptable.

5.5.5 TS 5.5 Ventilation System

TS 5.5, Ventilation System, states:

> <u>Specification(s).</u>
>
> a. The ventilation system shall operate in either normal or emergency mode. The ventilation system shall consist of ducts, blowers, dampers, flow and pressure measurement devices, and exhaust points above the roof of Rowland Hall.
>
> b. During normal operations, the ventilation system shall be capable of exhausting air or other gases from the reactor area at a rate of 3600 cubic feet per minute (cfm).
>
> c. During normal operation the ventilation system shall be capable of maintaining a minimum of 0.10 inches of water pressure differential between the reactor area and the control room, and between the reactor area and the outside air.
>
> d. During emergency situations involving release of radioactive materials into the air, an emergency exhaust with a high efficiency particulate arrestance (HEPA) filter shall be provided to exhaust a minimum of 240 cfm from the reactor area.
>
> e. Shutdown of the normal reactor area exhaust system and start-up of the emergency exhaust system shall be initiated by a high radioactive particulate count rate alarm signal originating in the reactor room, or a manual switch in the control room.
>
> f. During all modes of operation, the ventilation system shall exhaust at a minimum height of 90 feet above ground level.

TS 5.5, Specification a, generally describes the ventilation system (discussed in Sections 1.8, 3.1.1.1, and 4.1.1 of this SER).

TS 5.5, Specification b, provides the flow rate of the ventilation system in normal mode and is consistent with the flow rate used in calculations of radiological effects on building occupants and the environment.

TS 5.5, Specification c, provides for a negative pressure within the reactor facility to control the release of radioactive materials during normal operation.

TS 5.5, Specification d and Specification f, present the requirement to have a ventilation system with a filtered and controlled air pathway release point and release rate during emergency operation. As discussed in the licensee's SAR Section 3.6 (Ref. 20), the height of the exhaust stack helps to ensure dispersion and dilution of effluents released from the stack before they reach the ground.

TS 5.5, Specification e, provides for automatic initiation of the emergency exhaust system upon high particulate or gaseous radioactivity as a result of a fuel element failure.

TS 5.5, Specification f, further specifies the minimum height requirement for release of exhausted gases from the ventilation system, ensuring adequate dispersion with outside air. This specification helps limit the concentration of any airborne radioactivity released to the environment from reactor operations.

The NRC staff reviewed TS 5.5, Specifications a through f, and finds that TS 5.5 provides important design features of the ventilation system. These specifications support the SAR accident analysis, as supplemented by responses to RAI, and are consistent with the dose analysis performed to satisfy 10 CFR Part 20 requirements. The NRC staff also finds TS 5.5 consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 5.5 is acceptable.

5.6 Administrative Controls

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

The primary guidance for the development of administrative control for research reactor operations is NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's TSs are based on these standards. In some cases, the proposed wording of some TSs was not identical to that in NUREG-1537 and ANSI/ANS-15.1-2007. However, as discussed below, the NRC staff considered these instances and determined that the licensee's proposed administrative controls meet the intent of the guidance and are acceptable.

5.6.1 TS 6.1 Organization and Structure

5.6.1.1 TS 6.1.1 Structure

TS 6.1.1, Structure, states:

> The reactor facility is housed in the School of Physical Sciences of the University of California, Irvine. The reactor is related to the University structure of positions shown in the organization chart, Figure 1, where solid lines represent direct reporting responsibility, dashed lines indicate working relationships.

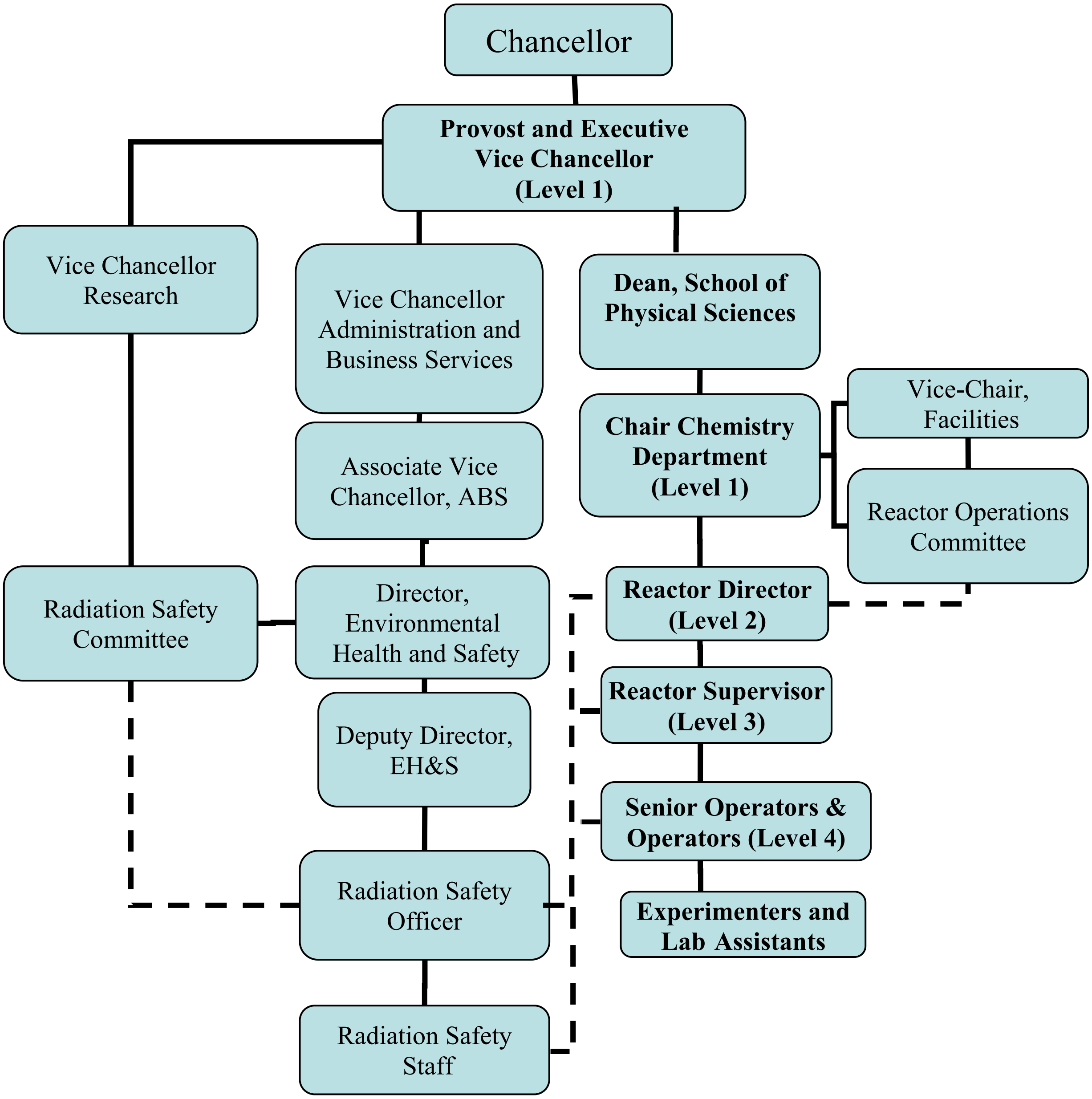

Figure 1 UCI Reactor Organization Chart

TS 6.1.1 ensures that the UCINRF organization structure is delineated, as described in TS 6.1.1, Figure 1. The NRC staff reviewed TS 6.1.1, Figure 1, and finds that Figure 1 identifies the reporting and communication relationships between the organizational units for the UCINRF and is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1.1 and TS 6.1.1, Figure 1 are acceptable.

5.6.1.2 TS 6.1.2 Responsibilities

TS 6.1.2, Responsibilities, states:

> a. The licensee of the reactor is the Board of Regents of the University of California, which has delegated authority for license matters to the Provost and Executive Vice Chancellor (Level 1) of the University of California, Irvine.
>
> b. The reactor facility is under the direction of a Reactor Director (Level 2) who shall be a tenure member of the University of California Irvine faculty. The Reactor Director shall report to the Chair of the Chemistry Department (Level 1), who, in turn, shall be responsible to the Dean of the School of Physical Sciences.
>
> c. Operations shall be supervised by the Reactor Supervisor (Level 3) who shall hold a valid senior operator's license for the facility. This position shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, the provisions of the Reactor Operations Committee and the provisions of the Radiation Safety Committee.
>
> d. Reactor operators (Level 4) shall be responsible for operation of the reactor and performing needed maintenance and surveillance, including radiological safety and necessary supervision of experimenters. Senior reactor operators shall assume duties for supervision of operators as required by the US Nuclear Regulatory Commission (NRC) in Part 55 of 10 CFR, and Technical Specification 6.1.3.
>
> (…)
>
> f. In the event of absence, or during filling of appointments to specific positions, temporary duties and responsibilities may be carried out by the person next higher or lower in line in the organization chart, provided the individual meets the basic qualifications for both positions.

TS 6.1.2, in conjunction with the organizational chart (Figure 1), presents both the responsibilities and organization of individuals associated with the facility. TS 6.1.2 shows the organizational responsibilities and arrangement for the radiation protection function. Figure 1 identifies the reporting and communication relationships between the organizational units. TS 6.1.2 also describes the organization and responsibilities of individuals in direct control of the facility. The NRC staff reviewed TS 6.1.2 and finds that the organizational responsibilities delineated in TS 6.1.2 are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, 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, states:

a. The minimum staffing when the reactor is not secured shall include:
   i. a licensed operator present in the control room; and
   ii. a second designated individual present within Rowland Hall able to carry out prescribed instructions and with the ability to check on the safety of the licensed operator and to act in the event of emergency; and
   iii. a licensed Senior Operator (SRO) readily available on call. Readily available on call means the SRO has been specifically designated, the designation is known to the operator on duty, the SRO can be rapidly contacted by phone by the operator on duty, and the SRO is capable of arriving at the reactor facility within 30 minutes under normal conditions.

b. A list of reactor facility personnel and other persons responsible for radiological safety and security on campus shall be kept in the reactor control room for use by an operator or experimenter. The list shall include telephone numbers of the Reactor Director, the Reactor Supervisor, the campus Radiation Safety Officer and other back-up radiological safety personnel, reactor operators, senior reactor operators, and personnel with responsibilities for maintenance in Rowland Hall.

c. Experimenters using the facility shall be certified by the campus Radiation Safety program as trained and authorized to use radioactive materials. The training shall include both general radiological training, including features of the ALARA program and specialized training in procedures for using reactor auxiliary experimental equipment (such as transfer systems), carrying out necessary surveys and record-keeping necessary for proper handling of radioactive materials within the reactor facility. Experimenters so trained and authorized shall be responsible for their own personal and sample/apparatus monitoring.

d. The following events shall require the presence in the facility of a licensed Senior Reactor Operator.
   i. Initial start-up and approach to power and final daily shutdown.
   ii. Fuel or control-rod relocations within the core region.
   iii. Insertion, removal, or relocation of any experiment worth more than \$1.00.
   iv. Restart following any unplanned or unscheduled shutdown, or unexpected power decrease of >10%.

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

TS 6.1.3, Specification b, describes the key personnel responsible for radiological safety and security on campus and their contact information for establishing immediate communication with the UCINRF operating staff.

TS 6.1.3, Specification c, describes mandatory training in radioactive materials required for experimenters using the UCINRF.

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

The NRC staff reviewed TS 6.1.3 and finds that the requirements of TS 6.1.3 are in accordance with the requirements of 10 CFR 50.54(k) and 54(m) and consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.1.3 is acceptable.

5.6.1.4 TS 6.1.4 Selection and Training of Personnel

TS 6.1.4, Selection and Training of Personnel, states:

> The selection, training, and requalification of operations personnel shall meet the requirements of ANSI/ANS-15.4-2007.

TS 6.1.4 establishes the criteria for the training and requalification program for operations personnel. The licensee used ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors" (Ref. 58), as guidance for selecting and training personnel. The NRC staff reviewed TS 6.1.4 and finds that TS 6.1.4 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1.4 is acceptable.

5.6.2 TS 6.2 Review and Audit

TS 6.2, Review and Audit, states:

> A Reactor Operations Committee (ROC) shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. Review and audit of radiological safety at the facility shall be carried out by the Radiation Safety Committee (RSC).

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

5.6.2.1 TS 6.2.1 ROC Composition and Qualifications

TS 6.2.1, ROC Composition and Qualifications, states:

> The ROC shall have at least five voting members, at least one of whom shall be a health physicist designated by the Office of Environmental Health and Safety of the University. The Committee as a whole shall be knowledgeable in nuclear science and issues related to reactor and/or radiological safety. The membership shall include at least two members who are not associated with the Department of Chemistry. Approved alternates may serve in the absence of regular members. Members and alternates and a chairperson for the committee shall be appointed by the Chair of the Department of Chemistry (Level 1) or higher authority. The Reactor Director and Reactor Supervisor shall be non-voting members of the committee.

The NRC staff reviewed TS 6.2.1 and finds the composition and qualifications for the ROC consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007, Section 6.2.1. On this basis, the NRC staff concludes that TS 6.2.1 is acceptable.

5.6.2.2 TS 6.2.2 ROC Charter and Rules

TS 6.2.2, ROC Charter and Rules, states:

> The following responsibilities shall constitute the charter of the ROC.
>
> a. Meeting at least annually, with provision for additional meetings when circumstances warrant to assure safety at the facility.
>
> b. A quorum shall consist of not less than a majority of the voting members and shall include the chairperson or his/her designee. A quorum shall not consist of a majority of operations staff.
>
> c. Review and audit of facility staff and operations as indicated in Technical Specifications 6.2.3 and 6.2.4.
>
> d. Designation of individuals to perform audits of facility operations and records.
>
> e. Preparation, approval, and dissemination of minutes of meetings.
>
> f. Preparation and dissemination of findings and other reports as needed to assure safe operations of the reactor.
>
> g. Approval of individuals for the supervision and operation of the reactor.

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

5.6.2.3 TS 6.2.3 ROC Review Function

TS 6.2.3, ROC Review Function, states:

> The following review functions shall be the responsibility of the ROC.
>
> a. Review and approval of all proposed changes to the facility, its license, procedures, ROC charter, and Technical Specifications, including those made under provisions of 10 CFR § 50.59, and the determinations leading to decisions relating to 10 CFR § 50.59 approvals.
>
> b. Review and approval of new or changed procedures, experiments, components, or instrumentation having safety significance.
>
> c. Review of the quality assurance program implementation applicable to the reactor components.
>
> d. Review of new experiments or changes in experiments that could have reactivity or safety significance.
>
> e. Review of violations of technical specifications, license, or violations of procedures or instructions having safety significance.
>
> f. Review of operating abnormalities that have safety significance.
>
> g. Review of actions and reports listed in Technical Specifications 6.6.1, 6.6.2, or 6.7.2.
>
> h. Review of audit reports, including reports from the campus Radiation Safety Officer, regarding the radiation protection program.

TS 6.2.3 establishes the ROC review functions to help ensure the safety of facility operations. The NRC staff reviewed TS 6.2.3 and finds that TS 6.2.3 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.2.3 is acceptable.

5.6.2.4 TS 6.2.4 ROC Audit Function

TS 6.2.4, ROC Audit Function, states:

> The ROC shall perform audits or review audits performed by designated individuals on its behalf at least annually. The audit shall include, but not be limited to, the following items.
>
> a. Facility operations for conformance to the Technical Specifications and applicable license conditions.
>
> b. Retraining and requalification of operators according to the Requalification Plan.
>
> c. The result of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, procedures or methods of operation that affect reactor safety.
>
> d. The facility Emergency Plan (EP) and implementing procedures including written reports of any drills or exercises carried out.

e. At least one of the auditors shall be familiar with reactor operations but not directly responsible for any portion of reactor operations.

f. Any deficiencies identified in an audit that affect reactor safety shall be immediately reported to the chairperson of ROC, and to the Level 1 administrator. A written full report shall be submitted to ROC within 3 months of any audit.

TS 6.2.4 establishes the ROC audit function's scope and independence requirements. The NRC staff reviewed TS 6.2.4 and finds TS 6.2.4 consistent with the guidance provided with NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.2.4 is acceptable.

5.6.3 TS 6.3 Radiation Safety

This TS is evaluated in Section 3.1.2 of this SER and is acceptable.

5.6.4 TS 6.4 Operating Procedures

TS 6.4, Operating Procedures, states:

> Written procedures, reviewed and approved by the ROC and the Reactor Director, shall be in effect and implemented for the following listed items. The procedures shall be adequate to assure the safety of the reactor but not preclude the use of independent judgment and action should the situation require such. Any changes to procedures shall be made in accordance with the requirements of 10 CFR § 50.59.
>
> a. Startup, operation, and shutdown of the reactor.
>
> b. Installation or removal of fuel elements, control rods, experiments, and experimental facilities.
>
> c. Maintenance of major components of systems that could have an effect on reactor safety.
>
> d. Surveillance checks, calibrations and inspections required by the technical specifications or that could have an effect on reactor safety.
>
> e. Personnel radiation protection, including provisions to maintain personnel exposures as low as reasonably achievable (ALARA).
>
> f. Administrative controls for operations and maintenance, and for the conduct of irradiations or experiments that could affect reactor safety.
>
> g. Implementation of required plans including Emergency (EP) and Physical Security (PSP) plans.
>
> h. Use, receipt, and transfer of by-product materials.

TS 6.4, Specifications a through h, help ensure that procedures are written, reviewed, and approved prior to performing any of the activities listed in TS 6.5. The NRC staff finds that TS 6.4 is consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.4 is acceptable.

5.6.5 TS 6.5 Experiment Review and Approval

TS 6.5, Experiment Review and Approval, states:

> Approved experiments shall be carried out in accordance with established and approved procedures. Procedures for experiment review and approval shall include the following requirements.
>
> a. All new experiments or class of experiment shall be reviewed and approved by the ROC and approved in writing by the Reactor Director. The review shall include analysis by the RSO or other designated radiation safety personnel.
>
> b. Changes to existing experiments or classes shall be made only after review by the ROC and RSO or the RSO designee, following performance of a 10 CFR 50.59 evaluation and the conclusion that the proposed changes do not require prior NRC approval.

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

5.6.6 TS 6.6 Required Actions

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

TS 6.6.1, Actions To Be Taken in Case of a Safety Limit Violation, states:

> In the event the safety limit on fuel temperature is exceeded:
>
> a. the reactor shall be shut down and the event reported immediately to the Reactor Director, the ROC chairperson, and the RSO. Reactor operation shall not be resumed until authorized by the NRC; and
>
> b. the event shall be reported within the next working day to the NRC Headquarters Operations Center; and
>
> c. a follow-up written report shall be reviewed by the ROC and sent within 14 days to the NRC (according to provisions of Technical Specification 6.7) describing:
>
>   i. applicable circumstances leading to the violation including, where known, the cause and contributing factors; and
>
>   ii. effects of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public; and
>
>   iii. the basis for corrective action taken to preclude recurrence.

TS 6.6.1, Specifications a through c, require the facility to shut down in the event that an SL is exceeded. The facility may not resume operation without authorization from the NRC. The violation must also be reported to the ROC and the NRC. The reporting requirements are detailed in TS 6.6.1, Specification c, and TS 6.7.2, specifying that the NRC must be notified not later than the following workday by telephone and a report is required to be submitted to the NRC within 14 days. TS 6.6.1, Specification c.iii, specifies that corrective actions are to be taken to prevent recurrence.

The NRC staff finds that TS 6.6.1 meets the requirements of 10 CFR 50.36(c)(1) for actions to be taken if an SL is exceeded. The NRC staff also finds that the actions the licensee proposes are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On this basis, the NRC staff concludes that TS 6.6.1 is acceptable.

5.6.6.2 TS 6.6.2 Actions To Be Taken in the Event of an Reportable Occurrence Other than a Safety Limit Violation

TS 6.6.2, Actions to be Taken in the Case of Events Other than a Safety Limit Violation, states:

> In the event of an occurrence of the type listed in items 1-7 below:
>
> a. the reactor shall be secured and the Reactor Director and Supervisor notified; and
>
> b. operation shall not be resumed until authorized by the Reactor Director and the ROC; and
>
> c. a follow-up written report shall be reviewed by the ROC and sent within 14 days to the NRC (according to provisions of Technical Specification 6.7) describing:
>
>   i. applicable circumstances leading to the violation including, where known, the cause and contributing factors; and
>
>   ii. effects of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public; and
>
>   iii. the basis for corrective action taken to preclude recurrence.
>
> 1. Release of radioactivity from the site above allowed limits.
> 2. Operation with actual safety system settings for required systems less conservative than the limiting safety system settings in these specifications.
> 3. Operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in Technical Specification section 3.
> 4. A required reactor safety system component malfunction that renders or could render the safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required.
> 5. An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from known cause are excluded.

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

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

TS 6.6.2 requires the reactor to be secured in case of a reportable occurrence. The event and corrective actions taken must also be reported to the facility management and the NRC. The NRC staff reviewed TS 6.6.2 and finds that the actions the licensee proposes in TS 6.6.2 are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007 for reactor safety. On this basis, the NRC staff concludes that TS 6.6.2 is acceptable.

5.6.7 TS 6.7 Reports

TS 6.7, Reports, states:

> In addition to the requirements of applicable regulations, and in no way substituting for them, reports shall be made to the NRC as listed below. All written reports shall be directed to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001.

TS 6.7 specifies the location for submitting required reports to the NRC. The NRC staff reviewed TS 6.7 and finds that TS 6.7 conforms to the reporting requirements in 10 CFR 50.4, "Written Communications," and is consistent with the guidance provided in NUREG-1537. The NRC staff concludes that TS 6.7 is acceptable.

5.6.7.1 TS 6.7.1 Annual Operating Report

TS 6.7.1, Annual Operating Report, states:

> A routine written annual report shall be submitted by the Reactor Director to the NRC, by September 30th each year regarding operations for the preceding academic year (July 1st through June 30th). The report shall include, but not be limited to:
>
> a. a brief narrative summary of operating experience (including experiments performed) and a tabulation showing the energy generated by the reactor (in megawatt hours), the amount of pulse operation, and the number of hours the reactor was critical; and
>
> b. the number of unplanned shutdowns and inadvertent scrams, including the reasons therefore, and corrective actions taken (if any) to reduce recurrence; and
>
> c. a tabulation of major preventive and corrective maintenance operations having safety significance; and
>
> d. a tabulation of changes in the reactor facility and procedures, and tabulations of new experiments, including a summary of the analyses

leading to the conclusions that they are allowed without prior authorization by NRC and that 10 CFR § 50.59 was applicable; and

e. a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the facility as measured at or prior to the point of such release or discharge. The summary shall include, to the extent practicable, an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient; and

f. a summarized result of environmental surveys performed outside the facility; and

g. a summary of radiation exposures received by facility personnel and visitors, where such exposures are greater than 25% of that allowed.

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

5.6.7.2 TS 6.7.2 Special Reports

TS 6.7.2, Special Reports, states:

> In addition to reports required according to Technical Specification 6.6, a report shall be made in writing to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001 within 30 days of:
>
> a. permanent changes in facility organization involving Level 1 or 2 personnel; and
>
> b. significant changes in the transient or accident analyses as described in the SAR.

The NRC staff reviewed TS 6.7.2 and finds that TS 6.7.2, Specification a and b, special reporting requirements, are consistent with the guidance provided in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.7.2 is acceptable.

5.6.8 TS 6.8 Records

TS 6.8, Records, states:

> In addition to the requirements of applicable regulations, and in no way substituting therefore, records and logs shall be prepared and retained for periods as described here. Records may be in a variety of formats.

TS 6.8 specifies the retention periods for different types of records generated by the licensee. The NRC staff reviewed TS 6.8 and finds that the contents of TS 6.8 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 for the types of records and their retention period. On this basis, the NRC staff concludes that TS 6.8 is acceptable.

5.6.8.1 TS 6.8.1 5-Year Retention Records

TS 6.8.1, Records to Be Retained for a Period of at Least 5 Years or for the Life of the Component Involved If Less than 5 Years, states:

> a. Normal reactor facility operation, but not including supporting documentation such as checklists, log sheets, etc., which shall be retained for one year.
>
> b. Principal maintenance activities.
>
> c. Reportable occurrences.
>
> d. Surveillance activities required by the Technical Specifications.
>
> e. Reactor facility radiation and contamination surveys.
>
> f. Experiments performed with the reactor.
>
> g. Fuel inventories, receipts and shipments.
>
> h. Approved changes in operating procedures.
>
> i. ROC records of meetings and audit reports.

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

5.6.8.2 TS 6.8.2 Records To Be Retained At Least One Certification Cycle

TS 6.8.2, Records To Be Retained for At Least One Certification Cycle, states:

> Records of retraining and requalification of all licensed operators shall be retained at all times each individual has duties as an operator or until his or her license is renewed.

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

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

TS 6.8.3, Records To Be Retained for the Lifetime of the Reactor Facility, states:

> The following records shall be retained for the lifetime of the facility. Applicable annual reports containing this information may also be used as records.
>
> a. Reviews and reports pertaining to a violation of a safety limit, the limiting safety system setting, or a limiting condition for operation as described in Technical Specification 6.6.
>
> b. Gaseous and liquid radioactive effluents released to the environs.
>
> c. Results of off-site environmental monitoring surveys.

d. Radiation exposures for all personnel that were monitored.

e. Drawings of the reactor facility and safety related components.

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

5.7 Conclusions

The staff reviewed and evaluated the UCINRF TSs as part of its review of the application, as supplemented, for license renewal of Facility Operating License No. R-116, NRC Docket No. 50-326. The UCINRF TSs define certain features, characteristics, and conditions governing the operation of the facility. The TSs are explicitly included in the renewed license as Appendix A. The NRC staff reviewed and evaluated the content of the TSs to determine if they meet the requirements of 10 CFR 50.36. Based on its review, the NRC staff concludes that the proposed TSs meet the requirements of the regulations. The NRC staff also reviewed the format and content of the proposed TSs for consistency with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and finds that the proposed TSs are consistent with these guidelines. The NRC staff concludes that the UCINRF TSs are acceptable for the following reasons:

- To satisfy the requirements of 10 CFR 50.36(a), the licensee has provided proposed TSs with the application for license renewal. As required by the regulations, the proposed TSs include appropriate summary bases. Those summary bases are included in the TS but are not specifications required by the regulations.
- The UCINRF is a facility of the type described in 10 CFR 50.21(c), and, therefore, as required by 10 CFR 50.36(b), the facility license will include the TSs. To satisfy the requirements of 10 CFR 50.36(b), the licensee has provided TSs derived from analyses in the SAR, as supplemented by responses to RAI.
- The TSs acceptably implement the recommendations of NUREG-1537, Part 1, and ANSI/ANS-15.1-2007 by using definitions that are acceptable.
- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee has proposed a TS specifying an SL on the fuel temperature and an LSSS for the reactor protection system to prevent reaching the SL.
- The proposed TSs contain LCO on each item that meets one or more of the criteria specified in 10 CFR 50.36(c)(2)(ii).
- The proposed TSs contain surveillance requirements that satisfy the requirements of 10 CFR 50.36(c)(3).
- The proposed TSs contain design features that satisfy the requirements of 10 CFR 50.36(c)(4).
- The proposed TSs contain administrative controls that satisfy the requirements of 10 CFR 50.36(c)(5). The licensee’s administrative controls contain requirements for initial notification, written reports, and records that meet the requirements specified in 10 CFR 50.36(c)(1), (2), (7), and (8).

The NRC staff reviewed the proposed TSs and finds the proposed TSs acceptable. Specifically, the NRC staff concludes that normal operation of the UCINRF 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 occupational exposures. The NRC staff also concludes that the proposed TSs provide reasonable assurance that the facility will be operated as analyzed in the SAR and that adherence to the proposed TSs will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 4, “Accident Analysis,” of this SER.

SECTION 6

CONCLUSIONS

On the basis of its evaluation of the application, as discussed in previous chapters of this safety evaluation report, the NRC staff concludes the following:

- The application for license renewal dated October 18, 1999, as supplemented on April 24 and June 2, 2000; January 27, March 23, May 17, July 14, August 25 (two letters), October 20 (two letters), and October 29, 2010; June 7 (two letters), June 24, July 7, August 1, October 3, and December 2, 2011 (two letters); January 12, March 1, and September 11, 2012; February 26, March 5, March 7, July 11, and October 8, 2014 (two letters); December 22, 2015 (two letters); and April 22, April 29, and May 13, 2016, complies with the standards and requirements of 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* (CFR).
- The facility will operate in conformity with the application, as supplemented, as well as the provisions of the AEA and the rules and regulations of the NRC.
- There is reasonable assurance that (1) the activities authorized by the renewed operating license can be conducted at the designated location without endangering health and safety of the public and (2) such activities will be conducted in compliance with the rules and regulations of the NRC.
- 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 the health and safety of the public.
- The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of NRC regulations, and all applicable requirements have been satisfied.
- The receipt, possession and use of by-product and special nuclear materials as authorized by this facility operating license will be in accordance with NRC regulations in 10 CFR Parts 30 and 70.

SECTION 7

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